IAEA-TECDOC-1020

# Design measures for prevention and mitigation of severe accidents at advanced water cooled reactors

*Proceedings of a Technical Committee meeting held in Vienna, 21–25 October 1996*

INTERNATIONAL ATOMIC ENERGY AGENCY IAEA

June 1998

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DESIGN MEASURES FOR PREVENTION AND MITIGATION OF
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# FOREWORD

Over 8500 reactor-years of operating experience have been accumulated with the current nuclear energy systems. New generations of nuclear power plants are being developed, building upon this background of experience.

During the last decade, requirements for equipment specifically intended to minimize releases of radioactive material to the environment in the event of a core melt accident have been introduced, and designs for new plants include measures for preventing and mitigating a range of severe accident scenarios.

The IAEA Technical Committee Meeting on Impact of Severe Accidents on Plant Design and Layout of Advanced Water Cooled Reactors was jointly organized by the Department of Nuclear Energy and the Department of Nuclear Safety to review measures which are being incorporated into advanced water cooled reactor designs for preventing and mitigating severe accidents, the status of experimental and analytical investigations of severe accident phenomena and challenges which support design decisions and accident management procedures, and to understand the impact of explicitly addressing severe accidents on the cost of nuclear power plants. This publication is intended to provide an objective source of information on this topic.

The meeting was conducted within the frame of the IAEA's International Working Group on Advanced Technologies for Water Cooled Reactors.

The IAEA scientific secretaries for this task have been J. Cleveland of the Department of Nuclear Energy and M. Gasparini of the Department of Nuclear Safety.

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## SUMMARY

### 1. BACKGROUND

The nuclear industry has, in general, achieved a high level of safety with the nuclear power plants (NPPs) operating today. However, it is the tendency of any industry to learn from experience and improve its product. A key finding of the IAEA's International Conference on The Safety of Nuclear Power: Strategy for the Future (Vienna, September 1991), was that:

> "Advanced reactor designs will explicitly incorporate design features that would permit the technical demonstration of adequate public protection with significantly reduced emergency planning requirements, e.g. relief from the requirement for rapid evacuation. Potential future owners of these designs have encouraged incorporation of such design features, and although no consensus has been established to totally eliminate emergency planning, many desire to eliminate the more onerous aspects of current procedures, particularly rapid action requirements. Such modifications to emergency planning should be considered".

The 1991 IAEA General Conference in its Resolution GC(XXXV/RES/553(para. 9)) invited the Director General to start activities on developing safety principles for the design of future NPPs. In 1992, the International Nuclear Safety Advisory Group (INSAG), a body advising the IAEA Director General on safety issues, proposed desirable features for enhancing the safety of future nuclear power plants (see the "Safety of Nuclear Power", INSAG-5, IAEA, 1992). They incorporate improved safety concepts including those addressing human factors and specific design features. Regarding plant design, the features state that it should, in particular, reduce the probability and consequences of severe accidents, have confinement systems to cope with pressures and temperatures occurring during severe accidents, and adequately protect against sabotage and conventional armed attacks. Consideration should also be given to passive safety features that are based on natural forces, such as convection and gravity, making safety functions less dependent on active systems and components like pumps and valves. In practice, some of these features are already being incorporated into modern plants that are under construction or have been recently commissioned. Incorporation of other features are envisaged in new designs being developed now. In June 1995, the IAEA published IAEA-TECDOC-801, "Development of Safety Principles for the Design of Future Nuclear Power Plants". The purpose of TECDOC-801 is to propose updates to existing safety objectives and principles which could be used as a basis for developing safety principles for the design of future NPPs. The TECDOC is a comprehensive document reflecting current trends in nuclear safety, especially those associated with the explicit consideration of severe accidents[1] already at the design stage. The proposed updates build closely on the INSAG-3 document "Basic Safety Principles for Nuclear Power Plants", Safety Series No. 75. Implementation of these principles should lead to reactor designs with a very high degree of safety, recognizing that as a practical matter these future designs must also result in economically competitive energy production if they are to be utilized.

[1] IAEA-TECDOC-801, *severe accidents* are described as nuclear power plant states beyond *accident conditions* including those causing significant core degradation. *Accident conditions* are described as departures from operational states in which the release of radioactive materials are kept to acceptable limits by appropriate design features (these deviations do not include severe accidents).

A key proposal of TECDOC-801 is that severe accidents beyond the existing design basis be systematically considered, and explicitly addressed if appropriate, during the design process for future reactors. Severe accidents are addressed by a ***Technical Safety Objective*** which is in part "... to ensure that for all severe accidents addressed in the design there are no serious radiological consequences; and to ensure that the likelihood of any severe accident that could have serious radiological consequences is extremely small." The focus here is on protecting public health and safety, and includes already explicit consideration of certain severe accident sequences. Severe accidents are further treated by a ***Complementary Design Objective***: "To ensure, in addition to meeting the technical safety objective, that severe accidents addressed in the design have no significant radiological consequences." This ***Complementary Design Objective*** accommodates the desire of many countries to demonstrate that no significant radiological consequences would occur outside the immediate vicinity of the plant, and thus no stringent off-site emergency response actions (such as prompt notification and/or evacuation, resettlement, etc.) would technically be necessary for designs that meet also this objective.

As noted in TECDOC-801 not all conceivable severe accidents can and should be explicitly addressed in the design. The selection of severe accidents to be explicitly addressed is to be based on a combination of best estimate deterministic analyses, probabilistic considerations (including the application of numerical safety targets as guidelines) and sound engineering judgement. It is recognized that as a result of this process, a decision will be made to exclude some severe accidents of extremely remote likelihood from the set of severe accidents to be explicitly addressed in the design. Reaching this final decision is an iterative process, with judgements made by the designer, based on the radiological protection goals to be met, followed by a careful analysis and then reviewed by utilities and regulators.

Finally, and very importantly, TECDOC-801 notes that design features that are provided to address severe accidents are not expected to meet the same stringent design criteria and requirements (redundancy, diversity, and conservative analysis and acceptance criteria) applied to the engineered safety features provided to cope with design basis accidents. However, design features for addressing severe accidents are still engineered in a way which would give reasonable confidence that they are capable of achieving their design intent.

In developing TECDOC-801 the participants recognized very soon that certain parallel or follow-up activities are necessary for sharing information and facilitating the understanding, use and implementation of the principles proposed in the TECDOC. The respective general topics are as follows:

- design approaches
- selection of accidents to be considered in design
- radiological consequences and
- use of probabilistic targets.

The first activity in this context was an IAEA Technical Committee Meeting (TCM) on Accident Prevention and Mitigation Capabilities of Future NPPs in November 1993. The participants of the TCM found that, prior to dealing with the proposed subject of the TCM, adequate plant conditions had to be identified and defined in consensus. However, this consensus was not reached, in particular on how to select and identify those severe accidents which should be explicitly addressed in the design of future NPPs.

Another TCM, 29 May-2 June 1995, was devoted to approaches to safety of future NPPs in different countries. At this TCM areas were identified where significant harmonization already exists (e.g. importance of defense in depth and safety culture, improved plant simplification, design margins), and areas where opportunities exist to improve harmonization but further work is still necessary (e.g. how to deal with external hazards, role of PSA, safety targets). The report of this TCM is published as IAEA-TECDOC-905, "Approaches to the Safety of Future Nuclear Power Plants" (September 1996).

Subsequently, a TCM, held 9-13 October 1995, dealt with Identification of Severe Accidents for the Design of Future Nuclear Power Plants. At this meeting the participants recognized that commonalities exist, such as the list of challenges to be considered in the design. However, they also acknowledged that there are significant implementation differences, together with areas for which further clarification would be useful, in particular when national approaches in addressing severe accidents differ (examples are the quantitative values associated with the term "non-significant radiological consequences", the screening/cut-off process for selecting relevant challenges and the consideration of external events more severe than those considered in the design basis).

Having this in mind, the participants nevertheless concluded that:

- The various plants have been designed to accommodate a set of design basis accidents, and this forms a foundation and a starting point for the identification of relevant severe accidents.
- Selection of those conceivable severe accidents that should be addressed explicitly in the design can be made with a combination of probabilistic, deterministic and engineering judgement methods. In most approaches, combinations of probabilistic and deterministic methods - however with some diversity in the balance between them - are used in the selection process. However, there is some diversity in the balance between the approaches. Specific values for cut-off limits which restrict the total number of initiators and sequences considered as well as the adequate processes to determine these limits are open to further discussion.
- A proper balance between prevention and mitigation should be maintained, however, priority should be given to prevention in particular for accident sequences with the potential for early containment failure. All operational states (including full power operation, low power and shutdown modes) should be considered. Accident management, including the prevention and/or mitigation of the degradation of the core in the vessel, of vessel failure and of containment failure continue to be important also for future designs.
- Many organizations utilize core damage frequency and large release probability limits with typical values of $10^{-5}$/reactor-year and $10^{-6}$ /reactor-year respectively.

A review of trends for water cooled reactors presently under development resulted in a set of severe accident phenomena to be considered, and addressed (prevented or mitigated), if appropriate, in their design. These phenomena, and associated candidate strategies for prevention and/or mitigation, are briefly summarized in Table 1. Depending on the design, phenomena related to processes such as reactivity transients, recriticality events, or missile generation might also need to be addressed.

TABLE 1. SEVERE ACCIDENT PHENOMENA AND CANDIDATE STRATEGIES FOR PREVENTION AND/OR MITIGATION

| Phenomenon or challenge | Associated candidate strategy |
|---|---|
| High pressure melt ejection and direct containment heating | Adequate primary circuit depressurization system |
| Hydrogen production in vessel | Avoid conditions allowing hydrogen production |
| Hydrogen production and combustion in the containment | Containment size and geometry, ignitors, recombiners, or containment inerting |
| Steam explosions in vessel | No consensus reached at TCM[a] |
| Steam explosions in the containment | No consensus |
| Core-concrete interaction in the containment[b] | Prevention through retention in vessel or mitigation through design features preventing interaction between corium and concrete |
| Containment bypass or loss of long term heat removal | Prevention through design |

[a] The probability of steam explosions in the vessel that may challenge the integrity of the containment is considered very low.

[b] Although not explicitly stated in this table developed by the TCM, this includes ex-vessel core melt spreading. An associated candidate strategy is to provide a distribution which allows long term cooling and which prevents basemat melt through.

## 2. PURPOSE

In order to examine the incorporation of measures for coping with severe accidents into the designs of future water cooled reactors, and to promote further discussion and information exchange amongst participating countries, the IAEA convened a TCM focusing narrowly on the Impact of Severe Accidents on Plant Design and Layout of Advanced Water Cooled Reactors from 21 to 25 October 1996. The objectives of this Technical Committee Meeting were to provide a forum for review and information exchange on national and international programmes or policies with respect to the severe accident issue, to describe design measures implemented or planned for addressing severe accidents, the research on severe accident phenomena which guide and support the design decisions, and the effect of addressing severe accidents on economic competitiveness.

## 3. SCOPE

The scope of the TCM included:

- Specific design provisions to cope with severe accident challenges for future plants.
- Impact of such design provisions on layout and costs.
- R&D in support of the validation of the proposed design provisions.

Because regulatory as well as utility requirements can differ from country to country, it was expected that presentations would allow to delineation of the specific links between these requirements and design provisions adopted for severe accidents.

While the scope of TECDOC-801 includes all reactor types, the scope of this TCM was limited to future LWR and HWR designs.

## 4. CONDUCT OF THE MEETING

The Technical Committee Meeting was jointly organized by the Department of Nuclear Energy and the Department of Nuclear Safety and held in Vienna from 21 to 25 October 1996. It was conducted within the frame of activities of the IAEA's International Working Group on Advanced Technologies for Water Cooled Reactors.

There were 25 participants representing 15 Member States, one International Organization (European Commission) and the IAEA. Meeting participants represented regulators (6), vendors or designers (6), utilities (6), as well as R&D laboratories (2) or State or international agencies (5).

The meeting chairman was M. Vidard of Electricité de France.

Following the paper presentations, reports on the following two topics were prepared during working group sessions:

- Design requirements and design measures for coping with severe accident phenomena and challenges, and
- Status of knowledge regarding severe accident phenomena.

All presented papers as well as the reports of these working group sessions are included in this TECDOC.

## 5. SEVERE ACCIDENT PROVISIONS IN ADVANCED WATER COOLED REACTOR DESIGNS

Design organizations have adopted various design features for future plants in order to cope with severe accidents. A broad range of candidate solutions for prevention and/or mitigation relying to various degrees on passive as well as active systems was described during the meeting for PWRs, BWRs and HWRs.

### 5.1. Prevention of severe accidents

The principle of accident prevention, as stated in INSAG-3, is:

"Principal emphasis is placed on the primary means of achieving safety, which is the prevention of accidents, particularly any which could cause severe core damage".

With regard to severe accidents addressed in the design, TECDOC-801 proposes to supplement measures for prevention of design basis accidents by design features and by procedures and appropriate training of staff to prevent a detrimental evolution of events outside the design basis. Various measures are incorported by designers of future water cooled reactors for severe accident prevention.

The 1500 MW(e) European Pressurized Reactor (EPR) being designed by Nuclear Power International incorporates design features to provide ample thermal margins and grace period, and the safety systems are designed with four fully physically separated trains and state-of-the-art man machine interface and I&C systems. Accident situations at reactor shutdown conditions are considered in the design by including provisions for backup of the two normal trains of the residual heat removal (RHR) system by two safety injection trains.

The design approach of the 1350 MW(e) KNGR, currently being developed by the Korea Electric Power Corporation (KEPCO), Republic of Korea, provides measures for increasing design margins, enhancing emergency safety system reliability and extending emergency system functions considering multiple failures beyond the design basis. Design features for the purpose of prevention and mitigation of severe accidents include a large pressurizer, large steam generators, a four-train safety injection system, a safety depressurization system, a passive secondary condensing system, an in-containment refueling water storage tank, a large double containment and a hydrogen ignition system.

Severe accident prevention is addressed in the design of future large and medium size WWERs (V-392 and V-407 series). These designs are being developed by three organizations: OKB "Gidropress", the Russian National Research Centre "Kurchatov Institute" and Leningrad Organization Atom - Electro Project. With regard to reactivity accidents, prevention of core melt is based on a highly reliable protection and trip system, together with means allowing to ensure substantial subcriticality margin after reactor trip. With regard to core cooling, passive systems and components are emphasized, e.g. high and low pressure accumulators and, for the specific case of the V-407 series, automatic depressurization of the reactor cooling system allowing water injection at low pressure from a water storage tank.

In Japan, strategies for core melt prevention in BWRs and PWRs are stressed. Use of all available makeup water, use of emergency power sources which are installed to cope with station blackout, and reactor coolant system depressurization are all generic strategies. To decrease risks, further specific strategies complementing the generic ones include scrubbing a vent for BWRs, secondary loop cooling for PWRs, use of containment cooling chiller or alternative auxiliary component cooling for PWRs. For future plants, and in particular for the Improved Evolutionary Reactor (ABWR-IER), further benefit for severe accident prevention is expected from more diversity in the ECCS design, increased margins, diversity in emergency power supply and simplicity.

The 1000 MW(e) SWR, which is a boiling water reactor with passive safety systems being developed by Siemens, incorporates automatic depressurization of the reactor coolant system on a low water level allowing water injection at low pressure through functionally redundant systems. Moreover, in case of total failure of the pressure relief systems, four emergency condensers are included in the design and have the capability to reduce the pressure to levels allowing the operation of an active decay heat removal system.

Highly reliable and redundant measures to prevent severe accidents are incorporated into the design of CANDU 9. The high pressure melt ejection accident (due to failure to shutdown), for example, is claimed by the designer (Atomic Energy of Canada, Ltd.) to be highly improbable ($10^{-8}$/reactor year) due to the presence of two diverse, redundant, physically separate, fully capable, independent, testable and dedicated shutdown systems. All credible core disassembly scenarios can occur only at low pressures and result in largely solid debris. Ample passive heat sinks surrounding the core allow sufficient time for the operators to initiate mitigating measures and terminate the accident progression early.

An Advanced Heavy Water Reactor (AHWR), currently being designed in India by Bhabha Atomic Research Centre, incorporates inherent characteristics and passive safety systems to prevent severe accidents such as a negative void coefficient, natural circulation core cooling at rated power, decay heat removal through passive isolation condensers and passive emergency core cooling through advanced accumulators and gravity driven water injection. Safety system arrangement is generally based on four independent trains, each capable of providing 50% of the requirement, and the large masses of water around the fuel /core provide adequate grace period.

The detailed discussions of the TCM concluded that choices on measures for core melt prevention play sometimes a key role in making decisions on what is actually needed for severe accident mitigation.

### 5.2. Mitigation of severe accidents

The principle of accident mitigation, as stated in INSAG-3, is:

> "In-plant and off-site mitigation measures are available and are prepared for that would substantially reduce the effects of an accidental release of radioactive material".

TECDOC-801 proposes that "even though accident prevention always has priority, it is important to include adequate measures in the design to mitigate design basis accidents and addressed severe accidents. Inclusion of such mitigation features contributes to defense in depth by providing more margin directed towards limiting the consequences of accidental release of radioactive material from the plant".

In general, accident mitigation provisions are of three kinds: design features, accident management features and off-site countermeasures. An important update to existing safety principles proposed in TECDOC-801 is "plants that meet the more restrictive standard of the Complementary Design Objective may be able to achieve commensurate reduction in emergency planning requirements". For severe accident mitigation, various measures are incorporated by designers of future water cooled reactors.

For the EPR, the design approach is based on the recommendations of French and German safety authorities as presented at the TCM on Identification of Severe Accidents for the Design of Future NPPs. This has led the designers to take the following design choices for severe accident mitigation: provision of a reliable depressurization system to preclude high pressure core melt sequences, a "dry" reactor cavity to avoid ex-vessel steam explosion, layout to eliminate direct connection between the reactor cavity and the upper containment compartments thereby preventing debris transport to the upper region of the containment in case of vessel rupture, a dedicated area for debris spreading and further cooling by water from the inside reactor building water storage tank (IRWST) to prevent basemat melt-through and limit non-condensable gas generation, provision of a hydrogen control system relying, at the current design stage, on passive autocalytic recombiners and some igniters, and a dedicated severe accident decay heat removal system. With regard to containment design, a large double wall (prestressed concrete and reinforced concrete) containment is provided for protection against both internal and external events. In the absence of a liner, and for limiting releases in the most extreme sequences, the design pressure of the inner prestressed concrete shell is designed for a pressure of 6.5 bar to envelop $H_2$ global deflagration and the containment atmosphere pressure increase without initiation of the containment heat removal system during 12 hours.

In the case of KNGR, the principal approach to mitigation is ex-vessel cooling of corium debris through providing a large cavity to capture and cool the debris. The adopted strategy is pre-flooding of the cavity with water from the in-containment refuelling water storage tank, through the opening of valves, with fusible plugs as backups in case of valve opening failure. High pressure sequences are addressed through a safety related manual depressurization system and cavity arrangement. Combustible gas control is achieved through provision of igniters to maintain the global concentration of hydrogen to below 10%. As for the EPR, the containment is of the double concrete wall type, but, with a steel liner provided on the inner shell. Compliance with ASME factored load category as the ultimate structural capacity is to be shown in case of burning of hydrogen resulting from a reaction of an equivalent of 75% of the active fuel cladding with water.

For large and medium size WWERs, an R&D programme is under way to identify which severe accident mitigative measures could be adopted. Experiments to investigate retention of molten corium inside the vessel by ex-vessel flooding with water have confirmed the feasibility of this approach for the medium size plant (V-407).

Work is under way in Egypt to evaluate a "Karlsruhe type" containment scaled up to 1400 MW(e) output. Fans are adopted to improve heat rejection and therefore to limit the pressure inside containment. Analyses are underway to quantify their effect on long term pressure a reduction after an accident with melting of the core.

In Japan, the Nuclear Safety Commission strongly recommends that the utilities voluntarily plan effective accident management. While efforts made in the past on prevention of core melt imply that the need for mitigative measures should be minimal, specific strategies have been defined for BWRs as well as PWRs. For BWRs, delivering water to the reactor cavity for ex-vessel debris cooling, containment venting to prevent its overpressure failure, or inertization - reinertization to cope with the hydrogen generation, have been implemented. For PWRs, adding fire water into the containment cavity for debris cooling, forced depressurization of the RCS or providing igniters for ice condenser containments are the most remarkable achievements. For future LWRs, special emphasis will be put on containment design.

However, the strategy has not yet been fully defined, as below some value(s) (cut-offs), further provisions are not actually meaningful for risk reduction.

For the SWR-1000, special attention has been paid to in-vessel retention of the corium, which is the basic strategy for limiting further progression of severe accidents, and to hydrogen related mitigation measures. Scoping studies performed by the vendor tend to show that flooding the reactor cavity with water up to the vessel lower head should provide for decay heat removal through the vessel wall without film boiling. In-vessel retention thus supports the claim that ex-vessel steam explosion is not a credible event. The design incorporates an inert containment, so for mitigation with respect to hydrogen production, emphasis has been on overpressure protection. Scoping studies show that, even assuming that 100% of the zirconium in the core reacts with water, the pressure inside the containment is limited to 0.75 MPa (7.5 bar), which is manageable at the engineering level. Risks resulting from other severe accident challenges have also been analyzed; in-vessel steam explosion was found not credible, as was high pressure melt ejection considering the redundancy of depressurization systems.

In the case of CANDU 9, though considered highly improbable, core melt progression has been analyzed in the calandria, then in the shield tank, and at last in the reactor vault. Dealing with debris cooling, the heavy water in the calandria and the light water in the shield tank have the capability to delay melt progression. Steaming in the calandria and then in the shield tank with relief to the reactor vault through rupture disks provides time for the operator to replenish these areas with water from the reserve water tank. The composition of concrete in the reactor vault is selected to minimize non-condensable gas production in case of corium-concrete interaction. The potential for steam explosion has not yet been assessed. The containment free volume is large (124 000 $m^3$) for pressure limitation, and both igniters and recombiners are incorporated into the design to limit hydrogen concentration in case of zirconium-water reaction. Containment coolers provide for long term pressure suppression. Severe accident mitigation is further enhanced in CANDU 9 by additional passive water source (the reserve water tank) and by the large containment free volume in addition to various engineered safety systems.

In the case of the Indian Advanced Heavy Water Reactor, the probability of a severe accident is expected to be negligibly small, due to the extensive use of passive safety systems, to ensure safety functions. Specific measures for coping with severe accidents are the double containment, the passive containment isolation, reactor cavity which gets filled with water following a LOCA and the passive containment cooling system.

### 5.3. Costs

Limited information on the impact on plant costs of addressing severe accidents in the plant design has been made available. Nevertheless, important qualitative insights on cost have been provided. Features to cope with severe accidents tend to increase plant complexity, and both preventive and mitigative features tend to increase capital cost. While for preventive features this capital cost penalty can be at least partly compensated by higher overall plant availability, this is not the case for features added for mitigation of severe accident consequences. Cost increases which result from features to address severe accidents which provide no benefit to plant availability should be limited to enable a compensation by an overall plant optimization.

Many potential future owners of advanced designs which pursue the Complementary Design Objective of TECDOC-801, expect that comparative risk assessment with other energy options and assessments of nuclear plant safety level will justify elimination of rapid action requirements in emergency plans for public protection. Of course, the most desirable approach would be to achieve technical justification of this without increasing plant cost, although clearly the enhanced safety requirements have a significant influence on investment cost.

## 6. R&D IN SUPPORT OF SAFETY ASSESSMENTS

In the United States of America, work sponsored by the US Department of Energy (DOE) on some severe accident issues is in part guided by risk oriented accident analysis methodology (ROAAM) for dealing with complex issues where uncertainties in knowledge are large. Specific areas, addressed by application of ROAAM to the AP600 design of Westinghouse are:

- in-vessel retention, where test programmes performed in the ULPU2000 facility and ACOPO were major elements for demonstrating that vessel failure was not a credible event.
- in-vessel steam explosion, where test results provided by MAGICO and SIGMA, as well as development of specific computer codes contributed to the analysis of vessel failure. The conclusion that vessel failure is not credible after an in-vessel steam explosion is still to be confirmed, the peer review of work done being still ongoing.

A programme dealing with debris cooling (MACE) is ongoing. The USA is also participating in the RASPLAV programme which deals with vessel cooling. This programme is performed in Russia in the framework of an international co-operation managed by the OECD Nuclear Energy Agency.

For the EPR, R&D programmes are focused on key issues for the assessment of proposed design solutions. Special emphasis is on the confirmation of the spreading concept, the validation of assumptions and methods used in the evaluation of the inner containment wall leak rate, material selection (in particular for the spreading area) and component qualification. The need for validated computer codes whose results are acceptable to licensing authorities is also stressed.

For the KNGR, work is underway on an in-core catcher for reactor pressure vessel protection, the development of analytical methods for debris cooling and hydrogen mixing and combustion, and conceptual studies of passive secondary cooling systems.

For WWERs experimental activities as well as developments of analytical tools are ongoing. For example, significant efforts are carried out on the interaction between corium and the vessel wall (RASPLAV), and on corium-water interaction in the LAVA test facility. The MELVES computer code is being developed for analyses of in-vessel retention.

CANDU 9 draws from extensive Canadian and international R&D efforts in development of analytical tools to predict abnormal behaviour. In additional, wide-ranging supporting experimental programmes have been underway in Canada for over 20 years to help predict abnormal fuel and channel behavior for CANDU reactors. Among the many new technologies developed to help better prevent and mitigate reactor accidents is the development of recombiners for long term hydrogen mitigation. Dedicated severe accident experimental programmes are underway to improve the understanding of the severe accident phenomena with greatest inherent uncertainties (i. e. core slumping). The Canadian nuclear industry continues its involvement in international severe accident research programmes such as RASPLAV.

## 7. R&D ISSUES

An R&D programme sponsored by the European Union on severe accidents for both future plants and current plants, in particular in eastern countries, is underway. In the fourth framework, extending over a 5 year period starting in 1994, emphasis has been put on the following:

- In-vessel core degradation and coolability, in particular in-vessel coolability and reactor pressure vessel behaviour.
- Ex-vessel corium behaviour and coolability, in particular corium spreading.
- Source term where both in-vessel and ex-vessel fission product behaviour are studied.
- Containment performance, with special emphasis on hydrogen distribution and combustion.
- Accident management.

Hydrogen generation during quenching is a specific R&D issue being addressed in Argentina. A theoretical model has been developed to improve the prediction of the hydrogen generation rate during quenching, assuming that there is a break of the oxide layer during core reflood. Results show reasonably good agreement between experimental data and numerical calculations.

In Russia, fission product release in the early phase of core degradation is being experimentally investigated together with comparison with numerical predictions. Significant isotopes such as $^{85}Kr$, $^{137}Cs$, $^{131}I$, $^{125}Sb$, $^{106}Rh$ and $^{144}Ce$ are being investigated.

With regard to the status of knowledge of major severe accident challenges, the Royal Institute of Technology (Sweden) presented the following views:

Containment survivability:

- There is a consensus on the very low conditional probability of containment failure in case of in-vessel steam explosion.
- Sufficient information allowing designing to preclude direct containment heating are available, both at the experimental methodological level and at the engineering level.
- $H_2$ detonation can be adequately addressed through mitigating devices (e.g. recombiners). Further research on $H_2$ mixing and the effects of $H_2$ detonation on specific containment designs are underway.
- Ex-vessel steam explosion, melt spreading and melt coolability are remaining challenges but ongoing R&D programmes will provide additional results.
- There is ongoing R&D dealing with containment survivability. Promising results have been obtained for the AP600. In-vessel quenching, which is central to accident management, remains a challenge.

Finally, further work is needed to better understand re-vaporization and re-suspension of aerosols from walls on structures on which they are deposited during an accident.

## 8. EXAMPLES OF VIEWS AND APPROACHES FOR SEVERE ACCIDENT PROVISIONS

In Belgium, starting from an analysis of probabilistic risk assessment (PRA) results, it was felt that the hydrogen issue had to be dealt with in current plants and it was decided to install passive auto-catalytic recombiners (PARs). Non-reliance on alternating current (AC) or direct current (DC) supply, together with self-starting of PARs at very low concentrations of hydrogen inside containment were important elements in the decision making process. Of particular interest is the analysis of constraints which had to be dealt with at the utility level, such as personnel security, radiation protection, theft protection, and maintenance. One of the main conclusions is that it is not always possible to install devices in optimal locations, and that other constraints can dominate the choice of locations.

In France, experience gained on current plants provides a background for the approach used by Electricité de France for assessing proposed design provisions which are specific for severe accident situations. At the implementation or decision making level, a cost analysis has to be factored in the process. Both construction costs and operation and maintenance costs must be considered. Though recognizing that a reasonable balance between prevention and mitigation should be maintained, prevention is in general more cost effective, especially considering the potentially very complex circumstances involved in a severe accident. Other important factors such as impact on routine operation, collective dose, and personnel security need also to be considered.

INVESTIGATION OF SEVERE ACCIDENT PHENOMENA
TO SUPPORT DESIGN DECISIONS AND
ACCIDENT MANAGEMENT PROCEDURES

(Session 1)

# ON THE HYDROGEN GENERATION DURING QUENCHING



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**Abstract**

The results of a diffusion model to calculate the hydrogen generation during the quenching of Zry in steam are presented. To simulate the break-away of oxide during the quenching, the model proposes a short-circuited diffusion process. The short-circuit process is a fissure propagation through the oxide layer. The diffusion model results with fissuration are compared with a proposed fast algebraic calculation procedure showing a good correlation for a cool down speed of 0.1K/s. The algebraic calculation procedure is compared with the experimental results for an average cool down speed of 100K/s giving also a good agreement.

## 1. INTRODUCTION

During a severe accident scenario in a nuclear power plant, different safety systems actuate. In particular, an accident management measure which delays the core uncovering is the water injection. This flooding action produces the highest temperatures in the uncovered portion of the core as a result of extended exothermic Zircaloy oxidation by steam right above the quench front[1, 2]. Up to 80% of hydrogen generated, during the CORA experiments, was produced during the flooding (quenching). Quenching may destroy, by thermal shock (quench-induced shattering), parts of the core and extend the debris bed formation. The thermal shock of the $ZrO_2$/oxygen-embrittled Zircaloy, generates new metallic surfaces by cracking and fragmentation. In most of the present Severe Fuel Damage (SFD) code systems, the quench behaviour is not considered or only treated by simplified user-specified criteria that are not validated against experimental data. As may be found[3], the hydrogen produced during flooding (rate and total amount) cannot be determined by the available correlation of Zircaloy/steam oxidation. A small scale quench rig apparatus[3] was designed to investigate the mechanisms of hydrogen generation.

In the present paper we propose a model to calculate hydrogen generation using a previously developed oxygen diffusion model[4], which was modified to perform transient oxidation considering the Zry $\alpha \leftrightarrow \beta$ phase transformation during heating and cooling.

## 2. THE MODEL

In a previous paper[4], we analyse the case of diffusion controlling process like the high temperature oxidation of Zry in steam during transients, in the temperature range where the system has always three phases. It was shown, that it is only possible to use the kinetic rate constant K, in the form $K^{oxide} = K_0^{oxide} \exp(-Q/RT)$, for every temperature transient speed and non oxide layer previously formed.

In another presentation[5] we extend the analysis to the transients starting at low temperature (two phases) up to sufficient high temperature (three phases) and cool down to low temperature (two phases).Then the $\alpha \leftrightarrow \beta$ phase transformation is considered.

To solve the several phase systems we use a modified previously developed code[7].

Because we considered a dissipative system, the system temperature will be imposed. That condition will permit us to compare the results with the quench rig experiment[2,3].

## 2.1 The influence of the β-phase

In a previous paper[6] we considered the appearance and disappearance of β-phase in Zry as a function of temperature. We give special attention to the precipitation during cooling of α-phase from β-phase. In the present model, to simplify the code, we did not consider precipitation during cooling of supersaturated β-phase. We keep as supersaturated phase up to its disappearances by diffusion process, according to the figure 2a of the aforementioned paper[6].

To modify the code[7] for transient oxidation through the $\alpha \leftrightarrow \beta$ phase transformation temperature we have considered that the concentrations at the interfaces are those of the Zry-O equilibrium diagram[8]. The equations used for the equilibrium lines may be found in a previous paper[4].

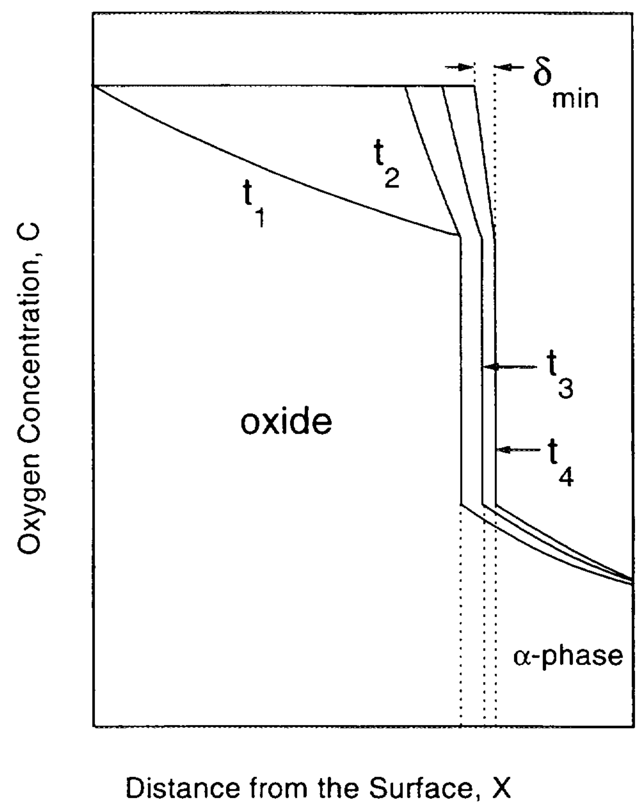


*Fig. 1. Schematic representation of oxygen concentration profile. $t_1$: just before the moment the fissure start. $t_2$: up to the moment the fissure propagates without disturbing the interface. $t_3$: the fissure movements bring the interface behind it. $t_4$: the fissure approach the interface a distance $\delta_{min}$.*

### 2.2. The oxide layer break

During the heat up transients the hydrogen production seems to be normal. That production, correspond to Zry oxidation without breaking the oxide layer. This behaviour can be understood because the stress between the $ZrO_2$ and the Zry relaxes with time, when the temperature increases. On the contrary, when the temperature decreases, the stress between the oxide layer and the metal becomes stronger as time (temperature) progresses (decreases) producing a stressed junction due to quenching.

To simulate the break of the oxygen layer we will introduce a new interface between the $ZrO_2$ short-circuited by the processes of fissure formation and the $ZrO_2$, that is still adhered to the metal. Oxygen diffuses through said interface, in the same way as we propose in a previous paper[9]. This new interface, is not of the Stefan type. The oxygen mass is not conserved through it. The fissure speed v is, by definition, a constant of the problem. We have a condition to start it, at a given time or temperature, and to stop it (v=0), for example, when a certain minimum distance, $\xi_{min}$, from that interface to the oxide/metal one is reached (figure 1).

To calculate the hydrogen production during transient, we integrate the oxygen concentration through the different phases, n. And for each time we have,

$$C_t = \sum_n \int C_{oxygen}(x,t)\, dx \qquad (1)$$

then,

$$C_H = 0.126\ \frac{C_{t+\Delta t} - C_t}{\Delta t}\ [mg\ of\ H / cm^2 / s] \qquad (2)$$

### 2.3. The algebraic calculation procedure

Because the computer time consumed in diffusion calculations is too large to use as a subroutine in an accident simulation code, we propose a fast algebraic calculation procedure. The algebraic calculation procedure is based in the well known kinetic rate constant equation.

As may be found in ref.[10] we show that a previous oxide layer of 15μm is more important than a high transient speed of 100K/s to produce a deviation of the $K^i_{oxide}$(instantaneous oxide kinetic rate constant) from the $K_{oxide}$(oxide kinetic rate constant). Then, it is possible to simulate a new transient oxidation (non previously oxide layer) with the kinetic rate constant, $K_{oxide}$.

## 3. RESULTS AND DISCUSSIONS

The diffusion code developed here was tested for different types of temperature transients. The transients are characterised by several parameters, figure 2, like the minimum and maximum temperature ($T_{min}$, $T_{max}$) the time at each temperature ($t_{min}$, $t_{max}$, $t'_{min}$) and the heat up ($S_{up}$) and the cool down speed ($S_{down}$).

### 3.1. The fissure through the oxide layer

At a given time or temperature during the quenching, $S_{down}$, we introduce a fissure with speed v, going from the external interface gas/oxide to the metal/oxide interface (figure 1). When the fissure approaches the oxide/metal interface at a minimum distance

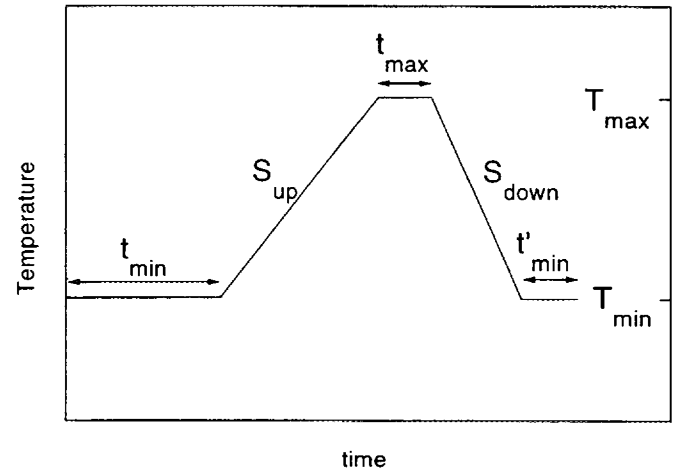


*Fig. 2. Schematic representation showing the characteristics parameter of a typical transient used in the present paper.*

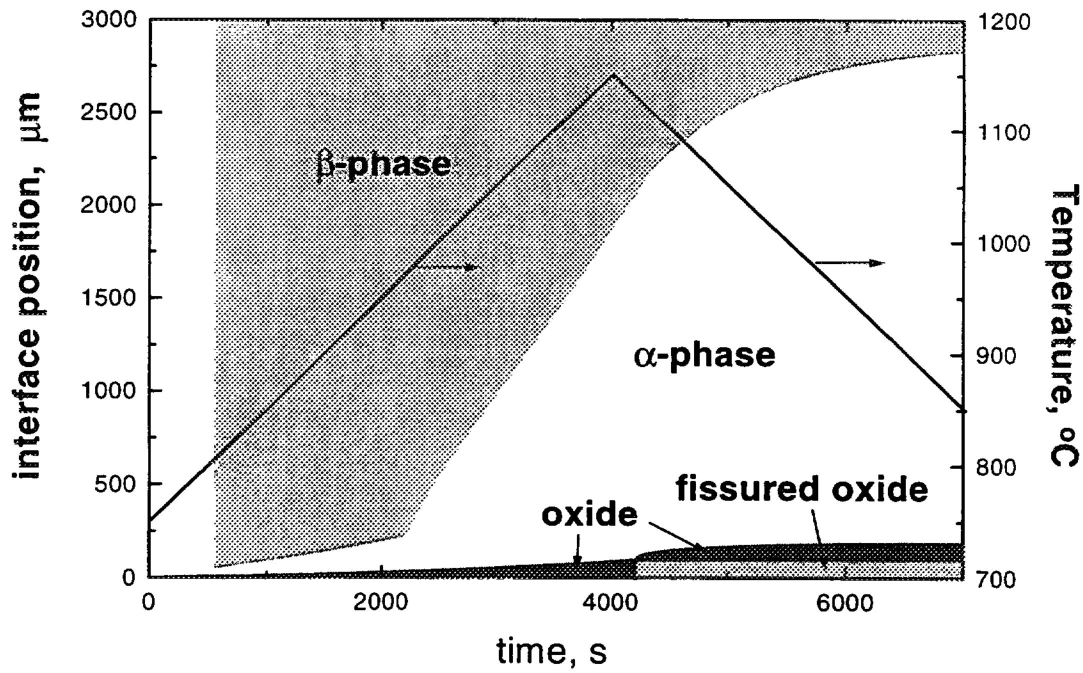


*Fig. 3. Interface evolution during a transient. Parameters characterising the transient: $t_{min}=1s$, $S_{up}=S_{down}=0.1K/s$, $t_{max}=20s$, $T_{max}=1150°C$ and $T_{min}=750°C$. The fissure was introduced at 1130°C during cool down up to the moment that the interfaces, fissured oxide/oxide and oxide/α-phase, approach at a distance $\delta_{min}=1\mu m$.*

$\xi_{min}$, the fissure is stopped (v=0). For every time step, even when the fissure travels from the gas/oxide interface through oxide/metal interface, we solve the diffusion problem for all the phases, included the oxide-phase non short-circuited.

In the figure 3 we show the results of re-oxidation at 1130°C by a fissure speed of v=10µm/s stopped when $\xi_{min}=1\mu m$. An important increase of oxidation speed is observed. Applying the equation (1) and (2) we calculate the hydrogen peak (figure 4).

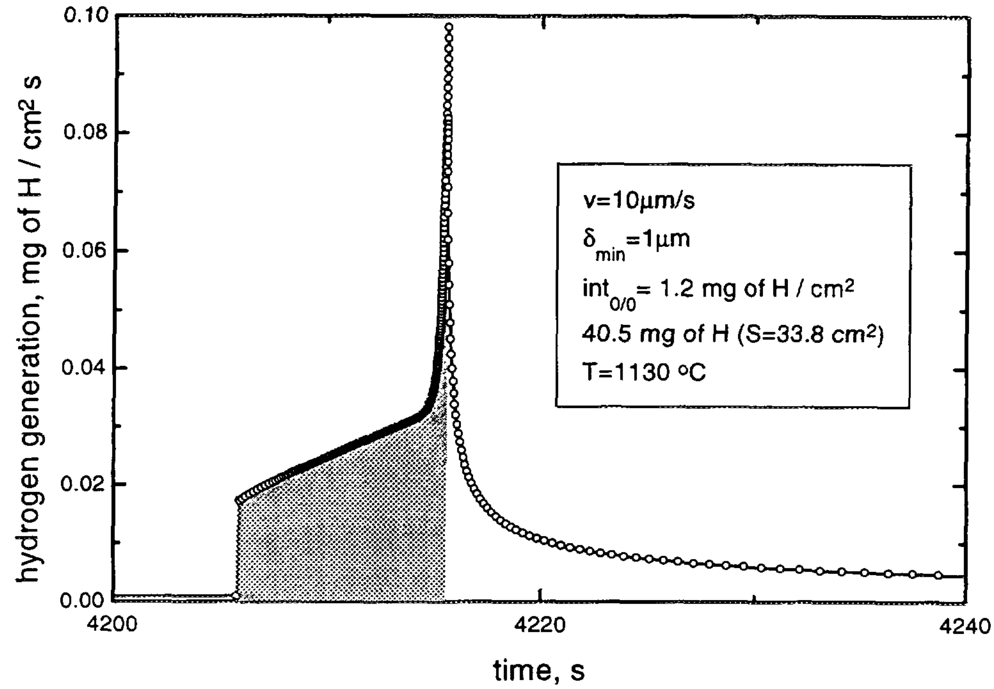


*Fig. 4. The hydrogen peak generated during the transient showed in figure 3. v is the fissure speed. The grey zone corresponds to the hydrogen generated during the fissure propagation.*

### 3.2. The hydrogen peak

The structure of hydrogen peak has a particular shape, figure 4, related with the diffusion problem showed in figure 1. It may be explained if we consider in detail the progression of the fissure (figure 5). From the start of the fissure (t=4206s) at the gas/oxide interface up to its arrival to (t=4215.5s) $\xi_{min}$ from the oxide/metal interface, the oxide/metal interface is accelerated producing the first step in the hydrogen peak (dot line added to show it, in figure 5). As both interfaces approach, the oxide/metal interface is accelerated up to the moment in which the fissure stops. Then, the hydrogen production is reduced abruptly by the growth of a new oxide layer formed over the metallic Zry surface. That reduction is in this case more abrupt because the temperature continues decreasing in the system considered, according to the transient imposed.

We integrate the hydrogen peak, to obtain the total hydrogen generation, from the start of the fissure up to the moment when the peak is reduced to the same initial value. We named that integral as (0/0) integral. When we integrate to 1/10 of the initial value, we named (0/10) integral, in the similar way that we show in figure 9. The (0/0) integral for the hydrogen peak plotted in figure 4 is 40.5 mg of H, for an effective surface of 33.8cm$^2$ that correspond to 10cm of Zry tube length used in the quench rig experiment[2,3].

### 3.3. The diffusion model results

The hydrogen generated for a cool down speed of 0.1K/s, figure 6, was calculated using the diffusion model with fissures introduced at temperatures of 950, 1000, 1050 and 1100 °C.

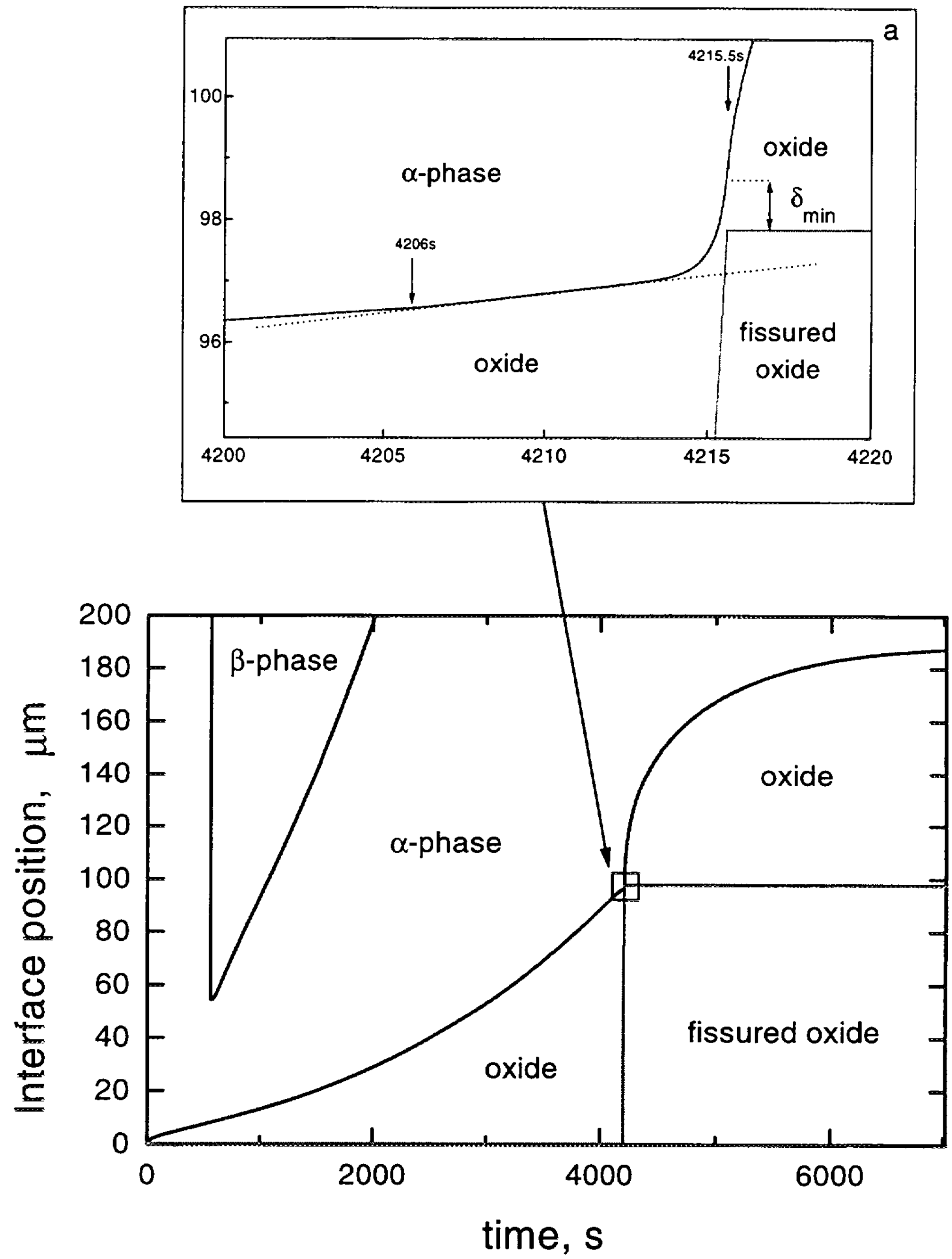


*Fig. 5. Shows the same results of figure 3 detailing the two oxide regions (down). (a)Enlargement the moment when the fissure approach to the oxide/α-phase interface. At 4206s the fissure starts at the gas/oxide interface, perturbing the oxide/α-phase interface grows (change of slope, showed by the dashed line). The fissure is stopped at 4215.5s, when the oxide layer thickness is less or equal* $\delta_{min}=1\mu m$.

To present the diffusion model results we integrate the hydrogen diffusion peak through the time in two ways: as the (0/0) and the (0/10) integrals, in the notation previously described.

The diffusion code predicts higher hydrogen generation than the experimental results in the temperature range selected for a quenching speed of 0.1K/s.

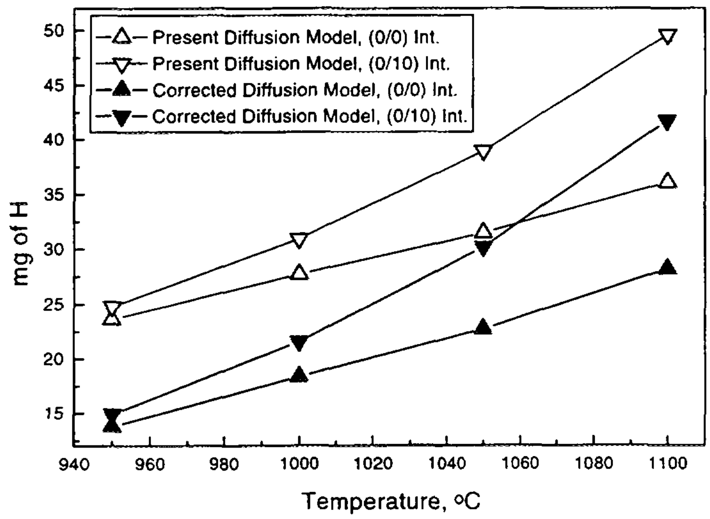


*Fig. 6. The hydrogen generated by the present diffusion model (quenching speed 0.1K/s) for fissures introduced during cool down at 950, 1000, 1050 and 1100°C and the corrected model diffusion values.*

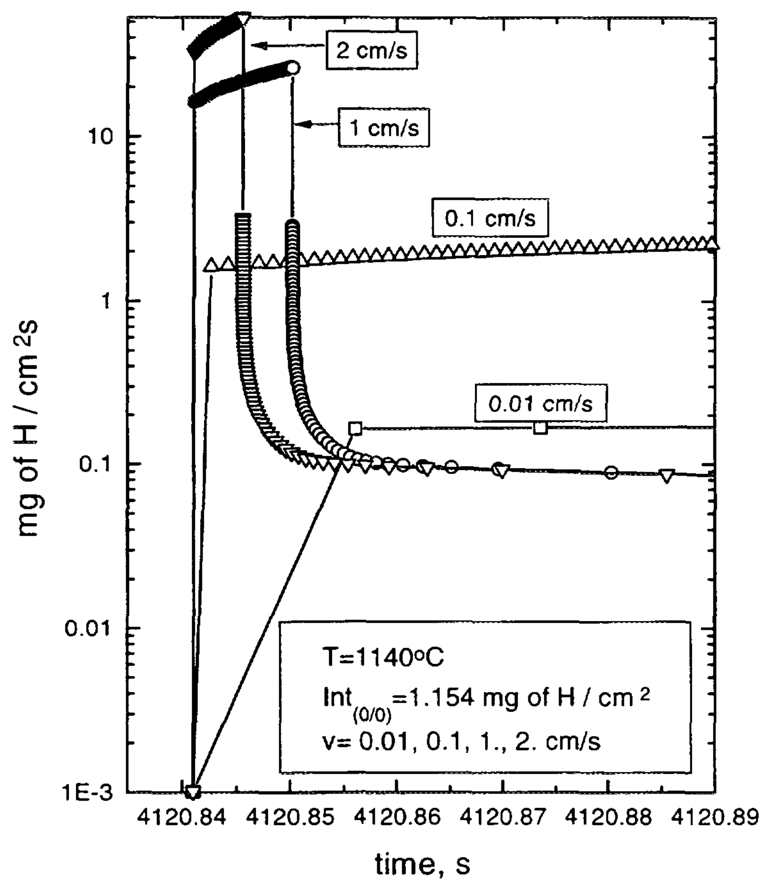


*Fig. 7. Different hydrogen peaks showing the influence of fissure speed, v, over its structure. The total hydrogen generated for each peak is the same and equal to 1.154 mg of $H/cm^2$. Quenching speed 0.1K/s.*

### 3.4. The fissure speed

At a first glance the fissure speed, v seems to be an important parameter. The influence of the fissured speed over the hydrogen peak is shown in the figure 7, together with the (0/0) integral.

The integral value for each peak is the same for all the different fissure speeds, v=0.01, 0.1, 1 and 2 cm/s. The value we obtain equals 1.154 mg of H / cm$^2$, for a fissure introduced at 1140°C and with a cool down speed of 0.1K/s. Then, no influence of the fissure speed over the hydrogen generation peak was found.

In the reality the fissure speed value must be very high. For our diffusion model it is necessary to use a v value compatible with the numerical calculation. In figure 7 we calculate the integral value between the start of the hydrogen peak and the maximum of the hydrogen peak. That corresponds exactly to the hydrogen produced during the time consumed by the fissure to go from the external oxide surface to near the oxide/metal interface. The hydrogen generated during that time can be considered as an error of our model (showed as a grey region in figure 4). In figures 4 and 5 the time is equal to 9.5s, from the 4206s to 4215.5s. We found that the hydrogen generated as an error of the model is also constant for different v (v=0.01, 0.1, 1 and 2 cm/s) values and equal to 0.2 mg of H / cm$^2$ for 1140°C, figure 7. This means that our value of 1.154 mg of H / cm$^2$ for 1140°C and integrated (0/0), is increased by approximately 20%. The real value is 0.954mg of H / cm$^2$. The correction for all the values are plotted in figure 6.

### 3.5. The algebraic calculation procedure

Due to the long computer time needed to calculate the hydrogen peak with the diffusion model it cannot be used in accident prediction codes. Then we*** and increase when the cool down speed growth we propose an alternative algebraic calculation procedure.

As we mention in a previous paper[4] and also here, it is possible to use the kinetic rate constant ($K_{oxide}$) under special conditions:

-non restrictions over the heat up and cool down speed,

-non previous oxidation before the temperature transients starts.

In the present model we considered a fissure that propagates instantaneously through the oxide during the cool down transient, starting a new oxidation over the new metallic surface exposed. Then, the previous conditions are fulfilled, because the re-oxidation process started by the fissure does not take into account the oxide previously formed.

The algebraic calculation procedure normally used is[4] :

$$\delta_n = \sqrt{\sum_{r=1}^{n} K_{oxide}(T_r)\Delta t_r} \tag{3}$$

with

$$t_{total} = \sum_{r=1}^{n} \Delta t_r \qquad \text{and} \qquad T = T(t) \tag{4}$$

In our case we must divide the calculation, before and after the fissure start. Then,

$$\delta_m = \delta_n + \delta_{m-n} = \sqrt{\sum_{r=1}^{n} K_{oxide}(T_r)\Delta t_r} + \sqrt{\sum_{r=n+1}^{m} K_{oxide}(T_r)\Delta t_r} \tag{5}$$

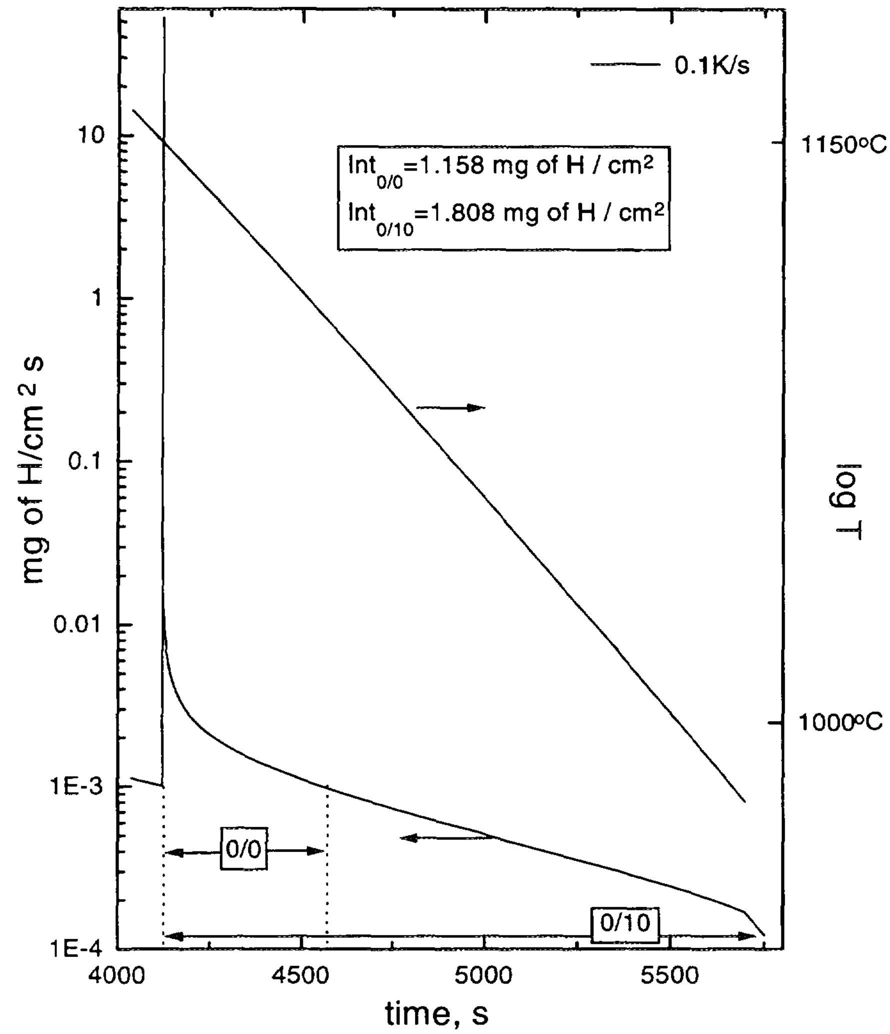


*Fig. 8. Hydrogen peak calculated with the proposed algebraic calculation procedure. For a fissured produced at 1150°C during a transient of 0.1K/s. Showing two types of integral limits, (0/0) and (0/10).*

The $\delta_{m-n}$ is the oxidation that produce the hydrogen peak, with

$$t_{fissure} = \sum_{r=1}^{n} \Delta t_r \quad \text{and} \quad t_{total} = \sum_{r=1}^{m} \Delta t_r \tag{6}$$

A typical peak is shown in figure 8, at 1150°C during a cool down transient of 0.1K/s. In our calculations to integrate the hydrogen peak we use a $\Delta t$=0.01s in the peak sharp region.

### 3.6. The corrected diffusion model and the algebraic calculation procedure

In figure 9 we plot the results from the diffusion model and from the algebraic calculation procedure. They have been obtained between 950°C and 1100°C for diffusion model results and between 950°C and 1150°C for the algebraic calculation procedure results. In the case of the algebraic calculation procedure we perform integrals (0/0), (0/10) and (0/100) types.

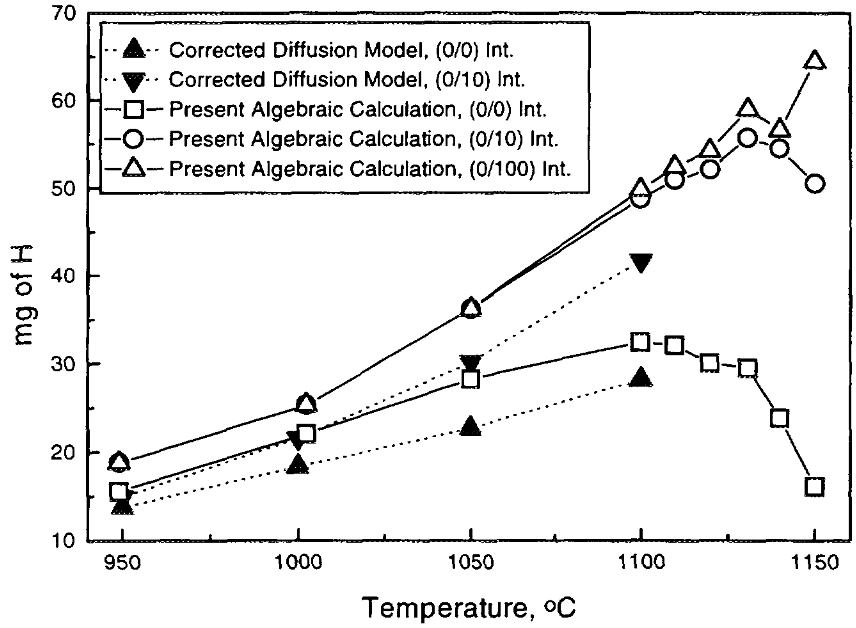


*Fig. 9. The same representation of figure 7 with the corrected diffusion model and the algebraic calculation procedure showing three types of integral limits (0/0), (0/10) and (0/100).*

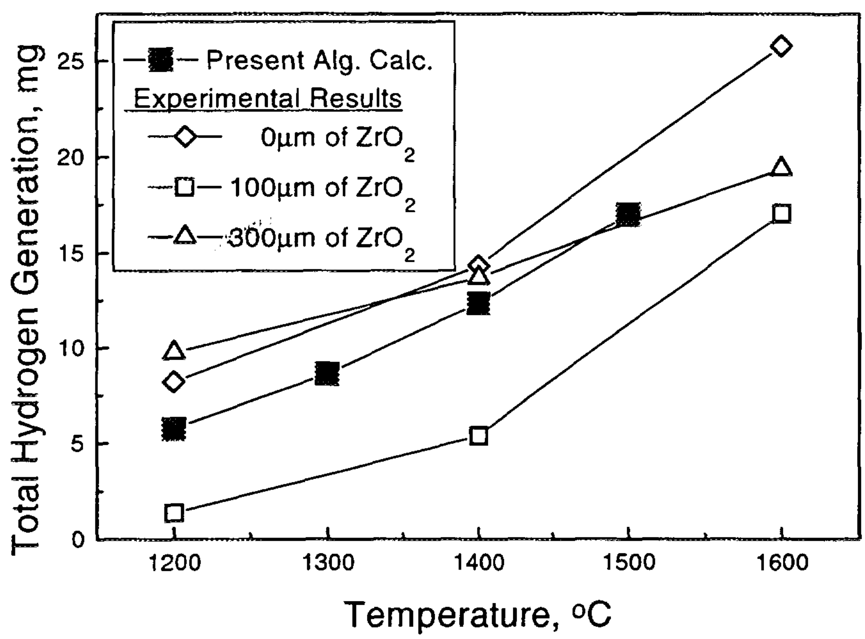


*Fig. 10. A comparison of hydrogen generated between the algebraic calculation procedure, for a cool down speed of 100K/s, with the last total hydrogen measurement in the quench rig experiment[11]. Sample effective surface 33.8 cm$^2$.*

The algebraic calculation procedure (0/10) and (0/100) type has the same behaviour that the diffusion model calculation. The differences between both calculations are small. This fact gives us the possibility to use the algebraic calculation procedure with small CPU time and to extend our calculation for high cool down speed.

### 3.7. Comparison between the algebraic calculation procedure for transients of 100K/s with the experimental results

In figure 10 we plot the results from the algebraic calculation procedure for a cool down speed of 100K/s and the new water quenching experimental results[11]. In order to have a good agreement with the experimental result we have introduced fissures at different temperatures 1200, 1300, 1400 and 1500 °C.

In the quench rig experiments the authors measured the hydrogen produced as gas and the hydrogen storage in Zry, for samples with different pre-oxidation treatments and from different quenching temperature. We plot their results: the total quantity of hydrogen produced during quenching, adding both values for each sample, as a function of the start quenching temperature.

## 4. CONCLUSIONS

The diffusion model with fissuration seams to be able to calculate the total hydrogen generation during transients.

The algebraic calculation procedure with small CPU calculation time is sufficient for having a good simulation of diffusion model with fissuration proposed.

The results of algebraic calculation procedure of hydrogen generation at high speed 100K/s (similar to the average quench rig experiment speed) are in good agreement with the experimental results when we use a temperature range for the break-away from 1200 to 1500°C in correlation with the experimental quenching temperature.


## ACKNOWLEDGEMENTS

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# EXPERIMENTAL AND COMPUTATIONAL INVESTIGATIONS OF FISSION PRODUCT RELEASE FROM WATER COOLED REACTOR FUEL IN THE INITIAL STAGE OF A SEVERE ACCIDENT

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**Abstract**

*The results of experimental investigations carried out in IPPE of fission products release from LWR fuel in helium and steam are described in this paper. Experimental results for 1000-2100°C are compared with ORNL and KFK data, as well as with calculations based on ORIGEN-2, MELCOR and SCDAP/RELAP/MOD 3.1 codes. Overall radiological hazard of fission products release from fuel at various temperatures including the core melt range, but without FP confining safety systems mitigation effect is estimated.*

## 1. INTRODUCTION

Generally accepted NPP safety principles require to limit the core melt frequency by the value of $10^{-5}$ /reactor-year. Last designs of advanced reactors trend to satisfy this requirement and even more, strict one excluding core melt sequence. As so, the precedent to core melt stage of accident deserve the attention. Firstly it is important for account of fission products (FP) release on this stage, as part of overall release, and secondly, in case of melt prevent this initial stage can convert into principal source of radionuclides entering into NPP containment compartment and environment.

Therefore in this paper attempts were made to compare the radiological hazard of FP release, as first part of source term for initial and final stages of severe accident. Characteristic for LWR fuel temperature intervals are considered :

- level ~ 1200°C - zirconium alloy vigorous oxidation in steam- zirconium reaction;
- level ~ 2000°C - zirconium alloy melt and uranium dioxide dissolution in zirconium;
- level ~ 2400-2700°C - fuel melt.

For this estimation the results of experimental investigations of Russian LWR fuel release FP, which are fulfilled in IPPE recently, and other laboratories ( ORNL [ 1] and KFK [ 2] ) data of PWR fuel are used.

## 2. MEASUREMENT of FP RELEASE FROM FUEL

The test facility scheme is shown in fig.1 . This facility consist of electric heater, filters chain for solid FP collecting and volatile FP accumulation system. Heater and filters are located in hot cell, the rest of equipment is placed in operator room. Zirconium, aluminum and silicium oxide powders are used as the filter materials. Registration of volatile FP is performed by sampling for periodical activity measurement with Ge(Li)-spectrometer and gas radiochromatography.

Fuel fragments of VVER, irradiated in IPPE's research reactor up to 9.9 MWtd/ kg U burnup (the $^{235}U$ enrichment is 3.6 %) have the dimensions : outer diameter - 7.53 mm, diameter of internal hole - 1.4 mm, height - 25 mm. Zirconium alloy fuel element cladding - 9.15-0.7 mm. The samples are placed in zirconium oxide crucible (cup). Measured and calculated FP inventory in the samples is shown in table 1.

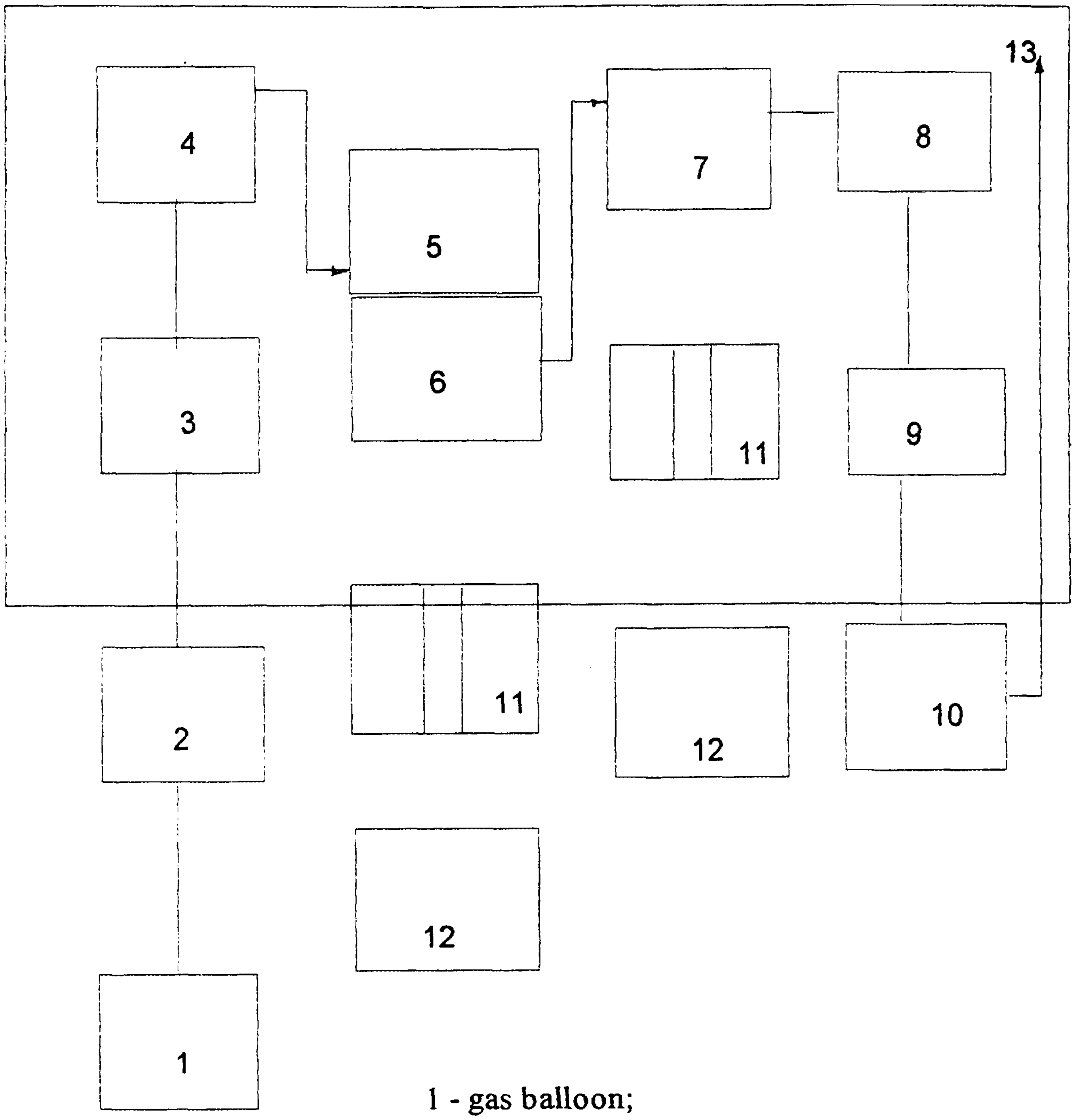
TABLE 1. FISSION PRODUCT INVENTORY, Ci / g U

| Nuclide | Experiment | ORIGEN-2 | Calc./ Exp. |
|---|---|---|---|
| $^{85}Kr$ | $4.1 \times 10^{-3}$ | $9.6 \times 10^{-3}$ | 2.33 |
| $^{133}Xe$ | $5.9 \times 10^{-1}$ | $8.6 \times 10^{-1}$ | 1.46 |
| $^{106}Ru$ | $1.8 \times 10^{-1}$ | $3.0 \times 10^{-1}$ | 1.66 |
| $^{125}Sb$ | $2.1 \times 10^{-3}$ | $7.4 \times 10^{-3}$ | 3.52 |
| $^{134}Cs$ | $4.6 \times 10^{-2}$ | $1.1 \times 10^{-1}$ | 2.39 |
| $^{137}Cs$ | $3.8 \times 10^{-2}$ | $1.1 \times 10^{-1}$ | 2.89 |
| $^{131}I$ | $3.7 \times 10^{-1}$ | $4.2 \times 10^{-1}$ | 1.13 |
| $^{144}Ce$ | $4.4 \times 10^{-1}$ | $5.8 \times 10^{-1}$ | 1.32 |

Experimental data are compared with code ORIGEN-2 ( ver. 2.1) calculation results [3]. Formation and accumulation of FP in fuel during 5-year irradiation in reactor is calculated with taking account of real reactor operation schedule.

For nuclides with maximal activity we have satisfactory coincidence of experimental and calculational data are seen. Nevertheless, their discrepancy some exceed the evaluated experimental error ( ~ 10-12 %). The data difference for nuclides with the small activity is greater, and in all cases the calculational results are higher than experimental ones. This fact is evident, because calculations don't take into account leakage of FP from fuel during reactor operation (fuel temperature operating level (calculational) is equal ~850°C).

The investigations of FP release from fuel in helium or steam are executed in IPPE. Flow rate of helium was equal 0.005 $m^3$ / min, of steam




1 - gas balloon;
2 - rotameter;
3 - hydraulic seal;
4 - steam generator;
5 - fuel heater;
6 - thermal depositor;
7 - filter;
8 - bubble condenser;
9 - steam condenser;
10- volatile FP-meter;
11- collimator;
12- spectrometer detector;
13- hot cell.

*FIG. 1. Fission product release and collection system.*

~0.08 m$^3$ / min. The fuel sample was heated with rate 6°C/min or 12°C/min to maximum temperature ~ 2100°C. After this the temperature was kept constant approximately 60 min and then cold down by natural way to the level ~700°C. The measurements of FP activity in gas or steam are executed with frequency (8 min)$^{-1}$ for heating rate 6°C/min and (4 min)$^{-1}$ for 12°C/min. Absolute FP release from sample was defined as difference of FP activity before and after annealing. Besides, for control FP activity in the filters, bubble condenser, measurements of Ge(Li)-spectrometer and activity on the pipes surface after heater are taken into account.

With view point of real accident process analysis reliability data for release in steam are rightful to larger extent than data for release in gas. The results of IPPE for overall release in steam at all time of experiment are compared with ORNL data [ 1] in table 2. The first value concerns heating rate 6°C/min, the second - 12°C/min. The data for ~ 2000°C differ slightly.

TABLE 2. FISSION PRODUCT RELEASE IN STEAM, %

| Nuclide | IPPE | | ORNL [ 1 ] | |
|---|---|---|---|---|
| | 1200°C | 2100°C | 2040°C | 2400°C |
| $^{85}$Kr | 46.1-28.1 | 97.8-94.5 | 75 | 100 |
| $^{137}$Cs | 13.7-12.7 | 84.5-75.1 | 80 | 100 |
| $^{131\,(129)}$I | 8.2-6.8 | 65.0-57.3 | 67 | 69 |
| $^{125}$Sb | 12.4-8.7 | 84.3-75.4 | 64 | 99 |
| $^{106}$Ru | 11.9-8.2 | 79.0-61.3 | | |
| $^{144}$Ce | 0.22 | 15.7 | | |
| Sr | | | 5.8 | 2.7 |
| Mo | | | 13 | 77 |
| Te | | | 63 | 100 |
| Ba | | | 32 | 30 |

It is necessary to note, that these data differ considerably from FP release in gas, as shown in table 3. The degree of this difference depends essentially on fuel temperature and nuclide. For example, in experiment [ 4] for non-volatile nuclides (Mo, Tc) at 1450-1750°C the release in steam is less, than release in gas. The information [ 3] for Sb at ~2700°C is analogous, but ORNL results for Mo and Te at the same temperature distinguish. The IPPE data ( table 2 and 3) for Ru release at 2100°C in steam is greater in several times, than release in gas. Data [ 5] for Cs release in steam at ~ 1000°C is considerably lower, than release in air, but according IPPE data ( table 2 and 3) Cs release in steam is greater, than in gas, that is distinguished with data of review [ 6].

TABLE 3. FISSION PRODUCT RELEASE IN GAS, %

| Nuclide | IPPE | SASCHA [2] | ORNL [1] |
|---|---|---|---|
| | 2100°C | 2150°C | 2100°C |
| | helium | air | hydrogen |
| $^{85}Kr$ | 98 | | 94 |
| $^{137}Cs$ | 64 | 26 | 96 |
| $^{125}Sb$ | | 4 | 6.4 |
| $^{99}Mo$ | | 25 | 7 |
| $^{106}Ru$ | 25 | | |
| Te | | 40 | < 46 |

Thus, on the whole, the data on medium influence on FP release are discrepant and incomplete.

## 3. FP RELEASE RATE FROM FUEL

The experimental results give the possibilities to draw the curves of time dependence of FP release from fuel. In result it is possible to have conception about release rate $\alpha(t)$. For example, fig.2 shows such dependence for iodine release from samples annealing with heating rate 6°C/min (a) and 12°C/min (b).

The experimental data for different nuclides were used for validation in IPPE of code MELCOR ( ver. 1.8.0), based on the diffusion model. The calculation take into account real geometric dimensions of fuel sample ( the ratio of surface to volume S/V = 853 $m^{-1}$ ). These characteristics differ appreciably from standard ones in given code for typical American fuel (S/V= 422.5 $m^{-1}$). Two parameter representations for $\alpha(t)$ are possible in MELCOR : $\alpha = A \exp(Bt)$ and $\alpha = K \exp(-R/T)$, which gives different results for many nuclides ( here: t - grad C, T - grad K ). On fig.2 calculational data are compared with experiment. The difference of data is substational. In particular, overall iodine release on MELCOR at the time of experiment (for two models of $\alpha(t)$) equals 100 % instead of 60 % in experiment (table 2).

Thus, dynamic release of FP is described by MELCOR with the recommended representations of $\alpha(t)$ not quite satisfactorily. For some nuclides ( I, Ru, Ce) code can't describe even overall FP release.

For correction of calculational model the piece - linear approximation of function $\ln \alpha = a + b/T$ , as follows from Arrenius's law, was used. At heating rate 6°C/min the coefficients a and b for three temperature ranges ( t<800°C, 800-2000°C, t>2000-2100°C ) were chosen. The result approximation has error, equal 5-10 %. However, if these

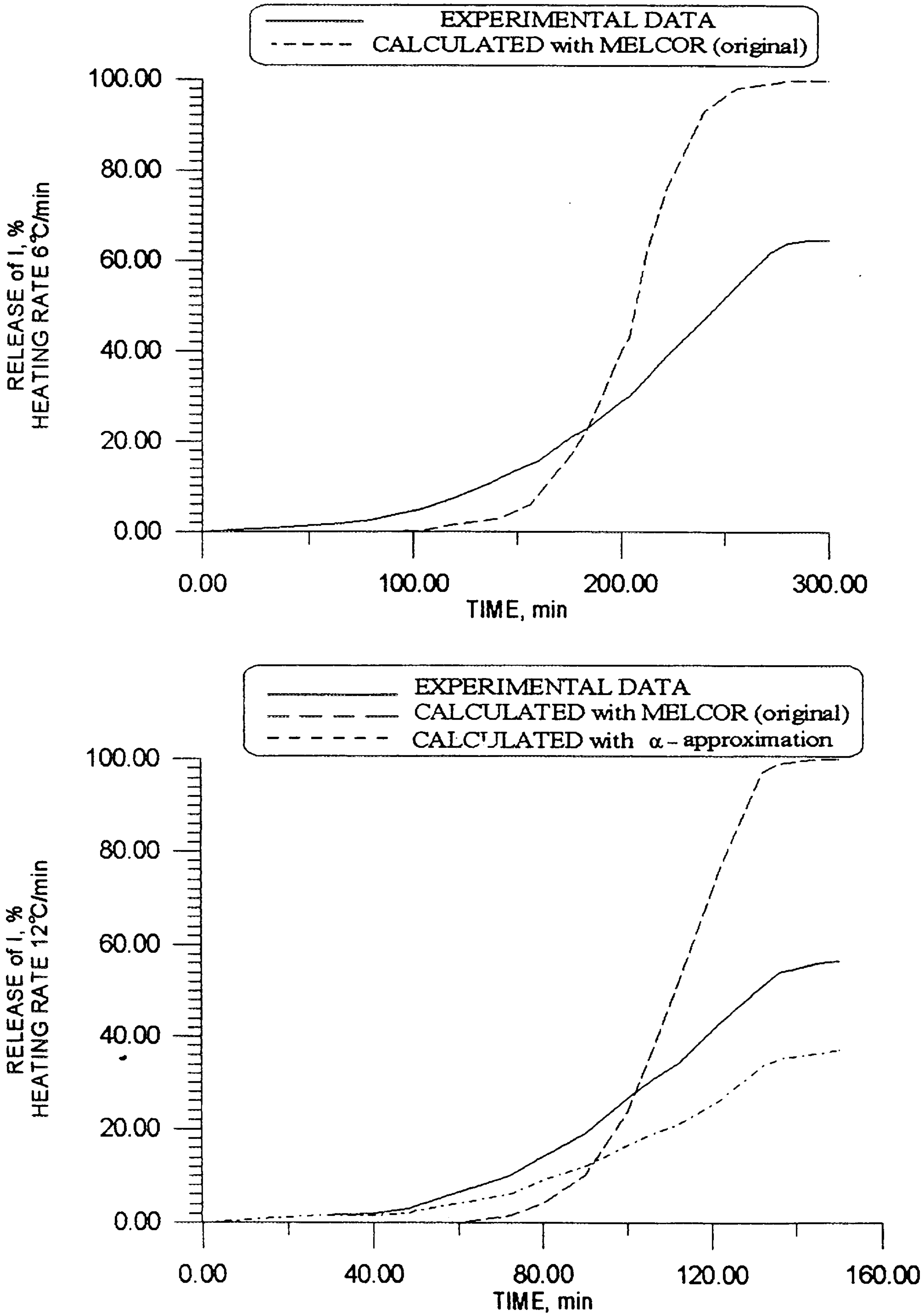



*FIG. 2. Iodine release in annealing.*

coefficients have been used for heating rate 12°C/min (additional curve on fig.2), the error rises to 30 % .

Hence it is possible to conclude that the parametrization of $\alpha$ from temperature only give non-correct results of code MELCOR ( ver. 1.8.0) calculational model. Real processes are very complex : FP diffusion in fuel grains, their migration in voids, FP thermophysical features change, the oxidation $UO_2$ and FP, fuel structure change at al.

This calculational result shows, that $\alpha$ must to depend on heating rate $dT/d\tau$. For improvement of calculational model next approximation was proposed :

$$\ln \alpha ( T, dT/d\tau ) = a_1 + b_1 / T + c_1 \, dT/d\tau$$

Here $\ln \alpha$ is linear function inverse temperature and her increase rate.

If the coefficients are chosen from release in steam, then coincidence of calculation and experiment for steam is sufficiently satisfactory, and discrepancy don't exceed the evaluated experimental error (~ 12%). However, if these coefficients are used to calculate FP release in helium, then the discrepancy equals approximately 3 times ( fig.3). Therefore the taking into account of medium characteristics ( gas, air, steam) must to become subsequent improvement of calculational model.

Besides MELCOR the more correct code SCDAP/RELAP/MOD 3.1, developed in INEL (USA), was used in IPPE for analysis of experimental data. This code take into account FP migration in fuel grains and voids, oxidation of fuel element cladding, dissolution $UO_2$ in zirconium alloy at al. For example, on fig.4 and 5 experimental data of Cs and I release are compared with given code results. Follows, that code SCDAP/RELAP/MOD 3.1 predicts volatile FP release with delay from experimental data. Nevertheless, the data of FP overall release at all period of experiment are sufficiently coincided.

## 4. INFLUENCE OF FUEL TEMPERATURE ON RELEASE RADIOLOGICAL HAZARD

The FP release information is initial data for their transport in NPP containment and environment at accident with taking into account the effect of safety system, cleaning filters, protective barrier et al. The result radiation situation depends on specific scenario and scale of accident.

The FP overall release from fuel at different temperature data are very useful for NPP safety philosophy and specified requirements to safety systems and plant design determination. For example, the calculation of radioactivity, spreading in environment from hypothetical water cooled reactor, sited near Obninsk, were fulfilled at different temperature fuel without taking into account safety localization system influence. The results

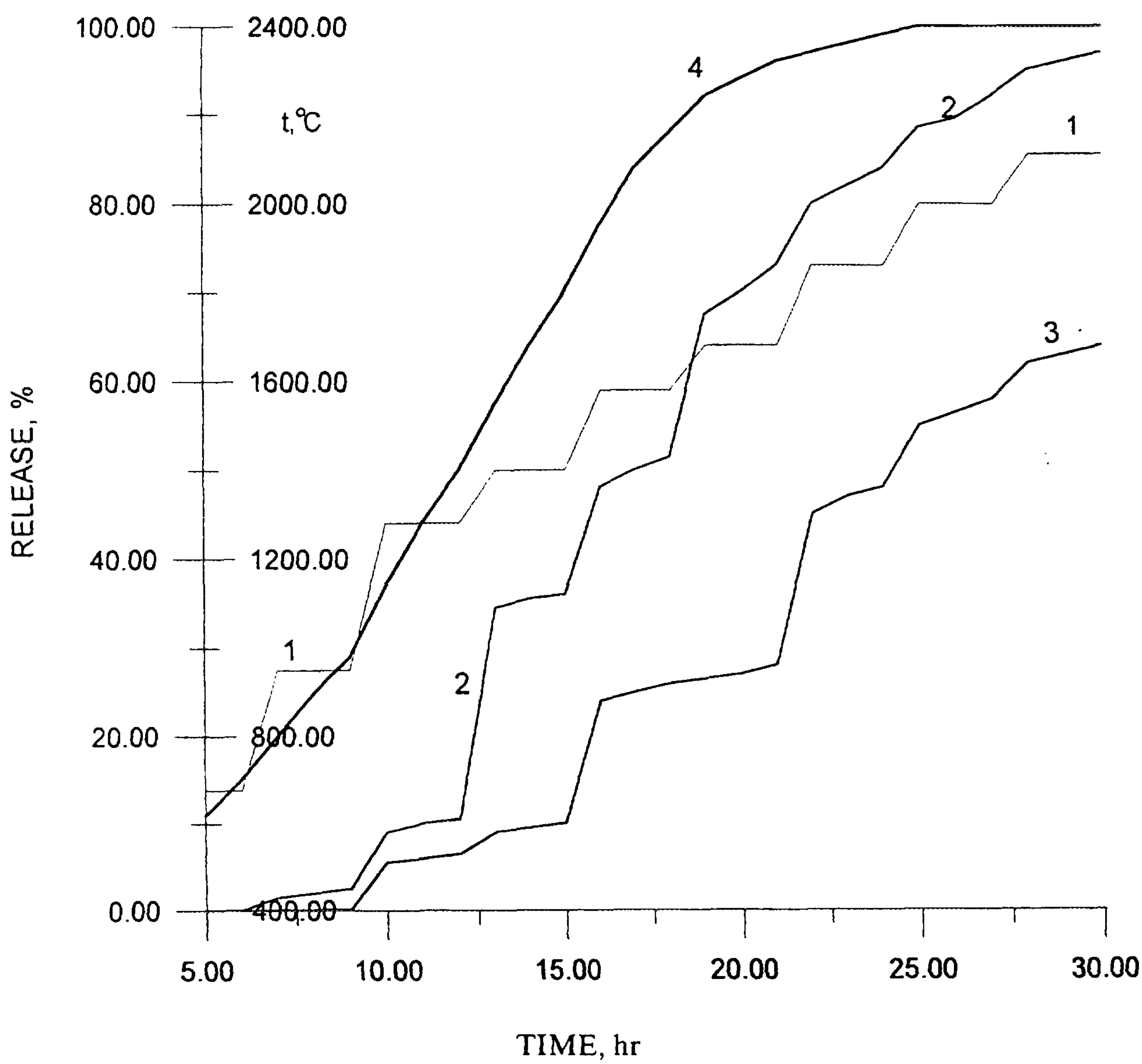



*FIG. 3. Kr-85 and Cs-137 release in helium:*

1 - temperature of heating,°C;
2 - Kr-85 release (experimental data);
3 - Cs-137 release (experimental data);
4 - Cs-137 release ( MELCOR with approximation ln$\alpha$ and "steam" coefficients )

of this calculation give the possibility to understand the influence of fuel temperature on NPP radiation safety.

FP release in steam above mentioned data of IPPE and ORNL were used in the calculation, and missing data for other nuclides are obtained by interpolation of different authors information. The Pu release was not taken into account. The Gauss diffusion model, recommended by IAEA, was used with atmosphere characteristic of given district. The different fuel temperatures are considered : ~ 1200°C, ~ 2100°C, ~ 2400°C. The last case

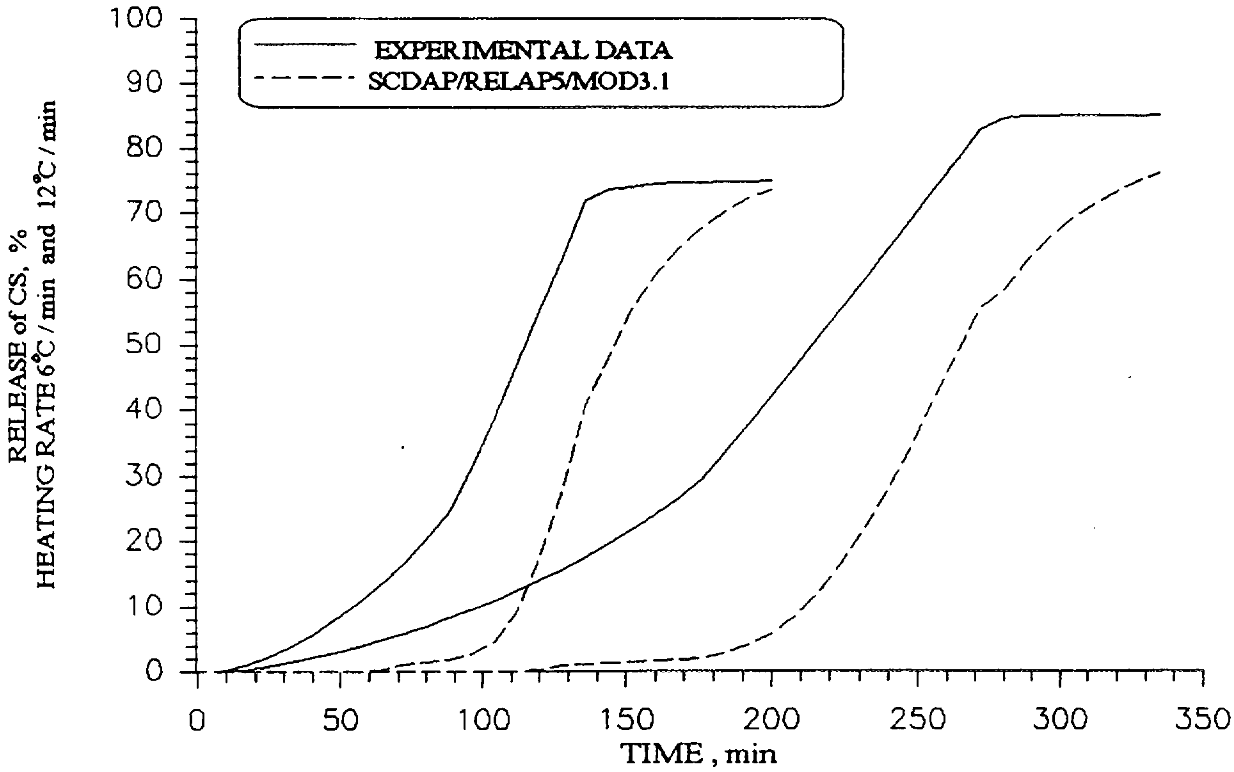

EXPERIMENTAL DATA
SCDAP/RELAP5/MOD3.1
RELEASE of CS, %
HEATING RATE 6°C / min and 12°C / min
TIME , min
0
10
20
30
40
50
60
70
80
90
100
50
100
150
200
250
300
350

*FIG. 4.*

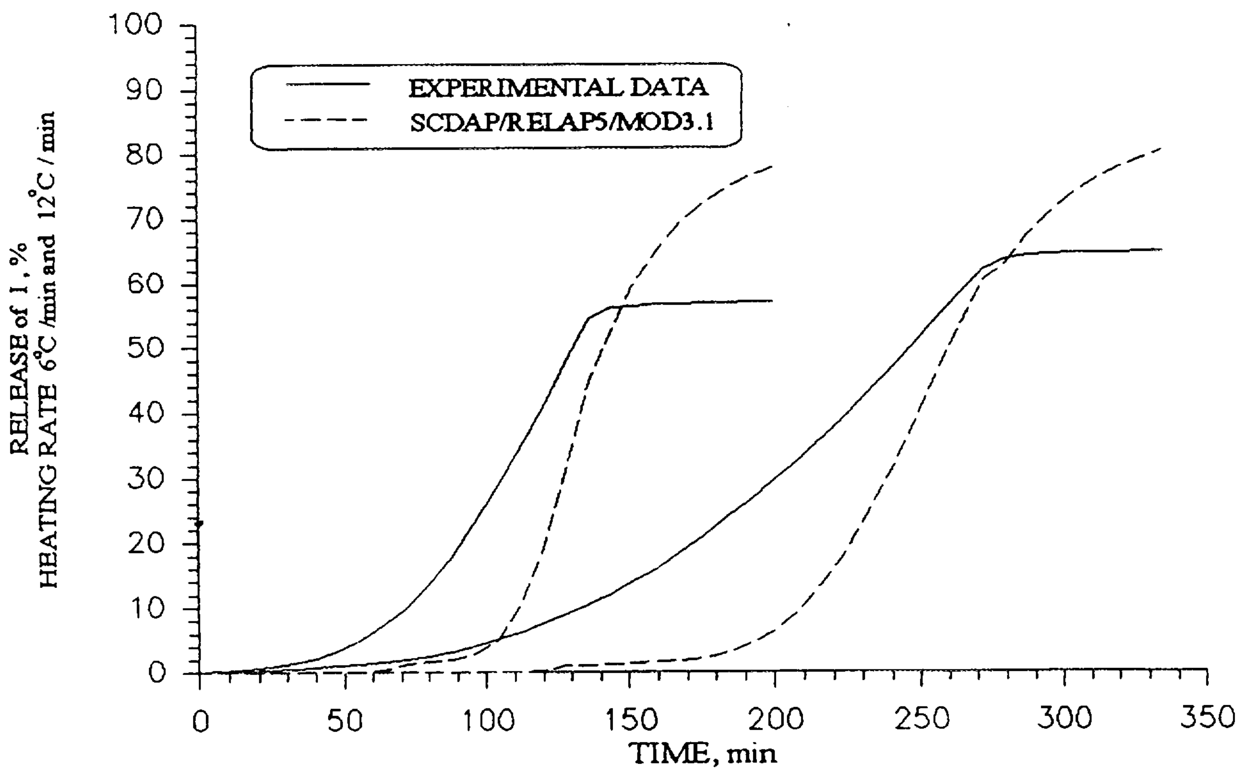

EXPERIMENTAL DATA
SCDAP/RELAP5/MOD3.1
RELEASE of I , %
HEATING RATE 6°C /min and 12°C / min
TIME, min
0
10
20
30
40
50
60
70
80
90
100
50
100
150
200
250
300
350

*FIG. 5.*

describes the beginning of fuel melt. The result effective equivalent dose on earth surface for the case with fuel temperature $t_{fuel}$ = 2100°C is lower at 2.1 times than one for fuel melt, and at $t_{fuel}$ =1200°C one is lower at ~ 8 times.

These results give the next summary :

1. The FP release on initial stage of accident was not negligible in comparison with fuel melt stage.
2. The NPP protective barriers preservation in comparison with fuel temperature factor is dominant.

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# THE DOE TECHNOLOGY DEVELOPMENT PROGRAMME ON SEVERE ACCIDENT MANAGEMENT

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**Abstract**

The US Department of Energy (DOE) is sponsoring a program in technology development aimed at resolving the technical issues in severe accident management strategies for advanced and evolutionary light water reactors (LWRs). The key objective of this effort is to achieve a robust defense-in-depth at the interface between prevention and mitigation of severe accidents. The approach taken towards this goal is based on the Risk Oriented Accident Analysis Methodology (ROAAM). Applications of ROAAM to the severe accident management strategy for the US AP600 advanced LWR have been effective both in enhancing the design and in achieving acceptance of the conclusions and base technology developed in the course of the work. This paper presents an overview of that effort and its key technical elements.

## 1. INTRODUCTION

A comprehensive severe accident management strategy which integrates safety goals and methodology of assessment into a unified tool for ensuring defense-in-depth and resolving safety issues has been developed. Known as the Risk-Oriented Accident Analysis Methodology [1], it provides rational approaches to focusing research and development efforts to obtain answers needed for commercial nuclear plant licensing.

The basis for development of ROAAM is found in philosophical and practical difficulties in quantifying likelihood in presence of large uncertainties in knowledge (epistemic uncertainty) [1]. Specifically, such difficulties have been encountered in addressing certain containment challenge mechanisms that generically became known as "severe accident issues," and ROAAM was specifically developed as a tool to facilitate their resolution. In this **issue resolution context**, ROAAM provides a synergistic collaboration among experts (nationally and internationally) on a particular issue; and facilitates that collaboration by a technical and procedural framework whose key elements are problem decomposition and explicit identification and treatment of any part that cannot be approached in a demonstrably quantifiable fashion ("intangibles"). Such an approach has been shown to be uniquely suited to achieving resolution.

However, resolution of individual issues is not particularly useful unless the results can help establish unambiguously whether an adequate level of safety has been achieved. The acceptance criteria must be derived from philosophically sound safety goals, and the path to closure must be **clear**, **consistent**, and **complete**. Accordingly, the methodology was also developed towards satisfying the needs in this direction. It is known as the **Integrated** ROAAM [1, 2, 8].

Key components of the Integrated ROAAM are: explicit *a priori* integration of probabilistic and deterministic elements, consistency among them, and utilization of this duality to achieve and demonstrate defense-in-depth. This leads to the safety goal that containment failure is "physically unreasonable" for all accidents that are not "remote and speculative". The term "remote and speculative" refers to frequencies based on reliability considerations, and "physically unreasonable" refers to verified applications of the basic laws of physics as implemented by ROAAM in its issue resolution context. Adoption of this evaluation technique leads to a way to evaluate rare, high-consequence hazards, which are difficult to quantity in a probabilistic manner and hence even more difficult to evaluate in a regulatory context based solely on a probabilistic risk assessment.

In its implementation, the integrated ROAAM begins with a complete systems analysis along the lines of a Level 1 Probabilistic Risk Assessment. This is used to define major accident classes and associated plant damage states, and to compute respective frequencies. A quantitative definition of a remote and speculative level is then made, and the resultant "screening frequency" is used to identify those accident classes which must be considered. For these classes, containment failure must be shown to be "physically unreasonable." This is defined as the severe accident management, or mitigation, window. The strategy can then be optimized by deriving the effect of system changes on the accident content within the window, and of containment hardware as they affect the physics of mitigation [2].

Considerable experience with ROAAM has been accumulated. Applications in its issue resolution context include the α-mode (steam-explosion-induced) containment failure [3], the Mark-I liner attack problem which is relevant to core melt accidents in boiling water reactors with a Mark-I containment configuration [4], and the direct containment heating (DCH) issue [5]. Resolutions have been obtained in all three areas [3, 4, 5, 24]. Further, in its integrated context, ROAAM has been applied to severe accident assessment and management for the Loviisa plant in Finland [8] and more recently to the AP600 design [2]. Current work includes evaluations of lower head integrity under thermal loads (referred to as "in-vessel retention") and under dynamic loads (referred to as "in-vessel explosions").

### 1.1 The use of ROAAM in the DOE approach to Severe Accident Management (SAM)

Using ROAAM in its integrated context, the DOE approach that has been adopted for the SAM strategy for advanced LWR designs is to focus on areas where a reliability-type approach is appropriate, and others where the phenomenology of the event itself is the central issue. This is done in terms of the prevention-mitigation interplay evident in the safety goal stated above. That is , on the one hand aiming to eliminate inherently uncertain scenarios, so as to allow the "physically unreasonable" to be clearly demonstrable, while on the other, subjecting the equipment procedures necessary for such elimination to the criteria that failure is "remote and speculative" (the "screening frequency level") subject to technological constraints of the specific reliability achievable. Using the Westinghouse AP600 design [6] as an example, this approach leads principally to three main reliability components, and two mainly phenomenological ones. They are:

<u>Reliability</u>

- Depressurize the reactor pressure vessel (prevent high pressure core melt ejection)
- Cool the containment - external spray cooling
- Control hydrogen gas buildup by passive autocatalytic recombiners

Phenomenology

- External coolability of reactor vessel containing relocated core melt
- Maintaining lower head integrity under steam explosions loading

The DOE program is focusing research efforts on opportunities for major advances in these two phenomenological areas. The objective is to provide rational approaches to addressing them, within accident scenarios relevant to particular reactor designs, and the data bases and computer codes needed to fully support these approaches. Emphasis is placed on focusing research to be efficient and effective, and on procedural approaches to involve the international expert community toward convergence and resolutions of the issues. For this, ROAAM is used in the issue resolution context for the two main research areas of **in-vessel core melt retention** and **in-vessel steam explosions**.

## 2. IN-VESSEL RETENTION (IVR)

The work on IVR involves two major experiments, ULPU and ACOPO, thermal and structural models of the debris and lower head, and an integration approach for assessing likelihood of failure. The basic document on this issue [7] was prepared and sent for a ROAAM review to 18 experts at the end of 1994. Resolution was reached after iterations with the authors, and the result is being used to support the AP600 treatment of debris coolability in the vendor's design certification application to the US Nuclear Regulatory Commission (NRC). The ULPU and ACOPO experiments described below played a key part in the resolution of this issue.

### 2.1 ULPU-2000 Experiment

The ULPU experiment [7] is a full-scale simulation of a nuclear reactor pressure vessel lower head, heated internally and submerged in a pool of water. The simulation is made in a vertical "slice" geometry, which allows a direct visualization of the heat transfer phenomena (Figs 1a , 1b). The experiment provides information on the coolability limits (critical heat flux, or CHF) as a function of distance along the arc length from the bottom to the upper edge positions on the hemisphere.

#### *2.1.1. ULPU-2000 Test Facility Design*

This experiment evolved from the original ULPU work which modeled phenomena characteristic of the Loviisa reactor. In that experiment the heater was limited by design and power to 1400 $kW/m^2$, and CHF could not be reached for the range of conditions investigated. In the present experiment, both the power and heater designs were upgraded to allow a peak heat flux of 2000 $kW/m^2$ , and the test section represents a downward-facing hemisphere.

Electric heater assemblies are mounted on the inner radius of the test section to simulate the internal decay heating from a molten pool of corium. The assemblies are made of individual copper block segments with embedded resistance heaters. Power shaping of individual heaters is used to simulate the axisymmetric geometry of the reactor lower head, and the instrumentation included surface microthermocouples. The walls of the vessel which confine the water are equipped with viewing ports through which the boiling and the vapor generated can be observed and recorded.

To understand the effects of local subcooling and vapor condensation effects which influence the CHF, experiments are conducted in two different modes. One mode of operation is to obtain lower bounds for the effects by running experiments at saturated, pool boiling conditions. The other is to allow for a natural convection flow loop in which the water in the downcomer (see Fig 1b), while saturated at the top of the facility, attains a subcooling equivalent to the gravitational head at the bottom of the test section.

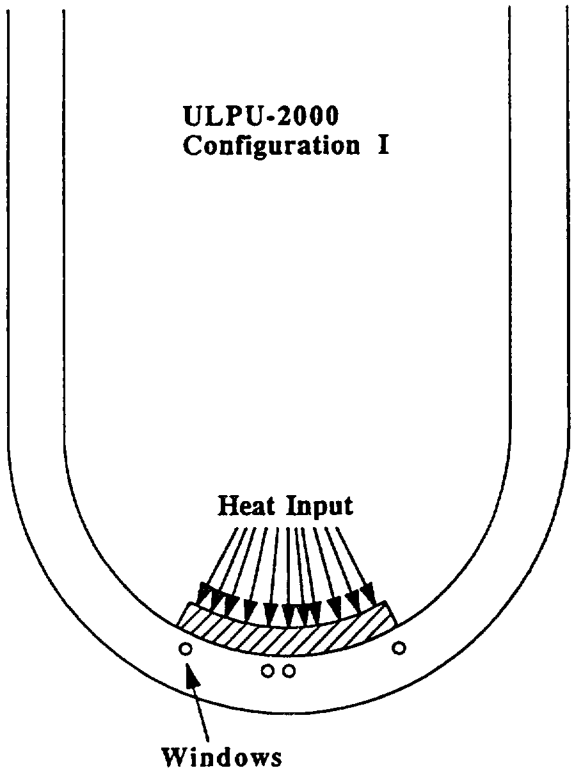



**Figure 1a** Schematic of Configuration I in ULPU-2000. The heater blocks extend over the region $-30° < \theta < 30°$.

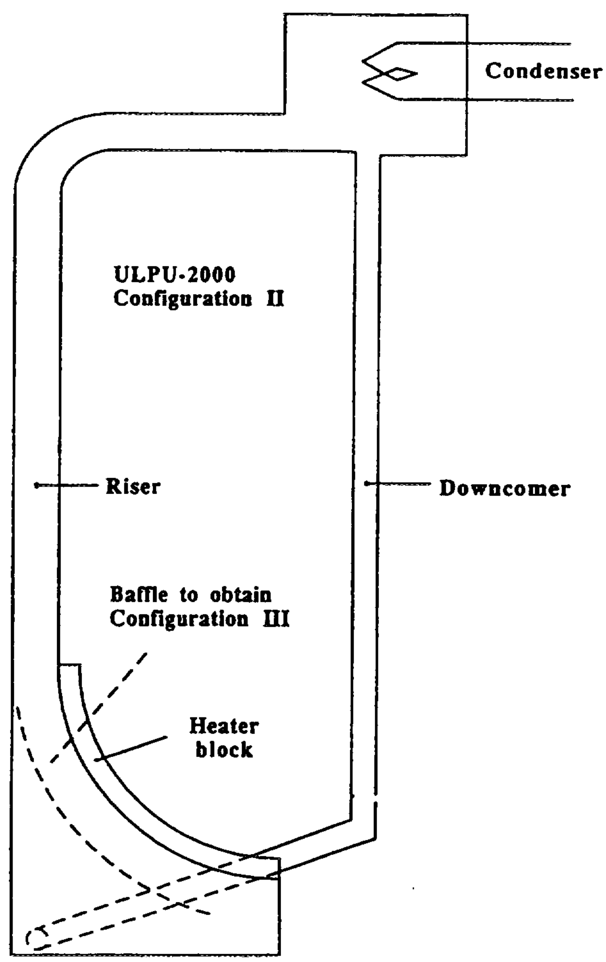



**Figure 1b** Schematic of Configurations II and III in ULPU-2000. The heater blocks extend over the region $0° < \theta < 90°$.

In relation to previous work on CHF, the present problem involves two unique aspects. One is that the vapor generated by boiling remains confined by gravity within a two-phase boundary layer all along an extended (large scale) heating surface. Within this boundary layer, flow velocities and phase distribution depend heavily on the local surface orientation to the gravity vector and on the cumulative quantities of steam generated in all upstream positions. The other unique aspect is that the thick heating surface (up to ~15 cm) has a very large thermal inertia. The ULPU-2000 was specifically conceived to represent these unique features.

*2.1.2. ULPU-2000 Test Programs*

The test programs were run in several configurations (see Figs 1a and 1b). Configuration 1 (C1) testing studied saturated pool boiling in $-30° < \theta < 30°$, and especially around $\theta = 0$ .° Configuration 2 (C2) simulated the complete geometry (a full quarter-circle) under both loop flow or pool boiling conditions. C2 represented an open-to-the-cavity geometry. Configuration 3 (C3) had a baffle to represent the thermal insulation that surrounds the reactor vessel, and the resultant channel-like geometry.

Most experimental runs were carried out in C2 to determine CHF as a function of power shaping, water subcooling, and recirculation flow rates. Additional tests examined the effect of surface wettability changes due to "aging."

*2.1.3 Conclusions Drawn from the Experiments*

An important first conclusion to be drawn from the experiments is that reliable full-scale simulations of the CHF distribution on the lower head of a reactor vessel submerged in water have been efficiently obtained with the ULPU-2000 facility. Secondly, the tests that were run indicate that the margins between predicted thermal load distributions and CHF are very large.

More recent experiments utilizing microthermocouples and high-speed videotaping have identified the presence of a new boiling transition regime with significant coupling between the overall system dynamics and the microphenomena [9]. These experiments lead to the conclusion that the coolability of the curved, inverted surface is controlled by microlayer evaporation, and by the time available between successive liquid contacts as dictated by system pulsations.

**2.2 Axisymmetric Corium Pool (ACOPO) Experiment**

The ACOPO experiment is a half-scale simulation of a nuclear reactor pressure vessel lower head in hemispherical geometry [10], as in the lower head of the US AP600 design. The test is designed to provide information on natural convection heat transfer (the thermal loading), from a simulated pool of molten corium in the lower head, over the range of prototypic Rayleigh numbers from ~$10^{15}$ to $10^{16}$. The ACOPO test is the successor to the COPO experiment [12], which provided data in the Rayleigh number range of interest but in a two-dimensional slice geometry, and to the mini-ACOPO, a one-eighth scale proof-of-concept for ACOPO. Previous notable work in this area was the pioneering research of Mayinger at the University of Hanover [14,23] and Dhir at the University of California-Los Angeles [11].

*2.2.1. ACOPO Test Facility Design*

The ACOPO facility is designed to help overcome difficulties in reaching the range of Rayleigh numbers of interest in the axisymmetric geometry. These difficulties arise from a strong dependence of the Rayleigh number on the characteristic length scale, and from the need to provide

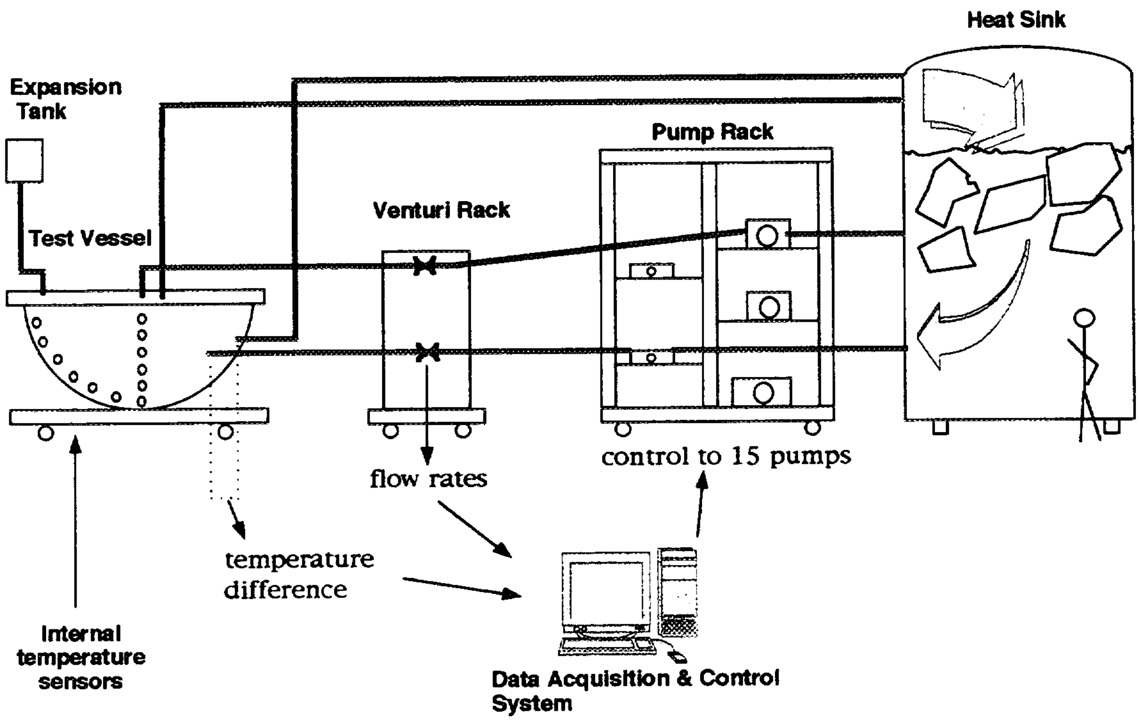
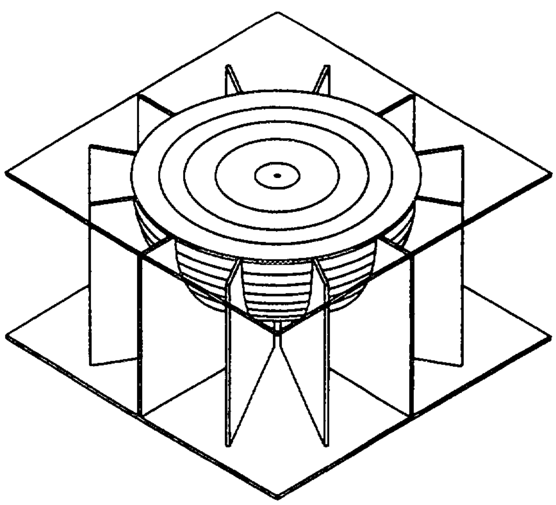



**Figure 2a** The ACOPO Half-Scale Facility



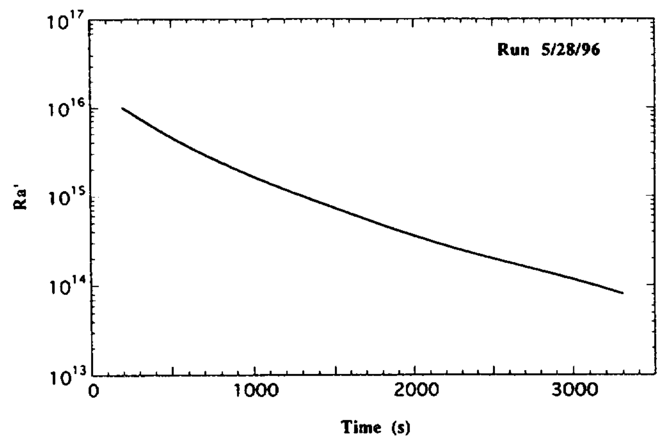



**Figure 2b** Schematic of ACOPO Test Vessel **Figure 2c** Rayleigh Number Transient in ACOPO Run

a uniform volumetric heating on a large scale and in a hemispherical geometry. The ACOPO design resolves these problems by the large scale of the test section and by using the internal energy of a preheated fluid to simulate volumetric heating. A rapid cooling of the boundaries of the hemisphere containing the heated fluid then provides a transient cooldown, which is interpreted as a sequence of quasi-stationary natural convection states. The earlier experiments with mini-ACOPO demonstrated and confirmed this approach.

The test section is a large (2 m) diameter hemisphere fabricated of square copper tubing cooling coils which are individually served by cooling units (Fig 2a). There are ten cooling zones on the hemispherical structure and five on the lid of the vessel (Fig 2b). Chilled water is used as the circulating fluid in the cooling coils; it is circulated through an ice-filled external steel tank to maintain cooling water temperature near 0° C. Individual cooling coils are controlled for flow rate and monitored for flow and temperature by venturis and thermistors respectively.

The copper hemisphere is filled with water and the contents are heated to ~ 95°C by recirculating through an external heater. Then, a topping procedure is carried out with heated and degassed water to ensure that no air is trapped under the vessel lid.

The apparatus is insulated on the outside, and special care was taken to avoid external heat flow paths to the cooling units. As a check, the energy balance between the calculated energy loss from the vessel contents and that transferred to the cooling units was found to be well within 10%.

### *2.2.2. ACOPO Test Program*

Experiments are run in the ACOPO by heating the vessel contents slowly with the external heater and final steam injection. The cooling circuits are then switched on to start the cooldown. Typically the experiment is continued for ~ 1 h.

As of the date of this presentation, five experiments have been conducted with highly reproducible results. The range of Rayleigh numbers from a typical run (dated 5/28/96) is shown in Fig 2c.

Analysis of the experiments provides upward and downward heat transfer data which are then compared with correlations based on the Nusselt number (Nu). The results of the ACOPO test runs are well correlated by

upward heat transfer: $$Nu_{upwd} = 1.95\ Ra^{0.18} \quad (1)$$

downward heat transfer: $$Nu_{down} = 0.3\ Ra^{0.22} \quad (2)$$

In the range $10^{15} < Ra < 10^{16}$, Eq. (1) is slightly lower than the well-known Steinberner-Reinecke correlation [14], and Eq. (2) is slightly above the Mayinger and mini-ACOPO [13] correlations. The heat flux shape was found to be in excellent agreement with that obtained in the mini-ACOPO.

The current work is oriented to more fundamental aspects, such as internal flow structure and its relation to the local heat transfer behavior.

### *2.2.3. Conclusions drawn from the experiments*

The in-vessel retention analysis for the AP600 design was based on the Steinberner-Reineke correlation and the mini-ACOPO correlation for upward and downward heat transfer, respectively. The difference between using these early correlations and the above experimental correlations shows that previously, the upward flux was underestimated by less than ~10%, while the downward flux was overestimated by less than ~6%. These variations are negligible in the context of the analysis and the margins to failure reported. *These experiments provide important confirmatory support to the validity of the in-vessel retention severe accident management strategy for AP600-type designs.*

## 3. IN-VESSEL STEAM EXPLOSIONS (IVE)

This work involves two major experiments, MAGICO and SIGMA, and two computer codes, PM-ALPHA and ESPROSE.m, to address the premixing and propagation, respectively, of steam explosions. Application of these tools were integrated (under ROAAM) with the melt relocation physics and the structural response of the lower head under impulsive loads to assess the likelihood of failure. This is the second component (IVR being the first) of a comprehensive SAM scheme based on lower head integrity. The basic document on this issue [15] was prepared and sent for a ROAAM review to 18 international experts in mid 1996. Supporting documentation on verification of the two codes was recently completed. The MAGICO and SIGMA experiments played a key role in the verification task, and the SIGMA experiment demonstrated the microinteractions concept, which is the key idea for ESPROSE.m. These experiments are described briefly below.

## 3.1 MAGICO-2000/PM-ALPHA

The MAGICO-2000 experiment [16], and the multifield code, PM-ALPHA [17], are the primary tools for studying the premixing phase of steam explosions (the multiphase transient obtained during contact of a high-temperature melt with a liquid coolant). The experiment involves well-characterized, high-temperature particle clouds mixing with water, and detailed measurements on both the external and internal characteristics of the mixing zone. The PM-ALPHA code, which is intended to simulate the thermalhydraulic transient, is used to aid in interpreting the experimental results with good predictive capabilities. The code results which are of most interest are the mixing zone compositions and associated length scales. These compositions are expressed as volume fraction distribution maps, evolving in time, which can be then used in a propagation code to compute a steam explosion.

### *3.1.1. MAGICO-2000 Experimental Apparatus and Program.*

The experimental objective is to generate a uniform cloud of particles at temperatures of ~ 2000°C which are released simultaneously into a water pool instrumented with thermocouples, videotape capability, and X-radiography. Particle clouds of $ZrO_2$, $Al_2O_3$, and steel were utilized in the experiments. A machined and drilled graphite heating block, electrically heated, contains and heats the particles before the drop into the water pool below. The overall arrangement of the heating block, and of the water pool, are shown in Figs 3a and 3b. Electrical resistance heating of the graphite block raises the particle temperature to the 1500 to 2000°C range in 7 to 10 h.

Experiments were conducted in two series: one, which addressed momentum interactions, was at room temperature ("cold" pours), while the second series was carried out with high temperature particles ("hot" pours) for phase change effects. Additionally, single-particle runs were done to test the techniques. Interactions during the pours were recorded on videotape; the particle cloud position and the liquid region interior boundaries were determined by flash radiographs. Measurements of the liquid swell, the particle cloud density, local void fraction, the water "hole" produced by the particle cloud in cold pours, and the height of the spray dome were all analyzed and compared with the predictions of the PM-ALPHA code.

### *3.1.2. Conclusions Drawn from the Experiments*

The MAGICO-2000 facility provides a unique capability to produce uniform particle clouds at high temperatures for study of detailed interaction with water pools. Interesting phenomena identified with the cold pours have included the formation of densely packed regions at the penetration front, formation of a cavity behind the dense particle clouds, and development of fingerlike instabilities at the penetration front. Hot pour phenomena which have been quantified include local voiding in the mixing zone, global voiding through the level swell, and the effects of pool water subcooling on the above. The use of the PM-ALPHA code has helped to interpret these

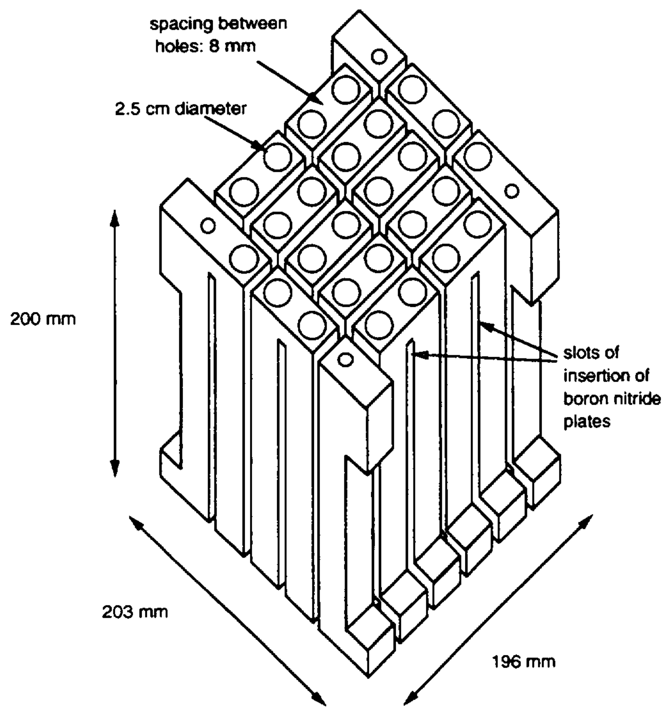



**Figure 3a** Heating element in MAGICO-2000

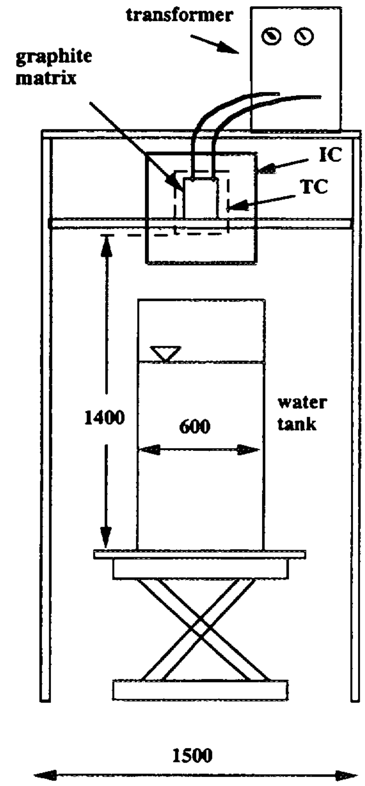



**Figure 3b** Schematic of MAGICO-2000. All dimensions in mm.

phenomena, and to establish a good predictive capability of the multifield aspects of the premixing phenomena. For the treatment of melt breakup and integral aspects of premixing with PM-ALPHA, see Ref. 25.

### 3.2 SIGMA/ESPROSE.m

The SIGMA facility [18] continues the investigation of steam explosion phenomena by experimental quantifications of the recently-introduced concept of microinteractions. In this approach, the fragmentation kinetics of hot liquid drops (e.g., molten corium) in another liquid (e.g., reactor coolant) under the influence of sustained pressure pulses are observed in the SIGMA hydrodynamic shock tube experiments. The underlying physics of the concept postulate that only a small quantity of coolant around each premixed melt mass sees the fragmenting debris coming from it (as opposed to a concept of homogeneous mixing of the fragmented debris with the coolant). The study of this in a controlled and quantifiable way is done by subjecting single drops of melt to a simulated explosion environment in the SIGMA facility.

The analytical formulation of this process is implemented in the ESPROSE.m code [19]. This code is designed to simulate the escalation and propagation of steam explosions, i.e., the wave dynamics through a given premixture and the surrounding medium, following an applied triggering pressure pulse. The outputs of the code are the pressure loads on adjacent structural boundaries, and kinetic energies of any mobile masses subjected to the explosion. The pressure loads can then be used in a structural analysis code to assess potential for failure; the kinetic energies are utilized for addressing mechanical damage from mobile object collisions.

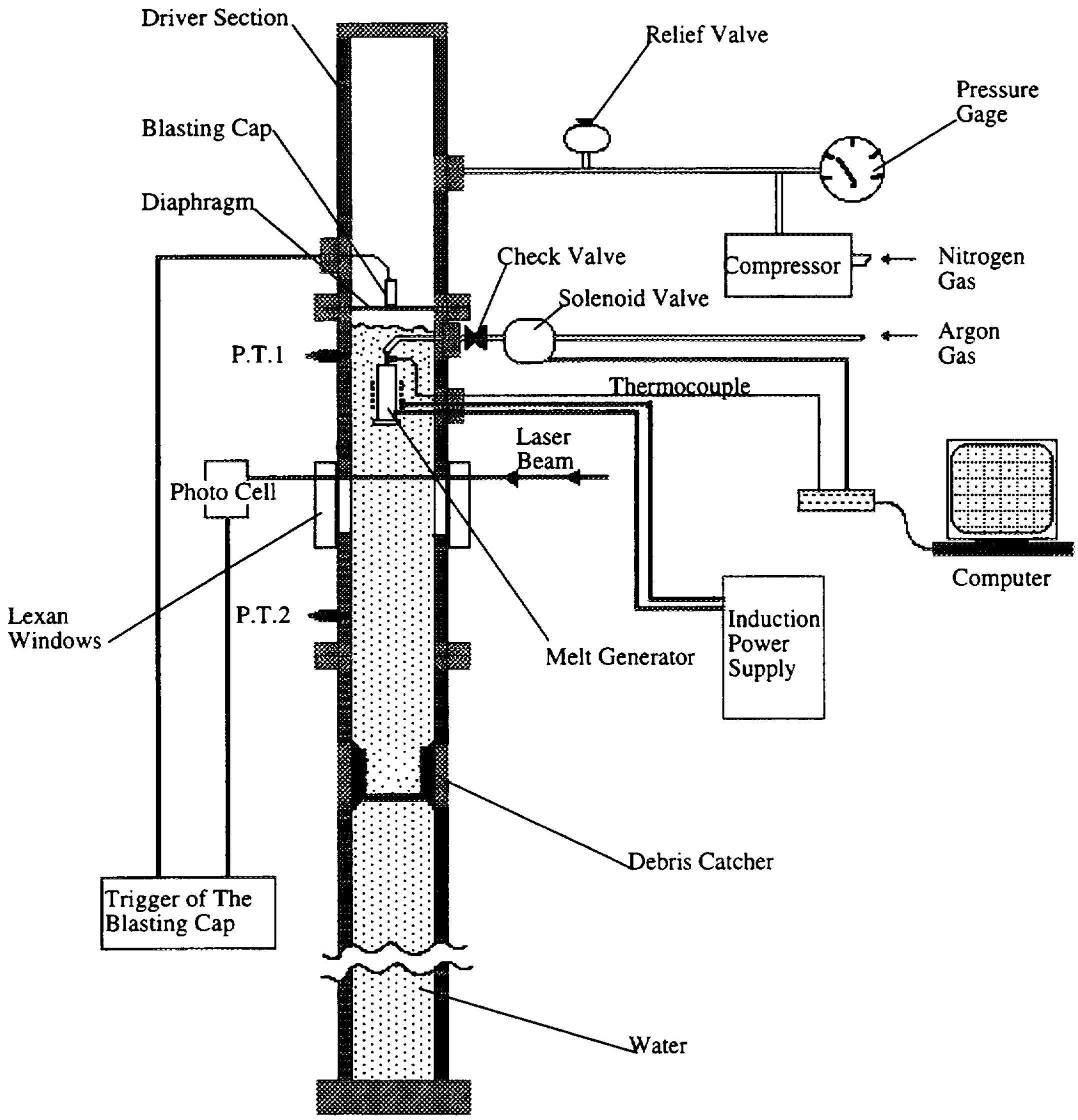



**Figure 4** Schematic of the SIGMA-2000 facility

### *3.2.1. SIGMA-2000 Experimental Apparatus and Program*

The basic component of the SIGMA facility is a shock tube with a 1 m long driver section and a 2 m long expansion section (see Fig 4). The design pressure is 1000 bar; with water in the expansion section, a pressure pulse of up to ~ 2 ms can be achieved before reflection of the shock wave from the tube end arrives back to the interaction location. The shock wave is initiated by rupturing a pre-scored diaphragm pressure boundary with an explosive cap.

A melt generator which is designed to melt and reproducibly release single drops of test material at temperatures up to 1800°C is placed above the viewing window. An induction coil heats a graphite crucible containing the test material. At the desired temperature, a single molten drop is released into the water-filled expansion section, and the explosive cap is triggered. The experiment is timed by means of a laser beam which detects the molten drop and initiates the firing sequence. High-speed movie and x-ray images provide information on the evolution of the microinteraction region, while transient pressures in the shock tube are measured with quartz piezoelectric transducers.

To date, the experimental program has utilized materials ranging from gallium and tin to aluminum and steel oxides, with varying melt and water temperatures, at shock pressures ranging from 68 bar to ~ 280 bar. Future experiments are planned with zirconium and uranium oxides.

### *3.2.2. Conclusions Drawn from the Experiments*

The SIGMA-2000 facility allows the observation of exploding molten drops under conditions resembling those in larger scale detonations. These experimental observations have provided input to the microinteraction models used to compute the propagation phase of steam explosions with ESPROSE.m. This derived capability to compute steam explosions has been verified by comparison to integral experiments, properly interpreting results ranging from weak propagations to strong detonations [26].

## 4.0 RELATED PROGRAMMATIC WORK

In addition to the experimental programs described above, the US DOE provides support to, or liaison with, other programs in severe accident research. These are the MACE program being managed by EPRI with ANL as the executing agency, and the RASPLAV program sponsored by theOECD-NEA with the Kurchatov Institute in Moscow as the executing agency. In addition, a contingency experiment in molten core retention (PACOPO) has been designed as a potential follow-on to the ACOPO test. All of these tests involve the use of prototypical corium mixtures as molten material.

### 4.1 Melt Attack and Coolability Experiment (MACE)

The MACE experiment [20], for which the US DOE is a major co-sponsor, is being conducted at ANL in cooperation with an EPRI-led international consortium which earlier was responsible for performance of the Advanced Containment Experiments (ACE). The MACE program addresses the use of water to terminate and stabilize a core melt accident at the ex-vessel stage, wherein molten corium is attacking basemat concrete. The M3b test is currently being prepared for execution in December 1996. In this experiment, a molten core/concrete interaction (MCCI) is created involving 2000 kg of molten corium (with a nominal composition of 57 $UO_2$/29 $ZrO_2$/6 Cr). The corium is generated at ~ 2500K initial temperature via an exothermic chemical reaction which takes place in ~ 30 s. The melt is thereafter internally heated at 300 W/kg $UO_2$ by use of direct electrical heating. The corium attacks and decomposes the underlying concrete basemat (which is composed of limestone and common sand for test M3b). At an ablation depth of 5 mm, water is flooded in to the test section. The M3b test is a repeat of a previously unsuccessful test, M3, which experienced moisture contamination in the corium reactants. Measures to strictly control and detect moisture contamination have been instituted for M3b.

### 4.2 The RASPLAV Program

The RASPLAV program being conducted at the Russian Research Center-Kurchatov Institute (RRC-KI) is sponsored by the OECD-NEA as a multinationally-supported project. [21]. The US NRC is the US participant in the program. At the request of the NRC program management, and with the approval of RRC-KI, DOE has supplied a contractor (University of California-Santa Barbara) to assist in the planning, evaluation and review of the program. The main experiment of this program involves 200 kg of prototypic melt contained by an externally cooled steel wall. The experiment is being conducted in "slice" geometry using molten corium at ~ 3000 K which is heated by induction through the parallel semicircular sides of the apparatus. The first such test has just been successfully completed. Other important aspects of this program are measurements of thermophysical properties of corium melts and natural convection studies using molten salts.

### 4.3 Prototypic Axisymmetric Corium Pool (PACOPO) Experiment

In late 1994 the idea was proposed for a DOE-sponsored prototypical material experiment to address heat transfer in an in-vessel molten corium pool by using the ACOPO experiment principle [22]. The experiment would utilize ~ 2000 kg of prototypic thermitic materials with an initial superheat of ~ 250 K and would be conducted in the same facility at ANL as the MACE test. A peer review of the ANL proposed design was undertaken in late 1995 which provided valuable guidance. However, budgetary restrictions and the imminence of results from the RASPLAV experimental program led to deferral of further work at this time on this concept.

## 5.0 CONCLUDING REMARKS

The work described above has been planned and executed with the intention of achieving closure to the concerns that arise in severe accident issues. We believe that significant progress has been made towards this goal, as evidenced by the wide acceptance of in-vessel retention as a key severe accident management strategy. This adds to the previous ROAAM results on $\alpha$-mode failure, Mark-I liner attack, and DCH, which were carried out under the sponsorship of the Nuclear Regulatory Commission, and helps to establish a new approach in addressing defense-in-depth at the containment integrity level. Special efforts have been made to involve the international safety community in this work and to synergize with it towards effective and widely accepted severe accident assessments and management approaches. We have confidence that the management goals can be efficiently achieved for advanced reactors with closure close at hand. Overall, this work is believed to be very beneficial for the future of nuclear power by removing uncertainties and thus increasing public acceptance. This work, focused so far on advanced reactors, could also benefit existing reactors, and we believe we can have an overall beneficial effect on the future of nuclear power.

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# ACCOMPLISHMENTS AND CHALLENGES OF THE SEVERE ACCIDENT RESEARCH

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**Abstract**

This paper describes the progress of the severe accident research since 1980, in terms of the accomplishments made so far and the challenges that remain. Much has been accomplished: many important safety issues have been resolved and consensus is near on some others. However, some of the previously identified safety issues remain as challenges, while some new ones have arisen due to the shift in focus from containment integrity to vessel integrity. New reactor designs have also created some new challenges. In general, the regulatory demands in new reactor designs are stricter, thereby requiring much greater attention to the safety issues concerned with the containment design of the new large reactors.

## 1. INTRODUCTION AND BACKGROUND

The light water reactor (LWR) systems engineered and constructed in the western countries followed a definite design philosophy for ensuring a very low level of risk to the public. Briefly, the plant systems are designed with the defense in depth concept. The systems are designed to withstand with a single failure and prevent a severe accident in which core damage could occur. The design goals for core damage frequency range from $10^{-4}$ to $10^{-6}$/reactor year. The plant systems are also designed to withstand the loadings due to the design-basis accidents and incidents, and specified external events, e.g., earthquakes, fires, tornadoes, floods etc. In addition, with characteristic foresight, the designers provided a strong containment system to contain any fission product radioactivity produced even in the beyond-the-design-basis accidents. The containment structures are designed to withstand pressures much beyond those imposed by the energy release during the design basis accidents. Mitigation measures are provided in the containment buildings e.g., the suppression pool in the boiling water reactors (BWRs) and the sprays, fan coolers and ice condensers in pressurized water reactors (PWRs) for long term heat removal from the containment buildings. The objectives of these containment safety systems is to keep the pressure low and protect the integrity of the containment in the beyond-the-design-basis accidents. In terms of public safety, it is perhaps self-evident that if containment integrity is not violated public safety is not compromised. The severe accident, even if it progresses to the core melt on the floor, will not be a life-threatening event from the point of view of public safety, if the containment remains intact and leak-tight. Adequate performance of the containment in the aftermath of a postulated severe accident, thus, is of vital concern. In particular, it has been determined that maintaining the integrity of the containment for the first few hours, after any fission product releases in the severe accident, can reduce the containment airborne radioactivity by orders of magnitude. This is a direct consequence of the time constant for aerosol deposition on the containment walls and floors. Early containment failure, thus, has to be obviated by design or by accident management. Late failure of the containment has also been questioned recently. Perhaps, the public anathema to evacuation and to even a minor land and water contamination is forcing a re-examination of the regulatory attitudes and safety philosophy. Consideration of the requirement of 24 hours as the time for containment leak tightness for the new plants in USA and the moves in Germany towards the design of the containment, which will not fail under extreme loadings, are indicative of these new attitudes and philosophy. These containment performance goals, laudable as they are, for the new plants, will be difficult to achieve if the old evaluation philosophy of using conservatism at each step is employed. Thus, it is imperative, that the new containment performance goals are accompanied by rational evaluation methodologies.

A severe accident by definition involves melting of the core and release of radioactivity. Clearly, the phenomena involved in a core-melt accident are very complicated, since the main characteristics of the accident scenario are the interactions of the core melt with structures, and water, and the release, transport and deposition of the fission product carrying vapors and aerosols. The interactions of core melt may lead to (i) ablation of structures (ii) steam explosions and (iii) concrete melting and gas generation. These phenomena involve the disciplines of thermal hydraulics, high temperature chemistry, high temperature material interactions, aerosol physics, among others. Predictions of the consequences of a severe accident have to be based on experimentation and models whose veracity may be limited by the scale at which the information about the phenomenology is derived. Scaling considerations become very important since large scale experiments are very expensive and difficult to perform.

Another aspect about severe accident consequences should be mentioned. The LWR safety systems for the design base accidents have an acceptance criterion: the peak clad temperature has to be maintained below 1200C, while employing conservative methods of analyses. No such criterion exists for severe accidents, which would focus the research adequately. Recently, the core damage frequency (CDF) $\leq 10^{-4}$ to $10^{-6}$, is becoming a criterion for severe accidents. This, however, is a probabilistic criterion and is subject to some interpretation. The CDF criterion also is not used as a design basis, but as a design goal. In the same vein, the research accomplishments are harder to evaluate, since there is no specific measure.

As mentioned above, it became clear quite early, and confirmed by the WASH-1400[1] and NUREG-1150[2] studies, that the containment had a central role in protecting the public against the consequences of a severe accident. Thus, the focus of the severe accident research, became the evaluation of the survivability of the containment for the various severe accident scenarios. More recently, the focus has shifted a little, due to the accident mitigation perspective, from the survivability of the containment to that of the survivability of the vessel. Vessel external flooding has been adopted in the AP-600 design[3].

In this paper, we will describe the progress of the severe accident research, in relation to the public safety issues posed by the hypothetical severe accident scenarios. Several issues were identified previously and the research work was focused towards resolution of those. New issues have been identified due to the changing attitudes about public safety, and by the designs of new reactors. We will attempt to describe the status of the research work focused towards the resolution of the new issues.

## 2. In-Vessel Accident Progression

A severe accident in a PWR starts with core uncovery initiated by loss of reactor coolant inventory and failure of some of the reactor safety systems. The in-vessel progression of the accident, from that point on, is determined by thermal-hydraulics and material interactions. If accident management actions are not successful, the rise in core temperatures due to undercooling leads to exothermic Zircaloy oxidation transient which delivers heat to clad and fuel at a very large rate (≅10 times the decay energy rate), a large amount of hydrogen is produced and released to the containment. Core temperatures rise at the rate of 1 to 10K/sec; melting starts with the structural and control rod materials and progresses in turn to clad, fuel eutectic, and fuel. Substantial loss of geometry takes place, and a melt pool may be formed within the original core boundary as happened in the TMI-2 reactor. Eventually, the molten core material may be discharged, as a jet, to the lower plenum as occurred in TMI-2. Alternatively, the core slumps and eventually attacks, thermally and mechanically, the core support structure. Failure of the support plate or core barrel brings the corium (molten fuel-structure mixture) in contact with water. In time, thermal attack on the vessel lower head occurs and, upon its failure, the melt material is ejected into the containment cavity to begin the ex-vessel phase of the accident.

It is perhaps instructive to delineate the time scales involved in the various phases of the in-vessel accident progression. The core boil-off and the initial heat-up process are relatively lengthy (2-3 hours), before significant core damage takes place. Accident termination during this time is relatively straightforward, if operator is able to add water to the reactor vessel. Clad melting, fuel melting, core blockage and core melt pool formation are relatively shorter duration processes (1/2 to 1 hour), during which access of water to some of the blockages and debris beds formed may become limited. The interaction of the core melt with the lower head water and structure, and the failure of lower head may be relatively longer duration (3 hours) processes if the melt quenches and reheats. Alternatively, if melt quenching does not occur, the lower head may fail relatively fast (minutes). The character of the melt discharged to containment is different in the two scenarios.

Accurate description of the in-vessel phase of the severe accident scenarios has assumed greater importance lately, since it has become evident that the assumptions made in its modeling determine the composition, amount and the rate of corium discharged to the containment, to which the containment loadings are directly related. In particular, if the projected loadings are severe enough to fail a containment soon after the vessel failure, e.g., due to direct heating or hydrogen detonation, the "source term" consequences of a severe accident can be very severe indeed. In addition to the predictions regarding the corium discharge characteristics, other parameters of great interest are:

a) the magnitude and rate of hydrogen generation,

b) the elapsed time before the onset of core melting,

c) the temperature levels of the reactor coolant system (RCS),

d) the fraction of the fission products retained within the RCS,

e) revaporization of the fission products from the RCS surfaces, and

f) the fission product chemical species.

Information about hydrogen generated (and released to containment) is required for its management and for establishing that detonations or transitions to detonation will not occur. Information about the elapsed time before onset of core melting provides the time window, available to the operator, for terminating the accident without the side effects of core damage or fission product release, i.e., before the risk to the utility's investment becomes high. During core-heat-up, a considerable fraction of energy generated may be transferred to the RCS, which may become hot enough to induce local failures. This could change the risk-dominant high pressure accident scenario, thus, accurate prediction of RCS temperature levels is essential in determining the consequences of some of accident scenarios. The fission products retained within the RCS during transport from core to containment are not available immediately as the source term. However, as the temperatures in the RCS rise due to the continued decay heat generation from the deposited fission products, the revaporization phenomenon becomes important, and in time much of the deposited fission products will leave the RCS and enter the containment. The impact of this phenomenon was not fully realized earlier; however, it has become quite clear that revolatilization may play a role in determining the fission product source term for the cases of late containment failure. Information about the chemical character of the fission product source term is not only required for modeling their transport, but also for predicting (i) their reaction with the structures in the RCS, and (ii) the propensity for their dissolution in the RCS or containment water.

Much research has been performed for the in-vessel melt progression phase of a severe accident. A representative experimental research program is CORA[4] in which several bundles representing PWR and BWR fuel arrangements were heated electrically and observations on fuel degradation were obtained. Previously, experiments were performed with the PBF[5] and LOFT[6] reactor facilities, and, currently, PHEBUS[7] experimental program is directed towards in-vessel melt progression.

Clearly, the above research programs have produced results which have reduced uncertainty. The state of knowledge with respect to the PWR in-vessel core melt progression confirms the picture conveyed by TMI-2. It is believed that a melt pool will form in the original core volume and will drain along the side of the core into the lower plenum to commence the loading on the lower head.

There is new information on the effects of accident management actions, e.g., water addition to a hot core. It was found in the CORA tests that this increases the core damage and the hydrogen generation, due to the increase in Zircaloy oxidation by the steam produced. A new experimental program CORQUENCH will investigate this further.

The state of knowledge regarding BWR in-vessel melt progression, particularly, for the higher probability depressurized dry core scenario, is relatively confused. Core wide blockage formation could occur, similar to that for a PWR; however, there is not enough data, or analysis to delineate the conditions, under which it could occur or not occur. Thus, it is conceivable that the BWR in-core melt progression may terminate with failure of the core support plate.

An accident management issue relative to the BWR accident management is that of addition of the cold water to the damaged core in which the control rods may have melted and the boron-carbide accumulated on the core support plate. Investigations on the reactivity effects of the above scenario are currently in progress in an EU project. The power spike, if any, will be mitigated by the Doppler feed back. Nevertheless, fuel damage may increase.

The attack of the melt discharged from the core region on the vessel lower head has not received as much attention as the in-core melt progression. There are, however, now, two EU projects performing experiments, and developing models, for description of the melt vessel interaction (MVI) and melt-water interactions (MFCI). The knowledge base in this area is increasing rapidly. Conclusions of some of the recent research are:

- It appears that immediate failure of the lower head due to the impingement of a melt jet may be physically unreasonable. If a steam explosion does not occur, the melt jet will fragment and form a debris bed in the lower head, which in time will remelt if water is not supplied. The vessel creep, due to the thermal loading, may produce a failure around a penetration, where the melt discharge will ablate the vessel and enlarge the vessel hole.
- In general, it appears that global vessel failure is physically unreasonable for both PWRs and BWRs.

## 3. Early Failure of Containment

The time span of interest is approximately 4 hours after the initial release of radioactivity that occurs during the core heat-up phase of the severe accident. This time span is sufficient to allow 99.9% of the aerosols in the containment atmosphere to deposit on the walls and the floors (and dissolve in water).

Conventional analyses indicate that during this time span, unless accident management actions are successful to keep the damaged core within the vessel, the melt may discharge into the containment and exert thermal, pressure and combustion loads on the containment, which may challenge its integrity. After a prolonged review of the severe accident scenarios, initially by the Containment Loads Working Group, formed by the USNRC and later by the expert panel working with the Sandia Laboratories on the NuREG-1150[2], the following major challenges, which may lead to an early failure of containment, were identified:

- direct containment heating as a result of melt discharge at high pressure from a vessel breach in a PWR,
- melt attack on the liner of the BWR Mark I containment,
- hydrogen detonation, and
- in-vessel and ex-vessel steam explosion.

Each of these challenges, in time, became a severe accident issue and led to several years of concentrated research. Some of these issues are resolved, or close to resolution, while others still are far from resolution. By resolution, we mean a technical consensus is reached on either the adequacy of the existing containment systems to meet the challenge posed with a very high degree of confidence, or, a technical consensus is reached on the necessary measures (accident management and/or back fit), which would impart that character to the existing containment systems.

## 4. Late Failure of Containment

The time span of interest is beyond 4 hours after the initial release of radioactivity in the containment. In this time span, if the melt is discharged into the containment, it is essential that a heat transport system is established within the containment, i.e., the containment heat removal systems, e.g., fan coolers in PWRs and suppression pool coolers in BWRs are functioning. Otherwise, the slow pressurization resulting from either the prolonged heat addition to the containment atmosphere, or the generation of steam from melt (debris bed) cooling, or the non-condensable gases generated from the molten corium concrete interaction (MCCI) can reach pressure levels at which the containment may fail or leak excessively. This may occur after several hours (more than 4), or a few days, depending upon the water availability, the type of concrete and the pressure-bearing capacity of the containment.

Another potential radioactivity pathway to the environment can result from the containment basemat penetration when the melt can not be cooled and it keeps attacking the basemat. This may occur after a day, or after many days, depending upon the heat removal from the melt debris, the type of concrete, and the thickness of the basemat.

The outstanding safety issues, identified for this time span are:

- melt (debris) coolability,
- concrete ablation rate,
- non-condensable gas generation rate, and
- performing of venting (filter) systems.

The most important of these issues is the melt (debris) coolability, since if water is available and the melt can be cooled readily the other issues become moot. Intensive research is currently in progress on melt coolability.

We shall review, in turn, the current status of these safety issues and briefly describe the results of the research performed recently towards their resolution.

## 5. Direct Containment Heating

The direct containment heating (DCH) issue has been around for a long time. Substantial experimental and analytical research, sponsored by the USNRC was performed in the late '80s and early '90s. Accompanied by a stringent peer-review-process this has resulted in a focused effort whose results have led to the resolution of this issue; for the Westinghouse pressurized water reactors.

The experimental research performed previously had employed a 1/30 scale facility, called CWTI[8], at Argonne National Laboratory (ANL) and a 1/10 scale facility, called SURTSY[9], at Sandia National Laboratories (SNL). There were substantial differences between the materials ($UO_2$) thermite at ANL and iron-alumina thermite at SNL and the containment representations used at these facilities. The results obtained, when extrapolated to the prototypic situations, produced contradicting conclusions about containment failure.

This period of confusion was followed by an activity, led by Dr. Zuber of USNRC which developed a severe accident scaling methodology (SASM)[10] and applied it to the DCH phenomena. Although not wholly successful, it crystallized the physics and the interrelated phenomenology of the DCH event, and pointed the way towards a realistic and extrapolatable experimental program. The resulting experimental program employed the same 1/10 scale SURTSY facility at SNL and a 1/40 scale CWTI facility at ANL and performed counterpart (same initial conditions, materials and containment representation) tests. These were complemented by a few tests in a 1/6 scale concrete containment (used earlier for structural-failure testing) and a few tests with iron-alumina thermite in the CWTI facility.

The most characteristic difference between the recent tests[8,9] and those performed earlier was the precise representation of the containment compartments for the Zion and the Surrey plants. The compartments, beyond the respective cavities underneath the vessel and their flow paths, were constructed to scale in both the SURTSY and the CWTI facilities. In addition, the melt was driven under relatively high pressure (7 MPa) by the properly-scaled volume of steam.

The data obtained from the tests performed in the two facilities and in the 1/6 scale facility have shown remarkable consistency and it appears that the DCH-controlling phenomena are equally active at all of these three scales. Additionally, the tests performed with the uranium-based thermite have shown somewhat lower pressurization than the equivalent iron-alumina thermite tests.

Perhaps, the most promising development[9] in the resolution of the DCH issue is the development of credible scaling methodologies based on the insights obtained from examining the experimental data. The convection limited containment heating (CLCH) model developed by Prof. Theofanous and colleagues at University of California, Santa Barbara, and the two-cell equilibrium model developed by M. Pilch at SNL[9] have been used by the authors to explain the experimental results obtained at different scales and their extrapolation to the prototypic situation. Their preliminary conclusions are that the models, validated by the data obtained in the Zion and Surrey representations, predict manageable loads for these containments for the high pressure severe accident scenario. Careful peer review of these findings was performed. Another finding[11] which has a direct bearing on the DCH issue is the high probability of unintentional depressurization occurring during the high pressure severe accident scenario. The reason is the establishment of natural circulation flow loops in the vessel, hot legs and the steam generators, which can transfer the energy from the core, during the heat-up phase, to the piping system. An elaborate program of 1/7 scale experiments performed at the Westinghouse laboratories, corresponding scaling analysis and the computer code simulations all point to the high expectation of the creep rupture of the surge line to the pressurizer before the vessel rupture. The depressurization induced will also bring water from the accumulators to the dry and hot core and change the high pressure scenario completely.

Other diversions of the classic high pressure (TMLB) scenario can occur. For example, one diversion is the failure of the pump seals. This small break LOCA can lead to clearing of the loop seal and also possibly greater thermal loading of the tubes in a dry-secondary-side steam generator, many plants have recently added the capability of depressurizing the primary system using relief valves operated with battery or steam turbine power.

A probabilistic safety analysis (PSA) of the high pressure scenario, with the potential of DCH, has been prepared by the Sandia Scientists. The resolution of DCH, conducted with the involvement of the cognizant technical community, has been successfully concluded.

## 6. MELT ATTACK ON BWR MARK 1 CONTAINMENT LINER

This safety issue was raised due to the short distance between the vessel and the containment liner in the Mark 1 BWR dry well. The contention was that the melt will be able to traverse that distance and melt the steel liner to fail the containment., soon after vessel failure. This issue stood as one of the major sources of risk for the Mark 1 BWR. The expert opinion obtained during the NUREG-1150 probabilistic safety analysis (PSA) work split on the assignment of the probability of the liner melt-through. The probability values, with water present in the dry well, ranged from 0.001 to 1.0. The authors of NUREG-1150 averaged these results to obtain a point estimate of 0.33, which certainly was a very arbitrary estimate of the probability of a sequence which has major source-term consequences for the Mark-I BWRs.

The resolution of this issue began with the report[12] prepared by Professor Theofanous and colleagues on this topic. They developed a formulation called risk-oriented-accident-analysis-methodology (ROAAM), which is structurally composed of a sequence of cause-effect relationships. This formalism, tailored to each process, employs probability estimates to specify the uncertain inputs or conditions and causal relations to consistently interconnect the intermediate stages of the process. The casual relations are based on physics of the particular phenomena in question with use of conservatism wherever appropriate.

The ROAAM methodology was employed to decompose the scenario into the individual components of melt release, melt spreading, melt concrete interaction and attack on the liner. The formalism employed three causal relations and five probability distribution functions to arrive at the probability of liner failure. The analysis was quite comprehensive and the causal relations employed phenomena models validated against experiments; with conservatisms added wherever model uncertainties dictated that. The conclusions derived were that the probability of liner failure, without water present in the dry well, is close to 1.0, while, with the water present in the dry well, the melt superheat and the liner submergence in the melt decreased to such an extent that the liner failure probability decreased to the range of 0.0001, which implied that the liner failure was physically unreasonable.

The ROAAM-based analysis was peer reviewed extensively which led to further investigations by specific working groups on the causal relations related to a) the metal content of the corium discharged from the vessel, b) the melt spreading in the dry well, c) the corium-concrete interaction and d) the creep-rupture failure of the liner. The conservatisms incorporated in the original analysis by Theofanous and colleagues were confirmed, except for the temperature assumed for the failure of the liner. The resulting modification (5) raised the liner failure probability, with water present in the dry well, to the level of 0.001, which can still be labeled as physically unreasonable. Thus, we believe this issue has been adequately resolved.

## 7. HYDROGEN COMBUSTION

The hydrogen combustion loads on the containment were the first to be addressed by the USNRC, since the hydrogen combustion event in TMI-2 triggered a heightened awareness of these loads. The hydrogen rule requires management of hydrogen concentration in the containment resulting from the oxidation of up to 75% of Zirconium clad. This has already been incorporated in the ice condenser, BWR Mark III and BWR Mark II and I plants. The large volume PWR containment were judged to be immune, since the hydrogen concentration did not reach high enough to produce combustion-induced pressure loads, which would threaten containment integrity. The hydrogen combustion loads issue for these plants relates to either high local concentration, or the transition to detonate, which can occur for special geometries (ducts, accelerating flow regions etc.) at relatively low (10%) hydrogen concentrations.

Hydrogen mixing research has been performed at several laboratories and several large experiments have been performed[13]. The recent work has been performed in Japan, with a multi-compartmented simulated PWR containment facility[14]. The overall conclusion derived from these experiments and from analytic studies is that hydrogen mixing is quite efficient and local non-homogenities do not persist for long periods, except when they are coincident with thermal stratification effects. Several experiments have been performed on the transition to detonation and there are instances where these events have been produced in laboratory.

There are, however, scaling difficulties and it is not clear that the prototypic geometries in containment would be prone to such transitions.

The most recent issue[15] with respect to hydrogen combustion is related to the DCH issue, i.e., whether the high temperature hydrogen formed by oxidation of the metallic components of corium, released during high pressure accident scenarios, is prone to detonation at relatively lower concentration. Experiments[16] conducted at Brookhaven National Laboratory have actually shown reduced propensity for hydrogen combustion and/or transition to detonation, at higher temperatures.

## 8. In-Vessel and Ex-Vessel Steam Explosion

The steam explosion loads on the containment were first considered in the WASH-1400 and, because of the assumptions made about the nature of this event at that time, the failure of containment (due to in-vessel steam explosion generated missile) contributed a substantial fraction of the probability for early containment failure. The work on steam explosions, since that time, led to more realistic estimates of the probability of containment failure due to in-vessel steam explosion. The current evaluation is[17] that this conditional probability (i.e., if there is a core melt) is less than 0.001.

Much experimental and analysis-development work is in progress, presently, on in-vessel steam explosions. Experiments have been performed with several kilogram quantities of heated particles and molten materials. An elaborate three-field analysis program: ESPROSE.m[18] and PM-ALPHA[19] codes have been developed. Some of the insights gained are (1) steam explosion probability is much reduced due to the extensive water-depletion that occurs around the fragmented particles of a jet, (2) super-critical steam explosions, however, can not be excluded, (3) the KROTOS experiments with Kilogram quantities of $UO_2$ + $ZrO_2$, show that this prototypic corium mixture does not explode readily. Further research work on in-vessel steam explosions is proceeding and significant progress has been made.

Recently an evaluation of the integrity of the lower head of the AP-600 reactor design subjected to an in-vessel steam explosion has been performed. It was found that the probability of failure of the lower head is ≤ 0.001[20]. This evaluation for the AP-600 vessel appears to be robust enough that it may be extended to higher PWRs.

Ex-vessel steam explosion loads on PWR and BWR containments have become an issue recently due to the accident management strategy of establishing deep water pools under the vessel prior to vessel failure. This strategy is employed in the Swedish BWRs and in the passive-advanced LWRs in the USA. There are many facets to the determination of the containment failure probability due to interaction of a corium jet with water in deep water pool e.g., jet characteristics, the corium composition, the extent of fragmentation, the strength of the trigger required, the pressure pulse generated in the steam explosion and the fragility of the containment.

## 9. Melt Debris Coolability

Melt coolability is perhaps the most vexing issue impacting severe accident containment performance for the long term. As mentioned earlier, melt coolability is essential to prevent both the base-mat melt through and the continued containment pressurization. Provision of deep (or shallow) water pools under the vessel may not assure long term coolability/quenchability of the melt discharges from the vessel. Interaction of the melt jet may lead to very small particles (in the event of a steam explosion), which will be difficult to cool in the form of a debris bed. On the other hand, incomplete fragmentation will lead to a melt layer on the concrete basemat under a water layer.

Coolability of a melt pool interacting with a concrete basemat by a water overlayer has been under intense investigation in the MACE project[21] sponsored by an international consortium and managed by EPRI. The experimental work is being performed at ANL. Three experiments have been performed successfully in which melt pools of 30 cm x 30 cm x 15 cm depth, 50 cm x 50 cm x 25 cm depth and 120 cm x 120 cm x 20 cm depth were generated on top of concrete base-mats and water added on top. The melt material contains Uranium oxide, Zirconium oxide, Zirconium and some concrete products. The decay heat generation in the melt was simulated through electrical heating. It was found that for these three tests, the effect of the side wall dominated the phenomena, since an insulating crust was formed, which attached itself to the side walls. The crust prevented intimate melt-water contact and the heat transfer rate slowly decreased from approximately 2 to 0.1 $MW/m^2$ which was less than the decay heat input to the melt. It was found that some melt-cooling, in that instance, was achieved through volcano-like melt eruptions into water. The results of the 120 cm x 120 cm test are being analyzed presently.

## 10. NEW REACTOR DESIGNS

The new reactor designs incorporate many improvements for reducing the core damage frequency, or the occurrence of a severe accident. Nevertheless, accident management and mitigation are also focused strongly. The major safety enhancement is either the provision of a core catcher in the containment, or the flooding of the vessel external surface to retain the core melt within the vessel. The former is being pursued for the EPR[22] and the latter for the AP-600 design[3]. The German design of the PWR containment also incorporates other features, e.g. protection against ex-vessel and in-vessel steam explosions.

The EPR core catcher design relies on the spreading of the core melt discharged from the vessel onto a large inert (specially designed) floor area, to a depth of less than 10 cm. The shallow melt pool can then be cooled by an overlayer of water. The same strategy is employed in the Karlsruhe containment design, except that the coolability is achieved by water addition from the bottom of the melt layer, which has been found to be effective in several small and medium scale experiments.

Research is currently under way to determine the spreading characteristics of corium melt that could be discharged from the vessel onto the core catcher. Various parameters which influence the spreading process are being investigated experimentally, and analytically, in the CSC project conducted under the auspices of the European Union.

The AP-600 strategy of in-vessel retention employs flooding of the containment to submerge most of the vessel. It has been established that the maximum heat removal capability (CHF) is greater than the maximum thermal loading imposed. Investigations are, currently, in progress to determine if the core melt of the larger power reactors (1200 to 1400 MWe) can also be retained in vessel by flooding the containment.

Considerable research has been conducted, or is currently underway, in estimating the thermal loadings on the lower head of a PWR or a BWR during the melt pool natural circulation that occurs if the melt is retained in the lower head. Experiments with simulant materials (mostly salted water) have been performed at the ACOPO facility[3] and at UCLA[23] in USA, the COPO facility in Finland, and the BALI facility in France[24]. Experiments with molten salts are being performed in the RASPLAV Project in Russia and in the SIMECO facility in Sweden[24]. More notably, experiments are in progress in the RASPLAV[25] Project with ≅ 200 kg quantities of molten corium ($UO_2$ + $ZrO_2$ + Zr). In all these experiments, the thermal loads on the vessel wall are measured. Considerable analysis efforts have also been successfully pursued. In general, it is found that the upward and downward heat transfer are scaled with the internal Rayleigh number and that the natural convection flows are highly turbulent at prototypic scale. Heat removal at the external surface of the vessel wall has also been measured at the ULPU facility in USA[3] and at the SULTAN facility in France[24].

## 11. CONCLUSIONS

The intensive research work on severe accidents initiated world-wide after the TMI-2 accident has borne fruit in several ways. The work identified new vulnerabilities for the LWR vessel and containments, but also provided answers to several questions and increased knowledge to the extent that a majority of the in-vessel and ex-vessel accident progression issues are resolved and the resolution of the remaining severe accident issues appears to be near. Issues defined by the severe accident management strategies employed in new reactor designs are being investigated and promising progress has already been achieved.

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# EUROPEAN UNION RESEARCH IN SAFETY OF LWRs WITH EMPHASIS ON ACCIDENT MANAGEMENT MEASURES

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**Abstract**

On April 26th 1994 the European Union (EU) adopted via a Council Decision a multiannual programme for community activities in the field of nuclear research and training for the period 1994 to 1998. This programme continued the EU research activities of the 1992-1995 Reactor Safety Programme which was carried out as a Reinforced Concerted Action (RCA), and which covered mainly research activities in the area of severe accident phenomena, both for the existing and next-generation light water reactors.

The 1994-1998 Framework programme includes activities regarding Research and Technological Development (R&TD), such as demonstration projects, international cooperation, dissemination and optimization of results, as well as training, in a wide range of scientific fields, including nuclear fission safety and controlled thermonuclear fusion.

The 1994-1998 specific programme for nuclear fission safety has five main activity areas: (i) Exploring Innovative Approaches, (ii) Reactor Safety, (iii) Radioactive Waste Management, Disposal, and Decommissioning, (iv) Radiological Impact on Man and Environment, and (v) Mastering Events of the past. The specific topics included in this work programme were chosen in consultation with the EU Joint Research Centres (JRC), and with experts in the different fields taking into account the needs of the end users of the Community research, i.e. vendors, utilities and licensing and regulators authorities.

This paper briefly discusses the objectives and achievements of the 1992-1995 RCA and also describes the projects being (or to be) implemented as part of the 1994-1998 programme in the areas of "Reactor Safety/Severe Accidents", particularly those related to Accident Management (AM) Measures. In addition to this, some relevant projects related to AM which have been funded via independent PHARE/TACIS assistance programmes will also be mentioned.

## 1 - Introduction

### Background of reactor safety research at the European Commission

Developing concepts and techniques aimed at improving the safety of nuclear power reactors has always been a key objective of the Research and Technological Development (R&TD) programmes of the European Union (EU).

The legal basis for EU research is set out in the EURATOM Treaty (1957) which specifies to the EU to "contribute to the raising of the standard of living in the Member States and the development of relations with other countries by creating the conditions necessary for the speedy establishment and growth of nuclear industries" (Title I, Article 1) and to "provide for the encouragement of progress in the field of nuclear technology" (Title II).

The latter Title assigns the EU among others to "promote and facilitate nuclear research programmes", in particular in the areas of physics and chemistry applied to reactor technology (Annex I), with the aim of "creating the conditions of safety necessary to eliminate all hazards to life and health of the public" connected to nuclear energy (preamble of the Treaty). More specifically, another obligation of the EURATOM Treaty for the EU is to "lay down basic standards for the protection of the health of workers and the general public against the dangers arising from ionizing radiations" (Title II, Chapter 3). These research tasks have been carried out either "indirectly" through shared cost or concerted actions involving the EU member states and coordinated by the European Commission (EC/Brussels), or "directly" through programmes executed in any of the institutes of the Joint Research Centre (JRC) devoted to nuclear reactor safety activities.

There are actually 3 main actors interested in reactor safety research, namely: the nuclear research organisations, the regulatory authorities, and the utilities and designers. The task of the EU within this frame consists in proposing a structure for integrated work programmes, providing partial funding of the joint research projects, coordinating the teams involved and promoting the exploitation of the results achieved.

On this basis the EU, acting in concert with these main actors, embarked in 1991 on a major "research and education programme" devoted to severe accidents analysis for existing as well as future light water reactors and executed in connection with existing national projects of the member states. This resulted in a specific programme which was executed as a Reinforced Concerted Action (RCA) in the period 1992-95.

Before the conclusion of this programme, originally planned for early 1995, the European Union (EU) adopted via the Council Decision 94/268/EURATOM of April 26th 1994, a new multiannual programme (to be executed under the EU 4th Framework Programme) for community activities in the field of nuclear research and training for the period 1994 to 1998. The Reactor Safety related activities of this programme extended the EU research about severe accident phenomena with more emphasis on the measures to prevent possible radioactivity release under these conditions. Other areas like Accident Management Measures, plants ageing phenomena, and exploitation of innovative approaches to improve the safety of new and operating reactors were introduced in this new programme.

To conclude the 1992-95 R&TD programme a symposium, called FISA-95, was organised on 20-22 November 1995 in Luxembourg [1] with presentations by the main actors of the programme. A mid-term review of the current programme (1994-1998) for all projects related to "Conceptual Design Features" and "Reactor safety - Severe Accidents", also under the form of a symposium, will take place in Luxemburg from November 17th to 19th 1997.

## 2 - EU research in the area of severe accident

One of the guiding principles for the research policy at the Community level is subsidiarity. Actions should be undertaken by the European Commission (EC) only under the following conditions: they should bring more value added to the project than if they were conducted separately at the single national level; they are of interest to the EU as a whole; they cannot be treated by a member state alone. The research activities in the area of severe accidents are generally just too complex and too expensive to be supported by a single country. Therefore they satisfy not only the above mentioned EURATOM obligations of EU commitment for nuclear reactor safety but also the subsidiarity principle of the Maastricht Treaty of 1992.

TABLE 1. PRELIMINARY CLASSIFICATION OF SEVERE ACCIDENT PHENOMENA IN LWRs

| | | |
|---|---|---|
| 1. | - | a severe accident is initiated by some event like a reactivity accident or the loss of coolant function, especially LOCA circulation failure or loss of heat sink, assuming that no safety system is working, and this initial phase is ending with the start of core uncovery |
| 2. | - | fuel heats up due to inadequate cooling and large core uncovery begins with subsequent oxidation of Zircalloy materials and hydrogen generation, till the start of core melting |
| 3. | - | extensive core melt occurs, leading to relocation of fuel, cladding and control rods, and debris beds are formed in the lower head (late phase core degradation phenomena) |
| 4. | - | in-vessel corium/water interactions like upwards ejection of water masses hitting the reactor vessel head and steam explosions (e.g. maximum release of 200-400 MJ of mechanical energy), with subsequent challenges to the integrity of the vessel (e.g. at the location of vessel penetrations) |
| 5. | - | in-vessel hydrogen burns or explosions with pressure peaks up to 5 bar and high temperature creep phenomena which further challenge the integrity of the entire primary circuit |
| 6. | - | relocation of large masses of molten core to the lower plenum with risks of additional steam and hydrogen explosions challenging the integrity of the primary circuit, with subsequent reactivity releases in the primary circuit |
| 7. | - | reactor pressure vessel failure causing core debris to relocate into the reactor cavity sump in the case of lower head failure or creation of missiles challenging the integrity of the containment and its equipment |
| 8. | - | sudden increased containment pressure and temperature due to ex-vessel hydrogen and steam explosions, and subsequent direct containment heating in the case of high pressure melt ejection (DCH / e.g. maximum containment pressure of 1 MPa and temperatures up to 700 °C for 45 tons of melt) |
| 9. | - | high containment pressure due to fires and steam effects, with subsequent reactivity releases with radiation challenges to the performances of the safety equipments in the containment |
| 10. | - | corium/concrete physical and chemical interactions in the sump, like formation of solid barrier crusts, early ablation of the basemat and releases of various gases which increase the containment pressure (e.g. hydrogen and carbon monoxide with increased risks of explosions) |
| 11. | - | loss of containment integrity due to high gas pressures and thermal stresses (accident scenarios with either short term dynamic or long term quasi-static effects) or local instantaneous missile impacts |
| 12. | - | basemat failure or melt-through and core debris release to the underlying ground |
| 13. | - | off-site radioactivity releases via containment leakage paths, containment by-passing, leaking or venting of gases. |

Considering the substantial amounts of funding and the large experimental facilities needed for severe accidents R&TD, coordinated research programmes are certainly most appropriate. International research programmes can offer indeed the most cost effective opportunity to investigate multidisciplinary and expensive problems like degraded core cooling or hydrogen risk mitigation. Moreover it is an optimum way to confront various experiences and to reach an international consensus about the understanding of severe accident phenomena and about the development of accident management strategies.

The subject of severe accident phenomena in LWRs is usually just too complex to understand physically and to predict numerically although there has been extensive research worldwide for over a decade. As a result of many discussions with the main actors, the preliminary classification indicated in Table 1 was agreed for the most relevant phenomenological events to be investigated in the work programme. Although it is believed that the qualitative aspects and the major possible problems in this area are well identified, there is still a need to quantify the risk and hence to reduce the many uncertainties still left in most severe accident scenarios.

It is worth mentioning that, in the EU, the main actors of the nuclear power plant community, namely: the research organisations, the regulatory authorities and the electrical utilities, have been able recently to sign different cooperation agreements, and that in all these agreements the need for additional research about severe accidents has been clearly pointed out.

Regarding extension of R&TD activities outside the EU , it should be added that some severe accident research projects with participation of East-European countries are being coordinated by the EC in the framework of the TACIS (Technical Assistance to the Commonwealth of Independent States) and PHARE (Poland-Hungary Assistance in the Restructuring of Economies) programmes.

## 3 - (1992-95) EU Research Programme on Reactor Safety

The 1992-95 EU Research Programme on reactor safety was focused on the understanding of scenarios and phenomena of beyond design-basis-accidents (beyond DBA) and on the development of accident management measures for Light Water Reactors (LWRs) of both the present and the future generation.

Special emphasis was put on the applications of this research programme to the development of measures for the mitigation of the consequences of severe accidents in LWRs, mostly of the evolutionary type in Western Europe, in the hypothetical case that the accident prevention measures fail. The attention was focused on the conduction of separate effects tests as well as large integral experiments of common interest to provide a database against which numerical codes could be validated in view of the extrapolation towards the real reactor situation. A few plant specific assessments have also been performed to compare the impact of several accident scenarios on different LWRs designs with emphasis on the source term issue [3].

This research programme was conducted through a 2.5 years joint effort from end 1992 till early 1995 and involved 20 contracting organisations coming from 9 EU member states, in cooperation with some Central- and East-European research organisations. A wide spectrum of severe accident scenarios and phenomena have been investigated both from an

TABLE 2. PROJECTS OF THE 1992-1995 EU RESEARCH PROGRAMME ON REACTOR SAFETY

**Area 1—accident progression analysis, i.e. the study of in- and out-of-vessel phenomena due to severe accident scenarios**

- project No. 1: modelling of core degradation progression based on large scale experiments like CORA and PHEBUS
  **1. CORE DEGRADATION (CORE)**

- project No. 2: problems related to the hydrogen behaviour including combustion phenomena and countermeasures like inerting and recombining techniques
  **2. HYDROGEN BEHAVIOUR ($H_2$)**

- project No. 3: molten fuel / coolant interaction with emphasis on the molten corium/water premixing and the potential for steam explosions
  **3. MOLTEN FUEL / COOLANT INTERACTION (MFCI)**

- project No. 4: reactor pressure vessel response including high temperature creep failure modes and in- as well as ex-vessel coolability techniques for melt retention
  **4. REACTOR PRESSURE VESSEL (RPV)**

- project No. 5: molten corium / concrete interaction with emphasis on ex-vessel corium retention devices
  **5. MOLTEN CORIUM / CONCRETE INTERACTION (MCCI)**

- project No. 6: radioactive source term behaviour including release from fuel and transportation in the reactor coolant circuit
  **6. SOURCE TERM (ST)**

**Area 2—behaviour and qualification of the containment system in order to evaluate the safety margins**

- project No. 7: investigation of containment integrity problems through studies of thermal and dynamic loading effects of short and long term types, the characterization of the containment thermalhydraulics and the identification of possible leakage modes through cracks, e.g. in the concrete structure
  **7. CONTAINMENT INTEGRITY (CONT)**

**Area 3—accident management support**

- project No. 8: signal validation under extreme accidental conditions and development of improved man-machine interfaces in advanced nuclear power plant control rooms with a view on new strategies for accident management support.
  **8. ACCIDENT MANAGEMENT SUPPORT (AMS)**

experimental and an analytical point of view. The work programme addressed 3 main areas, namely accident progression, containment integrity and accident management support, subdivided in 8 projects, each under the responsibility of a project coordinator nominated amongst the participants, as indicated further down in Table 2.

The 1992-1995 EU Research Programme was structured in 8 specific projects described further down in Table 2 . The final report of this programme, covering the outcomes of all 8 projects, will be published soon as a EUR report [2].

## 4 - The 1994-1998 EU Programme on Reactor Safety

The 1994-1998 EU Framework programme includes activities regarding Research and Technological Development (R&TD), such as demonstration projects, international cooperation, dissemination and optimization of results, as well as training, in in a wide rage of fields, including nuclear fission safety (441 Mio ECU) and controlled thermonuclear fusion (840 Mio ECU). The nuclear fission safety activities are broken down under direct actions under the responsibility of the EU Joint Research Centres (264 Mio ECU), and indirect actions coordinated by EC DGXII/F/5 and -6, Brussels.

The indirect actions (shared cost and concerted actions) of the Nuclear Fission programme (1994-1998) consist of five main activity areas:

1. Exploring Innovative Approaches
2. Reactor Safety
3. Radioactive Waste Management, Disposal, and Decommissioning
4. Radiological Impact on Man and Environment
5. Mastering Events of the past

The research projects of the Reactor Safety related areas will be conducted through a 3 years joint effort from beginning 1996 till end 1998, and will be funded between the Community (about ECU 30 million) and contractors from 20 organisations coming from 11 out of the 15 member states of the EU.

The Reactor Safety area is structured into 6 clusters, each containing several projects, and addresses in particular the field of Severe Accidents, that is:

i) In-Vessel Core Degradation and Coolability (8 projects) Cluster "INV"
Corium formation and behaviour
Molten corium coolant interactions
In-vessel corium coolability
RPV behaviour

ii) Ex-vessel Corium behaviour and Coolability (4 projects) Cluster "EXV"
Thermochemical modelling and data
Corium release and spreading
Corium retention and cooling

iii) Source term (10 projects) Cluster "ST"
In-vessel fission product behaviour
Ex-vessel fission product behaviour
Benchmark calculations

iv) Containment Performance and Energetic Containment Threats — Cluster "CONT" (10 projects)
Hydrogen distribution and combustion
Containment thermalhydraulics and cooling
Material data and structural response
Containment leakage

v) Accident management measures ( 5 projects) — Cluster "AMM"

vi) Ageing (7 projects) — Cluster "AGE"

Workshops, widely open to the international research Community along the lines of FISA-95, will be organised at mid-terms and at the end of theProgramme in order to present the results and conclussions.

## 5. PHARE and TACIS Programmes on Severe Accident Research Activities

Under the EU Framework Programmes, the Central and Eastern European Countries (CEEC) and Newly Independent States (NIS) research organizations have two main options for cooperating in EU sponsored nuclear safety research projects, i.e. either becoming subcontractors for projects of these programmes or joining a PHARE/TACIS assistance programme oriented towards the improvement of the safety level of NPPs of Russian design.

Recently, some research related projects of CEEC/NIS have been recently allowed to enter the PHARE and TACIS programmes in order to cover - further to the industrial assistance-also nuclear safety research activities to be mainly performed in CEEC/NIS countries. For the PHARE/TACIS 1995 and 1996 assistance programmes some of these research projects in the nuclear safety field have been discussed with the DGXII . However the funding as well as the coordination of these PHARE/TACIS activities in the field of nuclear fission safety falls under the responsibility of DGI-A.

## 6. Research related to Accident Management Measures

Through the 1992-95 Research Programme the European Community has undoubtfully contributed to the scientific basis for improving the evaluation of most of the challenges of severe accidents. As a result improvements are currently under discussion to optimize the balance between prevention and mitigation measures for the safety design against unlikely extreme events, should the prevention systems fail.

In addition to the traditional prevention techniques for the short term failure modes in the containment, measures for the mitigation of the consequences of long term failure modes have been investigated for existing as well as for future reactors. As a result it can be stated that the present knowledge about severe accident phenomena allows to extend the traditional defense-in-depth concept by introducing, as an additional line of defense, accident management procedures, and in particular mitigation measures for the consequences of severe accidents.

Successful accident management includes many tasks related to the use of information technologies, such as reliable identification of the actual plant and components state, information for assessing the accident progression and the plant response to operator action, and information for planning the mitigation strategy with potential uncertainties due to failed or misleading instrumentation. As a result of the fast progress in numerical techniques and the availability of very powerful computer systems with acceptable economic conditions, the area of accident management support is considered one of the most promising R&TD fields in which very effective solutions can be reached with reasonable efforts and in realistic time frame.

Following is a summary of the main projects which have been funded by the EU to assist the operators and the personnel of the Technical Support Centres in the application of Accident Management strategies in the event of a severe accident in a Light Water Reactor. One of them was performed as part of the 1992-1995 Research Programme, and therefore is already completed, and the others are being (or will be) performed as part of the 1994-1998 Framework Programme (see "AMM" Cluster of section 4) or as part of PHARE/TACIS assistance programmes.

### 6.1 *Project "Accident Management Support" (AMS)*

This project was the result of the combination of two originally envisaged projects of the 1992-1995 Programme, namely "Instrumentation and Signal Validation" and "Operator Assistance". Since the overall objective of these two projects was very similar, i.e. to provide comprehensive and reliable information for identification, prevention or planning of mitigation of an accident with up-to-date supporting technology for plant operators or emergency teams, only one common group was established to work in a multi-partner (10 parties) action coordinated by GRS/ISTec (Germany).

In order to be able to control and mitigate an accident, the operators should have at any time a realistic and correct picture of the accident and its progress. Hence instrumentation and signal validation are main issues under severe accident conditions, especially if instruments are working beyond their specification range. The objectives of the "Accident Management Support" (AMS) project were (i) to define, investigate and develop means and methods providing reliable information and diagnostics as well as support tools for accident management, and (ii) investigate the different signal validation methodologies with emphasis on the existing instrumentation rather than new instrumentation needs. The basic scope of this project was:

(1) To carry out investigations about real-time monitoring and decision-making techniques, using neural networks, advanced modelling and noise analysis techniques. Expert system based strategies were further developed for design and maintenance of emergency guidelines in connection with an "automatic operator model". Adaptative algorithms and extrapolations from recorded plant data have been developed with the aim of predicting critical milestones and optimising command/control under emergency conditions such as to enable the operator to take the optimal recovery actions in response to an accident.

(2) To conduct investigations in signal validation methodology and sensor modelling/signal processing. There are two basic approaches to match the requirements of reliabilility and validity of a plant signal: model-based methods realizing functional redundancy and noise diagnostic methods, using the signatures of inherent signal noise as "finger prints" of specific sensors and plant conditions. Sensor modelling activities were performed for a fission chamber model in a an extended range of faulty operating conditions. Noise diagnostic methods were developed using advanced signal identification techniques.

The work started with writing state-of-the-art reports (SOARs) in the two main areas: Operator support systems [4] and Instrumentation/Signal Validation [5]. In parallel to the compilation of the SOARs, specific research activities were performed in areas such as:

- Signal validation using advanced digitized techniques, i.e. model-based and noise diagnostic methods
- Sensor modelling and signal processing
- Man-machine interface: Operator role in advanced control room environment
- New tools, methods and computerised systems for accident management:
    . Development and Implementation of Accident Management procedures (DIAM)
    . Transient Analyzer for accident state assessment (TRANSAL)
    . Design and Maintenance of Emergency Guidelines (SAMARIA)
    . Recommendations for VDU display design and their use in accidents (INTERACT)
    . A decision Support system for containment release management (CRM)
    . Situation Related Operator Guidance (SIROG)
    . Knowledge based Operator Advisor system for use in severe accidents (OPA)

The main conclusion of the project was that, due to the availability of powerful information processing systems, substantial progress was made in the feasibility of advanced methods and systems such as signal validation, process and system state assessment, man-machine interface optimisation, and operator advising and assisting systems for diagnosis, execution of accident management procedures, and safety-function or situation-related decision-making.

### *6. 2 Project "Development of methodology for the evaluation of severe accident management strategies" (AMEM)*

This project of the 1994-1998 Framework Programme started in January 1996, and is being performed by a multi-partner team (4 organisations) under the coordination of NNC Ltd (Great Britain).

Accident management (AM) strategies with the potential to terminate or mitigate degraded core accidents are currently being developed and implemented at nuclear power plants worldwide. Decisions on their implementation are however, not straightforward as the actions may cause potential adverse effects and also involve physical phenomena that are not well understood. Current research emphasis has centred mainly on achieving a better understanding of the phenomena. Apart from the phenomenological issues, each accident management strategy also requires consideration of the following key interrelated issues :

- operator performance
- equipment availability and performance
- instrumentation availability and performance

The qualitative assessment and quantification of these thus entails a high degree of uncertainty. The framework for any assessment must therefore be able to address these issues in an integrated fashion and to allow the uncertainties to be addressed. A guiding principle must be that AM measures must have a certain robustness against uncertainties.

The objectives of the AMEM project are twofold. The first objective is to further develop integrated AM models for the assessment of the feasibility and effectiveness of potential severe accident management measures. The second objective is to apply these AM models in relevant case studies while contrasting the unique features and understanding the limitations of these models.

This project comprises the following tasks :

1. a detailed review of existing models and their recent applications in the assessment of the potential impact of severe accident management measures;
2. examination of the different criteria currently used for the assessment of the effectiveness of potential severe accident management measures and recommendations for the most appropriate one;
3. development of integrated AM models to consider key issues such as severe accident phenomena, operator response, systems availability and instrumentation availability;
4. on the basis of case studies agreed (and derived from different AM actions), demonstrate the proposed methodology, based on reference reactor designs (PWR, BWR, VVER).
5. perform an evaluation of the benefits and drawbacks of the different methodologies, as a result of the case studies selected

The final report for this project is expected by the end of 1998. Preliminary results will be probably presented at the FISA-97 Symposium.

### 6. 3 *Project "Algorithm Support for Accident Identification and Critical Safety Functions Signal Validation " (ASIA)*

This other project of the 1994-1998 Framework Programme is expected to start beginning 1997, and will be performed by a multi-partner team (4 organisations) under the coordination of NNC Ltd (Great Britain).

The AMS project of the 1992-1995 Research Programme mentioned above (see section 5.1) concluded that Critical Safety Functions (CSF) instruments may only survive for a limited period in a severe accident. It also provided recommendations on signal validation, and identified algorithms -based systems as holding great promise for validation of instruments and for identification of postulated accidents. The "ASIA" project will extend and build on the "AMS" work already done and will consist of three work packages:

1. Further development of operator aids based on physical models. Two particular aspects will be addressed: (a) validation of CSF measurements, and (b) understanding of accident progression, i.e. accident diagnosis.

The aim of the validation is to transform a set of raw measurements in a global consistent set of physical values; for faulty or non-measured values the system will attempt to provide substituted values. The solution of the system of relationships linking measurements will be carried out with search algorithms and constraint methods. The set of validated values will be then transmitted to a thermal-hydraulics module which will provide analytical redundancy. The results of the analysis could be then used to identify the behaviour of measurable CSF signals.

2. Study instrument survival in severe accidents. This will start by identification of a list of CSF measurements and complementary instrumentation needed for accident management for representative PWR designs.

Then, based on the environment expected after a severe accident as defined in the state-of-art reports produced in the "AMS" project, the survival of all neutron flux sensors will be assessed.

The output of this work package will be a document listing the measurements, instruments concerned, the survival times expected for accidents referenced, any potential for improvement in survival, the symptoms of failure expected and the potential for validation of the measurement.

3. Implementation strategy. The objective will be to study the needs of the operating staff, Technical Support Centre and Emergency Management for appropriate algorithms, and the formulation of recommendations for their implementation.

   The work will integrate the investigations on instrument survival of work package 2 with the algorithms investigated in work package 1. The output will be a strategic recommendation of the algorithms for each class of operational use, related to the different stages of progression of accidents from precursor stage through beyond design basis condition, and to severe accident conditions with fuel melting.

The final report for this project is also expected by the end of 1998. Preliminary results will be probably presented at the FISA-97 Symposium.

### *6. 4 "VVER 440-213 Beyond Design Basis Accidents Analysis and Accident Management" (PHARE Project 4.2.7.a)*

The project will cover various aspects of severe accident analysis for VVER-440/213 plants, identification and validation of preventive and mitigative strategies, and provision of the basis for future implementation of severe accident management guidelines. In addition to that, the appropriate analytical tools (codes and models) for accident analysis and for specific VVER (namely the MAAP/VVER code) will be supplied to all beneficiaries of the project together with the necessary training.

The project started beginning 1996 and has a duration of two years. The reference plant is Bohunice V2 440/213 (Slovakia), but tasks to evaluate the applicability of the results to both Paks (Hungary) and Dukovany (Czech Republic) are included in the work scope.

### *6. 5 "VVER-1000 Severe Accidents Management" (TACIS Project 93-3.8)*

The goal of this project is the transfer of Western technology and experience on Severe Accident (SA) analyses and phenomenology and on Accident Management (AM) strategies and procedures to the Russian counterpart, and the assistance and collaboration in the application and improvement of available SA codes to VVER plants as well as in the preparation of AM procedures.

This project includes the performance of a number of analyses for beyond design basis accidents (BDBAs) and SA analyses for the reference plant (Balakovo NPP, Unit 4) of the VVER-1000 scenarios which have been identified for the successful completion of TACIS Project 3.1 ("Probabilistic Safety Analysis") and which are also needed to support the development of AM measures for this type of reactor.

## 7 - Conclusions and future research needs

In conclusion the 1992-1995 EU Research Programme about severe accidents has proven to be very valuable for the improved understanding of some of the key safety problems of LWRs. This EU programme contributed to develop tools for further reducing the frequency and the consequences of core-meltdown accidents, while ensuring negligible releases from the vessel and from the containment under hypothetical beyond design-basis-accidents.

Activities performed as part of such programme have led to new cooperations in the research community, in particular with the industry. The background knowledge and the working methods of all partners were put together for the benefit of all. The main actors of this EU programme, that is: the research organisations, the regulatory authorities, and the utilities and designers, have expressed their satisfaction about the outcome of this Programme.

There seems to be a consensus to reduce the EU effort in some research areas like early core degradation which has been extensively investigated over the last decades, as well as high-pressure melt ejection and direct containment heating which are supposed to be "practically eliminated" for future reactors. Reversely there is a growing number of questions left open for future EU research about severe accidents, in particular in the areas of late phase molten corium coolability, hydrogen risk management, source term effects and containment bypass sequences as well as advanced accident management measures. This was taken into account in the objectives of the Reactor Safety area of the present 1994-1998 Research Programme of the EU as it has been reflected in the specific projects discussed in this programme.

Finally there is a common consensus among all the parties involved to extend the traditional defense-in-depth concept by introducing, as an additional line of defense, accident management (AM) procedures, and in particular mitigation measures for the consequences of severe accidents. Therefore the area of AM support is considered one of the most promising R&TD fields in which very effective solutions can be reached with reasonable efforts and in realistic time frame.

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DESIGN APPROACHES FOR PREVENTION AND MITIGATION OF SEVERE ACCIDENTS — LWRs

(Session 2-A)

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# IMPACT OF SEVERE ACCIDENTS ON THE EUROPEAN PRESSURIZED WATER REACTOR (ERP) DESIGN AND LAYOUT

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**Abstract**

The purpose of this presentation is to describe the impact of severe accidents on the EPR design and layout. After a summary of the safety requirements specified in accordance with the recommendations expressed by the French and German safety authorities, the main EPR features corresponding to the prevention and the mitigation of severe accidents will be described. Considerations with regard to R&D and cost impacts are also provided.

## 1. INTRODUCTION

To fully benefit from the outstanding and ever-growing French-German experience and to maintain a continuous development process avoiding discontinuity or break, the EPR is of an evolutionary design. This basic choice complies with the recommendations specified by the French and German safety authorities in their "proposal for a common safety approach for future PWR" [1]. At the same time, the EPR is an innovative product to combine competitiveness, improved operability and enhanced safety.

The enhancement of safety is achieved by the elaboration of a safety approach which follows the recommendations of the French and German safety authorities. These recommendations were already presented in the past, see for example [2].

The EPR approach is summarized hereafter, introducing the presentation of the EPR design features dedicated to the severe accident prevention, then to their mitigation. It must be underlined that the processes of elaboration of common positions by the safety authorities, and of development of the EPR, are presently in progress. Some modifications could be introduced in the future in the technical features.

## 2. OVERALL SAFETY APPROACH

### 2.1. General

A twofold strategy is pursued compared to existing plants: first, to improve the preventive measures against accidents, second, even if the probability of severe accident scenarios - up to core melt - has been further reduced, to implement additional features, mainly concerning the containment, to mitigate the consequences of such accidents.

This strategy is implemented by designing the plant with a strong "Deterministic Design Basis" and, beyond this basis, to consider "Risk Reduction" measures. It is a development of the well-known defence-in-depth principle.

Presentations were already made [3] and [4], therefore only a summary will be provided hereafter.

### 2.2. Deterministic design basis

The Deterministic Design Basis is based on systematically and deterministically chosen events. According to their anticipated frequency, those events are categorised in 4 Plant Condition Categories (PCCs). PCC 1 covers normal operation states, PCC 2 to PCC 4 envelop anticipated operational occurrences, infrequent and limiting accidents. The classification includes consideration of relative frequency of the events.

The progressivity-of-safety and defence-in-depth principles imply that the radiological consequences shall be commensurate with the frequency. The EPR project has proposed radiological limits accordingly, as well as design criteria. In the frame of the safety assessment, it is shown that those radiological limits and design criteria are met, considering dedicated systems and conservative assumptions including deterministic failure assumptions in the dedicated systems.

These dedicated systems, safety classified, are subject to extensive efforts in order to reach a high reliability and to keep the common mode failure potential at a low level, leading to an efficient prevention of severe accident.

### 2.3. Risk reduction

Nevertheless, to further reduce the risk of large releases from the plant, the EPR project has specified two Risk Reduction Categories corresponding to core melt prevention (i.e. Risk Reduction Category A or RRC A) and large releases prevention (i.e. Risk Reduction Category B or RRC B).

By reference to the INSAG 3 objective (probability of core melt < $10^{-5}$/reactor x year including all events and all reactor states), the following targets were specified by the French and German safety authorities, to be considered as "orientation values":

- for determining the adequate combination of redundancy and diversity, the designer can use probabilistic targets as orientation values ; in that case, orientation values of $10^{-6}$ per year for the probabilities of core melt due to internal events respectively for power states and for shutdown states could be used

- for those internal and external hazards the probabilities of which cannot be realistically determined, provisions have to be implemented by the designer to obtain a consistent design

A more specific and practical probabilistic decoupling value was defined by the EPR project, to be used as design target : the CMF per "family of events" (internal events for power or shutdown states) shall be less than $10^{-7}$ per reactor-year.

RRC A includes consideration of events with multiple failures and coincident occurrences up to the total loss of safety systems. The identification of such events is performed on a probabilistic basis, using realistic or of best estimate rules. Additional features must be provided in order to meet the CMF decoupling target specified before ($10^{-7}$ per reactor and year). The final safety assessment of RRC A is performed by a level 1 PSA.

RRC B, or severe accidents, constitute a new category of events for which provisions shall be foreseen in the EPR design, and the objective is to strengthen the design measures in such a way that extensive offsite countermeasures are not necessary, should such an event occur.

The corresponding design goals are specified by the French and German safety authorities in the following way:

- situations that would lead to large early releases such as containment bypass, strong reactivity accidents, core melt with reactor coolant system at high pressure or global hydrogen detonation, are "practically eliminated". When they cannot be considered as physically impossible, design provisions are introduced to design them out.

- all other situations, including the low pressure core melt accidents are "dealt with". The corresponding radiological consequences will necessitate only limited protective measures in area and in time. It means that there will be no necessity of permanent relocation, no need for evacuation outside the immediate vicinity of the plant, limited sheltering and no long-term restrictions in food consumption.

The safety assessment of RRC B is intended to be largely deterministic. The assumptions and design criteria will be as realistic as possible. With regard to the evaluation of the possible radiological consequences, regulatory acceptable dose limits do not exist, but the action levels recommended in the ICRP 63 publication (concerning relocation and evacuation) and the food marketing limits defined by the European Union can be used as references (Section 4.8).

## 3. MAIN FEATURES CONTRIBUTING TO THE IMPROVEMENT OF SEVERE ACCIDENT PREVENTION

Although the prevention of degraded core situations was already at a satisfactory level on the existing designs, additional features were introduced in the EPR to comply with the request of the French and German safety authorities to further reduce it. They are summarised hereafter.

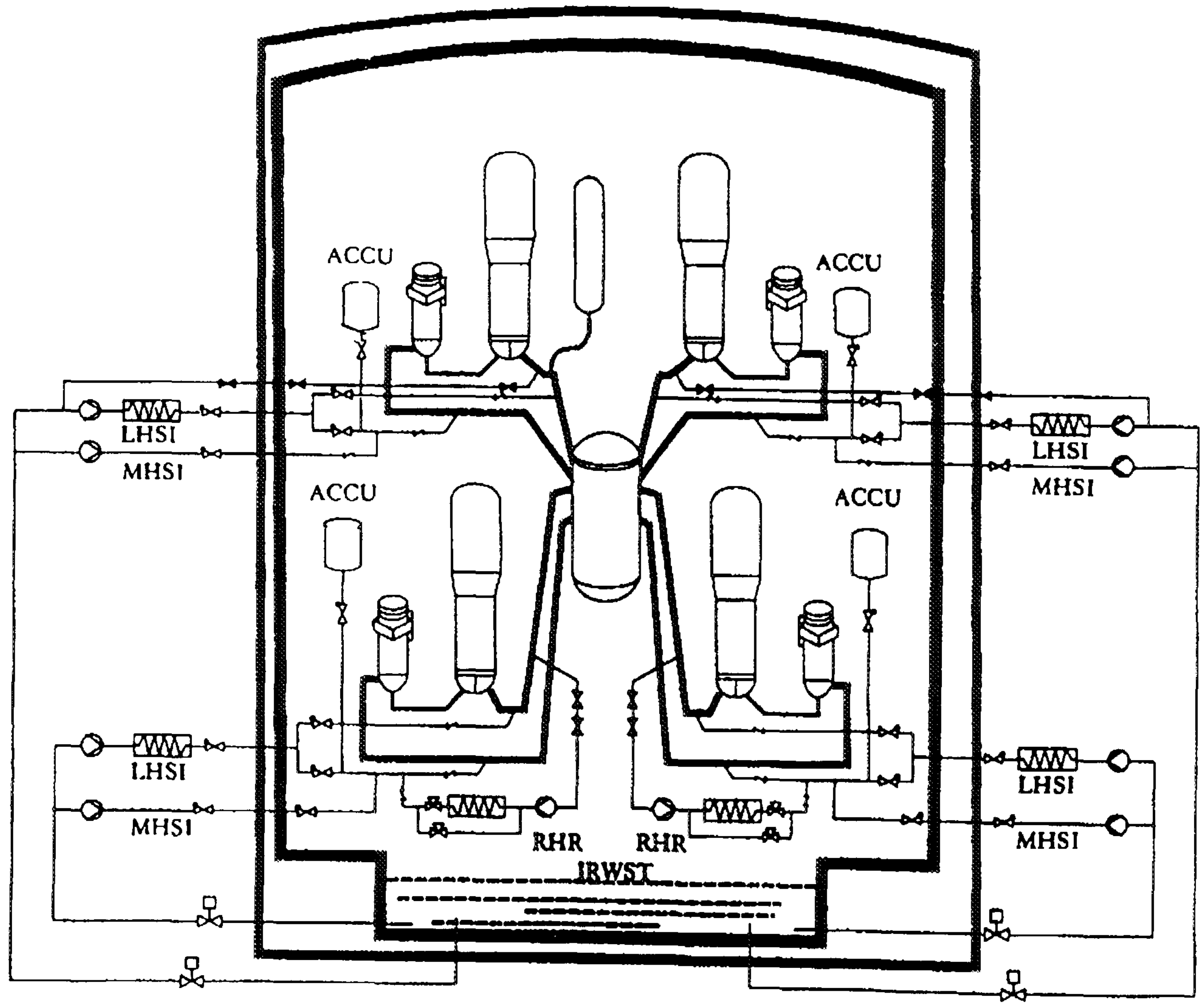
### 3.1. Main primary and secondary systems

Those systems are essentially the same as the ones existing on operating plants. The volumes of the main components (reactor pressure vessel, pressuriser, steam generators) are, however, larger in order to increase the "inertia" in case of disturbances. This inertia induces an increase of the operator grace period and a lower risk to open the primary and secondary safety valves.

To cope with the total loss of feedwater (one of the RRC-A situations), the pressuriser pilot operated valves are sized to permit bleed of the reactor coolant system and feed by the safety injection system, in order to avoid core melt.

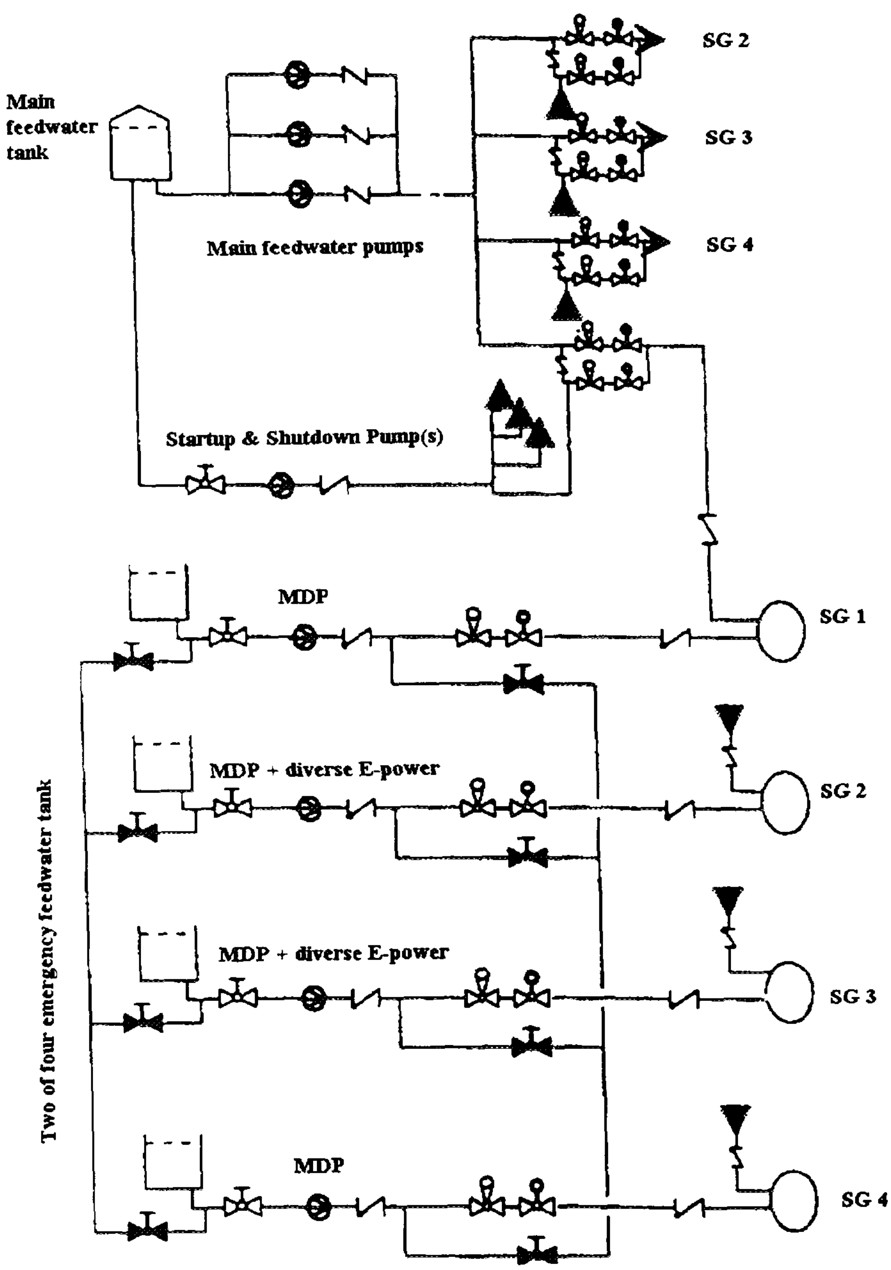
### 3.2. Main safety system organization

The Fig. 1 shows the primary side safety systems. They are organised in four trains that are fully separated, i.e. there is no header open between trains of a given system. The water




- 4 trains injection system
- In-containment refuelling water storage tank
- No need for containment spray for design basis accident
- Residual heat removal system inside the containment to minimaze the risk of containment bypass.

*FIG. 1 Primary side system design.*




- Four separate and independent trains
- The pumps are driven by electrical motors supplied by emergency diesel generators .
- Two out of four pumps are supplied by two diverse small diesel generators to meet probabilistic objectives
- To provide sufficient feedwater supply in case of :
  - Loss of normal feedwater
  - Small break LOCA and steam generator tube rupture
  - Feedwater line break

*FIG. 2. Emergency feedwater system.*

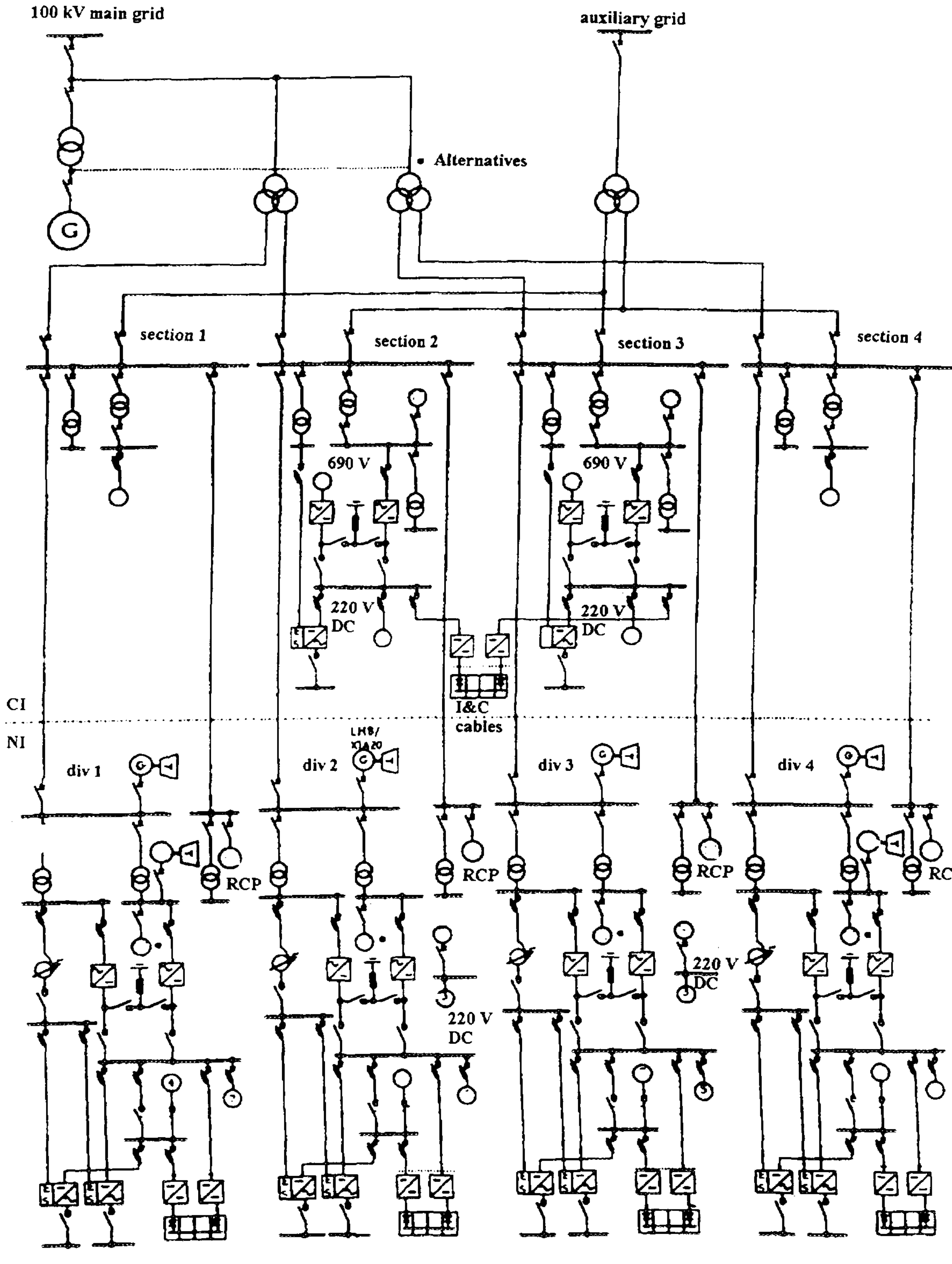



*FIG. 3. Single line diagram.*

TABLE. I. DIVERSIFICATION OF SAFETY-GRADE SYSTEMS.

| Safety-grade System function | Diverse System function |
|---|---|
| MH safety injection | Fast secondary-side pressure reduction<br>+accumulator injection<br>+LH safety injection |
| LH safety injection | Evaporation cooling with MH safety injection system<br>+ containmnent heat removal |
| Residual heat removal | Secondary side heat removal or LH safety injection |
| Fuel pool cooling | Evaporation cooling by fuel pool heatup + makeup |
| Secondary-side heat removal | Primary side bleed and feed |
| Reactor trip | Primary pressure limitation and boration |
| Emergency power Supply | Diverse small diesel generators |
| Instrumentation and control | Diverse and independent actuations of diverse system functions |

storage tank (RWST) for the safety injection is located inside the reactor building (IRWST) because it offers the following possibilities:

- avoidance of the injection mode switching from direct to recirculation, as it is the case when the RWST is outside,

- steam condensation in case of opening of the pressuriser valves, in order to avoid a direct release in the containment atmosphere,

- in case of core melt accident with reactor vessel meltthrough, availability of a large amount of water to initiate the cooling of the corium without any active action (Section 4.4).

The steam generator feedwater system includes (Fig. 2), in addition to the main system, a start-up and shutdown system and an emergency feedwater system. The latter comprises four trains normally completely separated, each with its own feedwater tank. "Passive" headers can be manually opened when necessary in the long term.

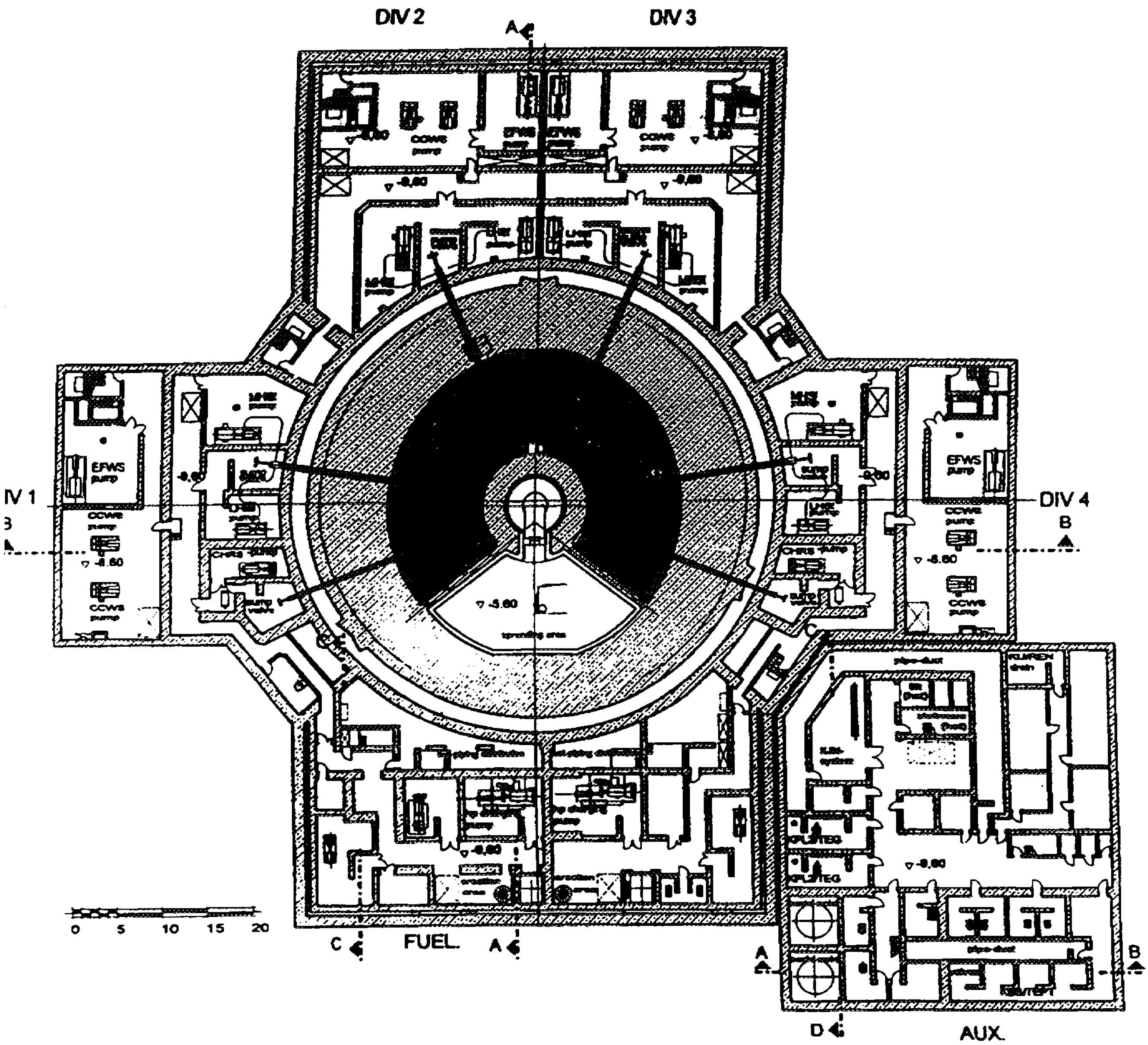



*FIG. 4. Building arrangement. Plant view-9.69 m.*

The onsite electrical power supply (Fig. 3) is ensured by one diesel generator per division (four in total). In order to practically eliminate the risk of core melt in case of total station blackout, two additional small diesel generators are also provided. They supply two emergency feedwater trains and the necessary I&C.

By this system organisation, the principle of simplification is fulfilled as well as the principle of diversification. As a matter of fact, any safety system function can be ensured by another system or a group of systems, as summarised in Table I.

### 3.3. Layout

The general building arrangement in four divisions (Fig. 4) also contributes to the prevention of core melt by providing a complete physical separation of the trains, contributing consequently to the elimination of common causes of failure between redundant trains due to internal hazards.

### 3.4. Man-machine interface and I&C systems

Due consideration is given to the human factor at the EPR design stage, taking into account aspects of operation, testing and maintenance. The general aim is to minimise the possibilities for operator errors. This is achieved by applying appropriate ergonomic design principles and by providing sufficiently long grace periods for the operator responses, thanks to the use of automated safety systems in the short term.

The necessary duration depends on the complexity of the situation to be diagnosed and on the actions to be taken. As a deterministic design basis, a grace period for control room actions of 30 min is used and of 1 h for local actions.

Sufficient and appropriate information is made available to the operator for a clear understanding of the plant status, including severe accident conditions, and for the clear assessment of the effects of his interventions. Emphasis is placed on the use of computer techniques for reliable diagnosis systems for operator support.

## 4. SEVERE ACCIDENT MITIGATION FEATURES

### 4.1. Overall strategy

Because of the stringent requirements imposed by the safety authorities summarized in the Section 2.3, the overall strategy with regard to severe accidents shall consist in maintaining the containment function far into the domain of hypothetical accidents with the following goals:

- avoidance of early containment failure or bypass,

- cooling of the corium in the containment,

- preservation of the containment functions (low leakage towards the environment, prevention of a basemat meltthrough, reliable isolation of the containment on demand, ultimate pressure resistance to cope with the majority of energetic events),

- pressure reduction of the containment by means of a containment heat removal system,

- collection of leakages in the reactor building annulus and release into the environment via the stack after filtration.

In addition, no active mitigation features shall be necessary in the short term.

The various features introduced in the EPR are described hereafter.

### 4.2. Primary system depressurisation

The phenomena associated with a core melt at high pressure, direct containment heating, very high forces on the internal containment structures and missiles, could cause an early containment failure. Because of its large potential, such a situation has to be practically eliminated by system design measures. In addition to the preventive features already described (Sections 3.1 and 3.2), the pressuriser pilot operated valves are used as depressurisation means, initiated manually by the operator in case of high primary temperature (~ 650°C) or low water level in the RPV. Consequently, the depressurisation system changes a high pressure melt ejection scenario to a low pressure core melt.

The capacity of discharge was fixed to be 900 t/h in order to limit the primary pressure to values below 20 bar when the RPV fails.

### 4.3. Reactor pressure vessel pit arrangement and design

It is the design concept of the reactor pit to be as dry as possible to avoid phenomena such as steam explosion when corium is released after meltthrough of the reactor pressure vessel. Therefore, water ingress into the reactor pit in case of LOCA is avoided. In addition, the reactor pit is designed in such a way that there is no direct connection between the pit and the upper containment compartments, thus contributing to the limitation of the direct containment heating risk.

### 4.4. Retention and cooling of the corium

In order to cope with the consequences of a postulated core melt accident, the EPR employs dedicated features for the retention and long term stabilisation of the melt inside the containment (Fig. 5).

After RPV failure and subsequent debris accumulation in the reactor pit, the melt is foreseen to spread evenly on a dedicated dry, flat surface outside the pit. Hereby, spreading is intended to occur in one event and to be followed by flooding and cooling of the melt from above using water from the IRWST. The proposed spreading area involves the following four elements:

- a sacrificial concrete layer,

- a sacrificial metal (cast iron) layer,

- a protective layer, covered by the above mentioned sacrificial concrete and metal layers,

- a basemat cooling.

The basemat cooling network provides an isotherm above the liner level in order to ensure that the basemat concrete will be kept below 100°C. The cooling device is placed directly below the protective layer. The cooling is ensured by the containment heat removal system that is activated 12 h after onset of the accident.

### 4.5. Hydrogen control means

For further details, see [4].

The major concerns due to the presence of hydrogen generated in the course of an accident are that high energetic chemical reactions may damage the containment or that important safety-related equipment may be impacted due to either pressure loads or high temperatures.

The most important potential source of hydrogen during a severe accident is the steam-zirconium reaction during core degradation. Oxidation of the total amount of zirconium present in the core would yield about 1680 kg of hydrogen. Calculations show that around 800 kg of hydrogen will be produced during the heating-up phase of the core. Since the zirconium/steam reaction is highly exothermic this amount will be released in a very short period of approximately 10 minutes. Consequently, the hydrogen control means must be able to cope with a hydrogen production rate of 1 to 2 kg/s. Afterwards, hydrogen is also generated but at a lower rate.

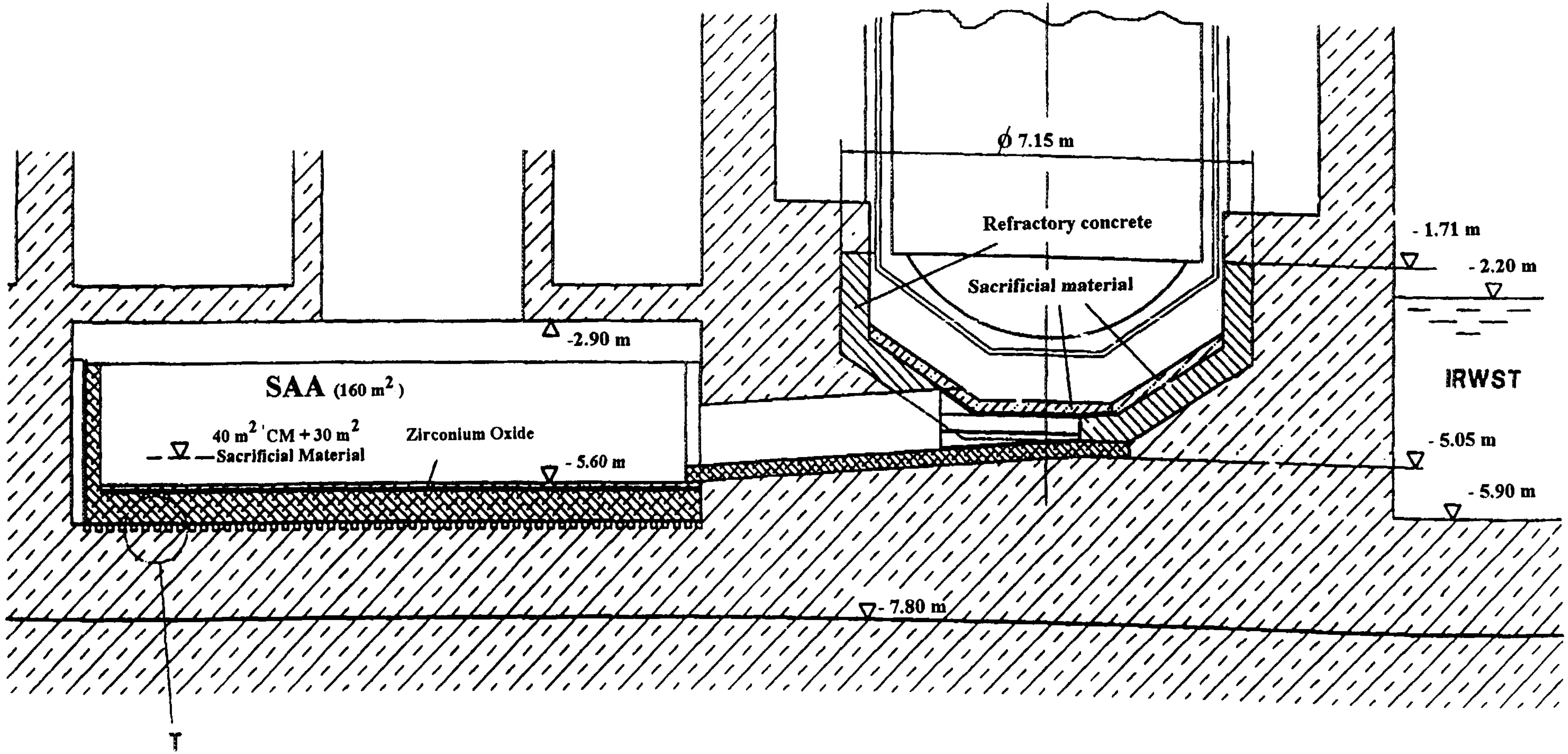



FIG. 5. Spreading concept.

Schematically, the hydrogen control system has to be designed to perform the three following tasks:

- the mean global hydrogen concentration in the containment shall not exceed 10% at any time,

- the mean local hydrogen concentration shall not exceed 10% at any time,

- the hydrogen concentration in the long term shall remain below the ignition limit of 4%.

These goals are achieved in the following manner (subject to verification and optimization studies):

- Catalytic passive recombiners are evenly distributed inside the containment. The number of needed recombiners is determined by the chosen requirement to reduce the average hydrogen concentration below the ignition limit in the long term, disregarding any hydrogen reduction by igniters. When taking a mass of hydrogen corresponding to the maximum calculated amount that could be contained in the containment taking into account the effect of the recombiners, the requirement to stay below 10% is fulfilled.

- Igniters are used to ignite the arriving gas cloud as soon as the local hydrogen concentration has reached the ignition limit. They will be placed where necessary, in particular directly at the relief openings of the IRWST.

**4.6. Containment inner wall**

The concept chosen for EPR is a double-wall confinement (Fig. 6). The outer wall is a reinforced concrete shell, resistant to the external hazards. The inner wall is a prestressed concrete shell resistant to the internal events and hazards.

In the course of a severe accident, the following major contributors of energy and mass transfer into the containment have to be considered:

- loss of primary coolant if the initiator is a LOCA,

- deflagration of the hydrogen produced during core degradation,

- corium heat transferred to the containment atmosphere,

- steam produced in the spreading area.

The characteristics of the containment inner wall were specified as follows:

- design pressure : the design pressure is fixed at 6,5 bar, this value being also compatible with the required specification that the grace period for any active actions shall be 12 hours,

- containment leaktightness : the rate of leakage from the inner wall must not be higher than 1% per day of the volume of gas of the containment at design pressure in accident conditions.

The chosen concept enables satisfactory leatktightness to be preserved for beyond those design conditions, in order to have margins and to guard against phenomenological uncertainties. As a matter of fact, such a prestressed concrete containment is estimated to remain leaktight up to

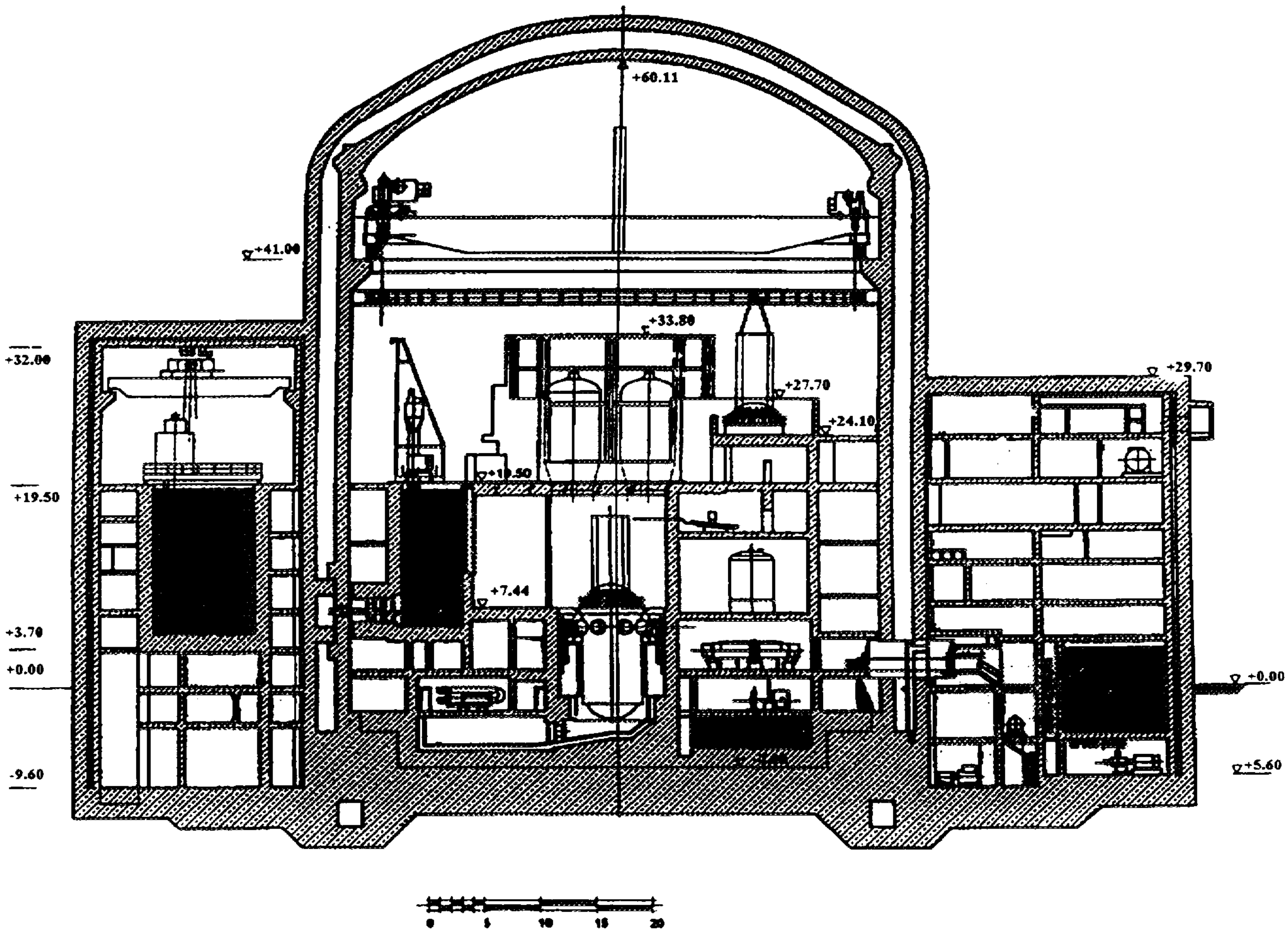



*FIG. 6. Building arrangement. Section A-A.*

a pressure of about 8,5 bar. Up to about 10 bar, the inner wall leakrate remains low (below approximately 3%). The mechanical strength limit (prestressed cable yielding) is estimated to be about 14 bar absolute.

### 4.7. Containment heat removal

As a first requirement concerning the containment heat removal, the safety authorities specified that the function must be ensured without venting device.

Considering the containment type selected for the EPR (double wall concrete containment), a containment spray with outside circulation was selected according to the following criteria:

- potential of pressure and temperature reduction in a reasonably short time duration (to reduce leakage and thus source term), together with a possibility of sufficient late actuation compatible with the hydrogen reduction concept,

- potential to return to a containment pressure near the atmospheric pressure,

- low sensitivity for the conditions which result from severe accidents inside the containment,

- potential to achieve low concentration of fission products in the containment atmosphere (from sump resuspension) in the long term,

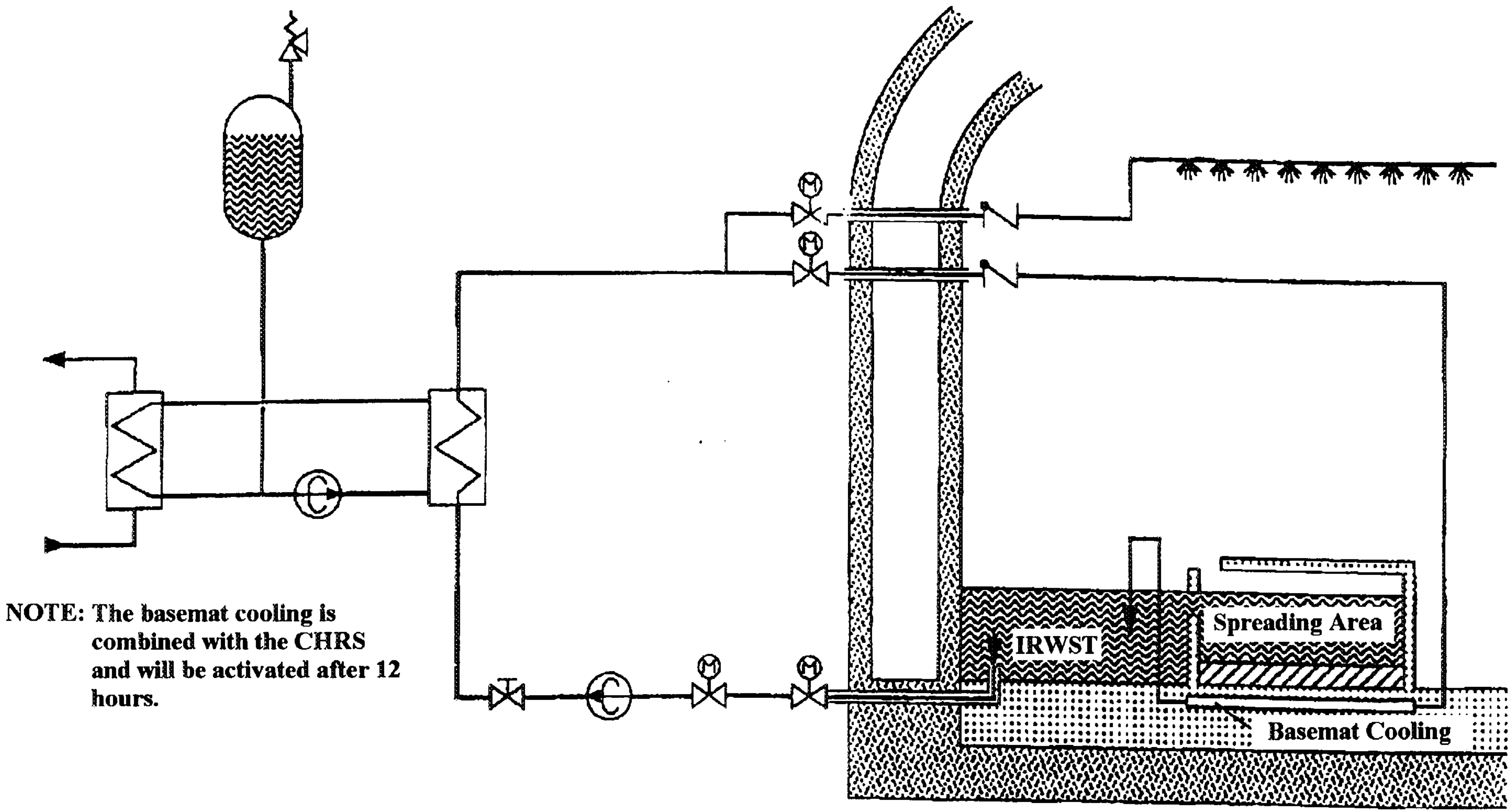



FIG. 7. Containment heat removal system.

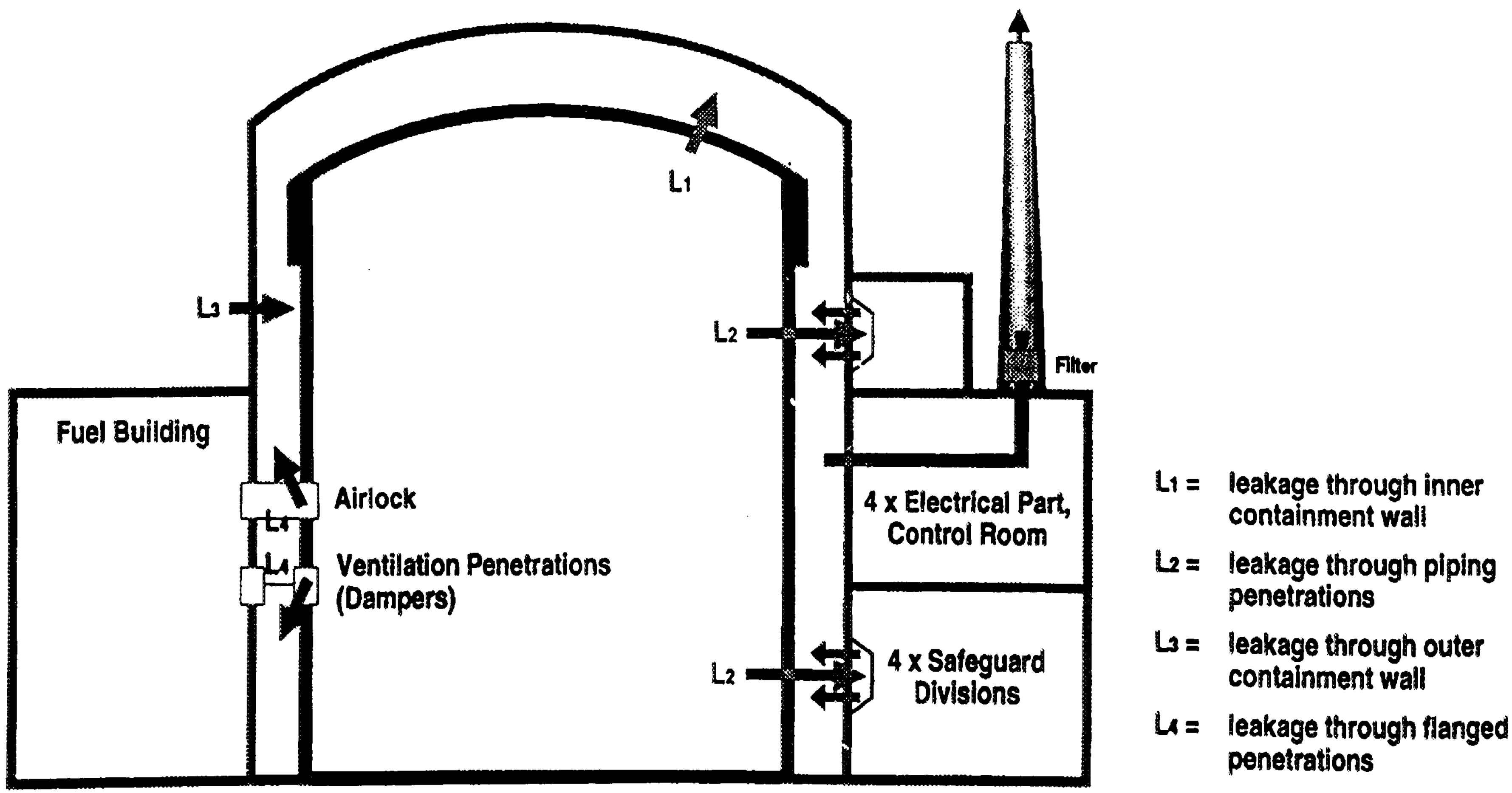



*FIG. 8. Leakage control.*

- low R&D needs,

- low operational constraints during normal plant life (test, maintenance), and in case of accidents.

The selected two train system (Fig. 7) has the following characteristics and performances:

- the containment pressure shall be reduced within 24 h to 2 bar with two trains. One train will be sufficient to maintain the containment pressure under 2 bar in the long term phase.

- the circulation of the water is ensured by taking suction from the IRWST.

- in parallel, approximately 5% of the flow rate will be transferred to the basemat cooling channels.

The backflow will be directed into the IRWST.

- the parts of the system outside containment (pumps, heat exchangers, valves) are located in a leaktight compartment (Fig. 8).

### 4.8. Confinement of the fission products

For ensuring that the radiological releases will be kept as low as required (Section 2.3), any direct leak to the environment must be prevented ("zero bypass" confinement). Therefore the following lines of defence are implemented:

- leaktight containment inner wall up to the accident pressure (Section 4.6), including basemat and state-of-the-art leaktight design of the systems and components passing through the containment building,

- recovery of potential leaks through the inner wall and the penetration sleeves in the containment inter-wall space, recovery in the peripheral buildings,

- transfer of the remaining leaks to the stack after filtration.

To assess the performance of such a confinement, a reference source term corresponding to a medium size LOCA was calculated taking into account the EPR features, i.e. leaktightness of the inner containment, filtration and stack release of the leaks. No effect of the containment heat removal system (spray) was considered. Table II provides typical results for 3 significant isotopes.

With such a source term, the objectives are met. No evacuation will be necessary even after a hypothetical core melt. Furthermore, no long term countermeasures, e.g. relocation, would be justified. Foodstuff produced beyond the first harvest after the postulated event in the immediate vicinity of the plant, could be commercially sold without any restrictions.

## 5. RESEARCH AND DEVELOPMENT

When selecting the features of the EPR, the continuity with the existing practices was an important criterion, limiting the cases where research and development activities were necessary.

TABLE II. TYPICAL RESULTS FOR SIGNIFICANT ISOTOPES

| | Stack release in TBq | |
|---|---|---|
| | Short term (< 24 hours) | Long term |
| Xe 133 | $2.10^4$ | $5.10^5$ |
| I 131 | 11,4 | 30 |
| Cs 137 | 1,8 | 2,7 |

For those of innovative nature, mostly concerning the severe accident case, the ongoing R & D, in particular in France and in Germany, is largely used. In all cases, the results were sufficient to take key orientations to specify the EPR conceptual features. No large R&D needs specific to the EPR were identified. In fact, the strategy in selecting these features was to remain as flexible as possible to adopt alternate solutions if necessary. This can be achieved, e.g. by decoupling certain areas from the general layout as it has been done with the spreading area for melt retention, where still different devices could be introduced.

Of course, complementary actions are essential to confirm these orientations, and possibly, to optimise the design. This is particularly true for:

- the confirmation of the spreading concept,

- the validation of the design assumptions and methods used in the containment design studies for evaluation of inner wall leak rates,

- the selection of materials,

- the qualification of components.

The EPR project also follows attentively the ongoing national and international programs, including those devoted to computer code validation.

## 6. CONCLUSION

The EPR shall simultaneously answer three main concerns:

- to be in conformity with the French and German safety authority requirements,

- to produce a competitive electrical power compared to other energy sources,

- to satisfy the various requirements expressed by the French and German utilities.

These objectives are to some extent, conflicting. Clearly, the enhanced safety requirements have a significant influence on the investment cost.

It must be recognised, however, that a direct compensation is generally possible when considering the preventive features. This is the case for the system organisation, which will provide more flexibility for the maintenance, and therefore, will contribute to increase the plant availability factor. Concerning the mitigative ones (containment design pressure, spreading area, hydrogen control devices, containment heat removal system), there is no benefit for the normal operation. Therefore, this investment cost has to be compensated by other design provisions. In addition to the design features increasing the plant availability, such provisions include:

- large unit size, to benefit from economy of scale effect,

- core design parameters, to enable flexibility in optimising the fuel management,

- increased steam pressure to improve the overall efficiency,

- component design and layout provisions to enable an extended life time.

It is also expected that the selection of the plant features will permit to benefit from a large standardisation effect.

Finally, the present preliminary economic evaluations show that this very challenging problem can be solved, ensuring the competitivity of the EPR.

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# PRELIMINARY THERMAL DESIGN OF A PRESSURIZED WATER REACTOR CONTAINMENT FOR HANDLING SEVERE ACCIDENT CONSEQUENCES

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## Abstract

A one-dimensional mathematical model has been developed for a 4250 MW(th) Advanced Pressurized Water Reactor containment analysis following a severe accident. The cooling process of the composite containment-steel shell and concrete shield- is achievable by natural circulation of atmospheric air. However, for purpose of getting higher degrees of safety margin, the present study undertakes two objectives : (i) Instalment of a diesel engine-driven air blower to force air through the annular space between the steel shell and concrete shield. The engine can be remotely operated to be effective in case of station blackout, (ii) fixing longitudinally plate fins on the circumference of the inside and outside containment steel shell. These fins increase the heat transfer areas and hence the rate of heat removal from the containment atmosphere. In view of its importance - from the safety view point - the long term behavior of the containment which is a quasi-steady state problem, is formulated through a system of coupled nonlinear algebraic equations which describe the thermal-hydraulic and thermodynamic behavior of the double shell containment. The calculated results revealed the following : (i) the passively air cooled containment can remove maximum heat load of 11.5 MW without failure, (ii) the effect of finned surface in the air passage tends to decrease the containment pressure by 20 to 30%, depending on the heat load, (iii) the effect of condensing fins is negligible for the proposed fin dimensions and material. However, by reducing the fin width, increasing their thickness, doubling their number, and using a higher conductive metal than the steel, it is expected that the containment pressure can be further reduced by 10% or more, (iv) the fins dimensions and their number must be optimized via maximizing the difference or the ratio between the heat removed and pressure drop to get maximum heat flow rate.

## 1. INTRODUCTION

One of the severe accident category - beyond design basis accident -is core meltdown which may take place following a loss of coolant accident (LOCA), where the emergency core cooling system (ECCS) malfunctions or fails to refill the reactor core. This type of accident is normally followed by dangerous consequences like steam explosion and hydrogen deflagration or detonation, where the containment pressure may rise beyond its design value (e.g. 15 bar). These effects might cause failure of the reactor containment followed by the release and diffusion of radioactive fission products into the environment thereby threatening the public safety.

Since the containment represents the final barrier confining the aerosol and noble gases it is very important to have a good designed containment at least adequate to mitigate those consequences. For this purpose the present study undertakes two objectives: (i)presenting a preliminary thermal design analysis for the containment of a pressurized water reactor, (ii)proposing some modifications to the current design containments for mitigating the consequences of the accident under consideration.

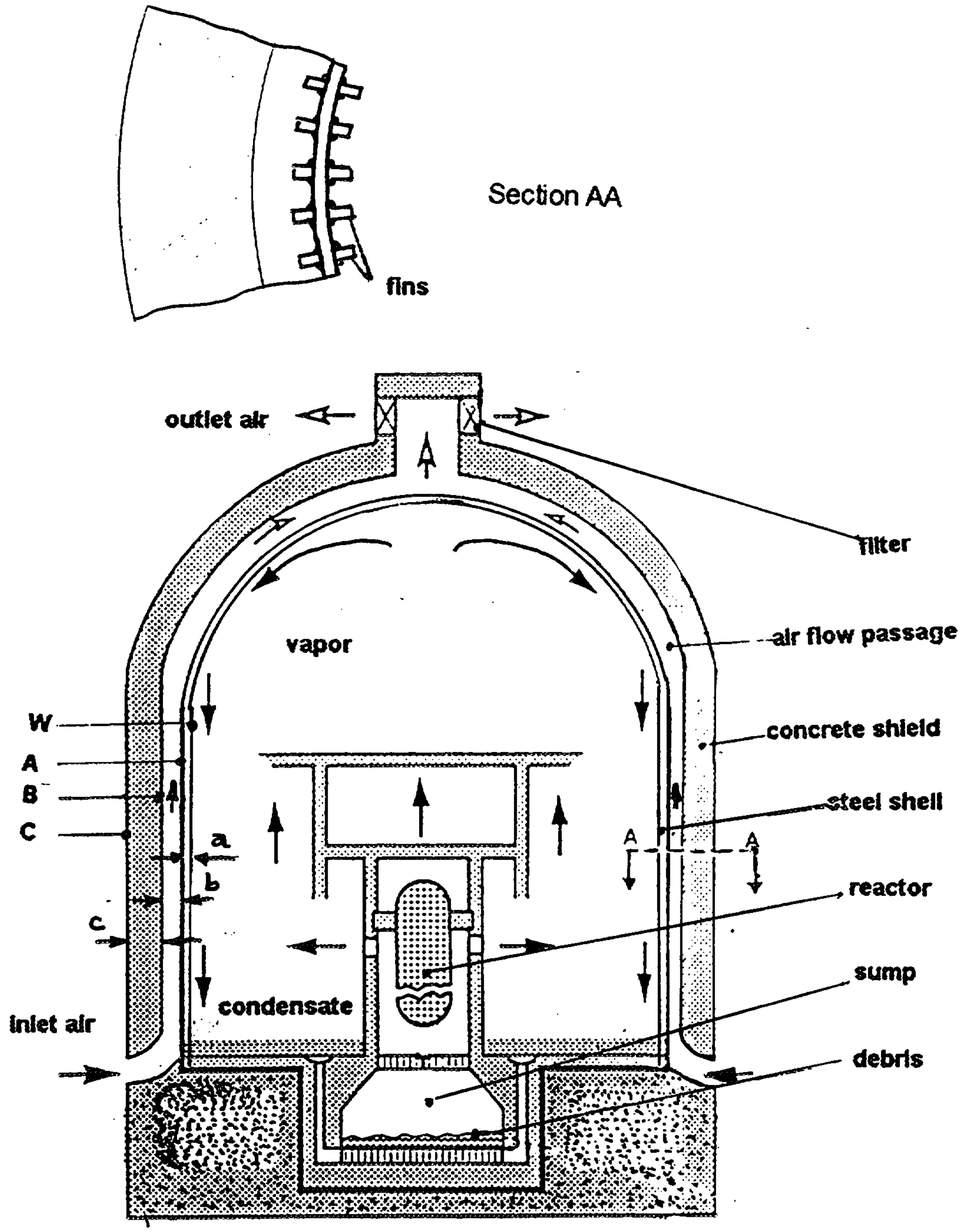


*Figure (1) Illustrative figure of a passive containment after a postulated severe accident.*

The present analysis is applicable to the today's double containment design. This composite containment, as shown in Fig.(1), consists of an inner steel shell 60 m diameter, 40 mm wall thickness, and reinforced concrete shield 2 m wall thickness. The annular gap between the two shells - which is about 80 cm width - represents a flow passage for atmospheric air. Thus the decay heat removal from the containment is based on a passive mechanism (e.g natural convection). However for getting higher degrees of safety, a diesel engine-driven air blower can be installed to force air through the annular space between the steel and the concrete shield. The engine which is energized by d.c. batteries can be remotely operated so that to be effective in case of station blackout. The free or forced air flow extracts the decay heat from the outside surface of the steel containment and transfers it by convection to the air passage and by radiation to the concrete shield where the latter dissipates the heat to the atmospheric air.

Several designs and computer codes for severe nuclear reactor accident containment analysis have been developed [1,2,3,4]. The CONTAIN integral analysis code [1,2] was used for studying the long-term thermal hydraulic conditions after a severe nuclear accident in a PWR where the characteristic time constant is on the order of days. Consequently, the post accident period which continues several weeks must be analyzed to calculate the time behavior of the containment thermodynamic variables (e.g. pressure, temperature, mass fractions of non-condensible gases, etc.) So CONTAIN côde therefore needs a long computing time for covering such long problem times. This is because the code uses calculational time-step size on the order of seconds to deal with the modeling approach used. So it is not practical to use CONTAIN code for parametric and conceptual studies as they are needed during the early design phase of future containment. Therefore, the simplified computer code TPCONT was developed for calculating the containment thermal-hydraulic conditions[3]. The advantage of this code is that the CPU time needed for a typical calculation over 40 days of problem time is 1 minute as compared with 5 hours, the time needed by CONTAIN. However, TPCONT has some uncertainties which need verification by experiments or by more detailed codes.

Another one-dimensional computer code PASCO has been developed [4], for predicting the thermal-hydraulic behavior of a reactor containment cooled by natural convection and to evaluate the experimental results obtained from PASCO test facility [5]. Good agreement between the measured and calculated variables(e.g. temperatures and air flow rates) has been obtained. However no comparison has been made between PASCO and the aforementioned codes.

In the present study the current design containment is modified in such a way that to permit forcing air-by a blower driven by diesel engine-inside the annular space between the two containments. To reduce effectively the containment pressure the steam condensation on the surface can be much enhanced by fixing highly heat conducting plates around the periphery of the steel containment. These plates act as fins to increase the condensation surface area. For plates 40 m high,50 cm width, and 0.5 m spaced, their total surface area may reach the cooled containment surface area. The plates act also as supporting or reinforced stays to the steel containment. Also it is possible to support longitudinal metallic plate fins on the periphery of the outside surfaces of the steel containment and concr te shield. For this purpose a one dimensional mathematical model has been developed for a PWR containment analysis following a severe core melt accident.

## 2.THE CONSEQUENCES FOLLOWING THE ACCIDENT

Current PWR plants have not been designed to withstand core meltdown and the ensuring accident events. Following the accident, the consequences may embrace some physical and mechanical processes. During the early period, which lasts about few hours after accident initiation, transient events likes blowdown, pressure vessel failure, steam explosion and hydrogen combustion or detonation may occur. The latter process may be followed by sudden and heavy load exerted onto the containment structure. The consequence would be mechanical failure (major leakage) of the containment and release of gases carrying radioactive materials into the environment . A very essential criterion for a future containment design must be the capability of the containment to resist the above mentioned process without failure.

## 3.MATHEMATICAL MODEL

In this study a one-dimensional mathematical model is adopted to describe the thermal and hydraulic processes which occur across the containment surfaces and the fluids boundary layer(condensate film and cooling air). After the occurrence of the accident the debris in the containment sump generates steam. Part of this steam condenses on the containment internal structure, steel installations, and the containment steel shell, while the rest accumulates gradually. Since the steel has relatively low heat capacity, high thermal conductivity and small volume (500 $m^3$), its temperature reaches the saturation condition in a few hours. As for the concrete internal structure, where it has low thermal conductivity and high heat capacity and large volume (13,200 $m^3$), it is heated slowly, as compared with containment atmosphere, and still absorbs heat for a relatively long period ( about 10 days) until it reaches a thermal equilibrium state with the containment atmosphere. At this time the internal concrete no longer absorbs heat with the result that approximately all the steam produced by decay heat -under certain equilibrium case- condenses on the containment steel surface and transmitted by the air. This represents the maximum rate of heat which can flow through the containment steel shell. From the view point of containment safety the time period preceding the equilibrium state (t<10 days) is not serious although the decay heat varies between 91.6 to 11.5 MW, and that the heating rate of internal concrete is about 20 °C per day. On the other hand the next period (t>10 days) is less safe. Since, the containment atmosphere pressure and temperature remain relatively high for a long period where the cooling rate is about 1°C /day. Thus any disturbance in the thermodynamic conditions of containment (e.g. hydrogen combustion, temperature rise in the atmospheric air or partial blocking of the air passage exit) may produce dangerous unsafe conditions. Based on the above discussion, the present study is therefore focused on the important long term period which comes after attainment of the equilibrium state between the containment vapor and internal structures (i.e for t>10 days). Therefore this system is treated as a quasi-steady state problem as described by the following equations.

### 3.1.Overall Heat Balance Inside The Containment

It is assumed that the containment water, atmosphere, and internal structure are at saturation conditions. Thus in the final steady-state case inside the containment the decayed heat is transferred to water present in the sump as a latent heat of vaporization. Meanwhile a part of produced vapor condenses and transfers its latent heat to the internal steel surface. The overall heat balance is as follows :

$$Q = A \cdot F \cdot F_1 \cdot h \cdot (T_{sat} - T_w) \quad . \qquad (1)$$

where $F_1$ is a factor of condensation heat transfer reduction due to presence of air with steam and can be determined [6] from the following equation :

$$F_1 = 0.4254 * \left(\frac{m_a}{m_v}\right)^{-0.573} \quad if \left(\frac{m_a}{m_v}\right) \le 0.05$$

$$= 0.17 \quad . \quad if \left(\frac{m_a}{m_v}\right) > 0.05$$

$$A.F = \eta \cdot A_f + A \quad .$$

and F is a factor of area increase due to the presence of the fins where $\eta$ is the fins efficiency and $A_f$ is the fins area[7].
h can be determined from McAdams filmwise condensation over a vertical plate

$$h = 1.13 * \left[\frac{g \cdot \rho_l \cdot (\rho_l - \rho_v) \cdot h_{fg} \cdot k_l^3}{\mu_l \cdot (T_{sat} - T_w) \cdot L_{eff}}\right]^{0.25}$$

and the steam mass can be determined as a function of state equation as follows:

$$P_v = \frac{m_v}{18} \cdot \frac{R \cdot T_{sat}}{V} \quad . \qquad (2)$$

where $m_v$ is the vapor mass and 18 is the water molecular weight.

The heat flow through the containment steel shell Q can be determined [8].
The only unknowns in the above equation are $T_{sat}$ and $T_w$, considering that water and steam properties can be correlated as a function of mean film temperature. Also the steam pressure can be correlated as a function of saturation temperature.

Steel inner surface temperature can be determined from the following equations assuming that in the steady state the heat transferred from the condensation process is conducted through the steel shell.

### 3.2. Heat Conduction Through The Steel Shell :

The heat flow is conducted through the steel shell by conduction:

$$Q = \frac{2\pi \cdot k_s \cdot L_{eff}}{\ln((D+2a)/D)} \cdot (T_w - T_a) \qquad (3)$$

### 3.3. Heat Flow at Surface "A" :

The heat conducted through the steel shell is convected to the flowing air through the air passage and radiated to concrete surface b as follows :

$$Q = h_a \cdot F \cdot \pi \cdot (D+2a) \cdot L_{eff} \cdot (T_a - Tm) + \pi \cdot (D+2a) \cdot L_{eff} \cdot \frac{\sigma}{\frac{1}{\epsilon_a} + \frac{1}{\epsilon_b} - 1} \cdot (T_a^4 - T_b^4) \qquad (4)$$

where the convective heat transfer is experienced to be by turbulent forced convection:

$$h = \frac{k_a}{L_{eff}} * 0.1 \ (Gr \cdot Pr)^{\frac{1}{3}} \quad , \quad Gr = \frac{g \cdot \beta \cdot (T_a - T_m) \cdot L_{eff}^3}{\nu^2}$$

The air properties are taken at the mean film temperature and all air properties are correlated as functions in the temperature.

### 3.4. Flowing Air Momentum Equation :

The buoyancy driving pressure is equated with the pressure drop of inlet air, acceleration, diversion through the air passage, friction, and the outlet pressure drops. It is assumed that the inlet and outlet manifolds are smooth. Therefore the momentum equation can take the following form :

$$g.L_{eff}.(\rho_{ain}-\rho_m) = \tfrac{1}{2}\,\rho_m.U^2.\left[2 + \frac{0.1364*L_{eff}}{D_h*Re^{\frac{1}{4}}}\right] \qquad (5)$$

The air properties can be correlated as functions in the temperature, reducing the unknowns to $T_m$ and $U$.

### 3.5. Heat Balance in The Air Passage :

The heat convected from surfaces "A" and "B" increases the air energy content. The equation describing the passage air heat balance is as follows :

$$\rho_m.U.\frac{\pi}{4}.\left[(D+2a+2b)^2-(D+2a)^2\right].(T_m-T_{ai}) =$$
$$h_a.F.\pi(D+2a).L_{eff}(T_a-T_m) + h_b.\pi.(D+2a+2b).L_{eff}(T_b-T_m) \qquad (6)$$

### 3.6. Heat Balance of The Concrete Shield "B" :

A part of the net heat radiated from surface "A" to surface "B" is convected to air flowing in the air passage. Meanwhile the rest is conducted to the surface "C" as shown below :

$$\pi.(D+2a).L_{eff}.\frac{\sigma}{\frac{1}{\epsilon_a}+\frac{1}{\epsilon_b}-1}.(T_a^4-T_b^4) =$$
$$h_b.\pi.(D+2a+2b).L_{eff}.(T_b-T_m) + \frac{2.\pi.k_c.L_{eff}}{\ln\frac{D+2a+2b+2c}{D+2a+2b}}.(T_b-T_c) \qquad (7)$$

Finally the heat conducted through the concrete shield is convected to the ambient air as given below :

$$\frac{2.\pi.k_c.L_{eff}}{\ln\frac{D+2a+2b+2c}{D+2a+2b}}.(T_b-T_c) = h_c.\pi.(D+2a+2b+2c)L_{eff}.(T_c-T_{ai}) \qquad (8)$$

The above eight equations contain eight unknowns namely $T_{mat}$, $T_w$, $T_a$, $T_m$, $T_b$, $T_c$, $m_v$ and $U$. These equations have been solved numerically to calculate the above unknowns.

## 4. Results and Discussion

The present work aimed to reinforce the containment engineering safety features. This is available through a smart design to enhance the performance

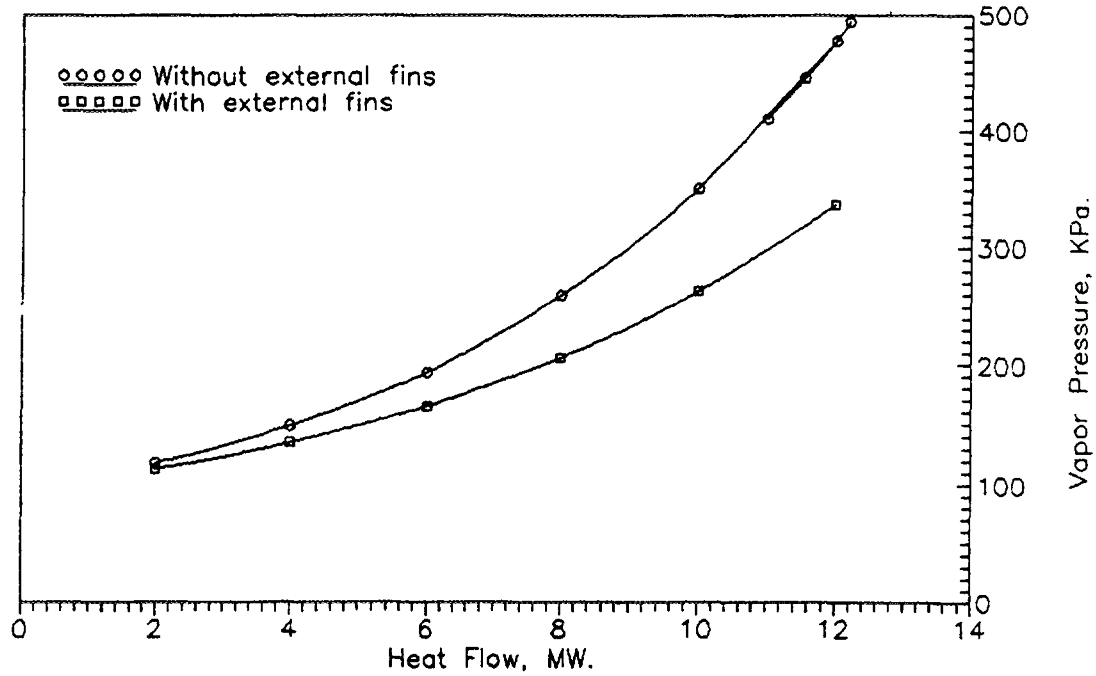


*Figure (2) Passive Containment Total Pressure with Removable Heat Load Variation with Fins Presence.*

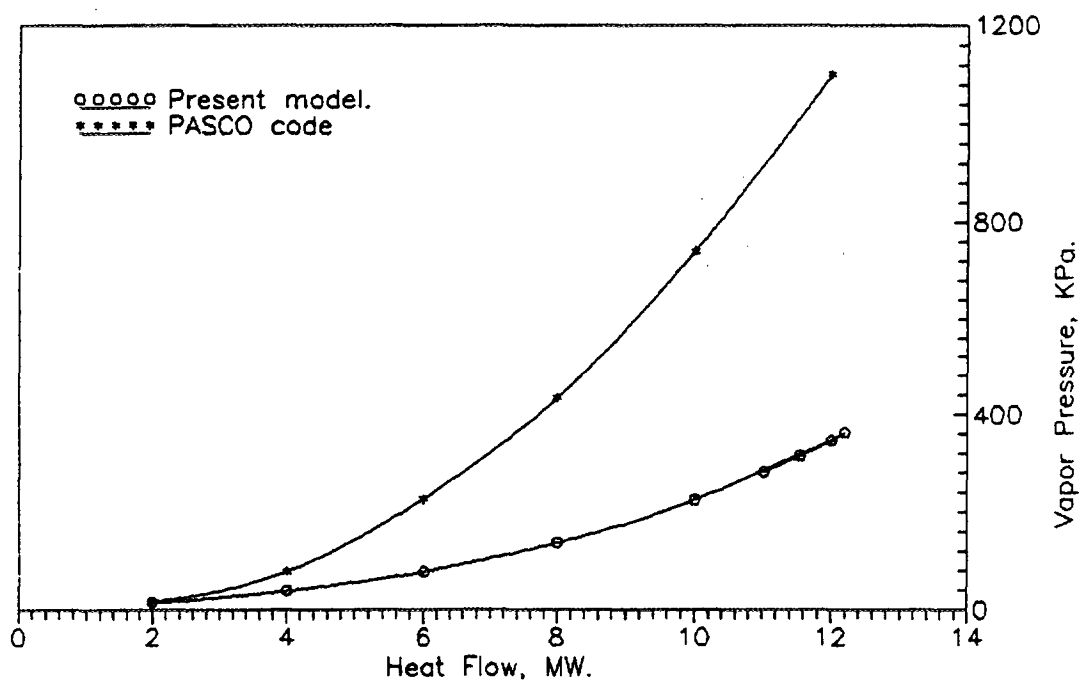


*Figure (3) Passive Containment Vapor Pressure with Removable Heat Load (without Air Filtering).*

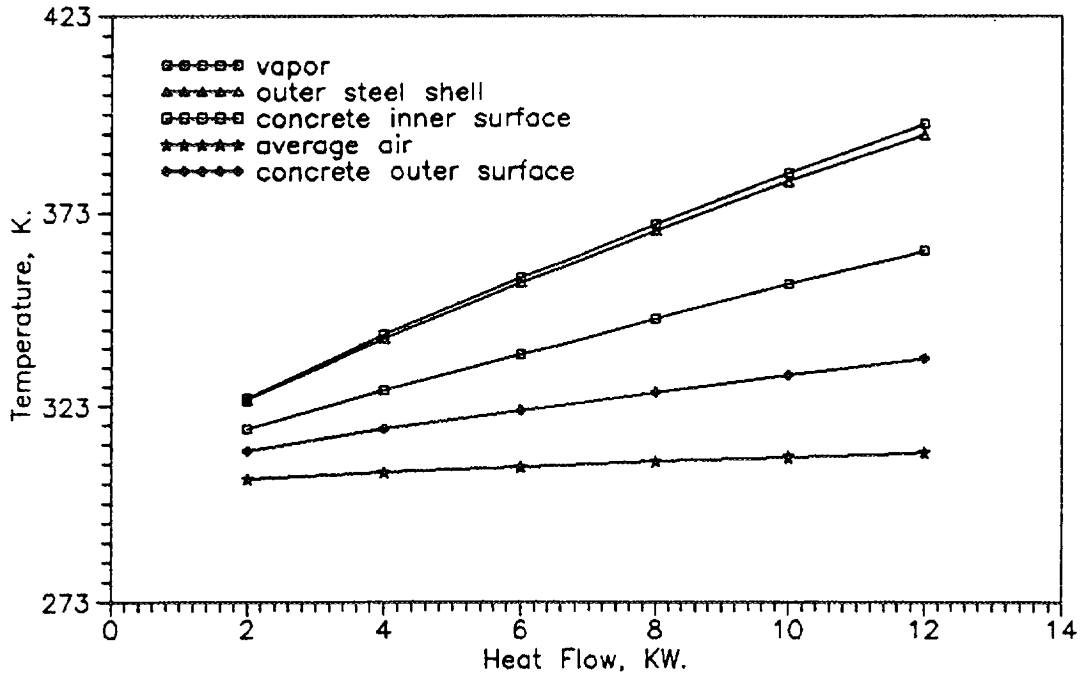


*Figure (4) Passive Containment Temperature Distribution (with external fins)*

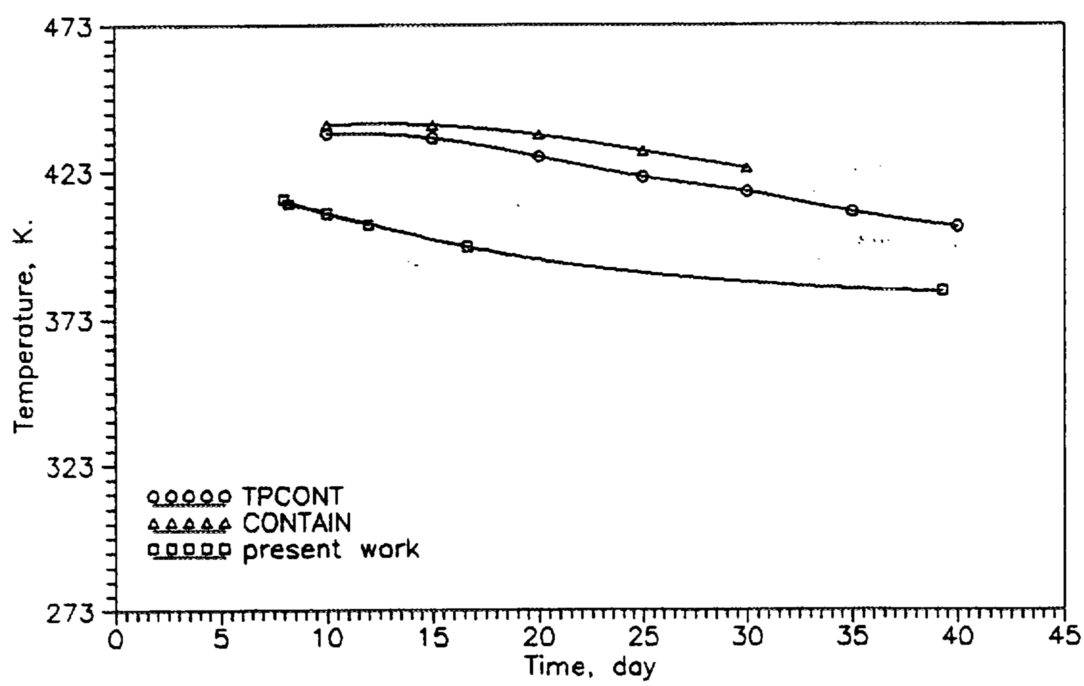


*Figure (5) Comparison Between Different Containment Codes Temperatures and The Present Model.*

of the double passive containment natural heat circulation after a postulated severe accident. The proposed design enhancement is to substitute the weakness of bad heat transfer coefficient of the flowing air by increasing the effective heat transfer area on the outer side of the containment steel shell. A simple one-dimensional mathematical model was employed and compared with by other works to stand up the validity of this model. Hence a comparison between the finned and unfinned outer steel shell was made.

The results presented here are based on the containment data reported in ref. 3 and 4. The system of the nonlinear algebraic equations derived , have been solved simultaneously for the unknown variables. The thermophysical properties of air and condensate film have been expressed analytically in terms of their respective temperatures. Due its importance as justified before at the beginning of the text, we are interested in the long term thermal-hydraulic and thermodynamic behavior of the containment. First, the effect of the longitudinally supported steel fins on the containment steel shell is investigated. The results -which are based on fin dimensions of 40 m height, 0.5 m width, 0.002 m thickness, and 0.5 m spaced- showed that the effect of external fins, which are supported on the outer surface of the containment steel shell, tends to reduce the containment pressure by about 20 to 30% as can be seen from Fig.(2), where the rate of pressure reduction increases with the removed heat load. On the other hand the calculated results revealed that the internal condensing fins have negligible effect due to the highly steam condensation coefficient. However, it is expected that these fins can be more effective if their number is doubled, i.e. 0.25 m spaced, with 0.25 m width, 0.04 m thickness, and made of a higher conductive material than steel. Fig.(3), shows the variation of the containment pressure with the removable load -which was taken as input variable- as compared with PASCO code. It can be seen that the differences between the two sets of results are relatively large. The higher pressure values of PASCO may be attributed to its low values of the air stream velocity (e.g. about 0.8 m/s). This diminishes the cooling rate of the containment atmosphere, thereby reducing the rate of pressure decrease. The variation of temperature distribution across the composite containment with the removable heat load is depicted in Fig.(4). These results are based on a finned surface in the air flow passage. It can be seen that the containment steel shell temperature reaches about 125 °C. The corresponding temperature without fins can be calculated to be 130 °C. These values correspond to a maximum heat flow of 11.5 MW as calculated in ref.[5]. The corresponding containment pressures are 4 and 5 bar respectively, which demonstrates that the fins reduce the containment pressure by about 20%. Finally, Fig.(5) shows the containment atmosphere temperature as compared with the TPCONT and CONTAIN codes. It can be seen that the higher temperatures detected by these codes may be attributed to the filters installed into the air pass during these codes calculations.

**NONEMCLATURE**

$a$ =containment steel shell thickness(m).
$b$ =air passage width (m)
$c$ =concrete shield thickness(m).
$c_p$ =specific heat(J/Kg.°K).
$D$ =containment diameter(m).
$F$ =factor of non condensible gases on heat transfer coefficient.
$Gr$ =Grashoff number.
$h$ =enthalpy(KJ/Kg), also heat transfer coefficient(W/m$^2$.°K).
$K$ =thermal conductivity(W/m.°K).
$L$ =containment height(m).
$L_{eff}$=containment effective height(L+D/2) (m).
$m$ =mass flow rate(Kg/s).

n =number of plate fins.
$n_v$ =number of vapor moles.
P =pressure(KPa).
Pr =Prandtl number.
Q =thermal power(W).
T =temperature(°K).
t =time(s), also fin thickness(m).
U =velocity(m/s).
V =containment free volume($m^3$).

**Greek Symbols**

$\beta$ = coefficient of thermal expansion( $°K^{-1}$).
$\eta$ = fin efficiency
$\mu$ =dynamic viscosity(Kg.s/$m^2$).
$\nu$ =kinematic viscosity($m^2$/s).9
$\rho$ =density(Kg/$m^3$).
$\sigma$ =Stefan-Boltzman constant(W/$m^2$.$°k^4$).

**Subscripts**

a =outside containment steel hull wall, also air.
ai =input air.
b =inside surface of concrete shield.
c =outside surface of concrete shield.
l =condensate.
m =mean value.
s =steel.
sat =saturation condition.
v =vapor.
w =inside containment steel shell surface.

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# CONSIDERATIONS OF SEVERE ACCIDENTS IN THE DESIGN OF KOREAN NEXT GENERATION REACTOR

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**Abstract**

The severe accident is one of the key issues in the design of Korean Next Generation Reactor (KNGR) which is an evolutionary type of pressurized water reactors. As IAEA recommends in TECDOC-801, the design objective of KNGR in regard with safety is to provide a sound technical basis by which an imminent off-site emergency response to any circumstance could be practically unnecessary. To implement this design objective, probabilistic safety goals were established and design requirements were developed for systems to mitigate severe accidents. The basic approach of KNGR to address severe accidents is firstly prevent severe accidents by reinforcing its capability to cope with the design basis accidents (DBA) and further with some accidents beyond DBAs caused by multiple failures, and secondly mitigate severe accidents to ensure the retention of radioactive materials in the containment by providing means to maintain the containment integrity. For severe accident mitigation, KNGR principally takes the concept of ex-vessel corium cooling. To implement this concept, KNGR is equipped with a large cavity and cavity flooding system connected to the in-containment refueling water storage tank. Other major systems incorporated in KNGR are hydrogen igniters and safety depressurization system. In addition, the KNGR containment is designed to withstand the pressure and temperature conditions expected during the course of severe accidents. In this paper, the design features and status of system designs related with severe accidents will be presented. Also, R&D activities related with severe accident mitigation system design will be briefly described.

## 1. INTRODUCTION

Korean Next Generation Reactor (KNGR) is an evolutionary type of pressurized water reactors with a rated capacity of 1350 MWe. As other ALWRs currently being developed in the world, KNGR also considers severe accidents in the design, following the fundamental safety principle of the defense-in-depth concept. Therefore, severe accidents are systematically addressed in the design by identifying the vulnerability of the current plant design and defining systems and equipment features or analysis works necessary for the safety enhancement.

In the KNGR design, severe accidents are considered in conjunction with safety goals. The KNGR design objective for safety is in the same track with the IAEA's safety objectives described in TECDOC-801. Thus, severe accidents are treated in the design to the extent that no significant radiological consequence to the vicinity of a nuclear power plant site can be technically assured. Whence, an immediate response outside of site boundary would be practically unnecessary for at least 24 hours after core melt occurs.

To practically implement the safety objective, probabilistic safety goals were established. As a safety goal to prevent severe accidents, the core damage frequency(CDF) shall be less than 1E(-5)/RY considering internal and external events except for seismic events which need to be dealt in a different way. For the mitigation goal, radiation release frequency which exceeds 10mSv(1 rem)/24 hours at the site boundary shall be less than 1E-6/RY. Further, a limit of the long-lived radioisotope release is required such that the frequency of accidents releasing more than 100 TBq of Cs-137 shall be less than 1E(-6)/RY to ensure the use of land around the plant site.

To meet the safety goals, specific design requirements have been developed[1,2]. For the prevention of severe accidents, increasing design margin and reinforcing the reliability of conventional engineered safety features (ESFs) are the major focus of requirements. For the mitigation of severe accidents, containment structural integrity under the severe accident conditions is almost important. To address the severe accident phenomena challenging the containment integrity, specific systems and their functional requirements were also developed. Especially, those phenomena challenging early containment failures, such as hydrogen detonation and high pressure core melt ejection, shall be explicitly considered in the design to virtually eliminate their threat to the containment.

In the next section, we will introduce the basic concept of KNGR to prevent and mitigate severe accidents. In Section 3, the system features and their design status for severe accident mitigation such as the containment general arrangement and cavity structure will be presented along with some of key design issues. In Section 4, R&D works related with severe accident mitigation to support the KNGR design will be briefly described and in Section 5, we will conclude the paper by presenting the results from preliminary probabilistic safety assessment of KNGR and future schedule.

## 2. DESIGN APPROACH OF KNGR TO DEAL WITH SEVERE ACCIDENTS

In this section, we will introduce the design approach of KNGR to cope with severe accidents in two steps: one for prevention and the other for mitigation.

### 2.1 Prevention of Severe Accidents

Preventing severe accidents generally means that the progression of initiating events is arrested in the category of design basis accidents(DBAs) so that the integrity of fuel can be maintained within the design limits. To prevent severe accidents, therefore, it is important to suppress accident initiators and ensure the proper function of engineering safety features.

Preventing severe accidents is implemented in the design by three steps: 1. increasing design margin to absorb abnormal transients, 2. enhancing the reliability of ESFs, and 3. extending ESF or other necessary system functions considering multiple failure conditions beyond DBAs. Table 1 summarizes the design improvements in KNGR to prevent severe accidents.

Table 1 includes design improvements for the prevention of containment bypass accidents which are another type of severe accidents. Prevention is more important for containment bypass accidents because they might result in unacceptable radiological consequences without core degradation. In KNGR, the Intersystem LOCA and multiple S/G tube rupture accident are explicitly considered in the design as noticed in Table 1.

### 2.2 Mitigation of Severe Accidents

If Engineered Safety Features(ESFs) fail to arrest the progress of accidents, the situation becomes severe accident conditions so that core degradation and successively reactor vessel failure could occur. The mitigation of severe accidents, therefore, are important measures to cool the degraded core and, thus, ensure the containment integrity.

**Table 1** KNGR design improvements for severe accident prevention.

| Category | Related Systems | Improved Features |
|---|---|---|
| Design Margin Increase | - Pressurizer | - PZR volume of 68 $m^3$(2400 $ft^3$){~0.017 $m^3$(0.62 $ft^3$)/MWth}<br>- No PORV installed<br>- No safety valve actuation for mild overpressure transient such as loss of load. |
| | - Steam Generator | - Dryout time of 30 minutes |
| | - Reactor Core | - Thermal margin of 10 ~ 15 % |
| ESF Reliability Enhancement | - SIS | - 4 Trains and dedicated system<br>- No realignment of suction line with the use of IRWST |
| | - EFWS | - 2 EFW tanks and dedicated system<br>- 2 Trains with 2 pumps per train |
| | - CSS | - Use of SCS pumps and Hx for back-up |
| ESF function extension/Other system reinforcement | - SIS and SDS with IRWST | - Feed and bleed cooling capability |
| | - SDS with IRWST | - Depressurization of RCS without containment contamination. |
| | - Reactor Protection System | - Use of a control grade alternate protection system for shutdown in case of ATWS |
| | - On-site Electrical Power | - Use of a gas turbine generator for AAC in case of SBO |
| Containment Bypass Prevention | - SCS and CVCS | - Higher piping design pressure for prevention of Intersystems LOCA |
| | - Steam Dump System | - No atmospheric dump in case of SGTR<br>- Use of N-16 detectors for early detection of the S/G tube rupture<br>- Higher reliability of steam dump valve operation |

In the KNGR design, the principal approach to mitigation is the ex-vessel cooling by providing a large cavity and cooling system. Whence, the cavity shall be designed to capture the corium and an associated cavity cooling system shall be provided to cool it. This concept is based on the defense-in-depth principle because the corium can be retained in the cavity which acts like an additional container. For cavity cooling, KNGR takes the pre-flooding strategy principally. In case that pre-flooding fails, however, fusible plugs are provided to enable the post-flooding.

Though the principal approach for severe accident mitigation in KNGR is the ex-vessel cooling using the cavity and associated system, a concept of cooling the reactor vessel exterior is being carefully examined for the feasibility of indirect cooling of corium relocated to the vessel lower region. For the ex-vessel flooding, the possible maximum flooding level without a change of the current plant layout design is found to be about 2.7m(8.9 ft) from the bottom of the reactor vessel. This level is good enough to flood the lower head of the vessel. However, more studies such as critical heat flux at the surface of the vessel and in-vessel melt progression are necessary to finally determine whether to take this concept.

Severe accidents involve various phenomena which can be categorized as High Pressure Melt Ejection (HPME), Direct Containment Heating (DCH), Hydrogen Deflagration and Detonation, Ex-Vessel Steam Explosion(EVSE) and Molten Core-Concrete Interaction (MCCI). The mitigation systems in KNGR focus on how to effectively manage these phenomena and are basically to ensure the containment integrity. Table 2 summarizes these design features and Section 3 will further describe the system design in detail.

**Table 2** KNGR design features for severe accident mitigation

| System | Major Functions & Related Phenomena | Performance Requirements / Design Criteria |
|---|---|---|
| Hydrogen Igniter System | - Combustible Gas Control | - Maintaining $H_2$ concentration lower than 10% with 100 % oxidation of active fuel cladding material |
| Reactor Cavity and Cavity Cooling System | - Retention of core debris<br>- Cooling of core debris to prevent MCCI and enable the long term cooling | - Sufficient area for core debris spreading ($0.02m^2/MWth$)<br>- Enough structural strength for steam explosion and DCH load<br>- Sufficient cooling water from IRWST |
| SDS and IRWST | - Rapid depressurization to prevent high pressure core melt ejection | - Depressurization capacity to 1.7 MPa (250 psig) before reactor vessel breach occurs |
| Containment | - Retention of radioactive materials under severe accident conditions | - ASME Factored Load Category as the ultimate structural capacity for the severe accidents |

## 3. KNGR DESIGN STATUS FOR SEVERE ACCIDENT MITIGATION SYSTEMS

### 3.1 Containment and General Arrangement

The containment is ever more important for severe accident mitigation since it is the last barrier against the release of radioactive materials. A preliminary design of the KNGR containment and general arrangement was completed and described in Ref.3 in detail. The KNGR containment is a large dry type and designed considering the load conditions due to severe accidents. The pressure capacity is sufficient such that the containment structural integrity can be maintained below the ASME Factored Load Category during the first 24 hours after core melt.

The KNGR containment is a double containment type as shown in Fig.1. The inner containment is a steel-lined, prestressed concrete cylinder with a hemispherical dome. The internal diameter of the inner containment is 45.7 m(150 ft). The nominal wall thickness is 1.2 m(4 ft) up to dome spring line elevation and the dome thickness is 1.1m(3.5 ft). The maximum containment height is 52.9 m(173.5 ft) above the operating floor. The 6 mm(1/4 in) thick steel liner plate is installed for leak tightness. The inner containment free volume excluding the volume of ICI cavity and drain sumps is 9.1E(4) $m^3$(3.2E(6) $ft^3$). The containment free volume is large enough not to exceed 13 v/o of hydrogen produced by 75% of the active fuel clad oxidation without any active countermeasure. The global hydrogen burning based on the adiabatic isochoric complete combustion model was found to result in the maximum containment pressure of 0.8 MPa(117 psia) with this free volume.

The outer containment is for biological shield and made of a reinforced concrete right cylinder with an inner diameter of 52.4 m(172 ft). The outer containment structure has sufficient strength for structural support and missile protection for which a local impact due to tornado-generated missiles is assumed. The KNGR containment and auxiliary building will be built on the common basemat. This provides an advantage for seismic design since it reduces the flexural shear loads in the auxiliary building shear walls and outer containment.

The annulus between the inner and outer containments is 2.1 m(7 ft) wide. The main function of the annulus is the collection of leakage from the inner containment. The leakage is filtered and recirculated back to the inner containment by the Annulus Ventilation System(AVS). The annulus also provides an access for installing, testing, inspecting, and tensioning the tendons. The annulus compartment is considered as a part of the penetration area. Thus, high energy lines in the annulus shall be enveloped by guide tubes or extended sleeves.

Fig.2 shows the general arrangement of the KNGR nuclear island in a plan view. The safety systems are located in the auxiliary building which surrounds the containment and the redundant trains of safety systems are physically separated by quadrant or symmetrical arrangement. As noticed in Fig.2, the SIS pumps are placed in each quadrant and the CSS pumps, EFWS tanks, and On-site Diesel Generators are symmetrically arranged. This arrangement is to prevent the propagation of external events such as fire and flood from one region to another. The containment internal arrangement was designed such that the mixing by natural circulation can be maximized and local accumulation of hydrogen can be prevented. Especially, the annular vent gap between the inner containment wall and operating floor was extended to 0.3 m(1 ft) for natural circulation from lower to upper compartments of the containment.

3.2 Reactor Cavity and Cavity Cooling System

The KNGR reactor cavity houses the reactor vessel and the in-core instrument(ICI) tubes. Additionally, the cavity has a role for the retention and cooling of core debris in case of severe accidents. The current cavity design considers pre-flooding strategy which fills the lower volume of the cavity before the reactor vessel failure occurs. Thus, the design issues of the cavity are:

1. the availability of the cooling water, 2. the cavity space enough to assure core debris spreading and coolability, and 3. the cavity structural strength to withstand the pressure load by steam explosion. The cooling water for the cavity is supplied from IRWST which is an enormous reservoir of cooling water. The pre-flooding strategy would ease the issue of core debris spreading and cooling due to the mixing effects by Fuel-Coolant Interaction. For pre-flooding strategy, it is considered that steam explosion is the most challenging issue. Accordingly, the cavity and reactor vessel support structural strength must consider the steam explosion load.

Fig.3 shows a schematic of the KNGR reactor cavity design along with the indication of vent path. The KNGR cavity has the approximately 84 $m^2$(906 $ft^2$) floor area which is about 0.021$m^2$ (0.23 $ft^2$)/MWth. To prevent direct containment heating due to the escape of core debris to the upper part of the containment, the cavity configuration is designed to minimize debris entrainment. Since there are seals around the reactor vessel head and ICI table in addition to the corbel and primary shield plugs which restrict the flow through the reactor vessel annulus, the steam and gases produced in the cavity escape mostly through the vent pathway as indicated in Fig.3. The vent pathway is designed to take many turns to knock off the entrained debris from the gas flow.

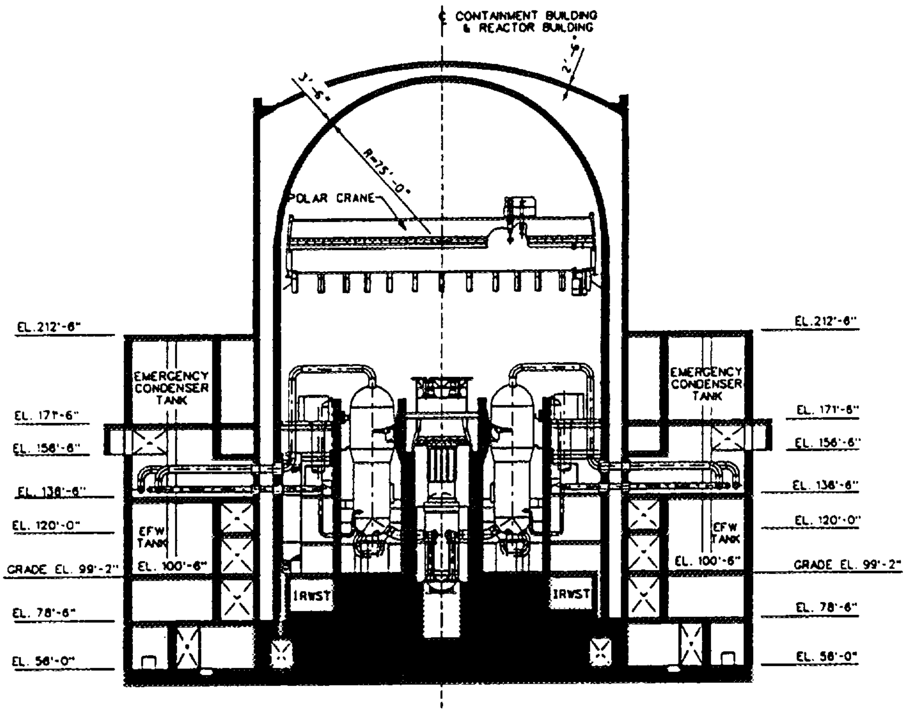


**Figure 1** A cross-section of the KNGR containment in the vertical direction

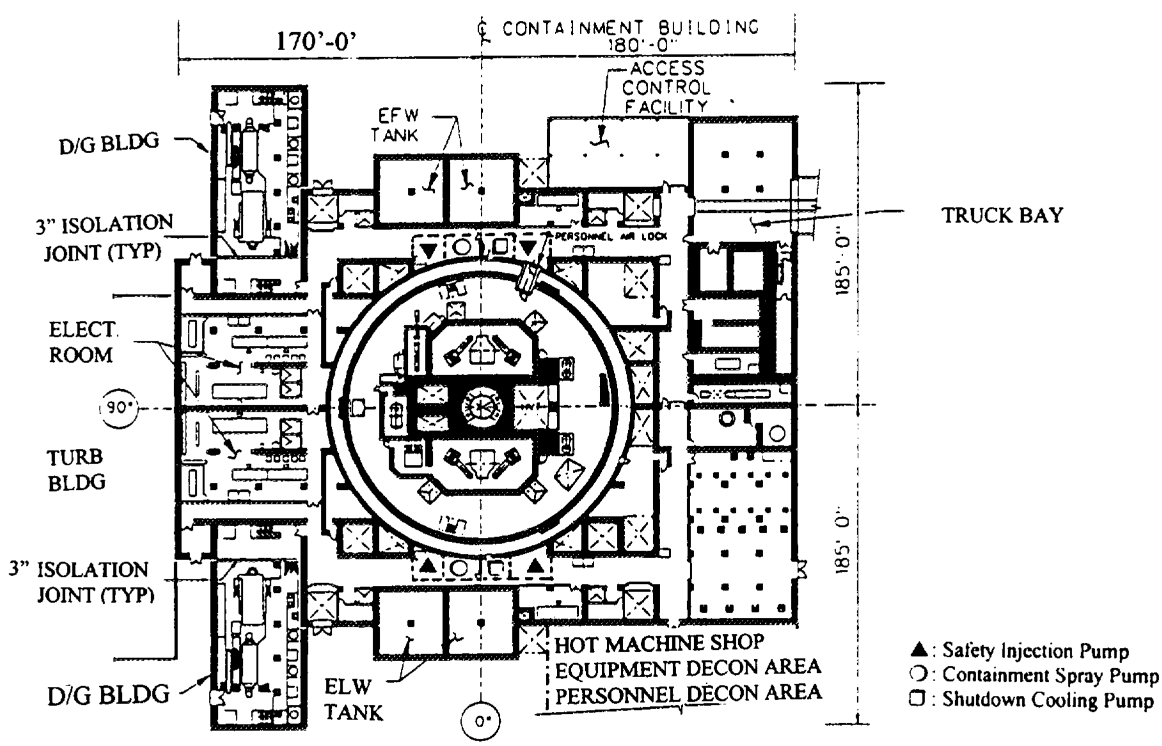


**Figure 2** A cross-section of the KNGR containment in the horizontal direction

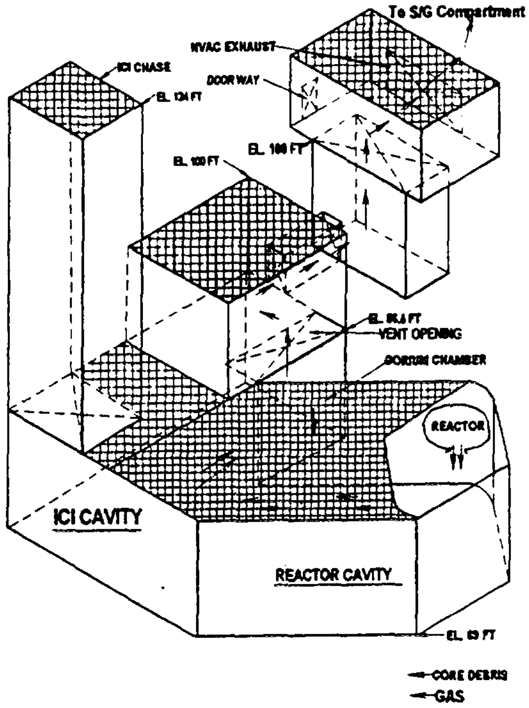


**Figure 3** Schematic of the cavity configuration and vent pathway(Not in scale)

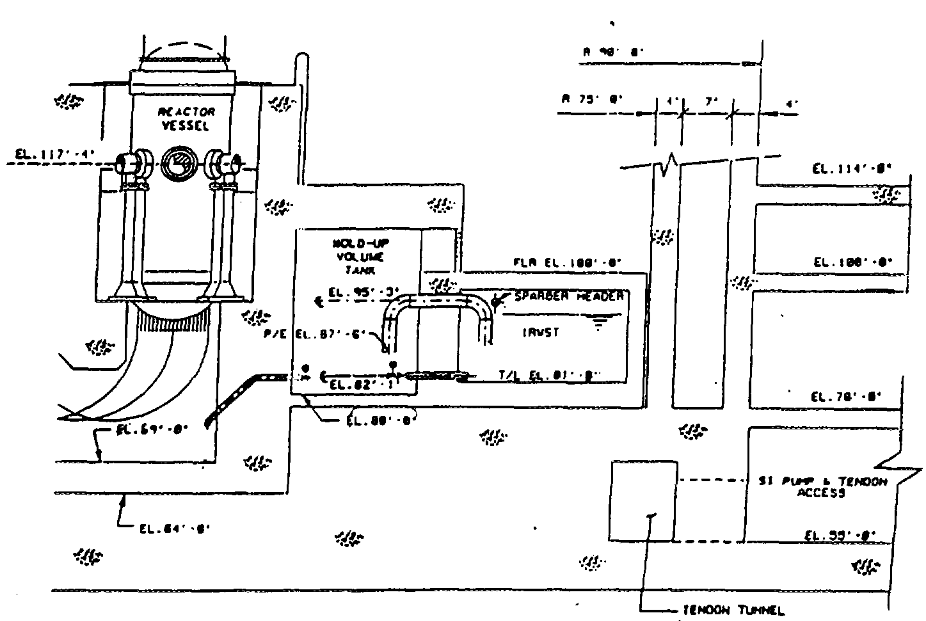


**Figure 4** Cavity water supply path

The cavity floor has an approximately 0.9m(3 ft) thickness sacrificial layer above the containment liner plate. The sacrificial layer is for the Molten Core-Concrete Interaction(MCCI) which will occur before a stable cooling of corium is established. The material for this layer needs to be highly resistant to MCCI without producing a large amount of non-condensible gases.

The cavity is associated with the Cavity Flooding System(CFS) to perform its function during severe accidents. Since KNGR takes pre-flooding strategy in principle, the cavity flooding starts by opening motor operated valves(MOV) manually when the core uncovery is detected. The core uncovery is determined by the core exit temperature and reactor vessel level monitors. Additionally, an abrupt change of neutron flux level may be used as an indicator.

The water supply path to the cavity is illustrated in Fig. 4. The water flows first into the Hold-up Volume Tank through the two 35.6 cm(14 in) diameter HVT spillways and then into the reactor cavity through the 25.4 cm(10 in) diameter reactor cavity spillways by gravity. The flow stops when the water level in the cavity equalizes to that in the IRWST. The equilibrium level shall be below the ICI plate under the reactor vessel lower head so that the contact of water with the ICI tube by inadvertent opening of MOVs can be excluded. In case that MOVs fails, the CFS can deliver water through two pipes with the fusible plugs. The fusible plugs are designed to melt due to the heat from the corium accumulated in the dry cavity so that the cavity flooding can start in a passive way.

3.3 In-Containment Refueling Water Storage Tank(IRWST)

The IRWST contains water for refueling operation and more importantly it plays many roles for accident mitigation. The IRWST provides water source for the safety injection and containment spray systems. It becomes a heat sink for the RCS inventory discharged from the pressurizer safety valves or safety depressurization system, and supplies water for cavity flooding. The water inventory shall be sufficient to perform these functions as well as refueling operation.

The IRWST locates below the basement floor slab and between the secondary shield and inner containment walls as shown in Figs. 1 and 5. In Fig. 5, the SDS discharge lines to IRWST and sparger locations are also illustrated. The SDS discharge lines are not so symmetrically arranged due to the restriction of the pressurizer location. However, the sparger lines are symmetrically installed to promote uniform mixing of discharged flow in the IRWST. There are two sparger headers and each header has six spargers. The cross sectional view and dimension of the IRWST is shown in Fig. 6 with the sparger location in the IRWST. The spargers shall be adequately submerged in the water to condense the discharged steam and properly located away from the IRWST wall in order to avoid high impact to the wall during the discharge. The normal water level of the IRWST is 3.7 m(12 ft) from the bottom, and with this level the water inventory is 2.54E(6) L(669,800 gallons). The minimum water level in the IRWST is designed to be no less than 1.75 m(5.75 ft), because the net positive suction head shall be maintained for the SIS and CSS pump operation.

There is approximately 1.2 m(4 ft) of freeboard space above the water surface in the IRWST. The freeboard space is to alleviate pressure in the containment and hydrogen buildup in the IRWST during DBAs and severe accidents as well as normal operation. During refueling operation, the containment low volume purge connections are used to remove hydrogen in the IRWST. For hydrogen control during DBAs, there are two connections to the hydrogen recombiner. Additionally, igniters will be installed in the IRWST for hydrogen control during severe accidents. Since the IRWST could be overpressurized when the RCS inventory is

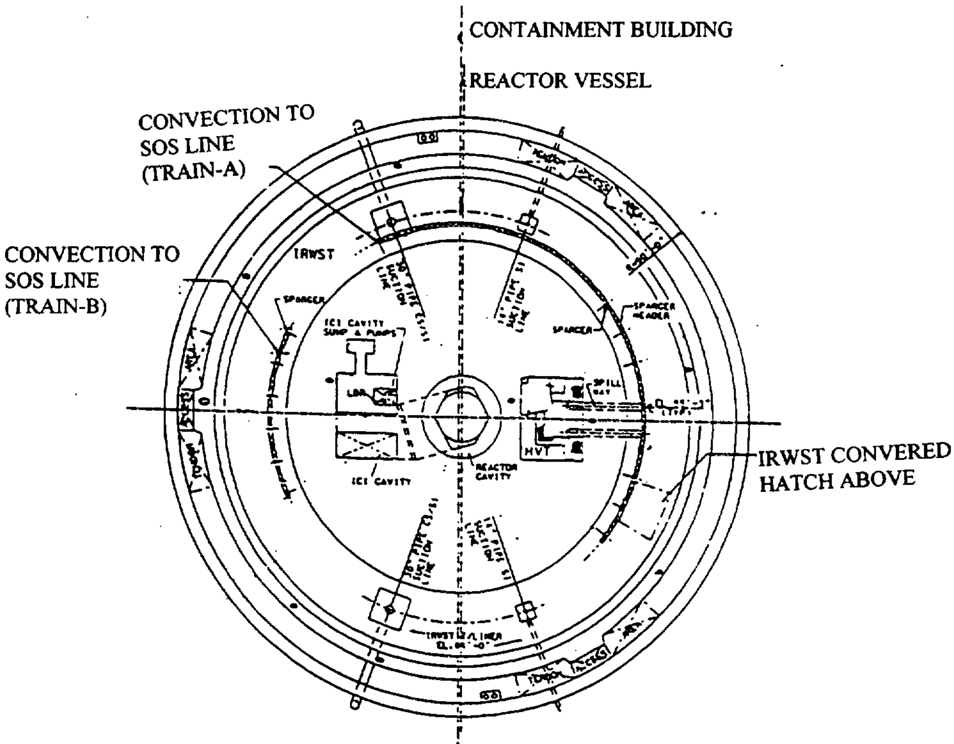


**Figure 5** A cross-sectional view of IRWST with a sparger line in the horizontal direction

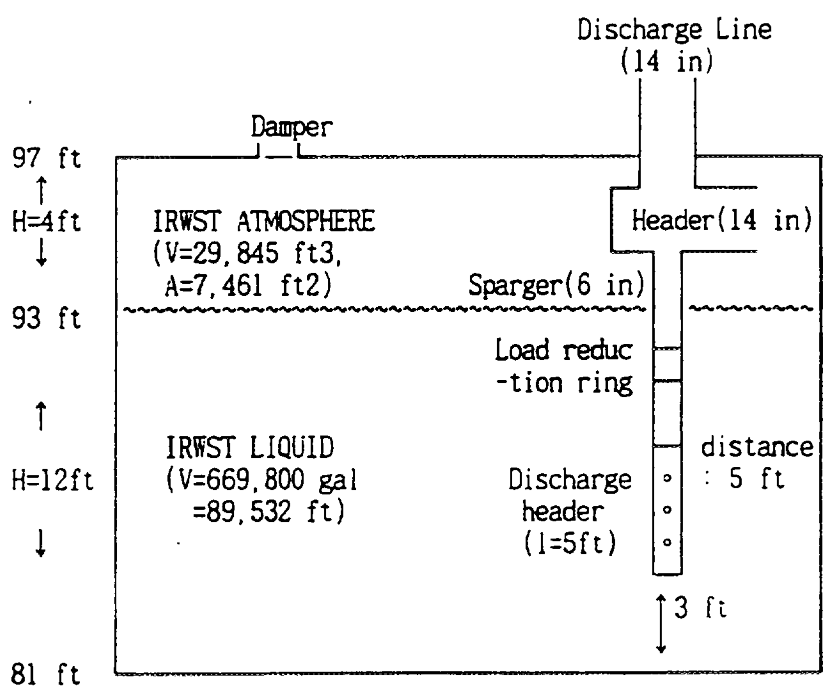


**Figure 6** A cross-sectional view of IRWST with a sparger line in the vertical direction

discharged into it, four relief dampers are installed at the IRWST ceiling to relieve steam and gases built in the IRWST. Also, hydrogen accumulated in the IRWST can exhaust through these dampers.

Due to the IRWST, the re-alignment of the SIS and CSS suction line to the containment sump is no more necessary for accident management. The water escaped from the RCS or sprayed by the CSS is collected on the basement floor slab and routed to the holdup volume tank. The water accumulates in the holdup volume tank before it begins to spill back into the IRWST through spillways in the tank wall.

## 4. R&D PROGRAMS FOR SEVERE ACCIDENT ANALYSIS AND SYSTEM DESIGN

In this section, some of the R&D works associated with the KNGR development for severe accident mitigation will be briefly outlined.

### 4.1 Conceptual study of an in-core catcher for the reactor pressure vessel protection

This R&D work has interests in the prevention of the reactor vessel breach by cooling the corium inside the reactor vessel. In severe accidents, the core may melt and relocate down to the vessel lower head. In this scenario, direct contact with molten core will heat up and deform the reactor vessel lower head, resulting in the rupture. The structure of the in-core catcher creates an engineered gap which will prevent the molten core from the direct contact with the inner surface of the vessel. Therefore, it is anticipated that the in-core catcher could firstly prevent rapid heating of the reactor vessel lower head and secondly help secure water cooling through the gap and hence prevent the failure of the reactor vessel lower head. The objective of this study is a conceptual design of the in-core catcher which is suitable to create the engineered gap under the core melt conditions.

### 4.2 Development of analysis method for core debris cooling

This R&D project consists of following subjects: 1. development of a heat transfer model for the core debris cooling in the cavity, and 2. experimental investigations of heat transfer mechanism of the reactor internal gap cooling and reactor vessel external flooding.

For the first subject, an analytical model is being developed to simulate heat transfer mechanism between core debris and cooling water dealing core debris like a porous medium. The experiments for the second subject focus on the critical heat fluxes in the two different situations. One is the critical heat flux attainable for a gap which is stipulated to form between the reactor vessel interior and molten core crust. The other is the critical heat flux on the surface of the reactor vessel in case that the reactor vessel exterior is cooled by flooding the reactor vessel.

### 4.3 Development of a 3-D mechanistic model for hydrogen mixing and deflagrations.

During severe accidents, a large amount of hydrogen generates by the oxidation of cladding material and Molten Core-Concrete Interaction. Therefore, it is important to control hydrogen concentration below a detonable level unless otherwise hydrogen detonation might occur and result in a containment failure.

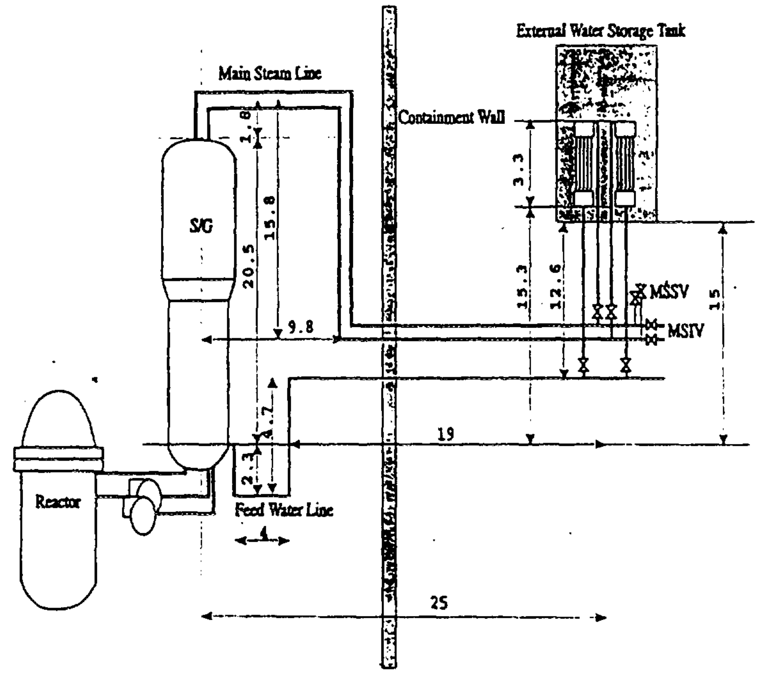


**Figure** 7 Schematic of passive secondary cooling system(unit in meter)

This R&D project develops a mechanistic model which can simulate the hydrogen mixing and distribution in the containment, and performs experiments to validate the model. The model will be implemented in the MAAP code if successfully validated. Also, appropriate energy levels for hydrogen ignition are experimentally investigated. The hydrogen ignition energy level depends on the igniter types and composition of hydrogen and steam in the air. If energy level is too high, it might cause a detonation. If it is too low, ignition may fail. Therefore, the ignition energy level is important for the hydrogen igniter system design.

4.4 Conceptual Development of Passive Secondary Cooling System(PSCS)

The PSCS consists of a water pool and heat exchanger located outside of the containment. This system is to back up the Emergency Feedwater System(EFWS). When the EFWS fails, the steam from the steam generator bypasses into the heat exchanger of the PSCS and is condensed by the cold water in the water pool. This condensed water is returned by gravity into the steam generator through the main feed water line. A schematic of the PSCS in conjunction with the S/G is shown in Fig. 7.

The scope of the conceptual development includes a system performance analysis using a system simulation code and large scale experiments, and separate small scale experiments focusing on the heat exchanger design. The main functional purpose of PSCS is to cope with total-loss-of-feedwater accidents and steam generator tube rupture events. However, we are examining a possibility to extend its function especially for severe accident mitigation, since it could be available for the water source to cool the containment.

5. Concluding Remarks

The KNGR design is currently in the second phase of which objective is to produce design details sufficient to confirm its safety. According to the current design schedule, the analysis and

**Table 3** Preliminary PSA Level 1 results of KNGR and comparison with YGN 3 PSA results.

| Initiator | KNGR(a) | YGN 3(b) | Percent Reduction (%) \|b-a\|/b*100 |
|---|---|---|---|
| LOCA | 1.35E-06 | 2.68E-06 | 49.6 |
| SGTR | 2.33E-07 | 7.05E-07 | 66.9 |
| ISLOCA | 3.01E-09 | 3.39E-08 | 91.1 |
| LOFW | 4.82E-07 | 1.25E-06 | 61.4 |
| Loss of Electric Power including SBO | 4.46E-08 | 2.01E-06 | 97.8 |
| LSSB | 2.09E-09 | 1.31E-07 | 98.4 |
| ATWS | 5.15E-08 | 3.87E-07 | 86.7 |
| Other Transient | 3.84E-07 | 6.23E-07 | 38.4 |
| Total | 2.55E-06 | 7.82E-06 | 67.4 |

system design for severe accidents will be completed by early 1999 when the standard safety analysis report is submitted to the regulatory authority for review. After that, more detailed analysis will be performed as necessary.

As mentioned in the introduction, the system improvements for severe accidents are related with the safety goal. The system improvements for severe accident prevention and mitigation are directly related with the core damage and large radiation release frequency goals respectively. According to the preliminary PSA Level 1 results[4], the CDF of KNGR due to internal events is about 2.55E-6/RY. Table 3 shows a comparison of the contribution of initiating events on the CDF between KNGR and Younggawang 3(YGN 3) which is a current standard design in Korea. As noticed in Table 3, contributions on the CDF from LOOP and LOFW events which were significant in YGN 3 were greatly reduced due to the use of AAC and dedicated EFWS as well as feed-and-bleed cooling capability in KNGR. For the CDF due to LOCA was also decreased remarkably, because the SIS reliability was greatly enhanced. The CDF by other events like SGTR, ISLOCA, LSSB, and ATWS was also significantly reduced due to the design improvements listed in Table 1. As the PSA level 2 analysis progresses, the effectiveness of the mitigation systems will be evaluated and accordingly accident management guideline will be developed.

In parallel with the design works, the R&D projects described in Section 4 will be reviewed for practical application to the design. If it turns out to be desirable and feasible to incorporate such features from both technical and economical point of views, their introduction will be considered in the detailed design phase.

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# SWR-1000 CONCEPT ON CONTROL OF SEVERE ACCIDENTS

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**Abstract**

It is essential for the SWR-1000 probabilistic safety concept to consider the results from experiments and reliability system failure within the probabilistic safety analyses for passive systems. Active and passive safety features together reduce the probability of the occurrence of beyond design basis accidents. Mitigative measures are incorporated into the SWR-100 design to cope with core melt accidents in order to limit their consequences in accordance with the German law. As a reference case we analyzed the most probable core melt accident sequence with a very conservative assumption. An initial event, stuck open of safety and relief valves without the probability of active and passive feeding systems of the pressure vessel, was considered. Other sequences of the loss of coolant accidents lead to lower probability.

## 1. Introduction

According to the German atomic law NPP's only will get an operating license if precaution measures have been incorporated in the design which practically prevent the occurrence of severe events. If hypothetically a core melt accident is being considered, the consequences to the environment do not lead to evacuation or relocation of the population living in the plant vicinity. The structures of the plant have to withstand the impact of the core melt accident without to permit release of radioactivity to the environment.

## 2. Frequency of core melt accidents

The safety concept of the SWR-1 000 fully complies with the specified requirements. Active and passive safety features together reduce the probability of the occurrence of beyond design basis accidents. On the other hand mitigate measures control the core melt to the highest extent possible to prevent consequences according to German laws. Preliminary probability concept analyses due to internal events during operation leads to core damage frequencies, which are much smaller as from external events. The plant can not be designed against extreme earthquake forces or other cosmic events leading to disintegration of the reactor building. As a very conservative assumption the most probable core melt accident sequence is being selected as a reference case in fulfilling the requirements of the 7th atomic law (deterministic assumption). For the SWR-1000 as initial event stuck open of safety and relief valves are determined without the possibility of active and passive feeding of the reactor pressure vessel. Other sequences of loss of coolant accidents leads to lower probability figures.

Within the probability analyses for passive systems the results from experiments and feasible system failures have to be considered.

*- Concept to control the core melt*

To control the core melt in the SWR-1000 design the following targets have to be considered:

- Retention of the core melt in the RPV by cooling the RPV from outside.
- Inertization of the containment with $N_2$ to prevent $H_2$-explosion or deflagration.
- Consideration of the $H_2$-content generated by 100% Zirconium-water reaction for the containment design.
- Passive release of heat from the containment
- Pressure reduction in the containment within a certain grace period of several days via the off-gas-system

In the following chapters the various countermeasures to control the various events are described.

*- Prevention of core melt under high pressure in the reactor pressure vessel*

The high pressure core melt can be prevented in the SWR-1000 design by the multiredundant and diverse features for pressure reduction. Compared with the RSK-recommendations for the EPR pressure release features could be defined as reliable if they work such as a safety valve for pressure limitation.

In all BWR-plants as a matter of principle reaching of a very low water level in the RPV or during pressure increase in the containment an automatic pressure relief function is being initiated by the reactor protection system enabling low pressure feeding of the RPV. The pressure limitation and reduction features of the SWR 1000 compared to current systems have been extended (see fig 1).

8 safety and relief valves with redundant and diverse pre-control valves are incorporated in the design.

For the function pressure limitation for each of the valves passives spring loaded pre-control valves and for each valve a magnetic pre-control valve are actuated by the I and C-system.

For the function pressure relief for each valve passive pulse generator operated membrane control valves are used and magnetic pre-control valves are actuated by the I and C-system.

During initiation of the pressure relief function the main valves remain open (mechanically blocked) as a back-up solution rupture discs are in stalled which start their function if the design pressure is reached. In case of a failure in the I and C-control system or loss of electricity from batteries the complete functions for pressure reduction are granted by that passive safety features. During loss of the pressure relief functions but well working operating pressure limitation the 4 emergency condensers take over the pressure reduction function. The capacity of the system enables reduction of the reactor pressure down to the operating level of the active decay heat removal systems (<10 bar). Core melt under high pressure is excluded by these means.

*- Prevention of steam explosion in the RPV*

Is the active and passive feeding being lost under low pressure the core starts melting. With progress of the core melt process the material flows into the lower plenum of the RPV, filled with water and the possibility of a steam explosion has to be considered.

A heavy steam explosion leading to a disintegration of the RPV needs an intensive mixture of core melt material with water. Following the literature the general probability of a heavy steam explosion is considered rather low and depends from the mode of the core melt process: Various possibilities how the core melt could get into contact with the water for initiating a steam explosion have been analyzed.

The core melt will solidify on the surfaces of the control rod guide tubes and the control rod drives or other structural materials. Countermeasures to prevent steam explosion are not necessary.

*- Retention of a core melt in the RPV; outside cooling.*

To prevent a disintegration of the RPV as a consequence of the accumulation of the molten material in the lower plenum of the RPV, an outside RPV-cooling system grants the integrity of the RPV walls and thus the release of the melt into the containment.

The cavity housing the RPV (pressure chamber) is flooded with water. Via respective piping systems the water is transferred from the flooding pool into the pressure chamber. The water level in the pressure chamber is being controlled on the same level as in the flooding pool which is adjusted to the level on which the emergency condenser is working. The details of that design are still under investigation. A passive initiation via pulse generators or melting barriers and manual interference may be feasible. In any case a malfunction during operation is prevented.

Heat transfer from the core melt collected in the lower plenum of the RPV to an outside located cooling system without exceeding the critical heat surface loads, i.e. without film boiling is possible.

The water inlet into the cooling features between reactor pressure vessel wall and insulation needs a space of 1 mm. Only for the central control rod drive tube nozzle a 10 mm space is necessary. 8 penetrations of 300 mm Ø in the RPV-support grants for an undisturbed release of the water-steam mixture. In the upper area the steam leaves the pressure chamber via the pipe penetrations, which carefully have to be designed for this purpose (see fig 2).

Investigations with the FE-code Adina-F concerning behavior of the core melt in the lower plenum leads to the expectation of rather small impact to the RPV-walls. Figure 3 shows the core melt in the lower *plenum of* the RPV with oxidized surface layer. In the area between metallic and oxidized melt the highest heat transfer transaction into the lower part of the RPV is predicted, which leads to rather small melting of the lower plenum RPV material. The pressure load capability of the RPV is not being reduced(see Fig. 4). Melting of the control rod nozzle or pump nozzle due to the outside cooling is not expected. The cooling capability of the control rod flushing water and the seal water for the main pump nozzles is not considered.

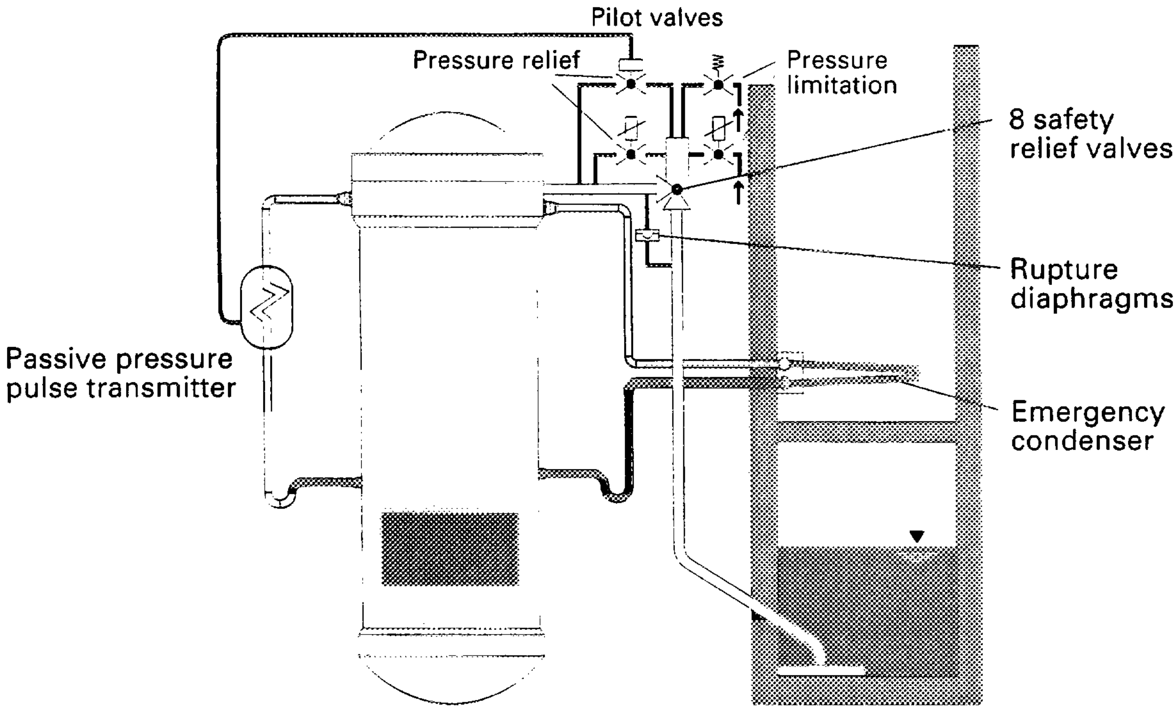


*FIG. 1. Pressure relief system for reactor pressure vessel.*

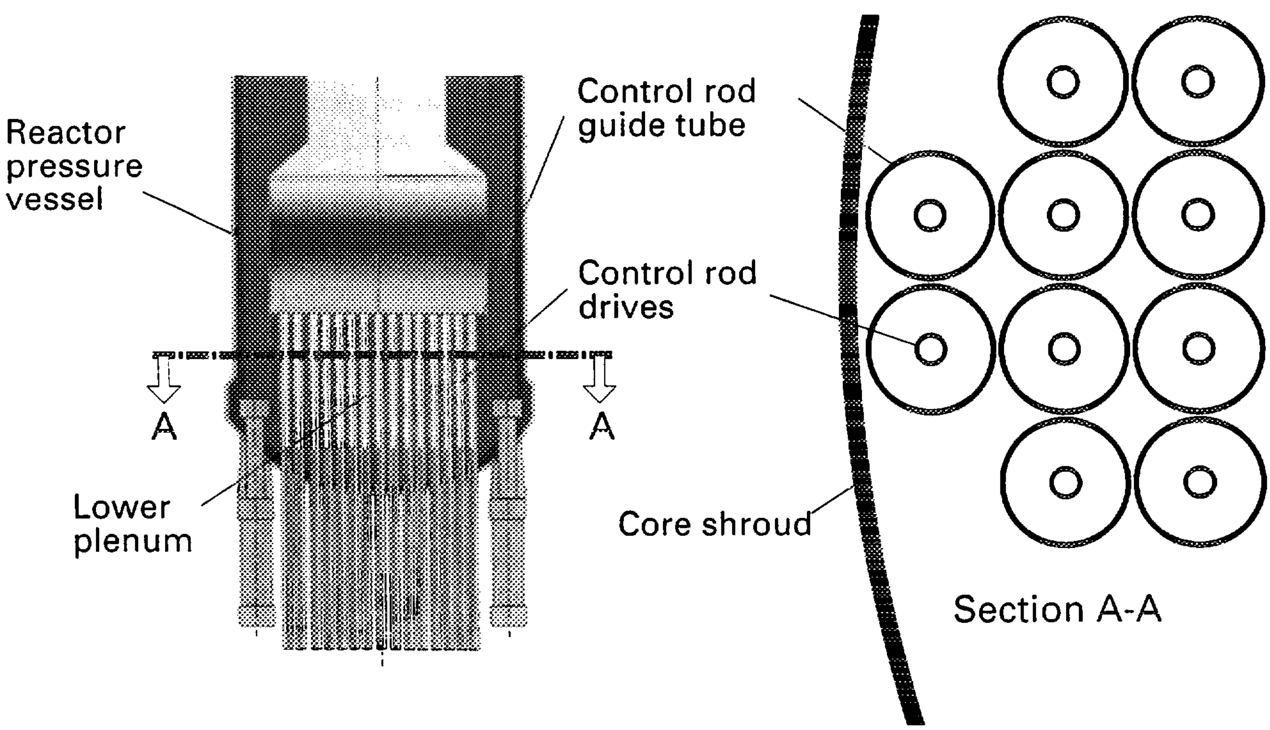


*FIG. 2. Core components in lower RPV plenum.*

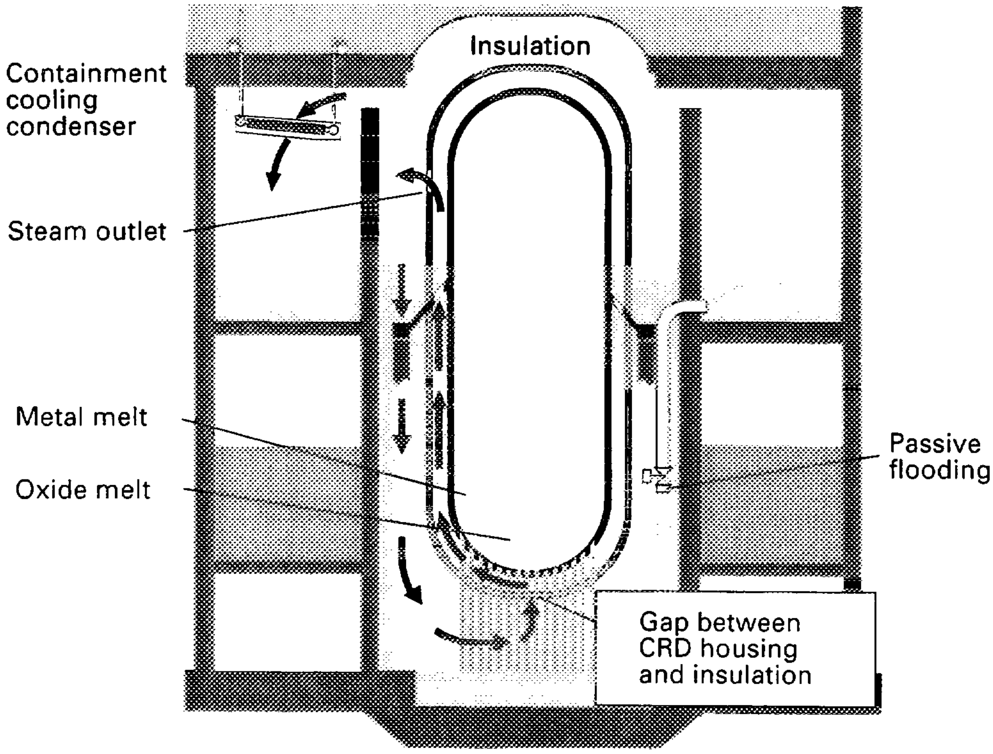


*FIG. 3. External cooling of RPV during core meltdown.*

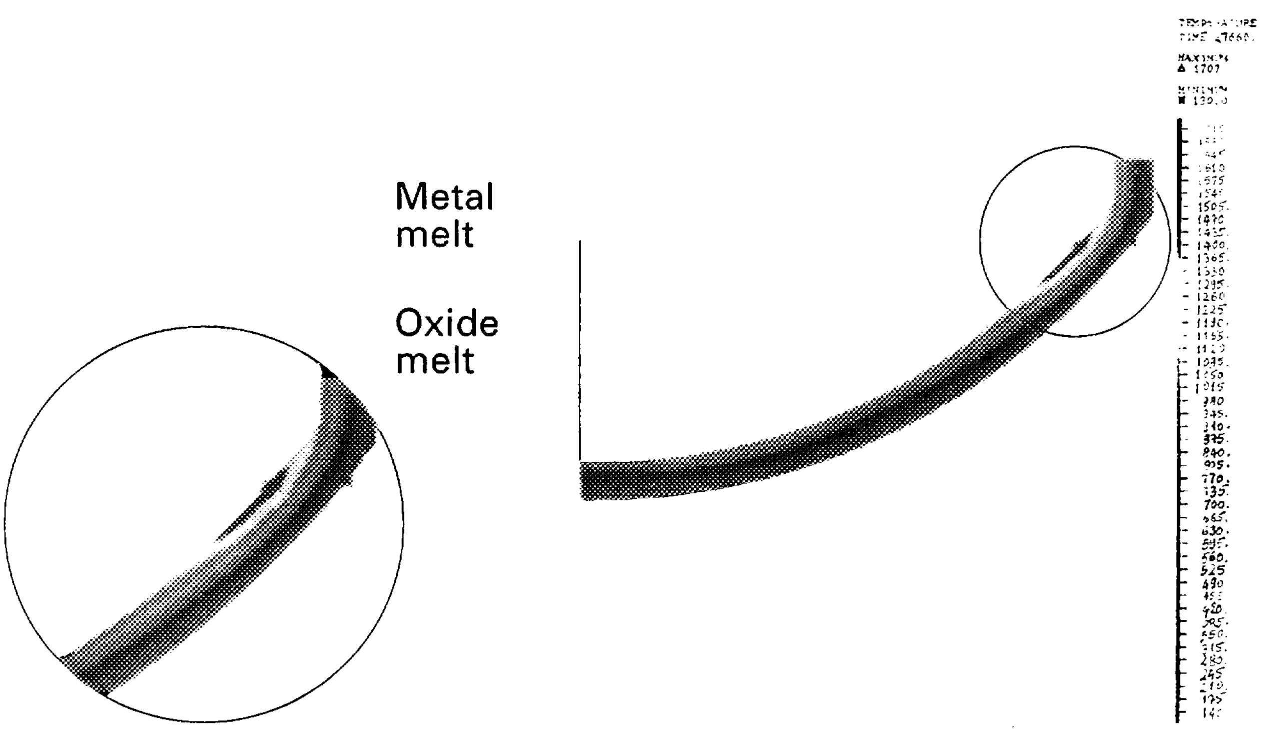


*FIG. 4. Temperature of RPV bottom head during core meltdown.*

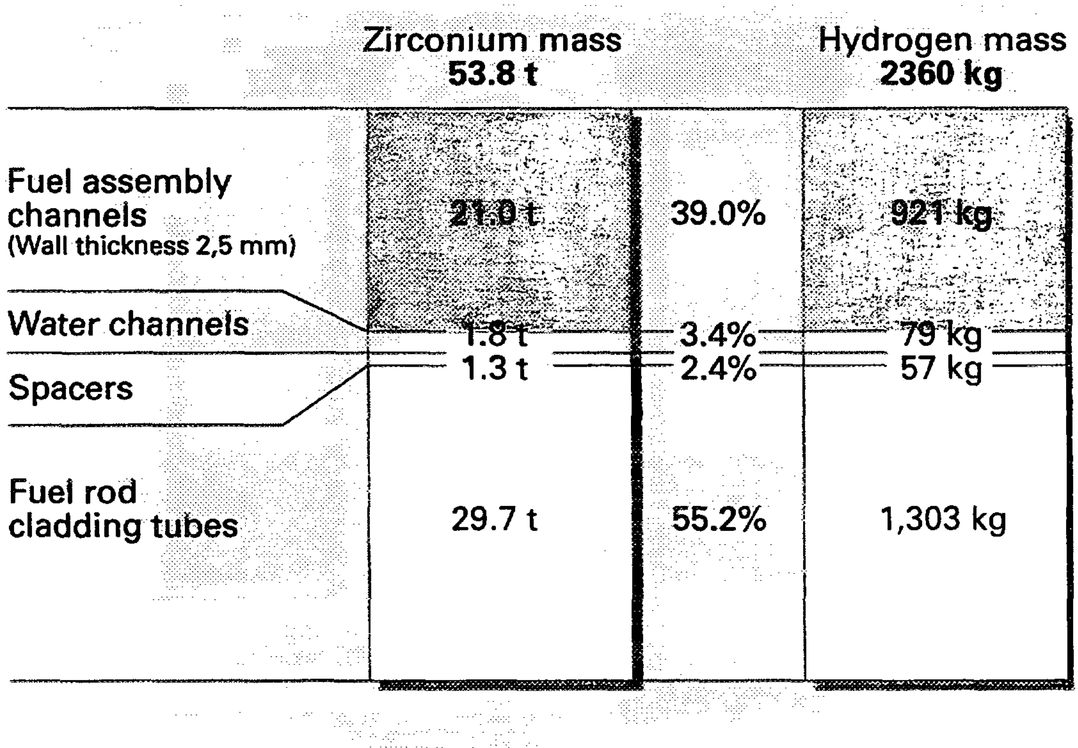


*FIG. 5. Hydrogen produced by zirconium-water reactor in reactor core.*

A release of the core melt from the RPV into the containment can reliably be prevented. Thus steam explosion or an interaction between concrete and the melt material is excluded.

- *Recriticality*

Recriticality after severe accidents only has to be considered if the geometry and composition of fission material, moderator and absorber are as favorable as before the disintegration of the core. This configuration is only feasible within a homogeneous fine fragmentation of the core material without significant parts of absorber material, but with an optimal moderator distribution inside the fragmented material.

For the BWR general a recriticality is highly improbable because of the unfavorable neutron economic structure of the core melt compared with the original core structure. The core may start melting from the lower core grid plate and the melt will drop or fragmented into the bottom part of the vessel. The early melting absorber material $B_4$ C-Fo from the control rods is located in the melting zone thus poisoning the fragmented material and the melt droplets. This kind of poisoned no homogeneous material remains uncritical. For the SWR-1000 recriticality of core melt material is excluded.

- *$H_2$-Production, Prevention of $H_2$-ignition*

In the core melt the Zirconium reacts with water and steam whereby $H_2$ is being generated. For the containment design pressure 100 % water/Zirconium reaction is being considered. A small quantity of $H_2$ is generated in the radiolyse-gas-production process. The

100 % $H_2$ /Zry reaction is a very conservative assumption, also other metal-water reactions are already covered by this assumption. With the core mass of 53,8 t (fuel cladding tubes, water channels; spacers and fuel channels) 2360 kg $H_2$ will be generated (see fig. 5). Core melt calculations usually consider a < 40 % Zry/Water-reaction.

To prevent a reaction of the $H_2$ with $O_2$ the containment will be inerted. Detonation or deflagration are prevented. To prevent an impermissible increase of $O_2$-concentration by the radiolytic process a number of recombiners will be installed (the number will be decided later).

*- Heat transfer from the containment*

For heat transfer out of the containment two active residual heat removal systems and four emergency condensers will be available. A complete passive heat transfer from the containment after 100 % failure of the active residual heat removal systems is possible via the water volume stored in the flooding pool. The steam generated from the outside cooling of the RPV will be condensed in the building condenser, the condensate is collected in the flooding pool. The condensation heat is transferred to the pool water, which is cooled via the containment condenser by natural circulation.

The $H_2$-quantity in the reference case could be stored in the condensation chamber thus the steam condensation in the buildings condenser is not being influenced. Assuming larger quantities of $H_2$ parts of the gas will be released into the pressure chamber. The containment condensers are designed for a inert gas mixture and are located at a level allowing collection of $H_2$. Deterioration of the heat transfer conditions leads to a pressure increase in the pressure chamber and flushing of $H_2$ in the condensation chamber. By these measures the passive heat transfer also in case of $H_2$ release is granted.

Interference of operating staff will only be necessary after evaporation of the 2200 $m^3$ water stored in the flooding pool. These conditions may be reached after 3 days after the accident. Reflooding of the pool easily can be performed by the fire brigade.

*- Design pressure of the containment*

The main parameters for which the containment has to be designed are the released heat from the RPV and the generation of steam by means of the decay heat which leads to a temperature and pressure increase in the condensation chamber and the flooding pool. Additional pressure increase following a core melt accident is the Zry/$H_2O$-Reaction, an exothermic reaction. Further increase of temperature and pressure has not to be assumed due to the inertization.

In the design of the containment the various inputs are considered. Realistic conditions of the gas distribution in the containment, the heat capability of the building structures and the 100 % Zry/$H_2$ 0 reaction leads to a design pressure of 7,5 bar. The pressure can be attained by using the passive safety systems and will decrease during the long term mode operation.

*- Pressure reduction in the containment*

With respect of the $H_2$ production during a core melt accident the pressure would not be reduced only by cooling. Specific features are necessary for pressure reduction in the

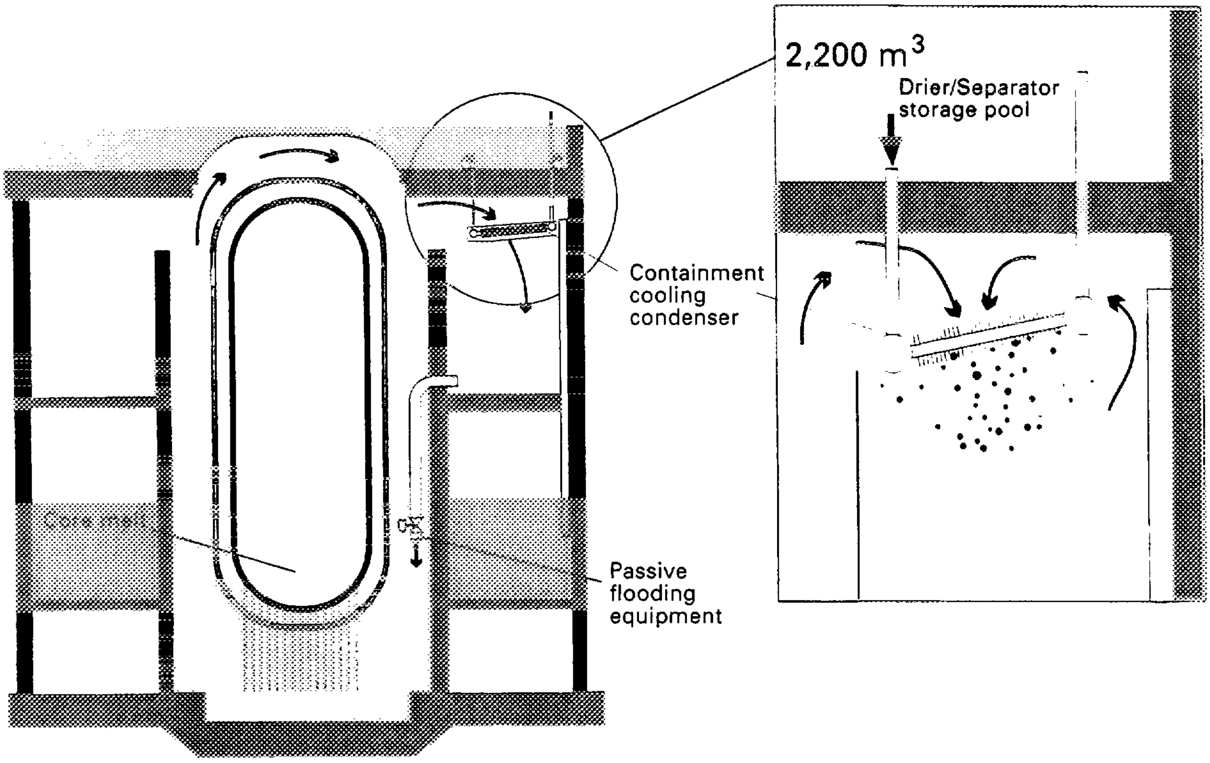


*FIG. 6. Passive heat removal from containment during core meltdown.*

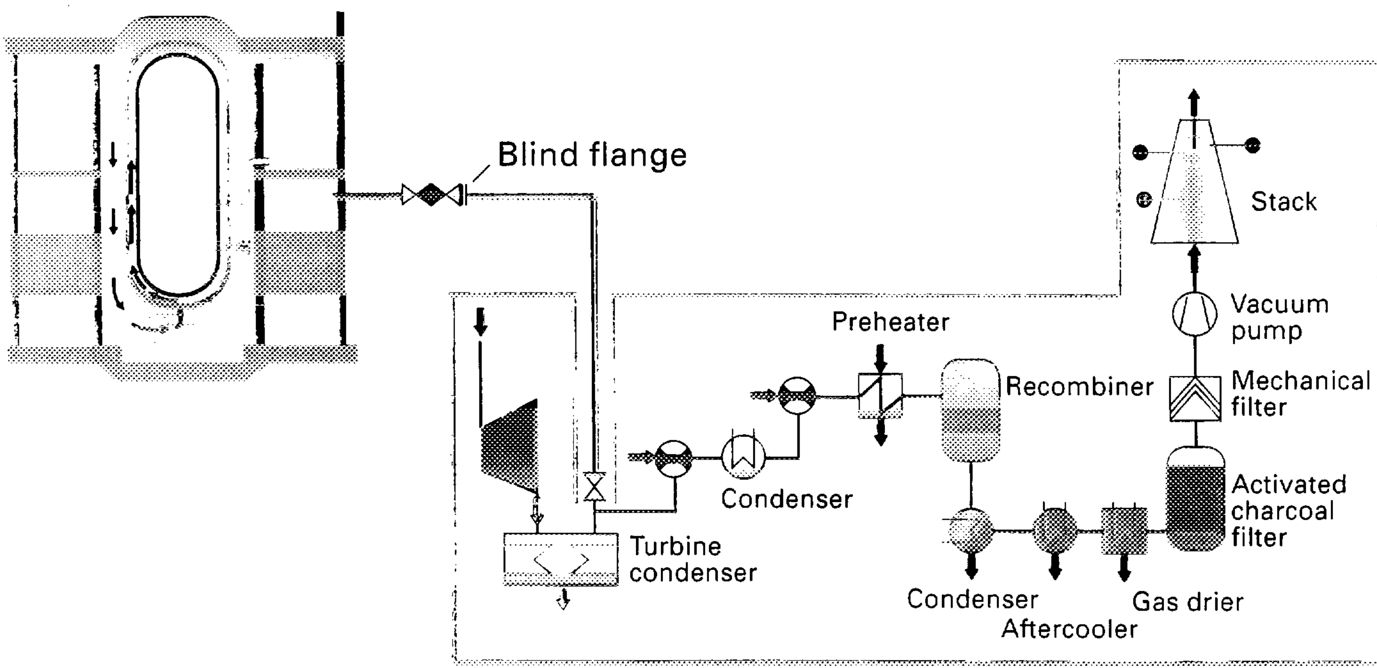


*FIG. 7. Reducing the pressure in the containment following a severe accident.*

containment (see fig. 6). For this purpose the off-gas system with recombiners and activated charcoal filters are being used. A small pipe connection between containment and off-gas-system is foreseen. This pipe is connected with the gas plenum of the condensation chamber and isolated by two valves and a blind flange. The pipe can be opened if needed. Aerosols are removed from the gas mixture by means of the washing process in the condensation chamber (99 %). The $H_2$ together with $0_2$ will be recombined in the recombiners to $H_2$ 0. The dry inert gases (noble gas) will be send to the activated charcoal filter for radioactive decays. The decay time for Krypton is approx. 3,6 days and for Xenon 60 days. After the pressure reduction phase which lasts approx. 40 h the offgas-system will be isolated for decay of the noble gases (see fig. 7).

## 3. Summary

The safety concept for the SWR-1 000 consisting of diversified passive and active safety features limits the probability of occurrence of sever condition to rather low figures due to internal events. The concept of control of postulated core melt is based on the retention of the melt in the RPV by external vessel cooling. For this purpose a passive flooding system is foreseen.

To prevent a core melt under high pressure in the RPV redundant and diverse safety features are installed for pressure control.

A steam explosion can be excluded during release of the melt and collection in the water filled lower plenum of the RPV due to the existing structures.

$H_2$-fire in the form of deflagration or explosion in the containment is prevented by inertization. A steam explosion in the containment or an interaction between concrete and core melt with all consequences is excluded.

The heat can be transferred out of the containment via the passive building condenser.

The pressure can be reduced via the offgas-system and the activated charcoal decay beds without to permit release of radioactivity into the environment.

This concept fully complies with new German law from 1994.

# DESIGN CONSIDERATION ON SEVERE ACCIDENT FOR FUTURE LWR

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## Abstract

Utilities' Severe Accident Management strategies, selected based on Individual Plant Examination, are in the process of implementation for each operating plant. Activities for the next generation LWR design are going on by Utilities, NSSS vendors and Research Institutes. The proposed new designs vary from evolutionary design to revolutionary design like supercritical LWR. Discussion on the consideration of Severe Accident in the design of next generation LWR is being held to establish Industry's self-regulatory document on containment design and its performance which ABWR-IER (Improved Evolutionary Reactor) on the part of BWR and Evolutionary APWR and New PWR21 on the part of PWR are expected to comply. Conceptual design study for ABWR-IER will illustrates an example of design approach for the prevention and mitigation of Severe Accident and its impact on capital cost.

### 1. Severe Accident Management

NSC(Nuclear Safety Commission)of Japan issued a statement on accident management (AM) in May 1992 to urge the nuclear Utilities to prepare accident management as a self -regulatory activity. Ministry of International Trade and Industry followed by issuance of a generic letter to ask the utilities to submit plant specific Probabilistic Safety Analysis (IPE as is called in the US.) and Accident Management strategies by the end of FY1993.

The regulatory position on severe accident issues is described in the NSC statement as follows; 'Adequate level of safety for the nuclear facilities has been kept through strict regulations based on the philosophy of defense in depth at the stages of design, construction and operation. The likelihood of the occurrence of severe accident is so low in probability that from the engineering point of view it is remote from reality, and the risk associated with the occurrence of severe accident is small. Accident management , if implemented will further contribute to reduce the risk arising from operation of nuclear power plants. Thus, NSC strongly recommends licensees to voluntarily plan effective accident management.'

Upon the recommendations, Utilities submitted IPEs, Accident Management strategies and implementation plan for all 51 LWRs in operation and construction to the regulatory body (MITI) in March 1994. The reports were reviewed by both MITI and NSC. MITI issued its evaluation report in October 1994, so did NSC in November 1995, both of which confirmed that the accident management strategies reported by the utilities were technically appropriate. Utility's Accident Management program consists of the following elements *(1)*;

- AM Procedures
- Plant modifications
- Check capability of instrumentations
- Training
- Organization

Plant modifications for Accident Management include the use of non-safety grade equipments in order to fully utilize existing plant capabilities to cope with beyond DBA situations, but considers system interface with safety system and reasonable assurance of function capability under Severe Accident condition. The proposed basic strategies are as follows (see Fig 1,2);

Generic BWR AM strategies ----- Prevention

1) Use all available makeup water driver & source to deliver water to RPV from non-safety water tanks via diesel-driven fire water pumps
2) Emergency power source connectivity to cope with Station Blackout for multi unit installation
3) Scrubbing vent to arrest TW scenario
4) Depressurization on water level signal and Alternate Rod-run-in logic to reduce ATWS / TQUX by system design

Generic BWR AM strategies ----- Mitigation

1) Ex-vessel debris cooling measures against , overheating, shell Attack , core-concrete reaction (Capability to deliver Fire-water & others to RPV/Containment/Cavity)
2) Avoid overpressure failure of containment & to allow time for containment heat removal function recovery by scrubbing vent
3) Inert entire containment & reinert after vent

Generic PWR AM strategies ----- Prevention

1) Use all available makeup water driver & source to deliver water to reactor from makeup water tank through RWSTand Alternative recirculation
2) Secondary loop cooling( Main feed water pump to cool primary loop (for ATWS, Turbine bypass system to depressurize primary loop)
3) Primary loop depressurization using relief valves in pressurizer and intact SG's to suppress coolant leakage and to utilize RHR.
4) Natural convection cooling of containment by containment cooling chiller
5) Emergency power source connectivity to cope with Station Blackout for multi unit installation
6) Alternative auxiliary component cooling

Generic PWR AM strategies ----- Mitigation

1) Firewater into the containment for debris cooling ,over-pressure protection
2) Forced depressurization of primary system
3) Igniter for Ice-condenser containment

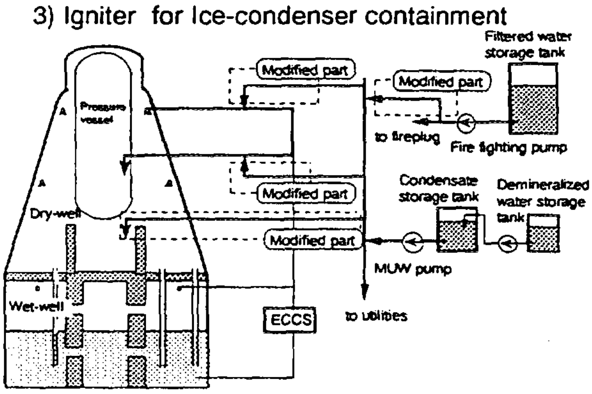


Fig. 1 Alternate water injection

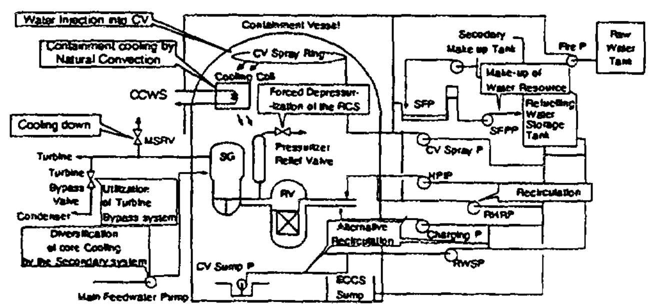


Fig.2 Typical AM Strategies and Facilities in PWR (Example)

For all LWRs, these AM strategies will be implemented by around the turn of the century. Each utility will conduct plant modifications while preparing infrastructures such as operating procedures including accident management guideline and training.

It is generally expected that future LWRs will have less complex Accident Management but nevertheless they are not free of such measures since Accident Management will provide flexibility to cope with unexpected situations.

## 2. ALWR Development

Various reactor concepts are proposed in Japan for future LWR. They include evolutionof ABWR, J-SBWR , RBWR(LWR breeder) on the part of BWR, andAPWR & its evolution and SPWR on the part of PWR. Supercritical LWR is a simple direct cycle LWR with elevated pressure and high thermal efficiency. This paper will focus on ABWR and its evolution as an example.

ABWR (Advanced Boiling Water Reactor)

Twin FOAK ABWRs were constructed by TEPCO. SECY-90-016 & 93-087 had been major elements for resolution in US of the capability to cope with severe accident. Japanese ABWR is essentially the same except that the fusible valve to supply water to the lower drywell is replaced by remotely operated valved to supply water from firewater tanks and that Gas turbine is not required for multiple unit installation with reliable power network. During the ABWR licensing process in US, it was analysed by GE that the critical structures would survive an ex-vessel steam explosion and that the containment can withstand pressurization by hydrogen if service level C is assumed in determining allowable stress.

ABWR-IER (Improved Evolutionary Reactor)

ABWR-IER is an evolution of ABWR and is intended to base its design on experiences of BWR operations and improvements while exploiting new and innovative design features available as technologies progress.

Pre-Phase(FY 1990 ) was intended to establish TEPCo's Utility Requirement *(2)* , which was followed by Phase I ( FY 1991-92)by BWR Utilities and NSSS vendors to investigate future technologies to apply to IER.

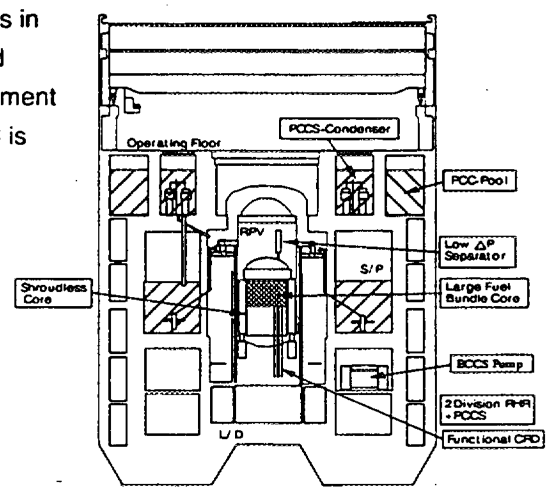


Fig 3 : ABWR-IER Concept

Phase II ( FY 1993-95) was intended to establish a reference plant concept *(3)* with focus upon nuclear boiler and engineered safeguard systems (Fig-3). Simple economic and safety evaluations were done to confirm the compliance to the Utilities requirements. Phase III( FY 1996-98) is intended to establish entire plant concept with emphasis on economic improvement to assure competitive edge over alternative power generating sources. T & D Program is also to be planned in this stage although some selected test programs are already going on such as effect of increased number of fuel rods in a bundle on heat transfer performance.

ABWR-IER design is based on ABWR, but has several improved safety features as follows.

1) Active & passive containment heat removal systems

In addition to traditional RHR systems to serve to remove heat from Reactor pressure boundary as well as from the containment, PCCS (Passive Containment Cooling System) as is developed for SBWR is being considered for ABWR-IER to provide additional margin to remove heat from the containment by passive means.

2) Diversity in ECCS (Fig-4)

ECCS network of IER adopts the same type of three independent division system (HP & LP in each division) as ABWR, increase in diversity in working principle is considered. ARCIC (Advanced Reactor Core Isolation System utilizing steam produced by decay heat ) now is equipped with its own small generator which replenish batteries with DC power for instrumentation and control in the event of station blackout. Motor-driven HPCF (High Pressure Core Flooder) and diesel-driven HDIS (High Pressure Diesel Driven Injection System) provide additional diversity.

3) Diversity in Emergency Power Supply (Fig-4)

The diversity of emergency power supply systems increases by allocating a diesel generator, an air-cooled diesel generator and a gas turbine generator for each safety division. This feature provides improved performance for station blackout sequences which probability would be dominated by common cause failure of some sort in conventional design.

4) Simplicity

Reducing complexity by reducing supporting systems such as component cooling system is expected to help simplify plant design.

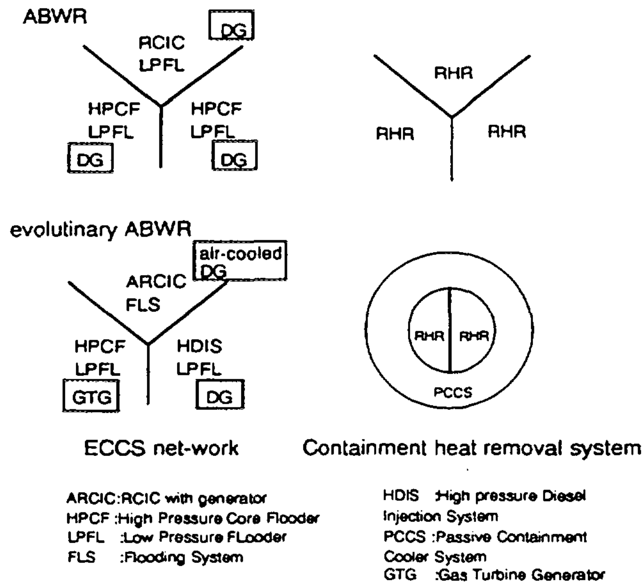


Fig 4 : ABWR & ABWR-IER ECCS/RHR

As for the mitigation of severe accident, such features as refractory material liner at the bottom part of lower dry-well, passive lower dry-well flooder, hydrogen absorber/adsorber are being discussed. In-vessel retention capability, if proved, will certainly add safety margin but will not totally eliminate measures to mitigate the consequences of severe accident.

**3. Containment for future ALWRs**

One aspect of design consideration for severe accident for future LWR will be severe accident mitigation especially in terms of containment design, although enhancement in prevention as discussed above for ABWR-IER will be visible in all future LWR design.

Japanese nuclear industry, with some advises from academia, is developing containment design document which will consist of performance target and design extension conditions so that appropriate design considerations on severe accident and other important issues are taken in the design of future containment In Japan. Design extension conditions are intended to provide additional margin to cope with severe accident although containment design basis itself is unchanged.

Containment performance target will probably include such targets as large release limitation, CCFP but also due consideration of identified threats to the containment as Design Extension Conditions.

The technical issues under discussion or to be discussed include;

• Hydrogen as pressurizing source for small containment
• In-vessel retention as additional margin, but not eliminate mitigation
• Debris cooling in reactor cavity
• Containment leakage characteristics
• External events

To raise some issues of concern;

1) Cut-off probability

The kind of scenario or phenomenon to be addressed in the containment design will certainly have limitations. One limitation arising from the imagination of designers. Another from balanced design approach, which essentially implies to what extent nuclear power should be prepared in the design for the highly unlikely events such as in-vessel steam explosion. Already certain cut-off probability has been utilized in the stage of screening of initiating events of external sources such as missile - the cut-off number hovers somewhere around 10(-7)/year.

Applying similar cut-off probability to the chain of events to eliminate certain scenario which may threaten the integrity of containment is a difficult task. Industry is discussing 10(-7)/year as such cut-off probability while requiring due consideration to every possible phenomenon in search for preventive and mitigative strategies.

2) Balanced approach

Given the decline of fossil price since mid 1980's and the deregulation of Utility business in US, Europe and Japan, the economic competition among alternative energy sources may push away nuclear power from future power market, if Utilities are much short-sighted without paying due consideration to nuclear power's importance in stable/secure energy supply and environment. *(4)*

For existing plants with advent in capital depreciation, efforts are made to preserve competitive edge by increasing availability while maintaining the same level of safety and reliability. For new installation, capital-intensive nuclear power may face difficulties if balanced approach and cost-effectiveness are not considered.

## 4. Conclusion

Severe Accident Management program in Japan is well under way to restrict risk arising from such highly unlikely events.Containment performance targets are being developed by Japanese industry as a self-regulatory guide with the aim of addressing severe accident issues in the design of future LWR.

Activities for the future LWR design is going on. ABWR-IER design will further reduce the likelihood of severe accident by diversity of systems & components to fulfill safety functions.

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XA9847568

# CONSIDERATION OF SEVERE ACCIDENTS IN DESIGN OF ADVANCED WWER REACTORS

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**Abstract**

Severe accident related requirements formulated in General Regulations for Nuclear Power Plant Safety (OPB-88), in Nuclear Safety Regulations for Nuclear Power Stations' Reactor Plants (PBYa RU AS-89) and in other NPP nuclear and radiation guides of the Russian Gosatomnadzor are analyzed. In accordance with these guides analyses of beyond design basis accidents should be performed in the reactor plant design. Categorization of beyond design basis accidents leading to severe accidents should be made on occurrence probability and severity of consequences. Engineered features and measures intended for severe accident management should be provided in reactor plant design. Requirements for severe accident analyses and for development of measures for severe accident management are determined.

Design philosophy and proposed engineered measures for mitigation of severe accidents and decrease of radiation releases are demonstrated using examples of large, VVER-1000 (V-392), and medium size VVER-640 (V-407) reactor plant designs. Mitigation of severe accidents and decrease of radiation releases are supposed to be conducted on basis of consistent realization of the defense in depth concept relating with application of a system of barriers on the path of spreading of ionizing radiation and radoactive materials to the environment and a set of engineered measures protecting these barriers and retaining their effectiveness.

Status of fulfilled by OKB Gidropress and other Russian organizations experimental and analytical investigations of severe accident phenomena supporting design decisions and severe accident management procedures is described. Status of the works on retention of core melt inside the VVER-640 reactor vessel is also characterized.

## 1. INTRODUCTION

Problems of severe accident consideration in large, VVER-1000 (V-392), and medium size VVER-640 (V-407) reactor plant designs are discussed in the paper. Descriptions of these plants are given in [1] and [2] where main features of design, equipment and main system characteristics of these plants are treated.

## 2. RUSSIAN NUCLEAR REGULATORY GUIDES REQUIREMENTS RELATING TO SEVERE ACCIDENTS

Requirements relating to beyond design basis accidents leading to severe accidents are treated in the Russian nuclear regulatory guides OPB-88, [3], PBYa RU AS-89, [4] and in Requirements on Contents of Safety Analysis Report for VVER Nuclear Plants, [5].
In accordance with documents, beyond design basis accidents are:

- accidents occurring due to initiator events not considered for design basis accidents,
- accidents accompanied by additional safety system failures as compared to single failure assumed for design basis accidents,
- accidents due to realization of erroneous personnel decisions leading to severe core damage or melting.

These documents prescribe to aim for the evaluated probability value of severe core damage or melting under beyond design basis accidents not to exceed E-5 1/reactor-year.

Moreover, for the sake of exclusion of public evacuation necessity outside the area specified by the regulatory requirements to NPPs siting, it is prescribed to aim for the evaluated probability value of occurrence of the largest specified by these requirements accidental release of radiactive materials not to exceed E-7 1/reactor-year.

If beyond design basis accident consequences analysis reveals that this requirement is not met, additional technical solutions should be provided in the design for management of beyond design basis accidents.

Mitigation of beyond design accident consequences is achieved by accident management and/or by realization of emergency measure programmes to protect population, personnel and environment.

Technical means and measures for beyond design basis accident management should be foreseen by the NPP design.

Any available in good working order means:

- designed for normal operation,
- designed for safety ensurance under design basis accidents

can be utilized to manage beyond desisgn basis accidents.
A special guide should be elaborated to manage beyond design basis accidents in accordance with the design documentation. A sequence to bring into realization of emergency measure programmes of protection of population and personnel in case of occurrence of a beyond design basis accident should be given in this guide.

Development of emergency measure programmes to protect population, personnel and environment and special instructions for personnel to manage these accidents should be realized on base of their analysis.

Analyses of beyond design basis accidents should be done in the NPP design. Moreover, conditions should be given in which fuel melting and/or exceeding of fuel rod destroying specific threshold energy are possible.

In accordance with [5], the list of beyond design basis accidents would be compiled on basis of review of all scenarii leading to exceeding normative personnel and population radiation exposures, radioactivity releases and/or radioactivity contents in the environment established for design basis accidents. Vulnerable features of NPP design, operational procedures and organizational structure of personnel activity that may become as the most probable causes of the conditions above would be determined on base of analysis of event trees.
Possible scenarii of severe accidents are split into groups in which plant process system operation required for accident mitigation is similar. Within each group representative scenarii are found in which four criteria as follows:
- maximum personnel and population radiation exposures
- maximum radioactivity release intensity
- maximum integral radioactivity release
- maximum damage extent of plant system and/or equipment

are fullfilled.

A categorization of beyond design basis accidents according to occurrence probability and to severity of consequences should be given in NPP design.

## 3. DESIGN DECISIONS FOR SEVERE ACCIDENT PREVENTION AND MANAGEMENT

In accordance with [3], [4], prevention of accidents and large releases outward reactor plant in designs of NPP with VVER-640 (V-407), [1], and VVER-1000 (V-392), [2] is provided due to a systematic realization of a defense in depth concept based on a system of barriers on the way of spread of ionizing radiation and radiactive materials to the environment and a system of technical and organizational measures protecting these barriers.

A set of defense levels are designed which provide protection of the plant and physical barriers from damage and preservation of the population and environment if some barriers would be damaged to some extent.

Passive heat removal systems for core residual heat removal during 24 hour plant blackout are available for provision of core integrity in design and some beyond design basis accidents. In LOCA accidents emergency core cooling system is flooding reactor from high and low pressure hydroaccumulators in succession. In the VVER-640 reactor plant an emergency pool connected by piping with refuelling water storage tank is created during this process.
Opening of the automatic depressurization valves secures core cooling through circulation circuit connecting refuelling water storage tank with reactor. Heat transfer to the atmosphere is realized with the containment passive heat removal system.

Substantiation of fulfillment of regulatory guides requirements in severe accidents can be divided into 4 work fields relating to provision of integrity of the following parts:

- reactor core
- reactor vessel
- reactor dry well and core catcher
- containment structure.

On this basis, development of the designs considered is conducted in direction of elaboration of devices aiming at:

- core damage prevention with aid of active and passive safety systems
- mitigation of corium-reactor vessel and corium-concrete well interaction
- provision of outer reactor vessel cooling (for VVER-640)
- corium catching in concrete well.

Concept of VVER-640 enhanced safety is based on corium retention inside reactor vessel. Possibility of outer reactor vessel cooling is ensured. A deflecting shell with a central hole is installed under the vessel. This shell directs the coolant flow along the bottom generatrices. Riser and lower channels are provided in the concrete well.

Works [6], [7],[8], [9] are devoted to investigation of possibility of core melt retention inside reactor vessel under conditions of postulated core meltdown. These works confirm theoretical possibility of core melt retention inside the VVER-640 reactor vessel.

Beyond design basis accident management is one of the defense levels and comprises a system of actions oriented to prevent accident progression to severe accidents or to mitigate severe accidents if they occur. Questions of VVER severe accident mitigation strategy are considered in [10].

Distinctive degrees of severity of possible accident progression based on condition of the defense barriers:

- reaching the maximum design basis fuel rod damage limit,
- a further core damage and/or melting,
- reactor vessel meltdown and/or destruction,
- containment damage

are considered when developing procedures for beyond design basis accident management.

Particular safety goals are formulated as related to each degree of severity and safety functions are determined of which fulfilment is required to attain these goals.

In case of failure to attain the goals for some degree of severity, the actions are determined which are oriented to delay a further damage progression on base of necessity of attainment of the safety goals for the next degrees of severity.

Measures for severe accident management are oriented to:
- prevention of core damage,
- retainment of the damaged or melted core if any inside the reactor vessel,
- preservation of containment integrity,
- limitation of radioactive releases into the environment.

The following means:
- any available efficient technical means designed for normal operation,
- systems designed for safety ensurance under design basis accidents

can be utilized for accident management to prevent the situation from progressing to a core melt.

In fact, the risk of core melt occurrence may be thought of as being directlly related to violation of two critical safety functions: reactor subcriticality function and reactor core cooling function.

However, fulfillment of primary boundary integrity function, primary coolant inventory function and ultimate heat sink function creates necessary prerequisites for fulfillment of the two functions above.

Reactor subcriticality provision implies:

- a timely reactor trip,
- provision of a sufficient subcriticality margin after reactor trip,
- prevention of inadmissable reactivity variations

A timely reactor trip for the VVER-640 and VVER-1000 is provided due to appropriate trip insrtrumentation and trip set-point adequacy.

A sufficient subcriticality margin after reactor trip is provided due to an adequate worh of shutdown rods and of emergency boron injection systems

Prevention of inadmissible reactivity variations is provided by timely reactor trip and boron injection into the reactor:

- from hydroaccumulators with actuation, if necessary, of the automatic primary depressurization system (VVER-640),
- from boron injection pumps and hydroaccumulators of the emergency core cooling system (VVER-1000),
- from the quick boron supply system in case of a reactor trip failure (VVER-1000),
- from the normal makeup system in conjunction with normal boron control system if they are efficient.

Fulfillment of the primary boundary integrity function or primary boundary damage limitation is provided by prevention of inadmissible thermal and mechanical loads on the primary boundary.

Prevention of inadmissible thermal and mechanical loads on the primary boundary in the course of accident management should be ensured on base of strict realization of operational and emergency procedures.

Primary coolant inventory is provided by the same systems as for reactor subcriticality provision.

Means for provision of the ultimate heat sink depend to a large extent on the concrete accident scenario.

Actual procedures of severe accident management at NPPs with advanced VVER are expected to be based on:

- Probability risk assessments defining scenarii with maximum contribution to core damage risk and revealing necessary mesures for accident management.

- Beyond design basis accident analyses necessary for understanding severe accident consequences and effectiveness of accident management measures.

- Instrumentation providing continuous parameter measurement featuring the unit critical safety functions mentioned above like reactor reactivity, hydrogen concentration in containment, contents of radiactive materials in primary and secondary coolant, outside containment radiation level etc.

- Systems giving to the operational personnel computerized information on condition of all safety and safety-related systems

## 4. EXPERIMENTAL AND ANALYTICAL INVESTIGATIONS OF SEVERE ACCIDENTS

### 4.1 Experimental investigations of severe accidents

The research and development works oriented to provision of measures and creature of means for prevention or mitigation of severe accidents leading to core damage beyond the limits prescribed for design basis accidents are being performed under special programmes. A systematic solution of design tasks is realized under these programmes on base of carrying out of computational, theoretical and experimental investigations.
Many scientific and research institutes jointly with OKB Gidropress participate in these programmes.

The approach adopted to design works for fulfillment of regulatory guide requirements in severe accidents determines contents of experimental and analytical investigations necessary for substantiation of design solutions. Experiments are aimed at process investigation of:
- core destruction
- corium transportation inside the core structure
- steam generation and steam explosions as a result of corium-water interaction
- in core melt bath
- heat removal from vessel outer surface
- corium-coolant, corium - metal of rector vessel and corium- concrete interactions.

The following experiments are planned to be performed in the first priority series:
- integral tests of fuel assembly destruction on PARAMETER test facility (Louch Scientific and Industial Association)
- investigation of radiactive fission product release following seal failure of fuel rods ( Obninsk Institute of Physics and Power Engineering)
- investigation of radiactive fission product release during corium melting on RASPLAV-2 test facility (Sosnovy Bor Science and Research Institute of Technology).

For investigation of reactor vessel behavior experiments are planned on:
- thermal and mechanical properties of the vessel steel ( Obninsk Institute of Physics and Power Engineering)
- boiling crisis on the vessel outer surface on PETLA test facility (Sosnovy Bor Science and Research Institute of Technology) and OKB Gidropress test facility
- water behavior on the surface of corium melt (Sosnovy Bor Science and Research Institute of Technology, Obninsk Institute of Physics and Power Engineering)
- physical and chemical processes in corium and its interaction with the vessel steel on RASPLAV programme (Kurchatov Institute Russian Research Center) and on RASPLAV-2 test facility (Sosnovy Bor Science and Research Institute of Technology, Obninsk Institute of Physics and Power Engineering)
- processes at core melt dropping into water on LAVA test facility (Louch Scientific and Industial Association), on BAK, VULCAN, TVMT test facilities ( Obninsk Institute of Physics and Power Engineering).

Tests on RASPLAV-2 test facility are planned for investigation of fission products and gases release in concrete- core melt - water composition.

### 4.2. Analytical investigations of severe accidents

Analytical investigations of severe accident processes are carried out using integral (considering reactor plant as a whole) and specific (for investigation of separate structures) computer codes. National and foreign computer codes are utilized for this goal.

Processes occurring during core drainage including seal failure of fuel rods, their destruction and melting taking into account of physical and chemical reactions, hydrogen and radionuclides release are analyzed with aid of RAPTA-SFD (Science and Research Institute of Inorganic Materials) and SVECHA (IBRAE) computer codes. In addition, SVECHA allows to evaluate core debris relocation into lower reactor plenum.

Analysis of heat transfer in "corium bath- vessel wall - water system" for substantiation of the conceipt of corium retention inside VVER-640 reactor vessel was performed [6] using MELVES computer code (Sosnovy Bor Science and Research Institute of Technology).

At elaboration stage are NARAL (Elektrogorsk Scientific Research Center) and TF PWR (Obninsk Institute of Physics and Power Engineering) computer codes that will allow to calculate processes of core melt bath formation taking into account convection, stratification and heat transfer .

Also elaborated is HITEF code (OKB Gidropress) that will allow to calculate heat up, deformation, melting and destruction of the bottom and cylindrical part of reactor vessel occurring as a result of its physical and chemical interaction with core melt.

RASPLAV computer code (IBRAE) allows to calculate processe starting from core melt bath formation up to vessel destruction.

DINCOR elaborated computer code will cover all the processes mentionned above taking into account possibility of non-symmetric corium formation and vessel bottom destructure.

Investigation of steam explosions during supplying water into reactor on stage of core degradation will be performed with aid of elaborated VAPEX computer code (Elektrogorsk Scientific Research Center).

Foreign computer codes like MELCOR, RELAP-SCDAP (USA), ATHLET-CD (Germany) and other are utilized for testing of ready-made or elaborated computer codes .
As an example participation of OKB Gidropress specialists from 1993 jointly with specialists of other scientific and research and project organisations of the Russian Federation in fulfillment of the International Standard Problem ISP36 [11].
This task is an investigation conducted with the aim of comparison of experimental and computational results of behavour of a VVER assembly model under conditions of an early core degradation stage.

The results of CORA-W2 experiment conducted in Karlsruhe nuclear research center in Germany were used as experimental results. Computations of the Russian specialists had been performed using RAPTA-SFD, MELCOR 1.8.2, ATHLET-CD MOD 1.1, ICARE2 MOD 1.0, SCDAP/RELAP5/MOD 3.1 computer codes.

It is noted in the report [11] on this work that:

- the utilized computer codes describe correctly the test bundle behaviour as a whole,
- a further improvement of the codes is required from the viewpoint of a more realistic models of fuel clad destruction, material interaction etc.,
- further efforts of the participants of the standard problem should be oriented also to study of a later stage of core destruction.

Works on computational and experimental justification are related with conduction of additional expensive investigations The complexity of investigations and generality of investigation goals for VVERs and PWRs makes international cooperation for fulfillment of these works advisable.

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# SEVERE ACCIDENT PREVENTION AND MITIGATION: A UTILITY PERSPECTIVE — EDF APPROACH

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**Abstract**

Current plants have excellent safety records and are cost competitive. For future plants, excellence in safety will remain a prerequisite, as well as increased cost competitiveness. When contemplating solutions to Severe Accident challenges, cost effectiveness is essential in the decision making process. This cost effectiveness must be understood not only in terms of capital cost, but also of Operation and Maintenance costs as well as absence of additional risks to plant operators. Examples are given to illustrate the recommended approach.

## I-INTRODUCTION

As outlined by many attendees to the meeting on "Identification of Severe Accidents for the Design of Future Nuclear Power Plants" (1), Severe Accident challenges were not explictly addressed in the design of current plants. However, it turned out that, due to good engineering practice, sufficient margin was imbedded in most designs to provide good or excellent protection against most Severe Accident challenges. To go further into the understanding of these remarkable results, it must be recognized that, starting from the evaluation of all available margins, very pragmatic studies were carried out and came to the conclusion that significant improvement in public protection could result from the implementation of adequate accident management and emergency planning provisions. Another very important conclusion was that, on operating plants, most design modifications contemplated to cope with specific challenges were difficult to implement, costly, and, in some cases, would be counterproductive through addition of unnecessary complexity to plant operation.

Moreover, these excellent safety records have been found to be compatible, at least in the case of EDF with acceptably low construction and operation costs.

To summarize the situation, one could say that current plants are safe, allow electricity generation at low and in general competitive costs and that severe accident provisions have a minimal impact on inspection and maintenance.

## II-REQUIREMENTS FOR FUTURE PLANTS

For future plants, it is generally agreed that Severe Accidents will have to be considered at a very early stage of the design. This approach has been reflected in recommendations issued by Safety Authorities as well as Utilities

Though there could be minor or easily explainable differences between requirements issued by Utilities, general trends could be identified:

*a balanced approach between prevention and mitigation was favored

*probabilistic risk objectives, either design objectives (Core Melt Frequency) or health objectives (Large Release Frequency) were based on internationally agreed recommendations.

A paper presented during the meeting by OPEN members, for instance, proposed to adopt INSAG3 recommended objectives, i.e.:

*Core Melt Frequency less than $10^{-5}$ per reactor year

*Large Release Frequency (Cumulated) less than $10^{-6}$ per reactor year

*Minimum Containment Performance, i.e. Large Release Frequency lower than Core Melt Frequency by at least a factor of ten.

For practical implementation, the following were recommended:

*consider all Severe Accident related challenges,

*address challenges as appropriate, i.e. those having a real impact on risk

*eliminate by design challenges with the potential for leading to early containment failure

*address Delayed Containment Failure through adequate Severe Accident Management procedures or guidelines or through specific features

*address Bypass sequences through prevention.

*Severe Accident sequences or challenges should not be Design Basis, meaning that:

+potential consequences should be assessed using best-estimate approach

+components needed to mitigate their consequences should survive under estimated accident conditions but should not be qualified for such conditions in the regulatory sense

+systems or components needed to mitigate their consequences should not be required to be redundant, though redundancy could be deemed appropriate, after analysis, on a case by case basis.

At last, one could add that Accident Management must remain one of the cornerstones of investment as well as public protection

This didn't mean that dramatic changes had necessarily to be anticipated at the design level. More simply, it was expected that through such a process, it would be possible to deal with Severe Accident challenges in a more consistent way, contemplated modifications ranging from minor adjustments in Accident Management procedures or guidelines to specific design provisions allowing to deal with clearly identified challenges.

## III-FURTHER CONSIDERATIONS

Utilities operating nuclear plants have to face two major constraints:

-plants have to be operated safely: Utilities operatoring nuclear units cannot contemplate a situation in which plant operation would result in risks unacceptable from a societal standpoint.

-electricity generation costs must be kept below levels corresponding to that of either competing energy sources, or of alternate means for producing electricity.

Up to now, in the French context at least, nuclear electricity has had an edge on other comparable energy sources. Though EDF are not necessarily in the best position to give an opinion on the causes of this cost advantage, there seems to be some consensus on the fact that standardization and good operating records were the two major factors

In the recent past, however, this cost advantage was decreased. Reasons for this narrowing are diverse, and range from external causes such as significant improvement in competitiveness of other energy production means or low cost of fossil fuels to additional internal burden resulting from increased maintenance as well as new regulatory constraints.

So, one of the most difficult issues for the future, for current plants as well as for future plants, is to improve plant economics while maintaining high safety records.

## IV-SEVERE ACCIDENTS AND COSTS

Before deciding what need to be done to consider in a first step, and address in a second step (and if appropriate) Severe Accident challenges, some specifics of such accidents need to be considered:

*the degree of complexity of Severe Accident sequences is much greater than that of Design Basis Accidents

*plant behavior cannot be actually considered deterministic. General phenomena can be anticipated, but detailed evolution is difficult to predict

*in-situ evaluation of plant status is difficult. For example, beyond a minimum level of core degradation, deciding whether the corium is still inside the vessel could prove problematic.

*as core degradation progresses, there are more uncertainties in the understanding of plant status.

It so appears reasonable to limit, as far as possible, core degradation through appropriate accident management procedures and deal with challenges to key component integrity (e.g. prevent vessel or containment structure)

For Utilities, though recognizing that mitigation is needed, prevention is emphasized. However, this must be done in a very comprehensive way.

*First, prevention must be controlled. In some cases, prevention results in added (and unnecessary) complexity which, in turn, can increase risk

*Then, costs incurred must be commensurate with risk reduction. If risk reduction is marginal, investing becomes questionable, and the question to be raised is: would the same (or a smaller) amount of money allow more significant risk reduction if invested to deal with other challenges?

*Some challenges can be adequately dealt with at minimal cost through mitigation. This is the case for hydrogen generation or vessel meltthrough for example.

*For all Severe Accident challenges, many candidate solutions exist. e.g. for Hydrogen, one can think of:

+inherent protection through an increase of the containment volume,

+containment inertization,

+mitigating devices such as recombiners or igniters.

Decisions must so be made on a case by case basis, after consideration of:

*the safety significance of the problem

*potential benefit resulting from implementation of each candidate solution

*costs incurred

At last, when dealing with costs, it must be kept in mind that not only investment cost is important, but also operation and maintenance costs.

## V-ANALYSIS PROCESS

When assessing the interest of candidate solutions to deal with specific challenges, many aspects have to be analyzed. Though other aspects could be found, the following must at least be considered:

### 5.1-Available experience on current plants

The first question to be raised is whether the challenge has been identified as a credible safety issue on current plants, and, if it were the case, how it has been addressed.

Pros and cons of solutions adopted for current plants have to be outlined to provide a reference for further work.

One important thing to look at is whether the possibility of preventing the challenge from occurring has been investigated, and understand the reasons which led to adopt the solutions existing on current plants.

### 5.2- Credibility of risk

The root cause of the challenge being investigated must be identified, and the associated risk evaluated. If the associated risk were deemed negligibly small as compared to potential costs incurred, addressing the challenge in the design would be highly questionable.

If, on the contrary, risk reduction was deemed "significant" the following would have to be investigated:

*is it possible to address the root cause of the problem

*candidate solutions for risk reduction (procedures, use of available systems, specific devices)

In all cases, reliability and cost effectiveness of contemplated possibilities should be assessed.

### 5.3- Reliance on plant vulnerabilities

The objective being risk reduction, from a technical standpoint at least, candidate solutions for specific challenges should not depend on plant vulnerabilities (systems or components which are significant or dominant contributors to risk) to perform their intended function.

For example if Station Blackout were a dominant contributor to Core Melt Frequency, relying on on-site Diesel Generators to limit melt progression would be a questionable option. Similarly, if a component such as a pump of a given type were the dominant contributor to one key system failure, it would seem unreasonable to rely on systems with identical components to recover from the situation.

This analysis must be very comprehensive, and the root cause of the vulnerability would have to be clearly identified; For example, it would be misleading to incriminate the reliability of a given type of pump if the problem were in fact on a lubricating system.

### 5.4- Impact on Normal Operation

The Utility objective being to minimize generation costs, and plant availability being key to low generation costs, candidate solutions contemplated to address specific challenges should have minimal impact on normal operation.

For example, spurious actuation of a depressurization system contemplated to deal with direct containment heating (DCH) and recovery time in case of spurious actuation should be sufficiently low not to affect significantly plant availability during the expected lifetime of the unit. In this respect, one can note that results on requested reliability for such a system would probably be different if the discharge were made directly to the containment atmosphere or to the In-Containment Refueling Water Storage Tank when one is provided.

### 5.5- Detrimental impact on more probable events

Candidate solutions proposed for Severe Accident related challenges should not degrade plant response in case of less improbable perturbed situations.

One of the principals adopted in plant design is that the most robust solutions are for the least improbable events. Proposing solutions for Severe Accident challenges resulting in decreased reliability of systems expected to operate in case of Design Basis Accidents would not be tolerated.

One example could be the decision to blow down the Steam Generators (SG) in case of Steam Generator Tube Rupture to prevent Containment bypass in case of a Severe Accident. Depending on the actuation signal chosen for SG blowdown, blowdown could occur in case of Reactor Coolant System break, situation in which it is recommended to maintain the SG inventory to prevent tube creep failure resulting from elevated temperatures on the primary side of the SG. In such a case, further assessment would be needed to evaluate the global impact of the solution, and not the specific improvement in case of a low probability sequence.

## 5.6- Impact on maintenance

Candidate solutions can have a detrimental effect on maintenance work in that they could add burden on the operators either for standard maintenance activities on other components, or for their specific maintenance.

### 5.6.1- Impact on normal maintenance

Most maintenance in the Nuclear Island is performed during outages. Some activities are on the critical path, both in terms of schedule and in terms of doses to the maintenance personnel. Contemplated solutions should not, to the extent possible, create problems on all standard activities. Examples of issues to be investigated are:

*no constraint exists on polar crane operation and (or) stud (de)tensionning,

*if intervention is needed on a candidate solution to allow easy maintenance on other components, this intervention is not on the critical path for the outage

*if such intervention is needed, impact on personnel exposure is negligible

### 5.6.2- Impact on device maintenance

Candidate solutions should be rugged enough to require no or minimal maintenance. When maintenance is nevertheless required, maintenance time should in any case be limited, and maintenance activities should bear no risk for operators.

A good example could be mitigating devices to deal with Hydrogen production. Before entering an extensive study to optimize the location of recombiners for example, the following should be investigated:

*areas available in the free volume of the containment after considering all possible paths of the trolley on the polar crane during maintenance

* in the remaining volume, areas were installation of the devices would create problems on other maintenance activities

*in the remaining volume, surfaces available to install the devices

*on these surfaces, identify locations were devices could be installed without creating risks for maintenance personnel (no need for very long ladders to have access to the devices, no risk of accident when operators pass by the devices, negligible dose,...)

*identify location where maintenance work can be done without too much difficulty.

## VI- CONCLUSION

Though current plants were not explicitly designed for Severe Accidents, they are quite safe and margins imbedded in the designs as well as implementation of adequate Accident Management procedures provide excellent resistance to many Severe Accident challenges.

The trend, for all future designs, is to consider all Severe Accident challenges at a very early stage of the design, and address them as appropriate. Taking advantage of lessons learnt on current plants, technical elements do exist to adequately address all challenges with risk significance.

However, considering that plant economics is becoming increasingly an issue for current as well as future plants, and that there are many possibilities to address specific challenges, thorough investigations will have to be performed to identify the best compromise between addressing specific risks and cost-effectiveness. In particular, candidate solutions will have to be analyzed considering their impact on all plant activities during normal operation to provide for assessment completeness.

DESIGN APPROACHES FOR
PREVENTION AND MITIGATION OF
SEVERE ACCIDENTS — HWRs

(Session 2-B)

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# SOME DESIGN MEASURES AND CONCEPTS FOR THE PREVENTION AND MITIGATION OF SEVERE ACCIDENTS IN THE INDIAN ADVANCED HEAVY WATER REACTOR

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**Abstract**

Thorium utilisation has been an important aspect of Indian Nuclear Power Programme. The strategy of realising large scale power generation from thorium has been clearly laid out right since the inception of Indian Nuclear Power Programme. As a part of this strategy an Advanced Heavy Water Reactor (AHWR) is being developed. This reactor would generate most of the energy from U-233, bred in-situ from thorium.

The paper describes the reactor configuration and passive safety features to remove core heat and also to ensure integrity of containment during accident scenarios. Some design measures and concepts for the prevention and mitigation of severe accidents in AHWR are described.

## 1.0 INTRODUCTION

Most of the nuclear power plants currently operating in India are 220 MWe Pressurised Heavy Water Reactors (PHWR) utilizing natural uranium as fuel. However, the long term nuclear power program in India needs to be based on the exploitation of its vast thorium deposits. As a part of this strategy, an Advanced Heavy Water Reactor (AHWR) is being designed and developed in India [1]. The main aims in the development of AHWR are thorium utilisation in a thermal reactor and an overall goal to enhance the safety of the reactor and also to improve the economics of the system [2]. The design of AHWR is based on well proven PHWR and BWR technologies. In addition, many passive safety features and engineering features are incorporated.

AHWR is a boiling light water cooled and heavy water moderated reactor. The core consists of $ThO_2$ and MOX fuel in which natural uranium oxide is enriched with plutonium oxide.

AHWR has a number of passive and advanced safety features. Through design measures, the probability of occurrence of severe accidents resulting in degradation of core, failure of containment, hydrogen production and combustion in the containment and core concrete interaction have been brought to a negligibly low value.

## 2.0 BASIC AHWR CONFIGURATION

Basic physics design and overall layout are in the process of optimisation and some of the features described below may be changed.

The AHWR core will use U-233 enriched thorium and Pu-239 enriched uranium as fuel. The basic core physics objective is to obtain around 75% or more power from thorium. AHWR fuel cluster is to have 52 fuel pins arranged in a square array. To be consistent with the basic design objectives of thorium utilization and negative void coefficient of reactivity, the cluster will incorporate MOX as well as $ThO_2$ fuel pins.

The reactor has a vertical pressure tube type of construction with calandria tubes surrounded by heavy water moderator. Coolant is boiling light water. The calandria is housed in a water filled reactor vault which acts as an effective radiation shield. End shields filled with steel balls and water are provided at both the ends (top and bottom) of the calandria. There is no outlet header in AHWR. Outlet tail pipes from individual coolant channels are connected directly to four steam drums. The separated steam from drums goes to the turbine. Hot water from each steam drum flows through four downcomers to an inlet header which is kept above the core elevation. Water from inlet header to core flows through inlet feeders. Fig.1 shows a schematic view of the AHWR.

A special feature is provided in the fuel cluster to direct the emergency core cooling flow to the individual fuel pins within the coolant channel. Eight perforated tubes, located on the periphery of the fuel cluster and running along its entire length are provided so as to direct the water jets at different elevations and in different directions.

The reactor will have two independent and fast acting shut down systems based on diverse principles and with sufficient redundancy.

## 3.0 PASSIVE SAFETY FEATURES FOR CORE HEAT REMOVAL

### 3.1 NEGATIVE VOID COEFFICIENT OF REACTIVITY

The fuel cluster of AHWR will consist of MOX as well as U-233 enriched ThO2 fuel pins. U-233 enriched thorium oxide will have a positive void coefficient but will be subcritical, whereas the MOX fuel will have negative void coefficient of reactivity. With proper combination of MOX and ThO2-U233 pins in a cluster it is possible to achieve overall negative void coefficient of reactivity under all operating conditions [3]. With this inherent feature, the reactor would be shut down automatically if there is any increase in void due to any transient or accident condition.

### 3.2 NATURAL CIRCULATION

The primary heat transport loop is designed to obtain full power flow by natural circulation. Primary circulation pumps are eliminated and the necessary flow rate is achieved by locating the steam drums at a height of about 32 m above the centre of the core [4]. Core cooling will not have to depend on external pumping power, moving parts or control instrumentation.

By eliminating nuclear grade primary circulating pumps, their prime movers, associated valves, instrumentation, power supply and control systems, the plant is made simpler, less expensive, and easier to maintain as compared to options involving forced circulation in the primary coolant circuit. A simplified PHT system flow sheet is shown in Fig.2.

### 3.3 ISOLATION CONDENSERS

Condensation of high temperature steam is an extremely efficient heat transfer mechanism, and direct removal of BWR core decay heat through condensing heat exchangers has always been viewed as an attractive option.

Removal of core decay heat in AHWR during normal reactor shut down condition is by passive means and isolation condensers are used for this purpose.

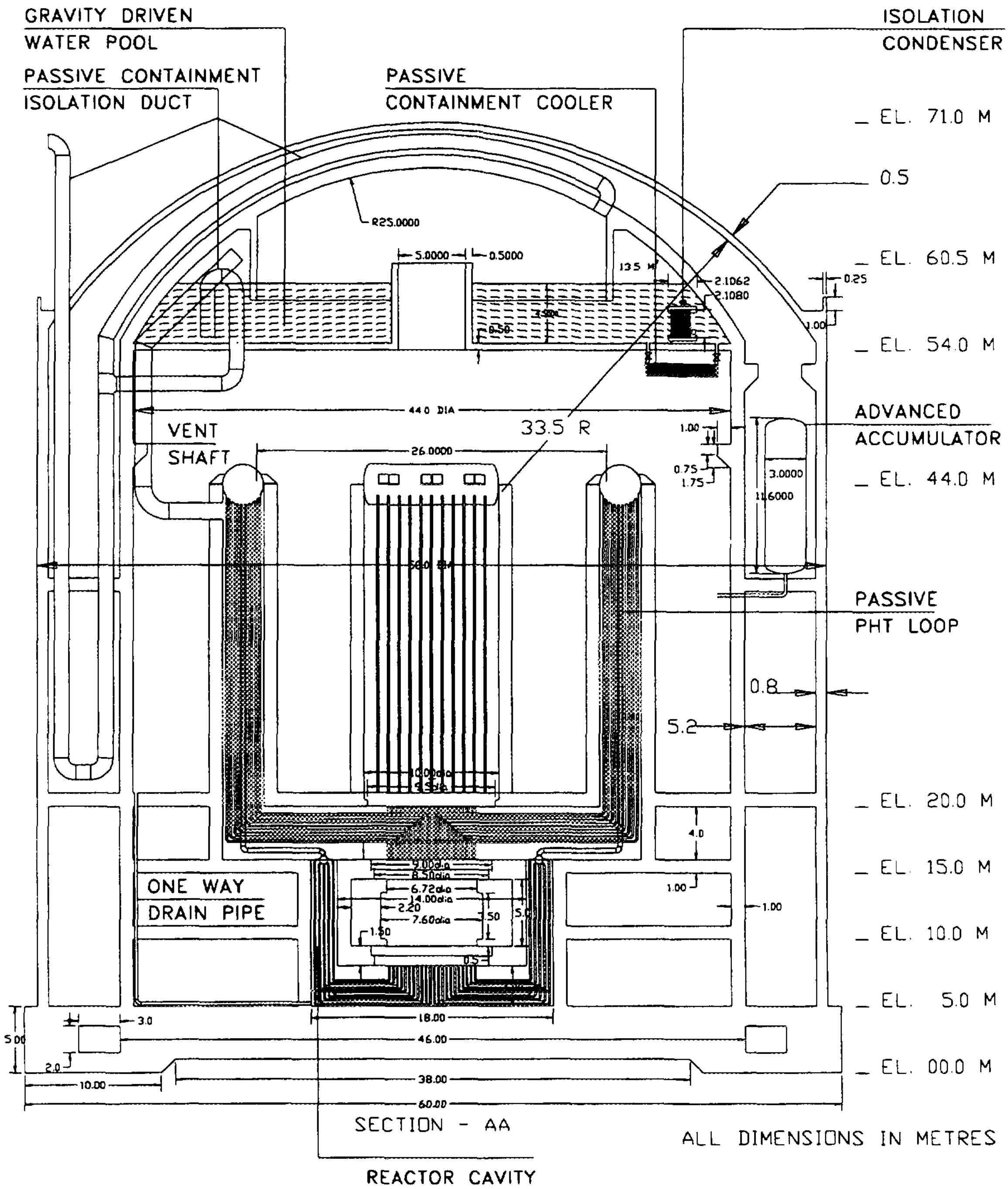


*FIG. 1. Proposed AHWR reactor building elevation.*

During normal reactor shut down, steam generated due to decay heat passes, by natural circulation, into the Isolation Condenser (IC) tube bundles submerged in the Gravity Driven Water Pool (GDWP) located high above the core in the dome region of the containment [4]. The steam condenses on the inside of the tubes and heats up the pool water. The condensate returns by gravity to the reactor. The water inventory in the pool is so chosen that it is sufficient to provide cooling for three days after reactor shut down. Schematic flow sheet for decay heat removal system is shown in Fig.3

Although the pool water is sufficient for three days cooling, a GDWP recirculation and cooling system is also provided with industrial grade heat exchangers, pumps, valves and interconnecting piping. Operator can switch on the GDWP cooling circuit if the need arises.

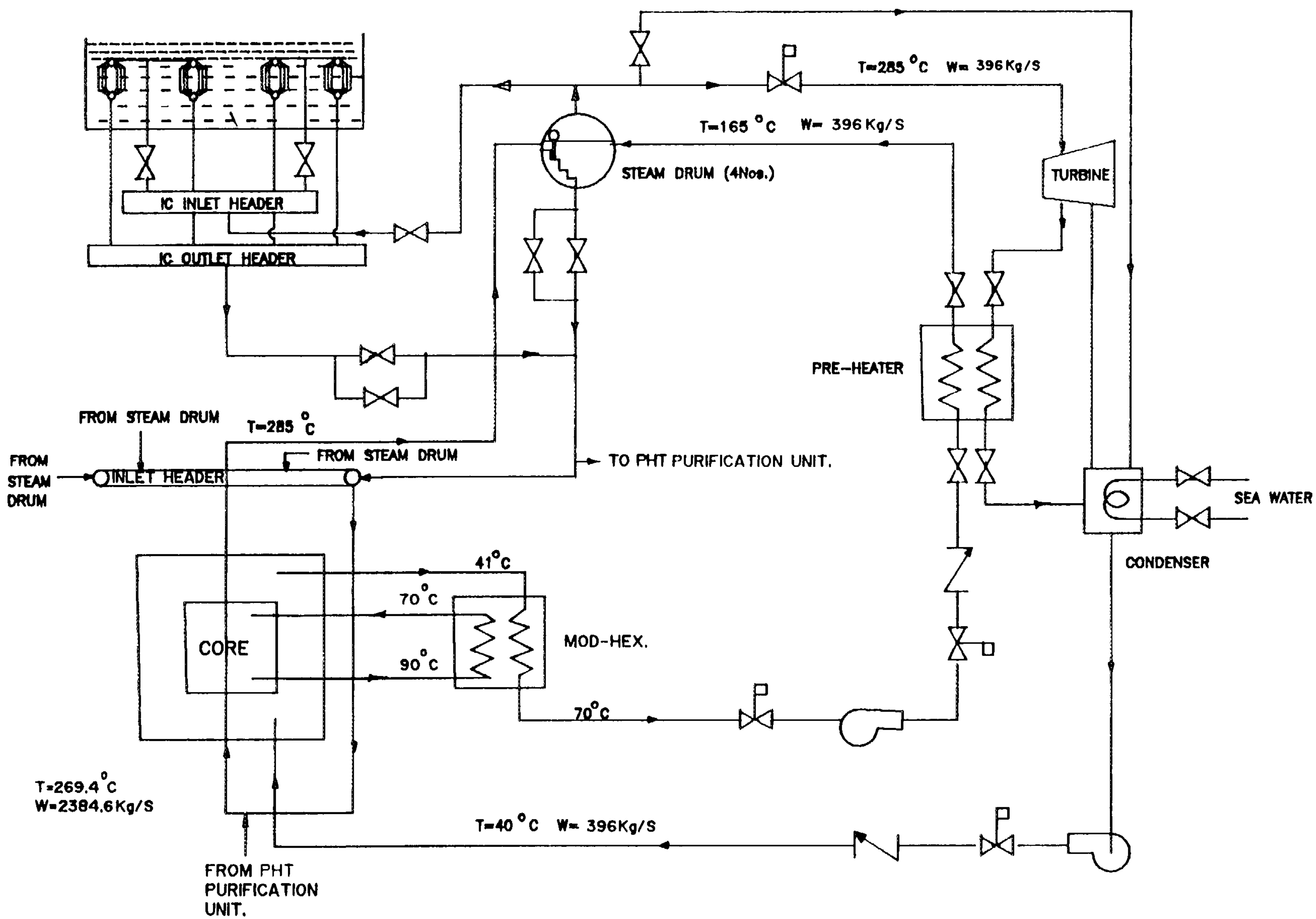


*FIG. 2. Simplified PHT system flow sheet.*

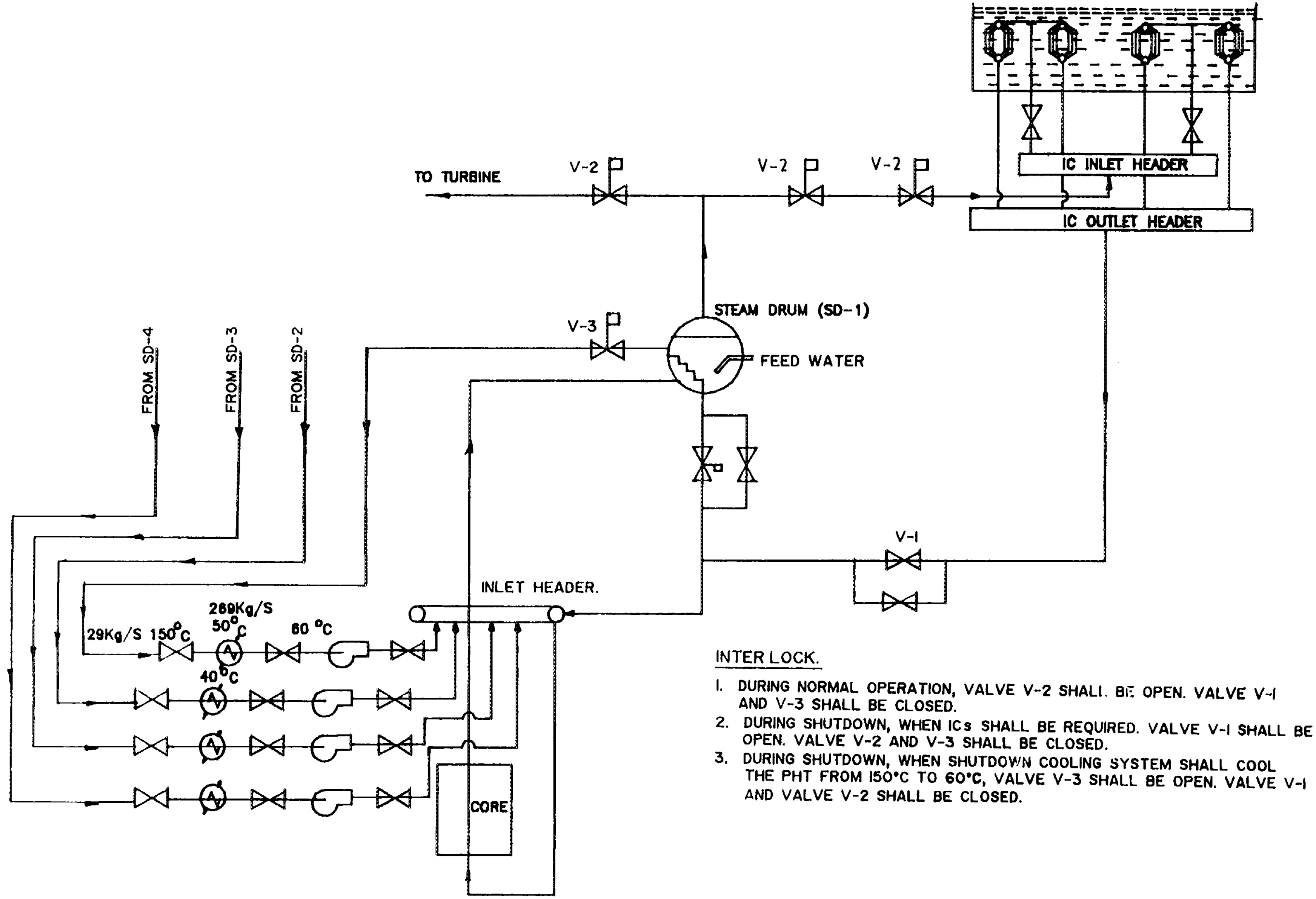


*FIG. 3. Decay heat removal system.*

### 3.4 RELIABLE EMERGENCY CORE COOLING SYSTEM

In the event of a loss of coolant accident (LOCA) due to a rupture in the primary heat transport system pressure boundary, emergency coolant injection would be necessary to keep the core flooded so as to prevent overheating of the fuel. In AHWR this is achieved through advanced accumulators. A series of rupture discs act as isolating interface between the advanced accumulators and the reactor core. These discs rupture when post LOCA depressurisation of PHT system reaches a pre-set level.

260 cubic metres of borated water is stored in four advanced accumulators pressurised to 4.8 Mpa by nitrogen. On the occurrence of a LOCA, the water from the four advanced accumulators is injected into the core and gets sprayed on the individual fuel pins as well as the coolant tube through eight perforated water tubes running along the periphery of the fuel cluster coolant tube to provide efficient emergency core cooling.

During a loss of coolant accident, water from an ordinary accumulator normally will last only for a few seconds. In advanced accumulator, with the installation of Fluidic Flow Control Device, a large amount of water will be provided quickly to the core in the early stages of the accident to flood the core and then a relatively small amount of water (consistent with the decay heat rate) will continue to flow into the core for a longer time of about 15 minutes, to remove the core decay heat. Fig.4 shows the advanced accumulator.

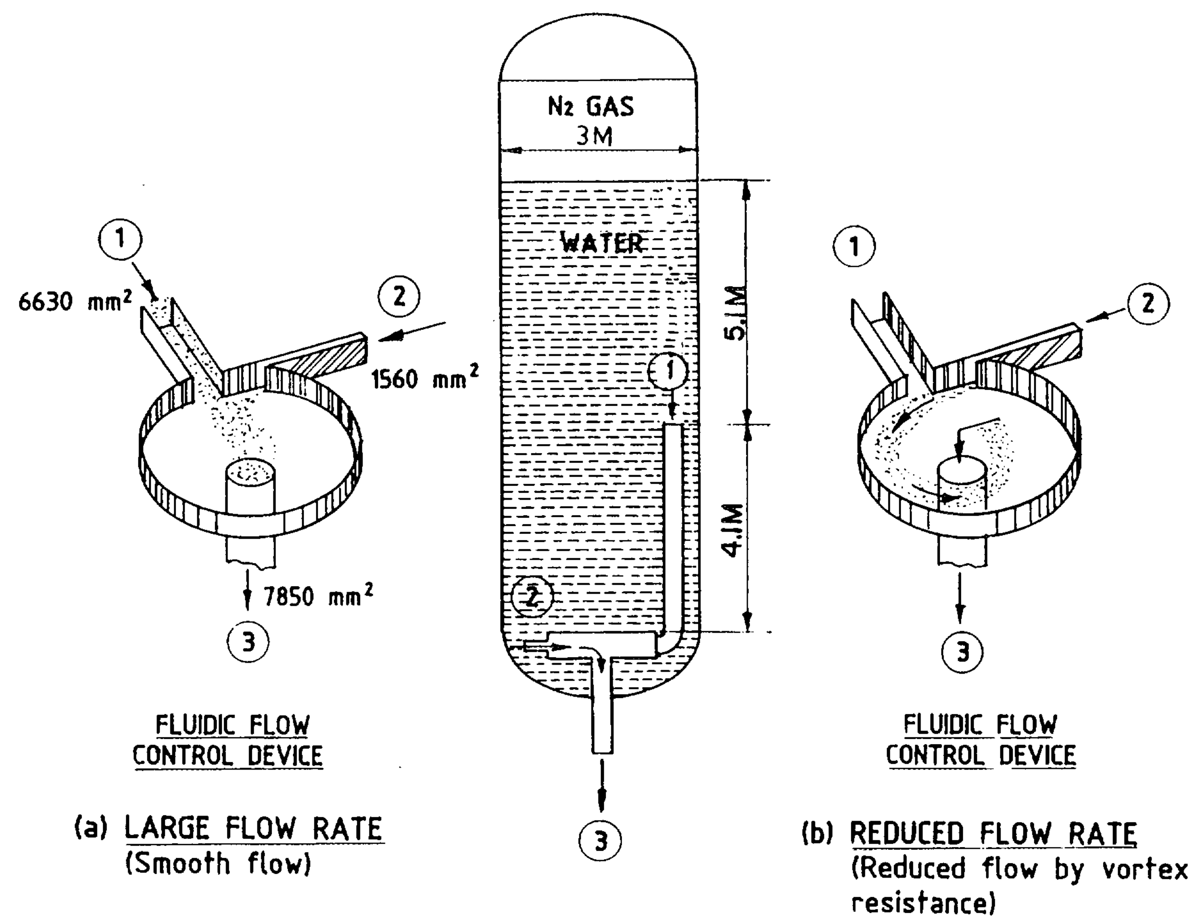


*FIG. 4. Advanced accumulator for AHWR.*

The probability of total failure of emergency core cooling system is expected to be negligible, since the injection of emergency coolant is from four independent advanced accumulators and then through gravity driven water pool, both of which are passive systems, not depending on external power, instrumentation and control interface or moving parts.

### 3.5 GRAVITY DRIVEN WATER INJECTION

Gravity Driven Water Pool (GDWP) is another passive safety feature [5] from where water will flow into the core by gravity after the water in the advanced accumulators gets exhausted. Total volume of borated water stored in GDWP is 5300 cubic metres. A number of outlet lines from the GDWP are arranged at different elevations of the water pool so that the water is released as per decay heat rate and core cooling is extended for more than three days without any operator intervention.

### 3.6 REACTOR CAVITY

After a loss of coolant accident, the water from the PHT system, advanced accumulators and the gravity driven water pool will be guided and will get collected in the space called Reactor Cavity [5] so that the core will be submerged under water. Provision is also made to pump water from the reactor cavity into the core in a long term recirculation mode.

### 3.7 REACTOR VAULT

The calandria vessel accommodating the core and moderator is housed in the reactor vault. The reactor vault contains about 350 cubic metres of water which acts as thermal shield. About 1.5 MW heat is generated in vault water and about 46.5 MW heat is produced in the moderator as a result of absorption of neutrons and gammas from the reactor. This heat is transferred to feed water system instead of being wasted as in most of the operating nuclear power plants, thus increasing the thermal efficiency of the plant.

In the case of a loss of coolant accident followed by a failure of emergency core cooling system (which is a very highly unlikely event), the inventory of water in the reactor vault and in the reactor cavity will serve as heat sink. Heat from the fuel will be dissipated to vault water by radiation and conduction and to reactor cavity water through coolant tubes by conduction, thereby preventing excessive temperature rise in the fuel. The large volume of vault water which is provided as thermal shield is also available to cool the core by pumping this water from the vault into the core.

## 4.0 PASSIVE SAFETY FEATURES TO ENSURE INTEGRITY OF CONTAINMENT

### 4.1 PASSIVE CONTAINMENT COOLING SYSTEM

The following two options are being evaluated for the removal of containment heat by passive means.

a) Passive containment coolers (PCCs), located in the GDWP at primary containment top region, are provided for long term cooling of the containment atmosphere in case of LOCA. These PCCs will help to mitigate the pressure and temperature build-up in the primary containment. Preliminary tests were done on a small scale model to understand the phenomenon of passive containment cooling [4].

b) In the second option, PCCs are taken out of the GDWP and are located under the gravity driven water pool in the form of finned tube units spread over a large area. Water from the pool is supplied to and returns from the finned tube banks by natural circulation via vertical headers.

The finned tubes are inclined to the horizontal, so that there is a preferred flow direction for the water. Water is supplied from GDWP via the header at one end, and heated water is returned to the pool via the header at the other end of each finned tube bank. In the long run, a boiling steam/water mixture is returned to the pool.

### 4.2 PASSIVE CONTAINMENT ISOLATION

To protect the population at large from exposure to radioactivity, the containment must be isolated following an accident. To achieve this, passive containment isolation in addition to the closing of the normal inlet and outlet ventilation dampers is being considered in AHWR. The Reactor Building air supply and exhaust ducts will be shaped in the form of U bends of sufficient height as shown in Fig.5. In the event of LOCA, the containment gets pressurised. This pressure acts on GDWP inventory and pours water by swift establishment of a siphon, into the ventilation duct U-bends. Water in U-bends acts as seal between the containment and the external environment, providing the necessary isolation between the two. Drain connections provided to the U-bends permit the re-establishment of containment ventilation manually when desired.

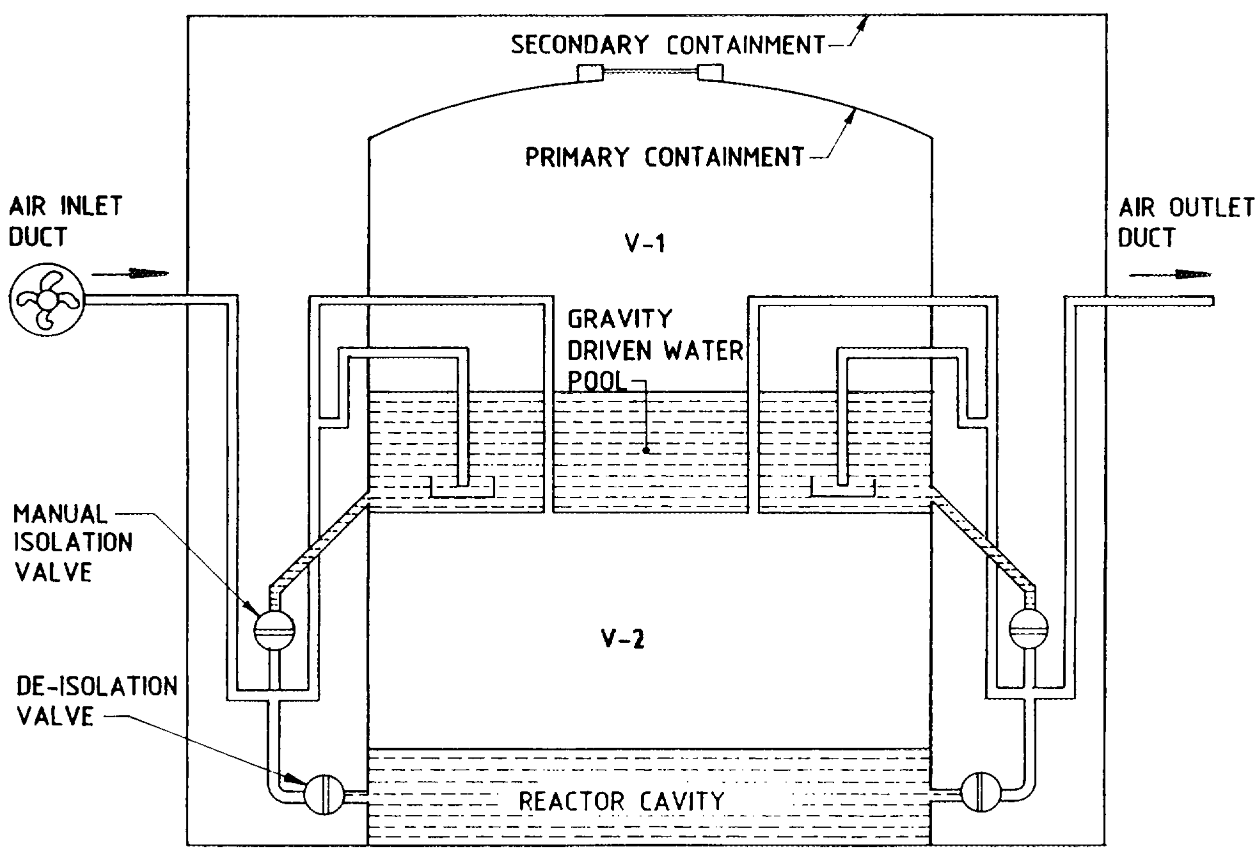


*FIG. 5. Principle of passive containment isolation.*

## 4.3 DOUBLE CONTAINMENT

The AHWR employs a double containment envelope instead of single envelope to reduce radioactivity releases to environment following postulated accidents. The concept of double containment is made to extend over all conceivable leak paths to ensure that all leakages from the primary containment are intercepted by the secondary containment.

The following features are provided in the design of AHWR to minimise the releases of fission products to the environment in case of an accident.

a) Passive containment isolation would be initiated in the event of an accident.
b) Reactor containment cooling system will cool down and depressurize the primary containment.
c)The large volume of the primary containment helps in minimising the peak pressure following an accident.
d) The Primary Containment Filtration and Pump back System provided will reduce iodine inventory in the containment after an accident.
e) A Purge System is provided in the inter-envelope space and will maintain negative pressure to have zero ground level release.

## 5.0 DESIGN BASIS ACCIDENTS

Because of the passive safety features and engineering features which have been incorporated in the design of AHWR, no significant radiological consequences to public would occur during design basis accidents.

**a) Large LOCA**:

In AHWR, all the large size pipes of reactor coolant system including the inlet header, are arranged well above the core to ensure core submergence in case of a large pipe rupture.

In case of LOCA, the reactor will be shut down by virtue of a negative void co-efficient of reactivity and two independent shut down systems (shut off rod and poison injection systems). The core will be cooled initially by injecting water into the core from advanced accumulators for at least fifteen minutes followed by the cooling of core through gravity driven water pool for three days without operator intervention.

**b) Large LOCA and failure of shut down system:**

In case of LOCA plus failure of the shut down system, the reactor would be shut down because of its inherent safety feature of a negative void coefficient of reactivity and the decay heat will be removed by the emergency core cooling system.

**c) Large LOCA and failure of ECCS:**

In case of postulated LOCA plus failure of ECCS (which is a very highly unlikely event in AHWR), the reactor will be shut down and core decay heat would be removed by conduction and radiation to moderator and vault water. PHT system water gets collected in the reactor cavity. The water from the reactor cavity can be pumped back into the core for long term cooling.

In the event of failure of ECCS piping or structural damage to GDWP, the inventory of GDWP will fill the reactor cavity.

**d) Large LOCA and failure of containment:**

Failure of either inlet ventilation dampers or outlet ventilation dampers of containment will not affect long term containment isolation because of the incorporation of passive containment isolation in the design of AHWR. With the provision of passive containment coolers, pressure build up in the containment is limited. Radiological consequences would be negligible because of the presence of double containment and other relevant engineering features.

## 6.0 CONSIDERATION OF SEVERE ACCIDENTS

The primary objective followed in the development of AHWR is to enhance the quality of safety by introducing passive systems for performing safety-related functions in the event of an accident. Passive systems are characterized by the fact that they utilize the laws of nature to ensure the safety of the plant. Hence the probability of occurrence of a severe accident in AHWR is expected to be negligibly low because of the presence of inherent and passive safety features as explained above in the paper. This could be established by a PSA when the design is finalised.

Pending the confirmation of the expected negligible probability of severe accident through PSA, we have examined the provisions made in the design to minimise the potential for severe accident. These features are described below:

### 6.1 DESIGN FEATURES FOR PREVENTION OF SEVERE ACCIDENT

Following design features are provided for the prevention of severe accidents.

**1) Negative void coefficient of reactivity.**

In case of failure of shut down systems which is a highly unlikely event, the reactor would be shut down due to the inherent feature of negative void coefficient of reactivity.

**2) Reliable emergency core cooling system with redundancy.**

During accident, core is cooled for 3 days, first by injecting emergency core coolant from four independent advanced accumulators and then through gravity driven water pool. Both the systems are passive and do not depend on any external power, instrumentation or moving parts.

**3) Availability of large quantity of water around the core to facilitate prolonged heat removal capability.**

In case of an accident, a large inventory of water is available in reactor vault and in the reactor cavity which will serve as heat sink.

### 6.2 DESIGN MEASURES FOR MITIGATION OF SEVERE ACCIDENT

Following design measures are provided in AHWR for the mitigation of severe accident.

**1) Double containment.**

Double containment is provided to minimise the release of fission products to the environment.

**2) Passive containment isolation.**

Inlet and outlet ventilation dampers along with passive containment isolation are provided for ensuring containment isolation. Provision of containment liner is also being considered.

**3) Passive containment cooling system.**

Passive containment coolers either located in the GDWP or located under the gravity driven water pool in the form of finned tube units are provided for the long term removal of containment heat. Passive containment cooling system will also help in lowering the temperature and pressure of the containment.

## 7.0 CONCLUDING REMARKS

The AHWR is being designed with a number of passive safety features which are explained in this paper. With these provisions, the potential for severe accident is expected to be negligibly low. This will be confirmed through PSA after the design is finalized. The need for further strengthening of the design provisions, if required, will be addressed at that stage.


## ACKNOWLEDGEMENT

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# ADDRESSING SEVERE ACCIDENTS IN THE CANDU 9 DESIGN

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**Abstract**

*CANDU 9 is a single-unit evolutionary heavy-water reactor based on the Bruce/Darlington plants. Severe accident issues are being systematically addressed in CANDU 9, which includes a number of unique features for prevention and mitigation of severe accidents. A comprehensive severe accident analysis program has been formulated with feedback from potential clients and the Canadian regulatory agency. Preliminary Probabilistic Safety Analyses have identified the sequences and frequency of system and human failures that may potentially lead to initial conditions indicating onset of severe core damage. Severe accident consequence analyses have used these sequences as a guide to assess passive heats sinks for the core, and containment performance. Estimates of the containment response to mass and energy injections typical of postulated severe accidents have been made and the results are presented. We find that inherent CANDU severe accident mitigation features, such as the presence of large water volumes near the fuel (moderator and shield tank), permit a relatively slow severe accident progression under most plant damage states, facilitate debris coolability and allow ample time for the operator to arrest the progression within, progressively, the fuel channels, calandria vessel or shield tank. The large-volume CANDU 9 containment design complements these features because of the long times to reach failure.*

## 1. OVERVIEW

CANDU reactors possess two inherent supplies of water close to the fuel: the moderator which surrounds the fuel channels, and the shielding water which surrounds the calandria. The short distance between the moderator and the fuel (1.5 cm), and the ability of the moderator to remove decay heat, allows the moderator to act as an emergency heat sink following a loss-of-coolant with failure of emergency core coolant injection. This heat removal path is efficient enough to prevent $UO_2$ melting. The shield tank in a severe core damage accident can remove heat conducted through the calandria shell. The shield tank cannot prevent fuel melting if all other heat removal systems, including the moderator, fail, but it can delay melt-through for hours and has the potential to indefinitely contain the melt within the calandria.

For this reason, we distinguish a *severe accident* in a CANDU, defined as one in which heat is not removed though the primary cooling system, from *severe core damage*, in which the pressure-tube geometry is lost. Severe accidents in which the moderator is available do not lead to severe core damage or fuel melting. Canadian safety practice has been to include the dominant-frequency severe accidents within the design basis - e.g., Loss of Coolant and Loss of Emergency Core Coolant (LOCA + LOECC). As a result, the frequency of severe core damage accidents has been reduced to the point at which they are residual risk events, typically less than $10^{-6}$ per year on an individual event basis.

For *severe accidents within the design basis*, typically a LOCA + LOECC, the fuel will heat up due to decay power and will heat up the pressure-tube through conduction, steam convection and radiation. At about 800°C, the pressure tube will start to plastically deform under the loads from the weight of the fuel and any residual coolant pressure, and strain or sag to contact the calandria tube. Since the calandria tube is cooled by the moderator, it will arrest the deformation of the pressure tube and provide a heat removal path to the bulk moderator. In this mode the pressure-tube acts as a fuse, deforming to allow efficient heat removal. The fuel bundles in such a sequence are severely damaged, with phenomena such as distortion of bundle geometry, oxidation of the clad, and, depending on the rate of oxidation, possible formation of a zirconium-uranium-dioxide eutectic at the clad/fuel interface. However the $UO_2$ itself does not melt.

As noted, severe core damage accidents beyond the design basis are residual risk events. A necessary requirement for severe core damage, defined for CANDUs as a widespread loss of channel integrity, to occur, is that the fuel channels not only be voided from within due to loss of HTS cooling and failure of ECC to inject, but that they additionally be voided from outside due to loss of moderator. In that case the fuel channels would gradually fail and collapse to the bottom of the calandria as the moderator boiled off. Blahnik (3) has characterized the degradation of a CANDU core with no cooling and gradual boiling-off of the moderator. The uncovered channels heat up and slump onto the underlying channels. Eventually, the supporting channels (still submerged) collapse and the whole core, still almost completely solid, slumps to the bottom of the calandria. Rogers et al (1,4) have developed an empirically-based mechanistic model that shows that the end-state of core disassembly consists of a bed of dry, solid, coarse debris irrespective of the initiating event and the core disassembly process. Heat-up is relatively slow, because of the low power density of the mixed debris and the spatial dispersion provided by the calandria shell, with melting beginning in the interior of the bed about two hours after the start of bed heat-up. The upper and lower surfaces of the debris remain well below the melting point and heat fluxes from the calandria to the shield tank water are well below the critical heat flux at the existing conditions. The calandria can therefore prevent the debris from escaping. Should the shield tank water not be cooled, it will boil off, and the calandria will eventually fail by melt-through, but this will not occur in less than a day, giving ample time for operator action such as flooding the shield tank from emergency supplies.

Because of the two redundant, diverse, physically separate, fully capable, independent, testable, dedicated shutdown systems, a failure to shutdown when required is a very low probability event, typically less than $10^{-8}$ events per reactor year, as predicted by the Probabilistic Safety Analysis. Therefore, severe core damage accidents resulting from failure of the control system and both of the two shutdown systems to shut the reactor down when required, are not considered. Additionally, severe core damage sequences resulting in core-wide high pressure melt ejection are irrelevant to CANDU reactors; simply put, the pressure tube again acts as a fuse and a small number of pressure-tube failures will relieve the internal pressure before much melting has occurred. References 1,2,3 and 4 confirm that severe core damage can occur only at low pressures and channel damage resulting from loss of all heat sinks results in predominantly solid debris.

Severe accident mitigation capabilities are being systematically addressed early in the CANDU 9 design process, which includes more explicit mitigation of severe core damage accidents, as well as meeting the traditional requirements for design basis accidents. Drawing from the methodologies used for severe accident analysis for similar operating reactors and the extensive research and development activities in support of the CANDU reactors, the CANDU 9 program for severe accident analysis is composed of the following elements:

- Systematic plant review,
- Probabilistic Safety Assessment (PSA) level I,
- Severe Accident Consequence Analyses (PSA level II),
- Severe Accident Design Assessments,
- Severe Accident Management Program, and
- Severe Accident Research Programs.

A *Systematic Review* of the Plant Design has been performed to identify the initiating events. A preliminary Level I *Probabilistic Safety Assessment* (PSA) is then performed and identifies the potential accident sequences that dominate risk. For design basis events, which as noted previously, include some severe accidents, the design organization then compares the results (frequency, consequences) to acceptance criteria, and determines whether further accident mitigation (such as further redundancy in process or safety-related systems) is **required**. In addition the PSA identifies beyond-design basis severe accidents, including severe core damage events. Those which lie in a frequency band between $10^{-6}$ and $10^{-8}$ events per reactor year are then examined in more detail, to estimate the consequences (*Severe Accident Consequence Analyses* or PSA Level II), and to determine whether further mitigation is cost- and risk-effective (*Severe Accident Design Assessments*).

The scope of the Level I PSA for internal events, performed in the pre-project phase, concentrates on the following initiating event classes: LOCAs and HTS leaks, feedwater and main steam line breaks, support system failures, moderator system failures, and failures following reactor shutdown. Failures include potential hardware failures and post-accident human errors. The PSA results guide the designers in the provision of appropriate redundancy, to meet reliability targets. They also assist in refinement of operator response guidelines, control centre design and the environmental qualification process. Some external events such as loss of off-site power are likewise also evaluated at an early stage. Other external events are analyzed later once a site is selected.

The CANDU 9 Severe Accident Consequence Analyses draw from the results of earlier severe accident analyses for the reference plant and other CANDU reactors (e.g. references 1,2,3,4 ) and concentrate on features new to this implementation. Thus a preliminary design assessment of severe accident mitigation features in the CANDU 9 reactor was undertaken. The first step was to assess the containment design against the dominant severe core damage sequences.

## 2. CANDU 9 DESIGN FOR SEVERE ACCIDENT MITIGATION

The CANDU 9 is a single-unit evolutionary heavy-water reactor based on the Bruce/Darlington plants with an electric output of 925 MW. Its major reactor and process systems use designs proven in the reference plants and in the single-unit CANDU 6. It also incorporates safety improvements especially for severe accident prevention and mitigation, and to increase the time available to the operator to arrest the accident progression early.

CANDU 9 uses the standard CANDU core arrangement of horizontal fuel channels cooled by heavy-water primary coolant, placed in a square lattice within a low pressure and low temperature heavy water moderator, surrounded by a large tank of light water for shielding. The 480 fuel channels, each consisting of a zirconium-niobium pressure tube in turn surrounded by a zirconium alloy (Zircaloy) calandria tube, contain twelve fuel bundles each about 0.5m in length. The 37 fuel element fuel bundles contain natural uranium sheathed in Zircaloy. The reactor structure assembly shown in Figure 1 illustrates the two additional water volumes (calandria vessel with about 330 Mg. of heavy water and shield tank with about 530 Mg. of light water, each with their own independent cooling systems)

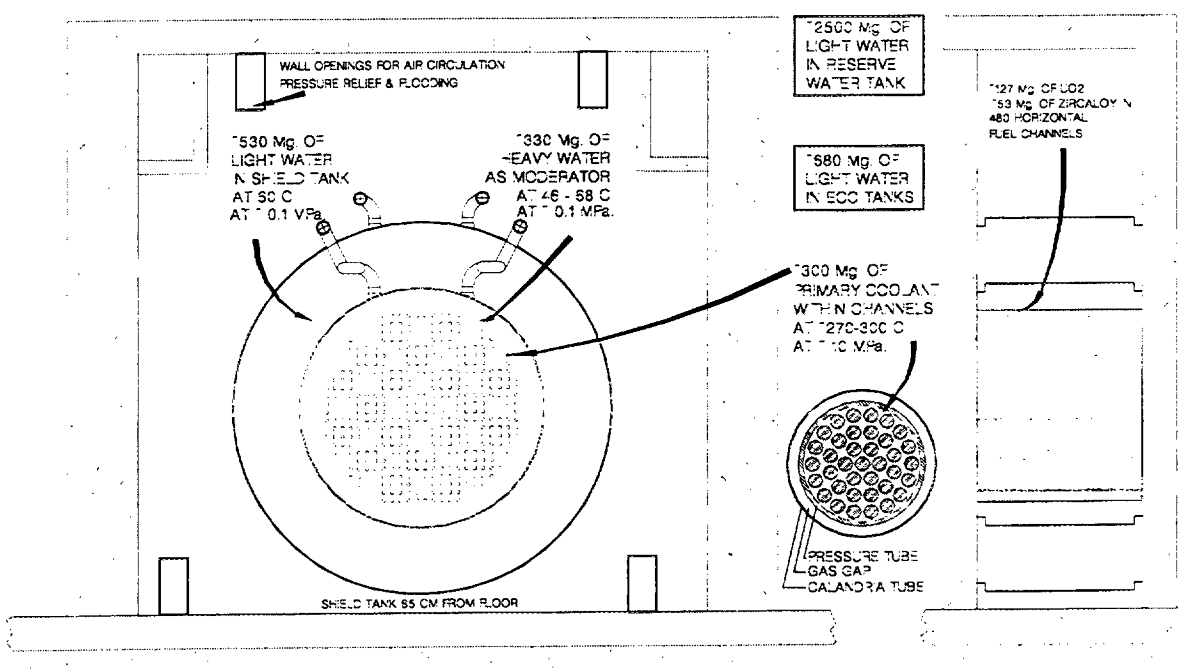


**Figure 1: CANDU 9 Reactor Core and structures.**

that, uniquely to CANDU, are instrumental in arresting the progression of severe accidents, such that potential debris is contained within the reactor structures (channels, calandria vessel or shield tank). CANDU 9 has an additional large tank of light water (about 2500 Mg.) located in the dome area of the containment. This Reserve Water Tank (RWT) supplies water automatically to the Emergency Core Cooling System pumps, and is also available to the operator to back up the normal heat removal systems and the front-line mitigation systems under accident conditions, specifically as emergency makeup to the steam generators, and to the heat transport system. This reduces the frequency of severe accidents. To reduce their consequences, the operator can use the RWT to keep the moderator and the shield tank filled with water, providing a means to cool the core even if the moderator and/or shield tank heat removal systems are unavailable. Water on the floor of the reactor building can be pumped back to the RWT to ensure the moderator and/or shield tank are always full, even if they leak.

The calandria-shield tank assembly, supported by two concrete reactor vault walls that span the two main concrete cross-walls of the reactor building internal structure, is located low, for structural stability against dynamic loads. This has the added benefit that the bottom of the shield tank is below the flood level.

## *2.1 CANDU 9 SEVERE ACCIDENT SCENARIOS*

The dominant severe core damage sequences from the PSA Level I analyses are characterized as follows: channels not only lose cooling through the primary side heat transport loop, they also lose moderator as a potential heat sink. Some of the severe accident initiating events involve a loss of all heat sinks at high pressures. If unmitigated, this leads to an in-core failure of a high power channel at high heat transport system pressures. Such a failure depressurizes the heat transport system and no debris formation nor melt ejection at high pressure occurs. All subsequent core damage occurs at low pressures. Thus dominant sequences all involve channel collapse at low pressures. As reported in references 3,4, there are four severe accident end states, defined by the terminal location for debris, which will stay stable indefinitely if the specified heat sink is maintained. Three are severe core damage states; the other is a severe accident with the damaged fuel contained in the channel:

| *Fuel / Debris location* | *Heat sink* | *Core Damage State* | *Illustration* |
|---|---|---|---|
| Fuel / debris in channels | Moderator water | CDS-1A | figure 2 |
| Debris in the calandria vessel | Shield tank water | CDS-2A | figure 3 |
| Debris in the shield tank | Base mat flood | CDS-3A | figure 4 |
| Debris in the Reactor Vault | Base mat flood | CDS-4A | figure 5 |

Furthermore there are two possible variations of each of the above severe accident end states: Dry, hot fuel/debris ('A' state) or Debris covered by water (flooded - denoted as 'B' states, in later discussion). The latter implies that some recovery action has taken place to introduce water onto the debris. The "flooded debris" alternative is mainly of interest for evaluation of containment response, because it potentially involves a short period of rapid steaming (i.e. steam surge associated with the quenching of a large mass of hot debris) while the containment pressure is perhaps already elevated by earlier events. The steam surge also determines the required surge relief capacity of the various vessels or rooms (HTS, calandria vessel, shield tank, reactor vault). The surge can occur at the most inopportune time from the standpoint of other containment challenges and is so considered in the containment analyses.

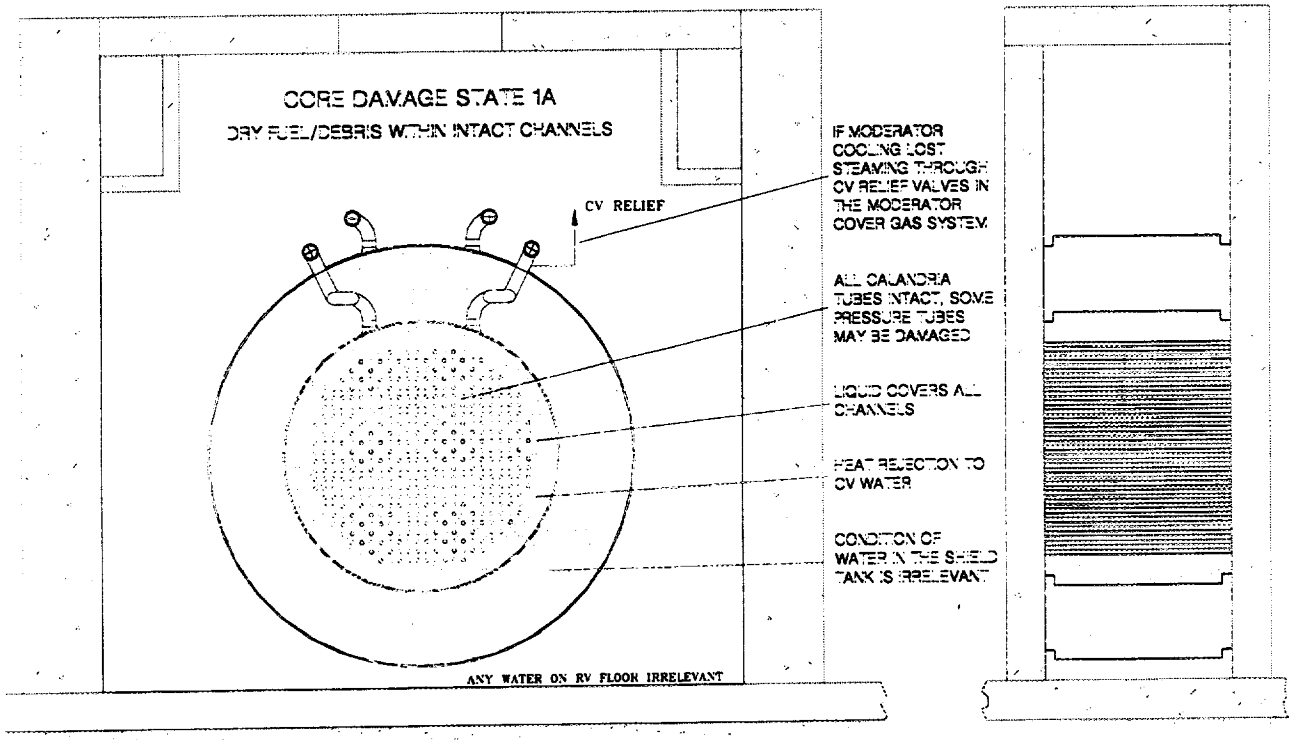



**Figure 2: Hot Fuel / debris in intact channels - CDS 1A**

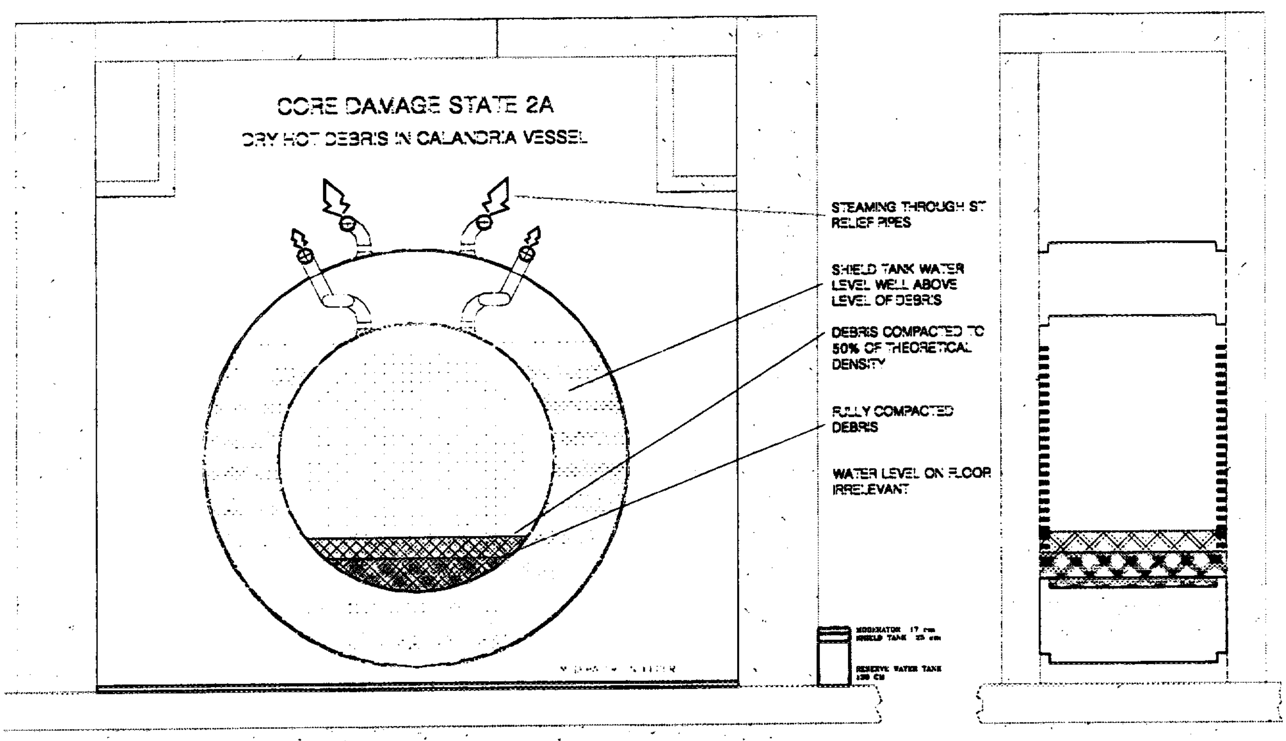



**Figure 3: Hot debris in calandria vessel - CDS-2A**

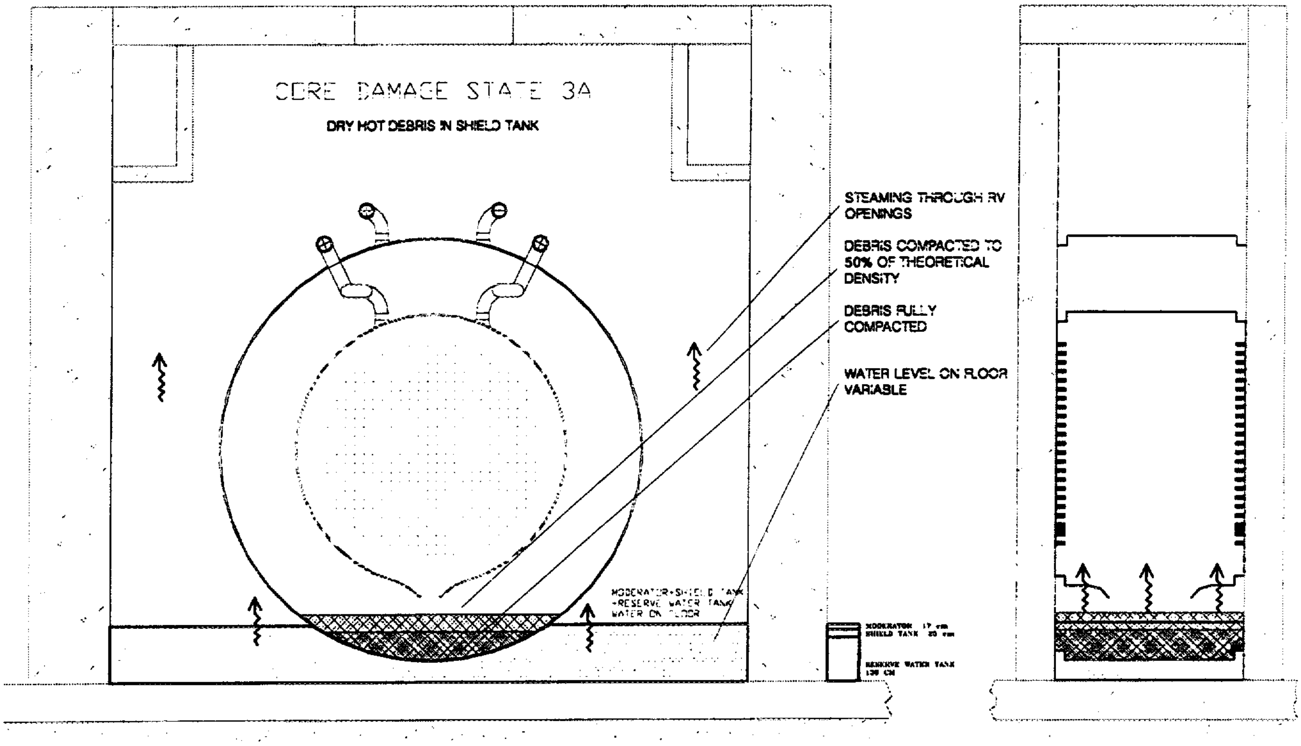



**Figure 4: Hot debris in shield tank - CDS 3A** *[Potential contributors to the flood level on the floor are listed in Table 1 ]*

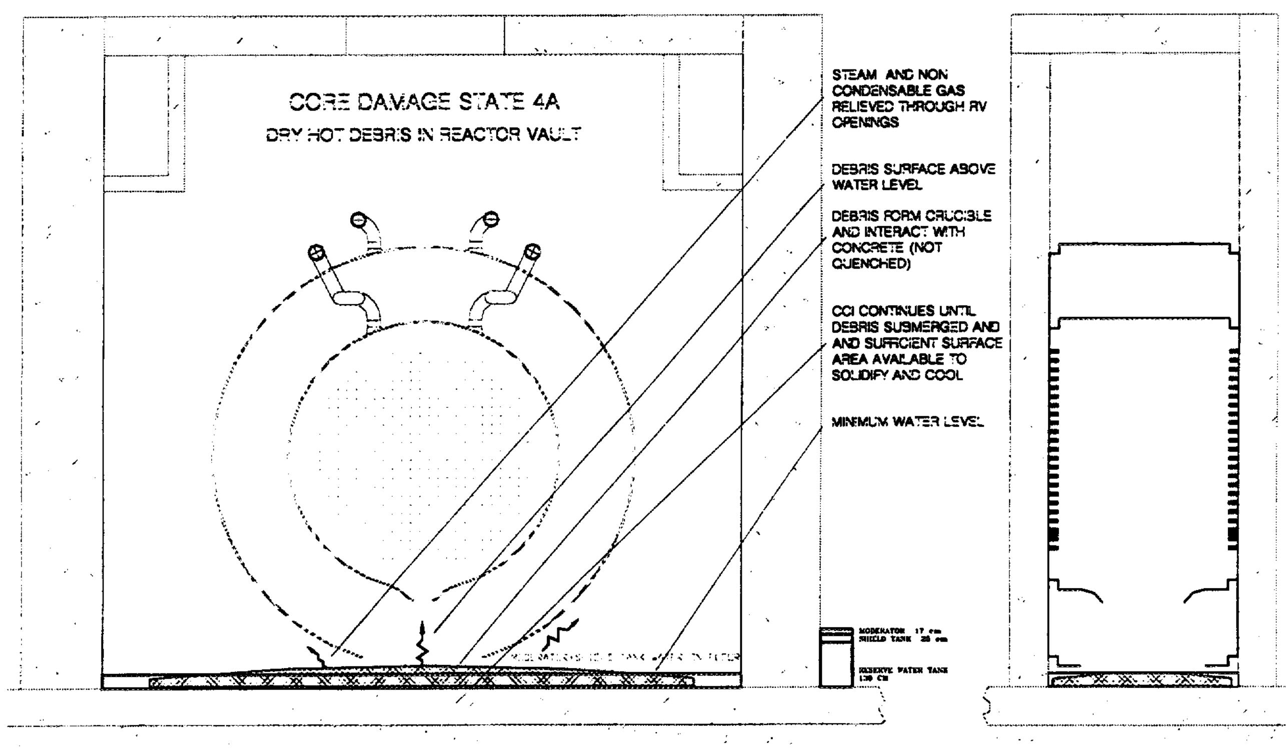



**Figure 5: Debris in Reactor vault - CDS 4A**

## 2.2 CONTAINMENT PHENOMENA

In order to assess the containment ability to mitigate severe accident events, the following potential phenomena are identified:

| SOURCE TERM | ACCIDENT PHENOMENA / NOTES |
|---|---|
| Steady release of steam/water mixtures. | Due to fuel/debris surrounded by water; for each of the core damage states in section 2.1 |
| Surge release of steam/water mixtures | Quenching of fuel/debris by water (operator actions or progression from one CDS to another). |
| Release and accumulation of non-condensable gases | Short term $H_2$ from steam reactions with zirconium, long term by radiolysis, corrosion, interactions with concrete (also CO). |
| Hydrogen combustion | Slow accumulation of $H_2$; igniters and recombiners limit the hydrogen concentration below the deflagration limit and permit only local burning. |
| Debris-water interaction within calandria and shield tank vessels | Energetic interactions precluded as debris not molten. $H_2$ estimates are low ( << 10% of molten zirconium may react - Reference 3) |
| Debris-water interaction on Reactor Vault floor | Floor always covered with water if debris on floor |
| Fission product (FP) interactions following release from debris | FP carry significant (up to 40%) decay heat. Steady containment heating by fission products can be simulated analytically; most of the fission products will be in the water pool on the floor. |
| Boundary (e.g. seal) failure due to prolonged high temperature/radiation exposure | Loss of containment integrity. |
| Mechanical impingement by jets, flying debris, insulation. | These are phenomena common to design-basis severe accidents, and are addressed in the layout, by provision of barriers, and in the design of the ECC sumps. |
| Hot gases, fires | To be evaluated |
| Vacuum | Coolers, sprays can induce vacuum for certain scenarios, induce structural loads. |

In the preliminary stages of the evaluations of the CANDU 9 severe accident mitigation capabilities, only the first two source terms are explicitly considered. The effects of some others are covered by the successful actions of the mitigating systems or dealt with in later analyses.

## 2.3 ACCEPTANCE CRITERIA

The following acceptance criteria are used in the preliminary assessments :

1) The maximum containment pressure is lower than the containment failure pressure for up to 24 hours after the onset of a severe accident. (This analysis uses $P_{max} \leq 450$ kPa (g), the pressure below which the steel liner stays intact; "true" containment failure pressure calculations pending.)
2) The hydrogen concentration remains below the limits for deflagration (a conservative value of 9.0% by volume is used in this analysis) in any given volume of the containment.
3) The maximum pressure/temperature/radiation field at containment seals, penetrations and doors are below the failure limits for the seals and the containment, whichever is lower.
4) The long-term heat removal capacity within the containment must exceed all heat sources such that conditions 1 and 3 are met.
5) The debris has adequate area to spread in the reactor vault ( a lower limit of 0.02 $m^2$/MWT debris spread area is targeted for some reactors -Reference 5) and any debris in the reactor vault are covered with water.

## 2.4 SEVERE ACCIDENT MITIGATING SYSTEMS

### 2.4.1 MODERATOR AS A HEAT SINK

In certain severe accident scenarios, the fuel channels are intact, hot and voided. This can occur, for example following a loss of primary coolant and a failure to initiate emergency core cooling. The moderator surrounds all the channels and removes the decay heat and the metal-water reaction heat from the hot channels (Figure 2). If moderator cooling is available or if any moderator inventory loss can be replenished in a timely manner, this core damage state can be maintained indefinitely. Various analyses (References 2, 3) have shown that the channels maintain their integrity as long as they remain submerged in the moderator. While the channel integrity is maintained, the fuel sheaths will fail and the bundles will slump in most locations, depending on the timing of the accident sequence. Even for the worst case (steam flow chosen to maximize the metal-water reaction in each channel), fuel remains below its melting temperature and only partial sheath melting is predicted.

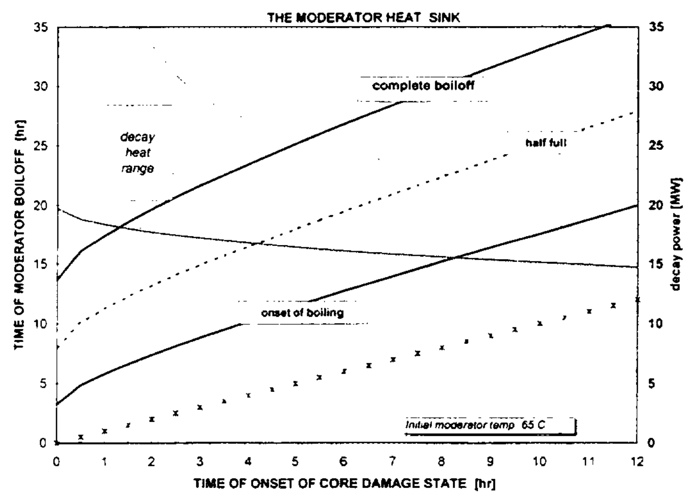



**Figure 6: Moderator boiloff by submerged channels** *[Example: If the moderator heatup starts at 7 hours after trip, it begins to boiloff at about 13.5 hours and is completely boiled off by 28 hours after reactor trip; during this period the decay heat is between 23 and 16 MW]*

If the moderator cooling is lost, the moderator begins to heat to saturation and then to boil. Pressure inside the calandria is relieved by rupture discs and/or by relief valves. After about three or four rows of channels are uncovered, the channels begin to fail by thermo-mechanical loads. The debris falls progressively into the moderator which eventually boils away. Conservative estimates of moderator boil-off time due to heat from the submerged channels is shown in Figure 6; the operator typically has many hours to replenish the moderator. This is a straightforward operation, consisting of opening the valves from the elevated Reserve Water Tank to the moderator, and refilling it by gravity. If the operator does not do this, estimates of moderator boiloff by debris collapse are shown in Figure 7; the debris collapse into the moderator can boil off a significant portion of the remaining moderator and induce a high steam surge load.

The calandria vessel over-pressure protection is provided by relief valves in the cover gas system and rupture disks at the end of four large pipes on top of the calandria vessel. The over-pressure protection system is designed to assure structural integrity of the calandria vessel (CV) against increase in pressure caused by in-core rupture of a channel, or loss of moderator cooling at full power. The rupture disk burst pressure is of course higher than the relief valve opening pressure. These relief systems also mitigate over-pressure in the calandria in severe core damage sequences, for example steam boil-off in a LOCA/LOECC/loss of moderator cooling triple failure.

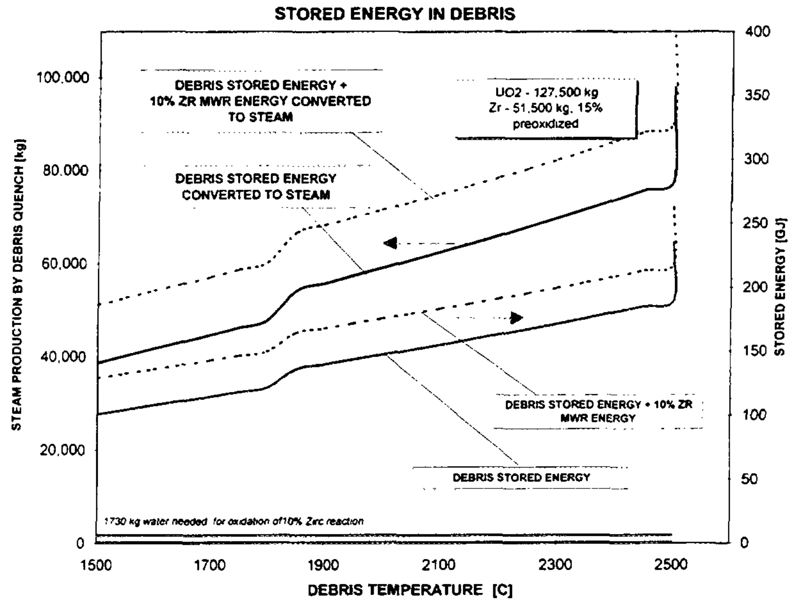



**Figure 7: Moderator boiloff by debris** *[ Example: Debris at an average temperature of 1900° C have a stored energy of about 140 GJ, enough to boiloff about 56 Mg of water by quenching]*

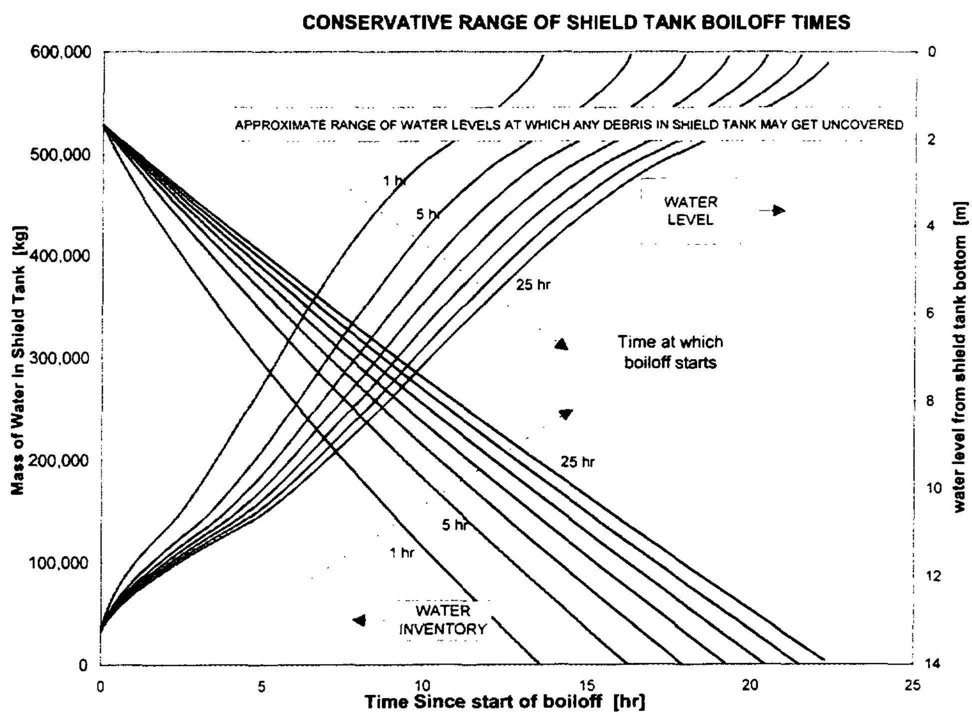



**Figure 8: Shield tank boiloff times as a function of time at which boiloff starts (CDS 2A)** *[ Example: If the Shield tank water begins to boil at 5 hours after reactor trip, its level drops to about 2m at after another 13 hours and is depleted by just over 16 hours after onset of boiloff]*

### 2.4.2 SHIELD TANK AS A HEAT SINK

In certain severe accident scenarios progressing to severe core damage, dry hot core debris may lie at the bottom of the calandria vessel (Figure 3) with decay heat removal by the shield tank water. This core damage state can be maintained as long as the shield tank water ( ~530 Mg.) surrounds the debris. Figure 8 shows the shield tank water boil-off estimates as a function of onset of the boil-off (which determines the decay power). Even without credit for the shield tank cooling system, which can remove about 0.3% of full power, the operator typically has more than 10 hours before the water level in the shield tank falls below the level of debris in the calandria vessel ( ~2m, see Figure 3). Replenishing the shield tank water is likewise a straightforward operation, consisting of opening the valves from the elevated Reserve Water Tank to the shield tank and refillling it by gravity. A shield tank over-pressure protection system prevents shield tank over-pressurization and allows decay heat to be released as steam to containment.

### 2.4.3 EXTERNAL FLOODING OF THE SHIELD TANK

Consider a more extreme case of Section 2.4.2 (Shield tank as a heat sink): In the unlikely scenario that the shield tank water is also lost (failure of operator to replenish the tank, or a break in the tank), debris may melt through the calandria vessel and end up in the shield tank. By this time the major liquid inventories (HTS, moderator and the shield tank water) are mostly on the reactor building floor (even in absence of coolers the majority of water is predicted to rain out) and along with potential contributions from ECCS and the Reserve Water Tank, flood the outside of the shield tank (see Figure 4) - there is no basement beneath the reactor as on operating CANDU plants. Thus, the CANDU 9 containment layout permits one more level of defence against vessel melt through by debris. The reactor centre line is at an elevation of 7.3 m from the basemat floor. With an external shield tank diameter of 13.3 m, the distance from the floor to the bottom of the shield tank is only 65 cm. The water level depends on the accident sequence, but with the ECC and RWT water inventories, can be as high as 2.5 m. The operator can also manually dump the RWT inventory on the floor to facilitate shield tank flooding.

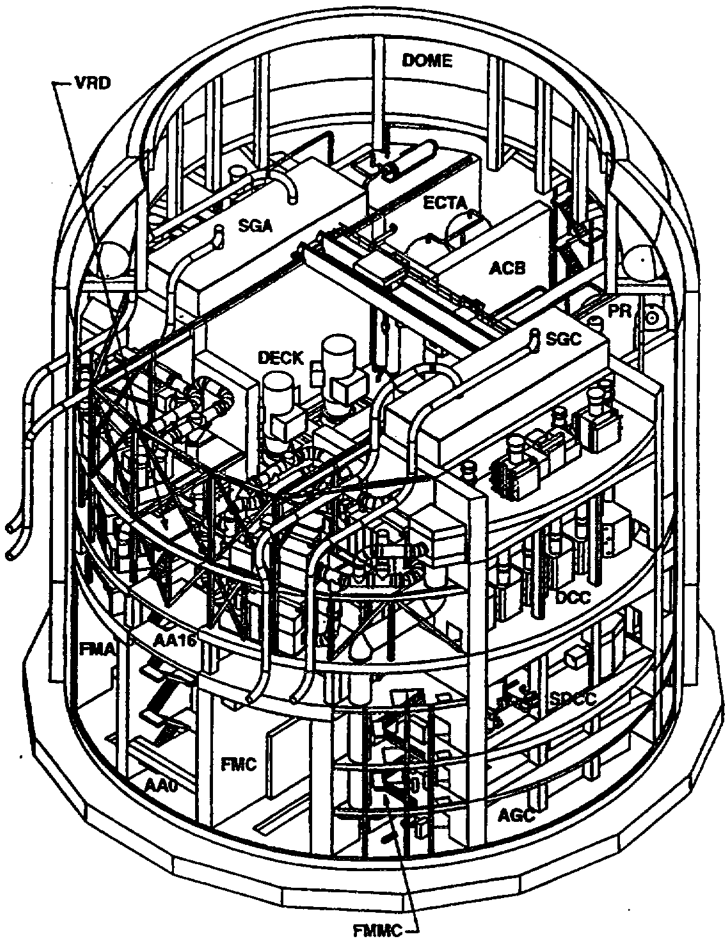
### 2.4.4 RESERVE WATER TANK FOR SEVERE ACCIDENT MITIGATION

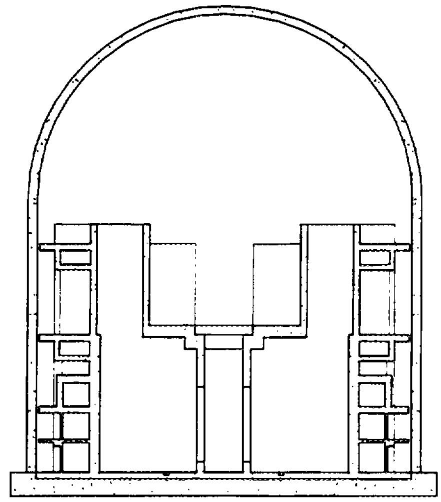
The Reserve Water System is a CANDU 9 innovation with significant accident mitigation capabilities. It is a passive, backup gravity-fed light water supply system that requires no pumps to deliver its inventory to critical locations. It consists of the Reserve Water Tank, located at a high elevation in the reactor building, and piping connections, with remotely actuated isolation valves, to the shield tank, HTS, calandria, Steam Generators and ECCS. The total capacity of the RWT is about 2500 $m^3$ and it can be replenished from the reactor building sump by two 100% recovery pumps. Injection from the Reserve Water Tank is initiated by the operator.

## *2.5 CONTAINMENT DESIGN FEATURES FOR SEVERE ACCIDENT MITIGATION*

### 2.5.1 CONTAINMENT LAYOUT

The general containment building schematics and equipment layout is shown in Figure 9. The CANDU 9 reactor building is a large dry containment, made of pre-stressed concrete with a full internal steel liner. It has a flat circular cylindrical base slab, it is 57m in internal diameter and it has a 42 m high circular perimeter wall, topped by a hemispherical dome for a total ceiling height of about 72m. The thickness of the exterior perimeter walls is 1.5m, with the dome wall thickness varying from 1.5 to 1.0 m. The internal structures are supported on the base slab and impose no load on the perimeter walls or the dome. The building is designed to maximize human access during operation to test, repair or replace components while minimizing radiation exposure. Areas containing potential heavy water leakage sources (reactor vault, fuelling machine vaults, steam generator enclosures, moderator pump and heat exchanger rooms and the shutdown bleed cooler area) are inaccessible while the reactor is on power and have a separate, controlled atmosphere. Blowout panels connect these sub-volumes in an accident to prevent local over-pressure and to ensure hydrogen mixing.



Cross-section across the reactor.

**124,000 $m^3$ free volume**

**57m I.D**
**71m High**
**1.25-1.5m wall thickness**
**Inner steel liner**

**Figure 9: A schematics of the containment layout**

The containment net free volume is one of the largest in the world, estimated to be about 124,000$m^3$ with an estimated equipment volume of about 11000$m^3$. The large net free volume limits the rate of rise in internal pressure and global hydrogen concentrations in accidents.

Ground-level openings between various structures and rooms allow for the unimpeded spreading of any water spilled in the reactor building. The openings between the reactor vault and the fuelling machine vaults extend to above the flood level. Special high level openings have been designed in the reactor vault walls near the feeder cabinets to relieve steam and gases released in the vault and to promote natural circulation in the absence of forced circulation by air coolers.

## 2.5.2 CONTAINMENT STRUCTURES AS HEAT SINKS

The large internal metal and concrete surfaces (over 50,000 $m^2$) provide significant heat sinks following release of steam into the containment. These estimates do not include floor and ceiling surface areas. They also help remove fission product aerosols by condensation, an effect not yet credited.

## 2.5.3 REACTOR VAULT DESIGN FOR SEVERE ACCIDENT MITIGATION

As noted previously, the reactor vault layout allows progressive containment of debris from a severe core damage accident in CANDU 9: first the calandria vessel, then the shield tank. In the most unlikely event that the debris melts through the shield tank and pours onto the floor, it will always fall into a pool of water (Figure 5) . The reactor vault has ample wall openings (> 30 $m^2$) to preclude pressurization of the vault. The large floor area of the CANDU 9 reactor vault (~ 116 $m^2$) is conducive to debris melt spreading: the floor area corresponds to about 0.041 $m^2$/MW of initial core thermal power. Guidelines for advanced reactor designs suggest a design target debris spread area of 0.02 $m^2$/MW rated thermal power (Reference 5) (higher values are better).

Severe accidents develop because the process and safety heat sinks become unavailable. In most cases that means that the water inventories that can potentially remove heat are either unavailable or are discharged into the containment by breaks or boiloff. Calculations for containment flood level estimates show that the basemat level surface area is about 1803 $m^2$. Table 1 lists the water inventories of major reactor systems. Also shown are the individual contributions of the various sources of water on basemat flood levels. If all the water from the HTS, calandria, shield tank, ECC tanks and the Reserve Water tank should end up on the floor, the flood level may reach about 2.5 m, enough to cover the lower portion of the shield tank. This estimate does not include potential fluid loss from non-seismically qualified systems (estimated to contain about 300 $m^3$ of water) and any losses from the feedwater system (estimated to contain about 2000 $m^3$ of water). While these two sources can add another 1.23m to the flood level, it is noted that actual flood levels are scenario specific and a simple addition of contribution from all service water, process and safety systems cannot be made. Further scenario specific calculations are pending.

**Table 1: Fluid inventories**

| SOURCE | MATERIAL | Volume [$m^3$] | LEVEL [m] |
|---|---|---|---|
| HTS | $D_2O$ | 363 | *0.22* |
| SG SEC. SIDE | $H_2O$ | 327 | *not credited* |
| MODERATOR | $D_2O$ | 307 | *0.17* |
| END SHIELD | $H_2O$ | 20.6 | *0.01* |
| SHIELD TANK | $H_2O$ | 529 | *0.29* |
| ECCS | $H_20$ | 680 | *0.38* |
| RESERVE WATER TANK | $H_2O$ | 2500 | *1.39* |

## 2.5.4 CONCRETE COMPOSITION TO MINIMIZE CORE CONCRETE INTERACTIONS

Core-concrete interactions are sensitive to accident specific details such as corium composition and attack characteristics and concrete properties, etc. However they are precluded by the various barriers described above. Nevertheless, the composition of concrete in the reactor vault (floor and lower sections of walls) is being optimized to minimize non-condensable gas production by interaction with solid and molten corium.

## 2.5.5 COOLERS FOR LONG TERM PRESSURE SUPPRESSION

Containment coolers and the ventilation system provide air cooling, exchange and distribution and maintain the containment pressure slightly sub-atmospheric under normal operation. The heat removal from the inaccessible areas is by two banks of ducted containment air coolers, each equipped with 4 air coolers on each side of the reactor and designed to maintain a temperature lower than the maximum permissible for equipment and concrete. These ducted air coolers draw air from the top of the steam generator enclosures (Figure 10), isolated from the dome area by blow-out panels, and discharge cool air at the bottom of the fuelling machine vault and reactor vault at two different elevations, thus ensuring good mixing. A common, separate environment is thereby maintained in the reactor vault, fuelling machine vaults, feeder cabinet areas and steam generator enclosures. The four unducted air coolers, under the dome, cool the air in the accessible areas.

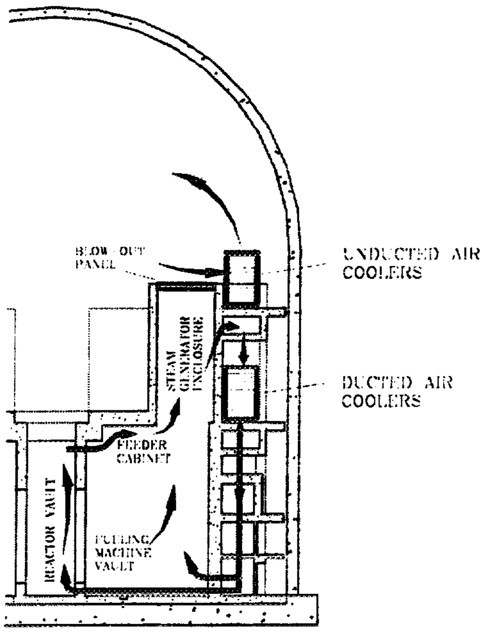



Figure 10: Air flow patterns in one half of the reactor under normal operating conditions

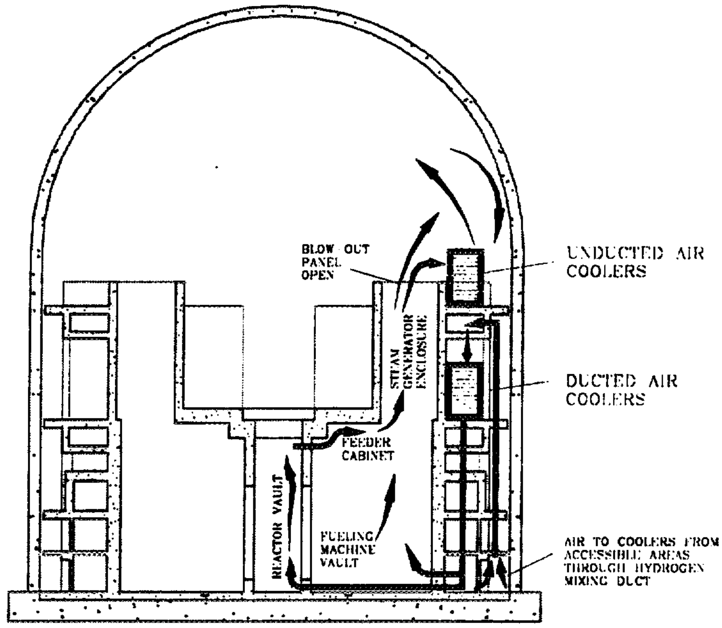



**Figure 11: Containment air flow patterns after an accident** *[Flow patterns in the other half are identical]*

In event of a loss of coolant accident, a high pressure or high activity signal on either of two fully independent containment isolation systems isolates the containment by the four valves in series in each of the ventilation duct penetrations in the reactor building wall. The blow-out panels and dampers between the steam generator enclosures and the accessible areas open and all coolers become available for containment cooling (Figure 11). In order to enhance mixing in the air environment, coolers begin to draw relatively cool and clean air from the accessible areas through the hydrogen mixing ducts located in opposite sides of the reactor building. The air coolers are important in evaluation of severe accident consequences. Enhanced heat removal capabilities of the air coolers under conditions of high humidity are well documented. Analytical models have been developed to compute heat removal capabilities under a wide range of steam concentrations and ambient temperatures. Analyses are underway to establish long term cooler survival and functionality under accident conditions.

## *2.6 HYDROGEN MITIGATION SYSTEMS*

During the course of a severe accident, hydrogen may be produced, in the short-term, within the fuel channels and by debris in the calandria vessel, shield tank or the reactor vault. Longer term sources of hydrogen include radiolysis, corrosion and core-concrete interactions. A large scale release of energy associated with hydrogen (and carbon monoxide) deflagration can pose potential threats to containment and equipment integrity. Efforts are underway to identify all short and long term sources of hydrogen and to mitigate them.

During severe accidents in CANDU reactors, concentrations of hydrogen build up slowly and reach combustible values over many hours. Therefore, the potential for hydrogen deflagration (at 9-10% volumetric concentration of $H_2$ at low steam concentrations) can occur only if spatial concentrations are allowed to build up. The containment layout precludes pockets or regions where hydrogen can accumulate. Hydrogen distribution in containment, in the absence of forced circulation, is governed by diffusion, condensation and natural circulation processes. The containment layout is conducive to natural circulation of gas mixtures in absence of forced circulation. Since these processes are not easily quantifiable, hydrogen mitigation systems that allow recombination or early ignition and burning are provided in various locations within the containment. Specifically the containment is equipped with both igniters and recombiners, to ensure that the hydrogen concentration remains below the critical value (9-10%) for deflagration. The design has been done before the confirmatory three-dimensional transient calculations, a technology which is only now available; the modelling of the hydrogen mitigation systems and estimates of hydrogen source terms is underway.

The igniters are made available from the onset of the accident to instigate local burns as soon as the local hydrogen concentration exceeds the ignition threshold (4-6 volumetric %, depending on the steam concentration). Igniters are not a panacea; their drawbacks include: a) their inability to operate in steam inerted environments, i.e., at steam volumetric concentrations > 55%, so that they may operate in some cases only after the steam has condensed and thus potentially initiate deflagration at high $H_2$ concentrations; and b) the potential to initiate deliberate ignition in a room with unknown, high hydrogen concentration.

The catalytic recombiners work over a wide range of hydrogen concentrations (from ~2%) and are unaffected by steam concentrations. They are a long-term hydrogen mitigation system and their $H_2$ removal capacity is of the order of tens of kg/hr per unit (typically 3-4 $m^3$ in size). Required and available recombination rates are generally small, and it would take hours to days to effect a measurable change in the containment hydrogen concentration. The placement of recombiners will be reviewed, once a detailed hydrogen distribution analyses for a range of severe accident scenarios is performed.

## 2.7 SEVERE ACCIDENT SPECIFIC INSTRUMENTATION

While some of the normal plant operation instrumentation can help ascertain the accident progression and reactor state, additional dedicated instrumentation is provided (12 qualified temperature, pressure, humidity and radiation monitors) to help identify the reactor state under accident conditions. Generic requirements for special instrumentation, dedicated to the monitoring of severe accident progression, and capable of surviving the anticipated harsh environment and operating in the range of anticipated extreme conditions are being developed as a part of this assessment. Further evaluation of the adequacy of the current post-accident monitoring instrumentation is planned.

# 3. CONTAINMENT RESPONSE TO STYLIZED SEVERE ACCIDENT CHALLENGES

In lieu of detailed analyses of core disassembly process for specific sequence of events, the containment response to a series of stylized loads is examined. Some sample results are presented here. In an initial simple simulation, the containment is subjected to a constant steaming load, representing any of the following core damage states:

- Hot dry intact channels (CDS-1A) submerged in boiling moderator
- Hot dry debris in calandria vessel (CDS-2A) with boiling shield tank water
- Hot dry debris in shield tank (CDS-3A) with boiling outside the submerged shield tank
- Submerged debris (CDS-4A) on reactor vault floor

In all these cases, it is conservatively assumed that all decay heat goes into boiling and that the initial pressure in the containment is atmospheric. The effect of pressure spikes due to the initial break and later quenching of fuel or debris is considered separately. The containment dome pressure and temperature transients are shown in Figure 12 and Figure 13 for a constant steam injection rate of 12.6 kg/s, corresponding to a constant decay power of 1%, typical during the long time-scales for CANDU. With 6 ducted air coolers operating, the containment over-pressurization is limited to less

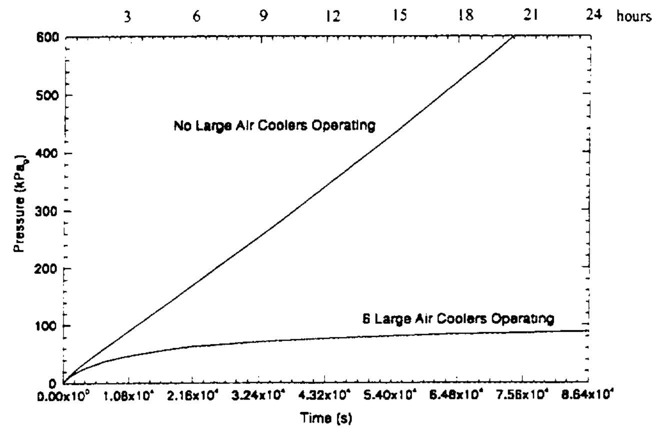



**Figure 12: Containment dome pressure transient for constant steam load at 1% decay power.**

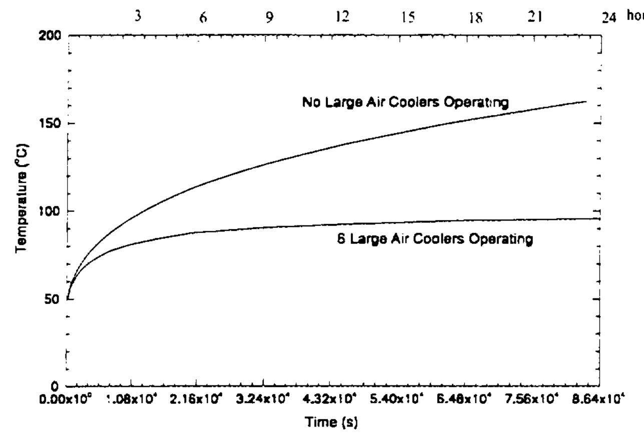



**Figure 13: Containment dome temperature transient for constant steam load at 1% decay power.**

than 100 kPa(g) and the temperature does not exceed 100° C. Without coolers, the containment pressurizes to the reference pressure of 450 kPa(g) after about 15 hours, at which time the gas temperature (Figure 13) reaches over 140° C. It is obvious that the coolers are important in the long term.

In another stylized scenario (Figure 14), a source of steam, consistent with decay heat production, is introduced into the containment at 15 minutes after reactor trip to simulate steam production from: Hot dry intact channels (CDS-1A) submerged in boiling moderator; or Hot dry debris in calandria vessel (CDS-2A) with boiling shield tank water. A subsequent 1800° C debris quench at 23 hours simulates:

- •Reflood of hot dry fuel in intact channels (CDS-1B) or core collapse into the moderator at 23 hours, or
- •Reflood of hot dry debris in the calandria vessel (CDS-2B) or debris melt through into the moderator at 23 hours, or
- •Reflood of hot dry debris in the shield tank (CDS-3B) or debris melt through onto the reactor vault floor at 23 hours, or
- •Debris dropping onto the reactor vault floor after melt-through of the shield tank (onset of CDS-4A) at 23 hours

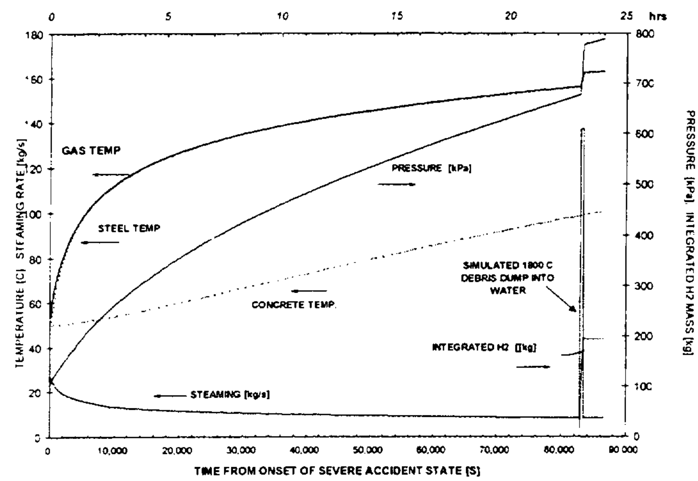



**Figure 14: Containment transients for early steaming and late debris quench.**

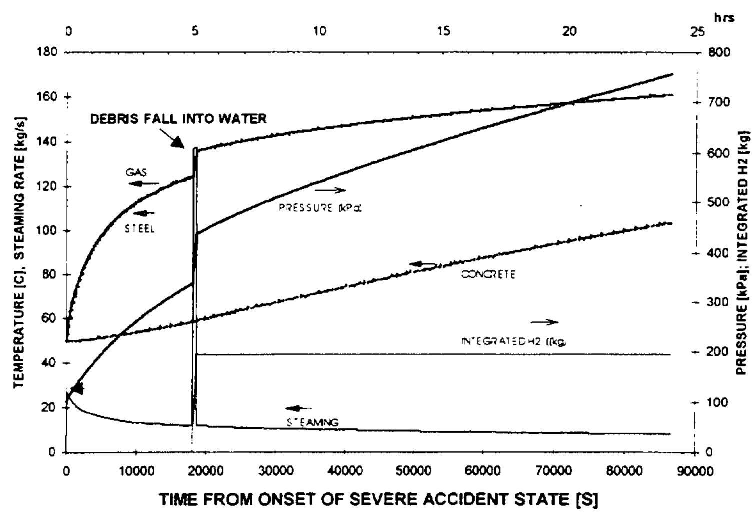



**Figure 15: Containment transients for early steaming and early debris quench**

A 10% oxidation of the un-oxidised Zircaloy is assumed to accompany the debris quenching process (to simulate the hydrogen source term). The containment coolers are assumed to be out of service. The containment response plotted in Figure 15 assumes debris quenching at 5 hours, instead of 23. Containment over-pressurization (pressure > 450 kPa (g)) occurs at about 10 hours instead of 15 due to early pressurization of containment by debris quenching at 5 hours. The debris quench contributes about 100 kPa to the containment pressure in both cases- i.e., the timing of debris quench has little effect on containment pressurization.

An operator action to reflood the debris at 5 hours, followed by restoration of cooling (termination of steaming from debris) is modelled. The containment response is plotted in Figure 16. The debris quench again contributes about 100 kPa to the containment pressure. The containment starts to depressurize as steam injection into it is terminated by operator action and the containment structures become the dominant long-term heat sinks. Containment response following a similar stylized severe accident following an early LOCA is presented in Figure 17. In this case, a steam surge by debris quench at 20 hours is simulated. In all cases the containment coolers are assumed to have failed at the onset of the accident. With coolers operating, containment over-pressurization is avoided with anticipated response similar to that in Figure 12.

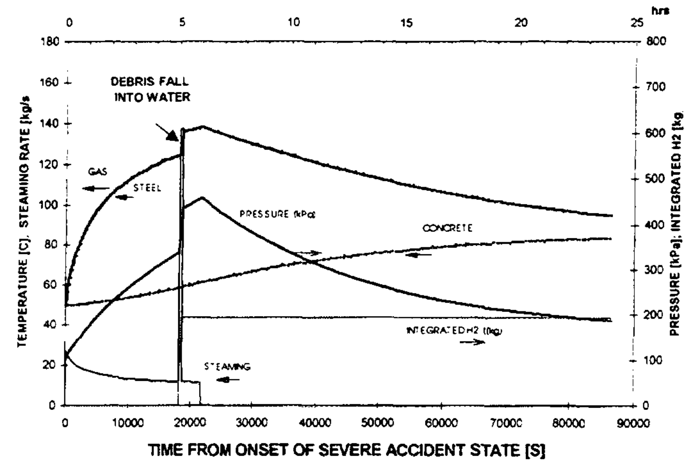



**Figure 16: Containment transients for early steaming, debris quench and accident termination**

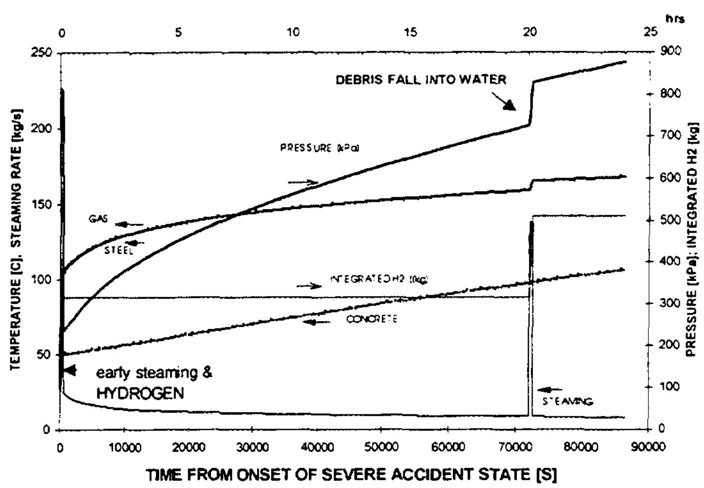



**Figure 17: Initial LOCA followed by core disassembly.**

The containment response to the stylized containment loads (typical of many severe accident sequences) presented above illustrates:

1. The coolers are effective in limiting containment pressurization,
2. If they fail, ample time is available to restore operation of the coolers, or initiate an alternative means of providing cooling or pressure suppression (now under review).
3. Overall containment pressurization is insensitive to the exact timing of events for a given class of severe accidents
4. Containment pressurization rate is relatively slow.
5. Structural heat sinks offer significant heat removal capability.

## 4. CONCLUSIONS

The CANDU moderator and shield tank water volumes provide unique severe accident mitigation capabilities. The reserve water tank in CANDU 9 affords additional time to arrest severe accident progression. Preliminary results confirm that containment air coolers are effective in avoiding containment failures for the whole range of accident progression pathways. Other features of the CANDU 9 containment include:

- The large CANDU-9 containment and the equipment layout results in large, open volumes with good potential for natural circulation and no apparent hydrogen traps.
- The pre-stressed concrete boundary with a steel liner results in high failure pressure.
- The large structural heat sinks significantly augment heat, humidity and fission product aerosol removal from the containment atmosphere by the air coolers.
- Reactor building flooding levels permit external cooling of debris in the shield tank and provide an extra boundary to arrest severe accident progression.
- Hydrogen mitigation systems allow systematic and timely dispersion and reduction of hydrogen.
- The reactor vault concrete floor composition and geometry minimize core-concrete interactions in the most unlikely event of debris arriving at the reactor building basemat.
- Instrumentation is provided for measurements and control under severe accident conditions

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## FINDINGS OF THE WORKING GROUPS

### 1. DESIGN REQUIREMENTS AND DESIGN MEASURES FOR COPING WITH SEVERE ACCIDENT PHENOMENA AND CHALLENGES

This section documents the results of Working Group 1. Each organization was asked to provide summaries of the design requirements in their country with regard to severe accidents, and to describe the design approach taken to cope with severe accident phenomena and challenges. The observed commonality in the design philosophy/approach taken by the various design organization represented at the TCM is also summarized in this section.

#### 1.1. Country-by-country summary

*1.1.1. Finland - Regulatory requirements with respect to severe accidents*

The requirements of the Finnish regulatory body Center for Radiation Safety (STUK) for severe accidents are summarized in Table 1. The requirements are mainly from YVL Guide 1.0 General Safety Criteria for the design of nuclear power plants[1]

The general requirement in the YVL Guide 1.0 is that the containment shall withstand the temperature and pressure loads in severe accidents. The release limit is 100 TBq of Cs-137 and no acute health effects to populations are allowed. The probability of exceeding these limits shall be extremely low. The limits are set in the Council of State Decision 395/91.

*1.1.2. France/Germany - The European Pressurized Water Reactor (EPR)*

(a) General design requirements

- Documents issued by the French and German safety authorities (on going process):
  - GPR/RSK proposal for a common safety approach for future PWR - May 25, 1993.
  - GPR/RSK recommendations concerning the common French - German safety approach for future PWRs - January 31, 1995.
- French and German Utilities requirements, generally translated in the European Utilities Requirements (EUR).
- Regulations, codes and standards applied in France and Germany presently considered in the internal designers' rules, specifying harmonized positions between French and German companies: These rules are described in specific documents, known as the "EPR Technical Codes" (ETCs), which will be submitted to the assessment of the French and German safety experts at the end of the Basic Design phase.

[1] The requirements have, in part, been developed for the revisions to the Loviisa plant.

TABLE 1. SUMMARY OF REQUIREMENTS OF FINNISH REGULATORY BODY (STUK) FOR SEVERE ACCIDENTS

## High Pressure Melt Ejection (HPME) And Direct Containment Heating

| Regulatory requirement in Finland | Objectives | Comments |
|---|---|---|
| HPME shall be considered in the design of containment. | Containment shall survive the loads caused by HPME. | No direct open pathway from reactor cavity to other containment compartments must exist. |
| Primary system shall have a system to reliably reduce the system pressure. | Reduce the pressure below 2 MPa. | Plant systems shall be designed both to survive HPME and prevent HPME. |
| The pressure relief valves shall remain permanently open. | Prevent repressurization. | |

## Hydrogen Production

| Regulatory requirement in Finland | Objectives | Comments |
|---|---|---|
| In the design of containment, 100% oxidation of core materials, which are readily oxidized, shall be assumed. | Early release is prevented. | Containment design pressure and volume shall be large enough to accommodate extensive hydrogen production. This requirement in fact determines containment volume (Containment design pressure and volume are not determined only by DBA LOCA). |
| No need for containment venting shall exist and containment integrity must not be threatened. | | |
| Hydrogen burns and explosions shall be considered in the design of containment. | Containment shall survive the pressure and temperature loads in severe accidents. | Double measures against hydrogen burns and detonation. |
| Formation of uncontrollable burnable or detonation of gas mixtures in containment shall be prevented. | Prevent energetic combustion phenomena in containment atmosphere. | Containment survival and prevention of burns and detonation. |

TABLE 1. (Cont.)

Steam Explosions in Containment

| Regulatory requirement in Finland | Objectives | Comments |
|---|---|---|
| Steam explosions and other energetic fuel-coolant interaction phenomena shall be considered in the design of the containment. Pressure and temperature in severe accidents must not lead to uncontrollable failure of containment. | Prevention of early catastrophic failure of containment. | Dynamic loads of steam explosions shall be studied in the design of the containment. Steam explosions cannot be excluded at this stage of knowledge. |
| Containment shall survive the dynamic loads from possible steam explosions. | | |

Core-concrete Interaction in The Containment

| Regulatory requirement in Finland | Objectives | Comments |
|---|---|---|
| It shall be shown with high confidence that core debris is coolable in containment. | Prevent basement melt-through. Show that severe accident can be stabilized in containment. | Melt coolability is not yet definitely shown experimentally, MACE and RIT experiments going on. |

Containment Long-term Heat Removal

| Regulatory requirement in Finland | Objectives | Comments |
|---|---|---|
| Filtered venting must not be the primary means to control containment pressure and temperature. | The releases shall be reduced as much as possible. Zero release is the target. | Containment cooling system for severe accidents is required. Passive system is preferred implicitly in the YVL Guide 1.0. |
| Containment shall be equipped with a filtered venting system to release the non-condensable gases. | Reduce containment pressure at long term. | |

TABLE 1. (Cont.)

## Chemical Phenomena in Containment

| Regulatory requirement in Finland | Objectives | Comments |
|---|---|---|
| Chemical phenomena under severe accidents shall be taken into account in the design of containment systems. | Prevent formation of exploding oxygen-hydrogen mixture in containment. | Hydrogen control systems required: inertisation, recombiners, igniters. |
| Radiolysis and interaction of core debris with containment materials are primary concerns. | Keep pH of containment water pools >7 to prevent formation of gaseous iodine and organic iodine compounds. | Containment pH control system required. |

## Recriticality

| Regulatory requirement in Finland | Objectives | Comments |
|---|---|---|
| Core and core debris shall be kept subcritical. | To prevent core power from exceeding capacity of containment heat removal systems. | Boron system capacity shall be large enough to achieve subcricality in total containment water volume, not only in primary system. |
| Instrumentation for neutron flux measurement under severe accident shall be available. | Detected recriticality. | |

## Primary System Pressure and Coolant Control

| Regulatory requirement in Finland | Objectives | Comments |
|---|---|---|
| Coolant relief outside primary system during normal operation is allowed only at the initial stage of the transient. | Keep coolant inside closed systems. | In BWRs, relief to condensation pool is no more allowed to remove decay heat from core. The requirement implies a need for passive primary system isolation condenser. PWRs can remove decay heat via steam generators. |

(b) Specific requirements with respect to severe accidents (as specified by the French and German safety authorities):

- One of the major objectives of the French and German safety approach for new reactors is the further strengthening of accident prevention through optimization of defence in depth. This is similar to those indicated in other countries for evolutionary reactors. With reference to INSAG-3, it is intended that the improvements in defence-in-depth should lead to global core melt probability of less than $10^{-5}$ per plant operating year.

- Regarding severe accidents the objective is the "practical elimination" of accident situations resulting in considerable off-site consequences. In particular, accident situations which would lead to large early releases, such as high pressure core melt, global hydrogen detonation, or containment bypass, have to be practically excluded. When they cannot be considered as physically impossible, design provisions have to be taken to "design them out".

- Low pressure core meltdown accidents have to be "dealt with", so that the associated maximum conceivable release would necessitate only limited protective measures in area and time. This would be expressed by no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, and no long-term restrictions in the consumption of food. For performing the assessment of the containment, the intervention levels proposed by ICRP 63 (for evacuation and relocation) and the EU limits (for food commercialization) can be used as references.

- Such small radioactive releases imply a very good performance of the containment, considering the low pressure core melt situations. For example:

  - there shall be no path of direct leakage from the containment building to the outside;
  - the design pressure and temperature of the containment inner wall must be such to allow a grace period of at least 12 hours without containment heat removal and to ensure its integrity and leak-tightness even after the global deflagration of the maximum amount of hydrogen which could be contained in the containment building in the course of low pressure core melt accidents;
  - the residual heat must be removed from the containment building without venting device;
  - the penetration of corium through the basemat of the containment building must be avoided.

Table 2 presents the EPR strategy for coping with severe accident phenomena and challenges.

*1.1.3. Germany - The SWR-1000 (BWR)*

(a) General design requirements:

- Reduction of the integral core melt frequency by use of passive systems for accident control in combination with a reduced number of active systems

TABLE 2. EPR STRATEGY FOR COPING WITH SEVERE ACCIDENT PHENOMENA OR CHALLENGES

| Phenomena and challenges | Strategy | Comments (*) |
|---|---|---|
| Phenomena associated with high pressure core melt (HPCM) | These phenomena are practically eliminated by: | |
| | 1. use of *highly reliable existing systems complemented* by use of bleed and feed of the primary system to prevent HPCM occurrence. | - *Criterion: reactor coolant system* (RCS) pressure $<2.0$ MPa (20 bar) at reactor pressure vessel (RPV) melt through. |
| | 2. transferring a potential HPCM situation to a low pressure core melt (LPCM) situation before RPV melt through. | - Basic solution is to use existing pressurizer relief valves; the alternative solution is to use dedicated depressurization valve(s). The *solution including a dedicated depressurization valve is now selected.* |
| Phenomenon associated with low pressure core melt in the short term (at RPV melt-through<br>- loads on the RPV support and cavity wall. | Dealt with by design of the RPV support and walls. | |
| - loads due to ejected core melt: direct containment heating (DCH). | Practically eliminated by elimination of HPCM and by design of the reactor cavity. | |
| Phenomena associated with $H_2$ combustion<br>- Global detonation | Practically eliminated by sizing $H_2$ control means and containment volume. | Maximum $H_2$ concentration would be maintained $\leq 10\%$, or $\leq 13\%$ at inerted conditions. Use of recombiners. |
| - Global deflagration | Dealt with by the design of the containment. | The containment design pressure of 0.65 MPa (6.5 bar) takes into account deflagration of the maximum $H_2$ that can be in the containment, taking into account the recombination operation. |
| - Local phenomena (detonation, deflagration detonation transition (DDT), fast deflagration) | Practically eliminated by controlling $H_2$ local concentration, or dealt with by design of the structures. | Local $H_2$ concentration is maintained $\leq 10\%$.<br>Use of igniters where appropriate. The prestressed concrete containment, designed for 0.65 MPa (6.5 bar), provides significant margin before unacceptable leakage ~1.0 MPa (10 bar). |
| - Long term ignition | Practically eliminated by controlling $H_2$ generation and concentration. | Recombiners are uses to control $H_2$ concentration. Very limited $H_2$ generation at low rate due to the design of the spreading area and its protective layer. |
| Steam explosion<br>- In-vessel | Practically eliminated because physically not credible. | Some energetic interactions could happen, but not to a level endangering the containment. On-going R&D is carefully reviewed. |

TABLE 2. (Cont.)

| Phenomena and challenges | Strategy | Comments (*) |
|---|---|---|
| - Ex-vessel | Practically eliminated by design provisions. | (See above)<br>- The reactor pit and spreading area are kept dry.<br>- In the spreading chamber water covers the corium after spreading, with no significant amount of water being present before spreading. |
| MCCI and basemat melt through | Practically eliminated by design provisions protecting the basemat, even in the very long term. | The basic solution involves spreading the corium in a dedicated chamber of approximately 170 $m^2$. Water from the IRWST then covers the corium for cooling. The solution includes a protective layer on the basemat to prevent MCCI and a cooling network to limit the temperature of the concrete.<br><br>Note: There is currently no evidence of the efficiency of reactor cavity flooding to achieve in-vessel retention for such a large reactor. |
| Continuous release of mass and energy into the containment | Dealt with by designing the containment and implementing a dedicated containment heat removal system (CHRS). | - The containment volume and the design pressure of 0.65 MPa (6.5 bar) are selected to provide a 12 hour grace period before the necessity of CHRS activation.<br>- The CHRS is a spray system, with its active components being outside the containment.<br>- Venting shall not be used.<br>- Prevention of MCCI (see above) limits the amount of non-condensable gases. |
| Containment bypass | Practically eliminated by design measures. | The very stringent objectives specified with regard to off-site consequences impose a "zero bypass" design of the containment.<br><br>- All penetrations (in particular ventilation, hatches, air locks) are equipped with double seals and leak recovery systems.<br><br>- These leaks and the leaks of the containment inner shell are collected in the containment annulus and sent to the stack after filtration.<br><br>- All penetrations are in communication with peripheral buildings, providing another barrier against leaks to the outside. |
| Recriticality | Practically eliminated because physically not credible. | In addition, recriticality would not lead to phenomena impairing the overall approach. |

(*) *The description reflects the present EPR features. The "Basic Design" is still in progress, and modifications could be introduced.*

- Low core power density and large water inventories in the RPV as well as inside and outside the containment
- Power generation costs competitive to fossil-fuel plants
- Codes and Standards
    - Atomic Law
    - Ordinances i.e. for radiation protection
    - Principles, recommendations and comments of German authorities
    - RSK - Guidelines
    - KTA - Safety standards
    - DIN - Standards for nuclear technology
    - Company internal rules and specifications for industry and NPP operators
    - IAEA Safety Guides
    - EUR (European Utility Requirements for LWR-NPP).

(b) Specific requirements with respect to severe accidents:

- Core damage cumulative frequency $< 10^{-5}$ per year
- Core damage cumulative frequency, $< 10^{-6}$ per year exceeding the limiting release
- Control of a severe accident (core melt) in such a way that the consequences of the accident remain restricted to the plant, thus precluding the necessity for wide-scale emergency response actions in the vicinity of the plant such as evacuation or relocation
- Control of a severe accident by passive means
- Retention of the core melt in the RPV by flooding the RPV vicinity
- Containment design for a pressure buildup due to the hydrogen released by a 100% zirconium-water reaction of the core's zirconium inventory
- Prevention of hydrogen-oxygen reactions (deflagration, detonation) by nitrogen inertization of the containment.

Table 3 presents the SWR-1000 strategy for coping with severe accident phenomena and challenges.

*1.1.4. Japan - Advanced LWRs*

(a) General design requirements (Japanese utility requirements)

- Availability >90%
- Competitive with alternative power generating sources
- 60 years life for non-replaceable components
- Improvements in
    - plant status information system
    - outage time
    - working environment.

(b) Specific requirements with respect to severe accidents (Japanese utility requirements)

- Core melt $< 10^{-5}$/reactor-year
- Large release $< 10^{-6}$/reactor-year
- A containment design guidance document for the next generation LWRs is being prepared by Japanese nuclear industry with advise from academic and research

TABLE 3. SWR-1000 STRATEGY FOR COPING WITH SEVERE ACCIDENT PHENOMENA AND CHALLENGES

| Phenomena and challenges | Strategy: features & measures | Comments |
|---|---|---|
| Core melt at high pressure | There are highly reliable, redundant and diverse depressurization devices for the primary system:<br>- redundant and diverse safety relief valves with diverse pilot valves<br>- rupture discs<br>- emergency condensers for depressurization of the primary system to the pressure for coolant injection with the Low Pressure Coolant System (LPCI) systems | Core melt at high pressure is technically excluded |
| $H_2$ production in the RPV and release to the containment | The containment is inerted by nitrogen to prevent deflagration and detonation and is designed for a pressure buildup due to the hydrogen release by reaction of 100% of the zirconium with water | |
| Steam explosion in the RPV | Due to the internals in the lower plenum of the RPV (control rod guide tubes), the water volume is divided in many sub-volumes. Thus a steam explosion jeopardizing the RPV integrity is not possible | |
| Recriticality of the core material | As a consequence of core melt fissile material is mixed with absorber material without moderator, thus recriticality is excluded | |
| Containment bypass | All the systems connected to the RPV are located inside the containment except the main steam lines (MSLs) and the feedwater lines (FWLs). MSLs and FWLs are equipped with 3 containment isolation valves per line | |
| Steam explosion in the containment and molten core - concrete interaction | The core melt is retained inside the RPV by cooling the exterior of the RPV by flooding, thus steam explosion in the containment and molten core - concrete interaction is excluded | |
| Containment over pressurization | Containment pressure is limited by passive heat removal by containment condensers. After a grace period of more than 3 days, $H_2$ release and recombination is accomplished via the offgas system and its holdup filters | |

institutes. This document will provide detailed performance targets and Design Extension Conditions for the containment. This document will include the following in addition to the above:

- The total probability of early containment failure should be $< 10^{-7}$/reactor-year with a conditional containment probability in the event of a severe accident of less than 0.1
- Identification of challenges associated with core melt accident. Strategies to cope with these identified challenges are considered in the design phase in search for possible prevention and mitigation, but consideration of cut-off probability, when reasonable is allowed.

Table 4 presents the ABWR-IER strategy for coping with severe accident phenomena and challenges. Table 5 presents the next generation PWR strategy for coping with severe accident phenomena and challenges.

*1.1.5. Korea, Republic of - Korean Next Generation Reactor (KNGR)*

(a) General design requirements

- Korean Atomic Law and other regulatory rules
- Safety and regulatory requirements for ALWRs currently under development

(b) Specific requirements with respect to severe accidents

- Regulatory position: Severe accidents must be considered for the safety of nuclear power plants. No specific regulatory requirement has been established yet, however.
- Utility position: Severe accidents must be considered in the design to the extent that the safety objective and goals are satisfied, and early containment failure mechanisms must be addressed in the design in the first place to practically eliminate their threat to the containment integrity.
  - Safety objective: No imminent off-site measures in case of severe accidents for at least 24 hours.
  - Safety goals: Core damage frequency $<10^{-5}$/Reactor years
    Large radiation release frequency $<10^{-6}$/Reactor years
    - Radiation exposure: $\leq 100$ mSv/24 hours
    - Land contamination: $\leq 100$ TBq of Cs-137

- Specific design requirements from the utilities:
  - Containment integrity criteria: ASME factored load category with severe accident load conditions.
  - Systems must be provided to prevent early containment failures.
    - $H_2$ control device must be provided to maintain $H_2$ concentration below 10% with 100% active fuel clad oxidation.
    - Primary system depressurization means must be provided to prevent high pressure core melt ejection.
  - Containment bypass accidents must be addressed by preventive measures in the design:
    - Design must consider intersystems LOCA and multiple steam generator tube rupture (SGTR) accidents.
  - Means must be provided to cool the molten core and to retain it in the containment:

TABLE 4. STRATEGY FOR COPING WITH SEVERE ACCIDENT PHENOMENA AND CHALLENGES FOR ABWR-IER

[Severe Accident]

| Phenomena and challenges | | Prevention | Mitigation |
|---|---|---|---|
| HPME/DCH | | - Depressurize | - Containment load carrying capability* |
| Hydrogen generation | | ----- | - Containment load carrying capability for pressurization<br>- Inert<br>- Recombiner to cope with long term radiolysis |
| Steam *explosion* | In-vessel | ----- | ----- |
| | Cavity | - Dry cavity policy | - Cavity's load carrying capability (pedestal can withstand impulse) |
| Re-criticality** | | - Standby Liquid Control System (SLC System) | ----- |
| Core-concrete reaction | | - Pour water to dry cavity<br>- *Refractory material* | - Pour water to cavity |
| Breach of RPV bottom | | - In-vessel retention expected due to heat sink (CR guide tube, etc)<br>- pour water to RPV | - Containment load carrying capability even for DCH/HPME |
| Heat-up of containment environment & pressurization of containment (heat transfer from, and pressure reduction in containment) | | - Pour water through containment spray system<br>-PCCS *in addition to residual heat removal* (RHR) | - Containment load carrying capability |
| Source term leakage (fission product leakage) from containment | | - Pour water | - Containment spray<br>- Suppression pool scrubbing |

[Design Basis Accident]

| Phenomena and challenges | Prevention | Mitigation |
|---|---|---|
| Pressure/temperature increase | ----- | - Suppression pool<br>- Containment spray<br>- RHR |
| Hydrogen generation | ----- | - Inert<br>- Recombiner to cope with long term $H_2/O_2$ production by radiolysis |
| Source term leakage (fission product leakage) from containment | ----- | - Containment spray<br>- Standby Gas Treatement System (SGTS) |

* Analysis of *DCH loads* indicates a small probability of containment failure, given vessel failure at high pressure for ABWR

** Narrow time window is available for the core to be critical because fuel rods will remain for a limited time after melting of Control Roads and the probability of recovery and startup of core injection system within this narrow window is very low. In case this is possible the additional heat generation associated with recriticality and the effect of injection of borated water as a part of accident management strategies will be considered in the assessment of the containment heat removal capacity.

TABLE 5. STRATEGY FOR COPING WITH SEVERE ACCIDENT PHENOMENA AND CHALLENGES FOR NEXT GENERATION PWR

[Severe Accident]

| Phenomena and challenges | | Prevention | Mitigation |
|---|---|---|---|
| HPME/DCH | | - Depressurize | - Containment load carrying capability (large free volume)<br>- Labyrinth concrete structure |
| Hydrogen generation | | ----- | - Large free volume (low $H_2$ concentration)<br>- Ignitor (option) |
| Steam explosion | In-vessel | ----- | ----- |
| | Cavity | ----- | - Cavity's load carrying capability<br>- Labyrinth concrete structure |
| Re-criticality* | | ----- | ----- |
| Core-concrete reaction | | - Pour water to cavity | - Pour water to cavity |
| Breach of RPV bottom | | - IVR by ex-vessel cooling (pour water to cavity) | - Containment load carrying capability (large free volume) |
| Heat-up of containment environment & pressurization of containment (heat transfer from, and pressure reduction in containment) | | - Decay heat removal by RCS submergence and SG cooling<br>-External spray | - Containment load carrying capability (large free volume) |
| Source term leakage (fission product leakage) from containment | | - RCS submergence by pouring water | - Passive annulus filter |

[Design Basis Accident]

| Phenomena and challenges | Prevention | Mitigation |
|---|---|---|
| Pressure/temperature increase | ----- | - Horizontal steam generator<br>- External spray |
| Hydrogen generation | ----- | - Large free volume (low $H_2$ concentration) |
| Source term leakage (fission product leakage) from containment | ----- | - Passive annulus filter |

* Narrow time window is available for the core to be critical because fuel rods will remain for a limited time after melting of Control Roads and the probability of recovery and startup of core injection system within this narrow window is very low. In case this is possible the additional heat generation associated with recriticality and the effect of injection of borated water as a part of accident management strategies will be considered in the assessment of the containment heat removal capacity.

TABLE 6. KNGR STRATEGY FOR COPING WITH SEVERE ACCIDENT PHENOMENA AND CHALLENGES

| Phenomena or challenge | Solution | Prevention/ Mitigation | Comments |
|---|---|---|---|
| High pressure core melt ejection (HPME) | Primary system depressurization system | Prevention | Depressurization to the cut-off pressure by using depressurization valves. |
| $H_2$ generation | Glow type igniters | Mitigation | Passive type devices are under investigation |
| Steam explosion | Robust structure of cavity and reactor vessel support | Mitigation | |
| Source term | Containment spray system.<br>Annulus Heating and Ventilation and Air System (HVAC) (filtered vent to the atmosphere or recirculation to the annulus area) | Mitigation | |
| Re-criticality | Addressed by the reactivity control and shutdown systems | Prevention | Emergency cooling water contains neutron absorber |
| Basemat melt-through (MCCI) | Reactor cavity with sacrificial layer; also water is provided to cool the corium accumulated in the cavity. | Mitigation | Special concrete resistant to MCCI is being investigated. |
| Long-term containment pressurization | Containment spray system.<br>Containment fan cooling system. | Mitigation | Containment spray system is a safety grade systems used for DBA management. |
| Containment by-pass | Increase of design pressure of piping (e.g. piping of the shutdown cooling system and the CVCS) connected to RCS.<br>No atmospheric dump of steam in case of steam generator tube rupture (SGTR). | Prevention | |

Table 6 presents the KNGR strategy for coping with severe accident phenomena and challenges.

*1.1.6. Russian Federation - The WWER-1000 and WWER-640 designs*

(a) General design requirements

- IAEA-design code (NUSS), safety principles (INSAG-3, 5)
- Russian Federation nuclear energy laws and Gosatomnadzor regulations

(b) Specific requirements with respect to severe accidents
[ no information submitted ]

Table 7 presents the WWER-1000 strategy for coping with severe phenomena and challenges, and Table 8 presents the WWER-640 strategy for coping with severe accident phenomena and challenges.

*1.1.7. Sweden - BWR 90*

(a) General design requirements

- Design to meet licensing design basis, basically US NRC rules and General Design Criteria (GDC), + supplementary rules (30 minutes grace period for safety-related operator actions, 80 $cm^2$ hole in RPV, N-2 arrangement of safety systems).
- Design for flexible and reliable power operation with performance characteristics in correspondance with "Performance requirements of thermal power plants connected to the Nordic grid" by NORDEL - a co-ordination committee for the power generating utilities in the Nordic countries.
- Design for short refuelling outages (internal rules).
- Design for low occupational radiation exposure (proper material selection, water chemistry control, installation, accessibility, ventilation, etc.) (internal rules).
- Lowering core melt probability to less than $10^{-5}$ per year, in line with the expectations in INSAG-3 (internal rules).

(b) Specific requirements with respect to severe accidents

- Containment must be laid out to collect and contain possible molten material ejected from the RPV.
- Containment must be provided with over-pressure protection devices discharging through high-efficiency filters (filtered venting) to minimize releases to atmosphere and land contamination.
- Necessary support systems must be provided to enable controlling the situation for a number of hours. After that time, additional support systems, e.g., mobile water and electric power supply systems, may be credited.

Table 9 presents the BWR 90 strategy for coping with severe accident phenomena and challenges.

TABLE 7. WWER-1000 STRATEGY FOR COPING WITH SEVERE ACCIDENT PHENOMENA AND CHALLENGES

| Phenomena and challenges | Strategy | Comments (*) |
|---|---|---|
| Phenomena associated with high pressure core melt (HPCM) | These phenomena are practically eliminated by: | |
| | 1. Use of highly reliable existing systems complemented by use of emergency gas removal and bleed and feed of the primary system to prevent HPCM occurrence. | Basic solution: existing pressurizer safety valves and emergency gas removal system. |
| | 2. Transfer HPCM situation to LPCM before RPV melt through. | |
| Phenomena associated with low pressure core melt in the short term (at RPV melt through)<br>- loads due to ejected core melt (DCH) | Practically eliminated by elimination of HPCM and design of the core melt catcher. | |
| Phenomena associated with $H_2$ combustion.<br>- Global detonation | Practically eliminated by sizing $H_2$ control means and containment volume. | The containment volume is 60000 $m^3$. |
| - Global deflagration | Dealt with by designing the containment. | Containment design pressure is 0.5 MPa (5.0 bar). |
| - Local phenomena (detonation, DDT, fast deflagration) | Practically eliminated by controlling $H_2$ local concentration, or dealt with by designing the structures. | Use of igniters where appropriate. Prestressed concrete containment, with design pressure of 0.5 MPa (5 bar), provides significant margin before unacceptable leakage ~0.7 MPa (7 bar). |
| - Long term ignition | Practically eliminated by controlling $H_2$ generation and concentration. | Recombiners are used to control $H_2$ concentration. Very limited $H_2$ production is anticipated from MCCI due to the design of the core melt catcher. |

TABLE 7. (Cont.)

| Phenomena and Challenges | Strategy | Comments (*) |
| --- | --- | --- |
| Steam explosion<br>- In-vessel | To be eliminated by design provisions if necessary. | Careful follow-up of on-going R&D. |
| - Ex-vessel | Practically eliminated by design provisions. | (See above)<br>- The reactor cavity is maintained dry.<br><br>- In the core catcher water covers the corium after spreading, with no significant amount of water being present before spreading. |
| MCCI and basemat melt through. | Practically eliminated by design provisions protecting the basemat, even in the very long term. | - The basic solution involves spreading the corium in a dedicated chamber. Water from the IRWST then covers the corium for cooling. The solution includes a protective layer limiting the risk of MCCI and a cooling network to limit the temperature of the concrete. |
| Continuous release of mass and energy to the containment. | Dealt with by designing the containment and implementing a containment heat removal spray system. | - The containment design pressure is 0.5 MPa (5.0 bar).<br>- Venting shall not be used. |
| Containment bypass | Practically eliminated by design measures. Unavoidable residual leaks are collected in the containment annulus and sent to the stack after filtration. | The very stringent objectives specified with regard to off-site countermeasures impose a "zero bypass" design for the containment. |
| Recriticality | Practically eliminated because physically not credible. | |

(*) *The description reflects the present WWER-1000 features. The "Basic Design" is still in progress, and modifications could be introduced.*

TABLE 8. WWER-640 STRATEGY FOR COPING WITH SEVERE ACCIDENT PHENOMENA AND CHALLENGES

| Phenomena and challenges | Strategy | Comments (*) |
|---|---|---|
| Phenomena associated with high pressure core melt (HPCM) | Practically eliminated. | |
| | 1. The strategy involves use of highly reliable of existing systems complemented by use of emergency gas removal and bleed and feed of the primary system to prevent HPCM occurrence. | |
| | 2. The strategy involves transfer of a potential HPCM situation to a low pressure core melt (LPCM) before core melting. | Basic solution: emergency depressurization system decreases the primary pressure to the value in containment. |
| Phenomena associated with low pressure core melt.<br>- loads on the RPV | Dealt with by design of the RPV emergency cooling system. | |
| Phenomena associated with $H_2$ combustion.<br>- global detonation | Practically eliminated by sizing $H_2$ control means and containment volume. | The containment volume is 60000 $m^3$. |
| - global deflagration | Dealt with by designing the containment. | The containment design pressure is 0.5 MPa (5.0 bar). |
| - local phenomena (detonation, DDT, fast deflagration) | Practically eliminated by controlling $H_2$ local concentration or dealt with by designing the structures. | The design pressure of the prestressed concrete containment is 0.5 MPa (5 bar). This provides significant margin before unacceptable leakage ~0.7 MPa (7 bar). |
| - Long term ignition | Practically eliminated by controlling $H_2$ generation and concentration. | |

TABLE 8. (Cont.)

| Phenomena and Challenges | Strategy | Comments (*) |
| --- | --- | --- |
| Steam explosion<br>- In-vessel | Practically eliminated by design provisions if necessary. | Careful follow-up of on-going R&D. |
| - Ex-vessel | Practically eliminated by design. | The strategy involves retention of the core melt inside the RPV. |
| MCCI and basemat melt through. | Practically eliminated by design. | The strategy involves retention of the core melt inside the RPV by flooding the reactor cavity. |
| Continuous release of mass and energy into the containment. | Dealt with by designing the containment and implementing a passive containment heat removal system. | - The containment volume is 60000 $m^3$ and the design pressure is 0.5 MPa (5 bar).<br>- Venting shall not be used. |
| Containment bypass | Practically eliminated by design measures.<br><br>Unavoidable residual leaks collected in the containment annulus and sent to the stack after filtration. | The very stringent objectives specified with regard to off-site countermeasures impose a "zero bypass" design for the containment. |
| Recriticality | Practically eliminated because physically not credible. | |

(*) *The description reflects the present WWER-640 features. The "Basic Design" is still in progress, and modifications could be introduced.*

TABLE 9. BWR 90 STRATEGY FOR COPING WITH SEVERE ACCIDENT PHENOMENA AND CHALLENGES

| Phenomena and challenges | Strategy | Comments (*) |
|---|---|---|
| Phenomena associated with high pressure core melt (HPCM) | Core melt at high pressure is technically excluded, by depressurization of the RCS. Depressurization is performed by two diversified systems, each capable of preventing core melt at high pressure. | The two systems for depressurization of the reactor to a low pressure are: the normal safety relief system with own-medium operated main valves that are activated by electric pilot valves; and the emergency discharge and relief system with motor-operated ball valves. |
| Phenomena associated with low pressure core melt in the short term (at RPV melt-through):<br>- loads on the RPV support and cavity walls;<br>- loads due to ejected core melt (DCH). | The loads on the RPV support and cavity walls will be limited as long as the melt-through will be local; in BWR 90, a local melt-through is likely due to the large number of penetrations through the bottom of the RPV.<br>The probability for a DCH is extremely low due to the diversified systems for depressurization. Even if a DCH were to occur, it is not likely to jeopardize containment integrity since lower drywell volume is significant and the relief areas to the upper drywell are large. Besides, the containment is inerted and that mitigates DCH effects. | |
| Phenomena associated with $H_2$ production and combustion:<br>- Global detonation/deflagration<br>- Local phenomena (detonation/fast deflagration/ DDT)<br>- Long term | The containment atmosphere is inerted (filled with nitrogen) to prevent deflagration and detonation of combustible mixtures of $H_2$ & $O_2$.<br>Combustible concentrations of oxyhydrogen are further prevented by the containment atmosphere being treated in a recombiner system to limit the oxygen content. In the event of excessive hydrogen generation, a depressurization via the filtered vent system will occur. | |
| Steam explosion<br>- In-vessel explosion | The lower plenum of the RPV is divided into a rather large number of sub-volumes (by control guide tubes, etc.), and a steam explosion that would jeopardize the integrity of the RPV is not credible. | |

TABLE 9. (Cont.)

| Phenomena and challenges | Strategy | Comments (*) |
|---|---|---|
| Steam explosion<br>- Ex-vessel explosion | In the present BWR 90 design the lower drywell (the cavity below the RPV) is filled with water prior to RPV melt-through.<br>If the energy from a significant amount of corium is transferred to the pool water in a short time, significant dynamic loads may obviously result, but it is considered, however, that the advantages of a water pool with respect to fragmentation of the melt and improved coolability are larger than the disadvantages with respect to ex-vessel steam explosion since the probability for a significant steam explosion is extremely low. | The possibility of ex-vessel steam explosions following ejection of molten material into water pools has received much attention in the research community, following pilot tests using molten $Al_2O_3$ that yielded very intensive reactions.<br>It may be noted, however, that experiments with molten $UO_2$ at ISPRA, Italy indicate that the steam explosion risk has been much exaggerated. |
| Molten core-concrete interaction (MCCI) and basemat melt-through | The lower drywell (the cavity below the RPV) in BWR-90 is filled with water before RPV melt-through and thus efficient fragmentation of the corium is expected before it reaches the bottom of the lower drywell. The corium-concrete interaction will thus be limited and radioactive matter released from MCCI will be effectively quenched. | |
| Mass and energy continuous release in containment | Temperature and pressure rise is reduced by continuous operation of the drywell spray.<br>The long-term strategy for providing a stable final state implies filling up the containment with water and depressurizing it via the filter. This yields a situation where all corium is covered with water to keep containment pressure and temperature low.<br>Over-pressure protection is provided by a filtered vent system: a vent line, blocked by a rupture disk, leads from the compression chamber of the wetwell (which contains cleaned gases from the drywell) to the stack via a high-efficiency filter; and a second line, provided with two normally closed, manually controlled valves, leads from the drywell to the filter. | The containment spray system comprises two subsystems, the wetwell spray and the drywell spray subsystems. The former is normally in use also during plant power operation, whereas the latter is taken into operation only in accident situations to reduce drywell pressure.<br>In addition, the containment can be sprayed by the containment water filling system which has connections to the fire fighting water system and for outside mobile sources.<br>Restart of the RHR system is assumed to occur after 24 h. |

TABLE 9. (Cont.)

| Phenomena and challenges | Strategy | Comments (*) |
|---|---|---|
| Containment by-pass | The function of a typical BWR containment can be jeopardized by internal as well as external by-passes. In BWR 90, internal by-pass between drywell and the wetwell compression chamber has been practically excluded. | The lines to the condensation pool in the wetwell from the safety relief valves and the letdown pipes are run through a separate shaft with horizontal connection to the pool and no longer through the partitioning floor between dry- and wetwell. |
| | With respect to external by-pass BWR 90 follows the strategy of its predecessors; isolation valves and penetrations are checked with respect to leaktightness regularly, and special procedures apply when any barrier is broken (e.g., during refuelling). | All pipes penetrating the containment are provided with isolation valves that close upon signal from the safety system; only a few valves necessary for safety functions remain open or are re-opened. |
| Recriticality | In case of reflooding of a degraded core, operator actions shall include start-up of a high-capacity boron injection system to prevent recriticality during reflooding.<br>Melting of the core implies that fissile material will mix with absorber material (with no moderator), and therefore recriticality is excluded. | |

(*) *The description reflects the standard BWR 90 design. Continued design improvement efforts may lead to modifications.*

### *1.1.8. USA - The AP-600*

(a) General design requirements

- AP600 is designed to meet U.S. NRC regulations and General Design Criteria (GDC). Also designed to meet the Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document (URD).

(b) Specific requirements with respect to severe accidents

- Regulatory requirements: In the United States, Part 52 of the 10 Code of Federal Regulations requires a probabilistic risk assessment (PRA) as part of an application for design certification. The PRA provides a detailed evaluation of the design, including plant, containment, and typical site analyses that consider both internal and external events. An objective of the PRA is to demonstrate that the plant core damage frequency and severe release frequency meet the U. S. NRC safety goals. The NRC PRA safety goals are a calculated plant core damage frequency less than or equal to $10^{-4}$ events per year and a calculated severe release frequency less than or equal to $10^{-6}$ events per year for sequences resulting in a greater than 25 rem whole-body dose over 24 hours at one-half mile from any individual reactor. The URD PRA requirement is the frequency be less than or equal to $10^{-5}$ events per year for core damage and less than or equal to $10^{-6}$ events per year for severe release events.

- The AP600 PRA core damage and severe release frequencies meet and exceed the NRC and URD safety goals. Specifically, the AP600 core damage frequency is calculated to be approximately $2 \times 10^{-7}$ events per year including both power and shutdown conditions. The AP600 large release frequency is approximately $3 \times 10^{-8}$ events per year including power and shutdown conditions.

- AP600 design features that reduce risk include:

  (i) More lines of defense to current US operating plants, which provide more success paths following an initiating (potential accident) event. More lines of defence also provide redundancy and diversity to overcome common cause-related concerns. For example, AP600 has at least five lines of defence for core heat removal.

  (ii) AP600 uses passive safety-related systems to mitigate design basis accident and reduce public risk. The passive safety-related systems rely on natural forces such as density differences, gravity, and stored energy to provide water for core and containment cooling. These passive systems do not include active equipment such as pump. One-time alignment of safety-related valves actuates the passive systems using valves such as squib (explosive) valves, check valves, and air-operated valves.

  (iii) AP600 has a lower core power density.

  (iv) Use of canned reactor coolant pumps avoids seal LOCA issues and simplifies the chemical and volume control system.

(v) Loss-of-offsite power event is much smaller contributor for AP600 than for current or evolutionary plants because AP600 passive systems do not depend on AC power, including diesel generators.

- AP600 mitigation features that reduce risk include:

(i) For RCS inventory make-up during LOCA events, passive core cooling system uses three passive sources of water to maintain core cooling through safety injection.

(ii) An automatic depressurization system is incorporated into the design for severe accident depressurization of the RCS. The system can also be used in a partial depressurization mode to provide long-terms RCS cooling with normal residual heat removal injection.

(iii) AP600 has safety-related Class 1E DC power that has a large battery capacity to support all frontline passive safety-related systems for 72 hours.

- AP600 features that reduce severe accident risk include:

(i) AP600 design has a unique severe accident feature that virtually assures in-vessel retention of molton core debris following core melt events. The in-vessel retention (IVR) feature on AP600 is a severe accident management strategy that involves cooling the external surface of the reactor vessel by flooding the reactor cavity to prevent reactor vessel melt-through following a core melt scenario. This strategy offers significant potential for mitigating severe accidents by preventing ex-vessel severe accident phenomena such as core-concrete interaction, high-pressure melt ejection, and ex-vessel steam explosions.

(ii) AP600 passive containment cooling system provides protection to the containment pressure boundary by removing the decay and chemical heat that slowly pressurise the containment. This system reduces the potential for decay heat pressurization of the containment.

(iii) The use of hydrogen igniters reduces the already low probability of containment failure due to hydrogen combustion.

(iv) The use of passive safety-related systems provide a reliable and diverse RCS depressurization which significantly reduces the likelihood of high pressure core damage events.

The AP600 strategy for coping with severe accident phenomena and challenges is summarized in Table 10.

### *1.1.9. Canada - The CANDU-9*

CANDU 9 is a single unit evolutionary heavy water reactor based on operating single and multiple CANDU power plants. It is in advanced stages of design and incorporates a

TABLE 10. AP-600 STRATEGY FOR COPING WITH SEVERE ACCIDENT PHENOMENA AND CHALLENGES

| Severe accident phenomena or challenge | Strategy for AP600 | Design features |
|---|---|---|
| In-Vessel Retention of Molten Core Debris (IVR) | - Depressurize the RCS<br>- Submerge the reactor vessel - Provide natural circulation pathway<br>- Risk-Oriented accident Analysis Methodology (ROAAM) analysis to demonstrate vessel failure is physically unreasonable | - Low power density core<br>- Automatic depressurization system (ADS)<br>- No lower vessel penetration - Reactor vessel reflection insulation designed to maintain natural circulation pathway |
| In-Vessel Steam Explosion | - ROAAM analysis to demonstrate lower head integrity for in-vessel fuel coolant interaction (FCI) | |
| High Pressure Melt Ejection | - Depressurize the RCS | - Automatic depressurization system |
| Ex-Vessel Steam Explosion | - IVR prevents debris relocation from vessel | - See IVR |
| Core-Concrete Interaction | - IVR prevents debris relocation from vessel | - See IVR |
| Melt Attack on Containment Structure | - IVR prevents debris relocation from vessel | - See IVR |
| Containment Pressurization from Decay Heat | - Passive containment cooling | - Passive containment cooling system which removes decay heat through containment shell via water or air cooling |
| Hydrogen Combustion | - Release hydrogen from RCS at low pressure - Mix hydrogen in containment air<br>- Ignite hydrogen at low concentration | - Automatic depressurization system<br>- Stage 4 ADS release to steam generator compartments<br>- Hydrogen igniters |
| Elevated Temperatures | - Analysis to assure that hydrogen burning does not heat up containment pressure boundary | - Stage 4 ADS release to steam generator compartments |

number of additional severe accident prevention and mitigation features along with its use of proven designs for its reactor and process systems.

(a) General design requirements

- Meet Canadian regulatory requirements for lincensability in Canada. The projected doses for design basis accident must meet all regulatory requirements.
- Reduce severe core damage frequency.
- Additional active and passive measures for severe accident mitigation, early termination of accident progression and reduction of severe core damage consequences.

(b) Specific requirements with respect to severe accidents

- Some severe accident sequences [e.g. LOCA + loss of emergency core cooling (LOECC)] are already considered as design basis accidents.
- Severe core damage requires additional failure of the moderator cooling system.
- There are no explicit requirements by Canadian regulators for consideration of severe core damage accidents in the design basis.
- Preliminary PSA (level I) is performed to identify internal failure permutations that lead to plant damage states with potential for severe core damage.
- The cumulative core damage frequency for internal and external events targeted as below $10^{-5}$ per reactor-year.

The CANDU-9 strategy for coping with severe accident phenomena and challenges is summarized in Table 11.

*1.1.10. India - The Advanced Heavy Water Reactor (AHWR)*

(a) General design requirements

- The main aims of development of the boiling light water (pressure tube vertical type), heavy water moderated AHWR are:
  - to utilize thorium in a thermal reactor
  - to enhance the safety of the reactor by providing passive safety features
  - to bring the probability of occurrence of a severe accident to negligibly low value, and
  - to ensure integrity of containment during accident scenario.

(b) Specific requirements with respect to severe accidents

Trough appropriate design measures such as inherent safety features and passive safety systems, the probability of occurance of severe accidents is brought to a negligible low value. This will be confirmed through PSA after the design is finalized.

Design measures and passive safety features provided for the prevention of severe accident:

- negative void coefficient of reactivity (to shut down the reactor)
- two independent and fast acting shut down systems (to shut down the reactor)

TABLE 11. CANDU-9 STRATEGY FOR COPING WITH SEVERE ACCIDENT PHENOMENA AND CHALLENGES[1]

| Phenomena or challenges | Prevention | Mitigation |
|---|---|---|
| High pressure melt ejection | 1. High pressure melt ejection probability is reduced to less than $10^{-8}$/year by reducing the probability of loss of shutdown when required by provision of two diverse, independent, fully capable and dedicated shutdown systems. | |
| | 2. High pressure melt ejection by flow blockage in a single channel is considered as a design basis accident and consequences are shown to be acceptable. | |
| | 3. Severe core damage results in formation of solid debris at low pressures in the calandria vessel surrounded by over 500 Mg (500 tonnes) of cold water. | Pressure relief is achieved by rupture disks in the calandria vessel. Reactor shutdown and water makeup to the moderator and shield tank. |
| Direct containment heating | Water volumes in the moderator and shield tank prevent direct ejection of hot melt into the containment (for an illustration of these protective water volumes, see figure 3 from the paper "Addressing Severe Accidents in the CANDU 9 Design" in these Proceedings). | The probability basemat attack by molten fuel is further reduced by low location of shield tank and external flooding of shield tank and the large flat basemat design. |
| Hydrogen source terms | The hydrogen source term by zircaloy oxidation is limited by inherent CANDU channel design which provides for predominantly solid debris formation. | |
| | The hydrogen source term by radiolysis and by corrosion of structural components is reduced by judicious choice of surface coatings and finishes. | |
| Hydrogen combustion | The large containment volume limits hydrogen concentrations. Equipment layout and design of the containment internals avoid pockets where $H_2$ may accumulate. Coolers provide forced circulation mixing. The containment layout promotes natural circulation. | Hydrogen mixing is enhanced by coolers and natural circulation pathways. Igniters are incorporated into the design for early ignition of local concentrations, and recombiners are incorporated for long-term hydrogen reduction. Locations for igniters and recombiners are judiciously selected. |

TABLE 11. (Cont.)

| Phenomena or challenges | Prevention | Mitigation |
|---|---|---|
| Steam explosion in vessel | Not yet resolved.<br>No significant melt formation is expected during channel failures leading to debris formation. Probability of conditions required for occurance of in-vessel steam explosion is considered low. | There are pressure relief pathways in the calandria vessel and shield tank. |
| Steam explosion in containment | Retention of debris in the calandria vessel is facilitated by presence of a large mass of water in the shield tank enveloping the vessel. Additional water makeup to shield tank is provided by reserve water tank. The generic issue of steam explosions in containment is not resolved. | Large openings in the reactor vault allow pressure relief due to a range of energetic water-melt interactions. |
| Core-concrete interactions | The containment design (e.g. flat base mat and low elevation shield tank) assures a water layer on any core debris in the containment. | The concrete composition on reactor vault base mat floor is chosen to minimize production of hydrogen and carbon monoxide and to delay concrete ablation. A large flat basemat area ($>0.04$ $m^2$/ MW) is provided for spreading of debris. |
| Long-term pressurization of containment | The large containment volume and ample structural heat sinks delay pressurization. The pre-stressed thick walled concrete containment with steel liner has a high failure pressure. | Containment coolers maintain pressure well below failure pressure. Hydrogen mitigation systems are designed to avoid global hydrogen burns. |
| Recriticality | Natural uranium fuel debris mixed with shutdown rods and absence of moderator (pre-requisite for severe accident) precludes re-criticality. | |
| Containment by-pass | Containment isolation.<br>Containment bypass probability reduced by design measures. | |

[1] *The CANDU 9 design is evolutionary and still under development.*

TABLE 12. AHWR STRATEGIES FOR COPING WITH SEVERE ACCIDENT PHENOMENON AND CHALLENGES[a]

| Phenomena and challenges | Strategy | Comments |
|---|---|---|
| 1. High pressure melt ejection and direct containment heating | - Natural circulation capacity at rated power and negative void coefficient along with two independent shutdown systems reduce the probability of this event to a lower level.<br>- Direct containment heating in AHWR is not possible. | - If there is any melt, it would be arrested by the end shield and reactor cavity water below the core.<br>- Moderator, vault water, water in end shield and reactor cavity will prevent direct injection of hot melt into containment. |
| 2. Hydrogen production and combustion in the containment | A large containment volume of 50,000 $m^3$ is provided. $H_2$ concentration in the containment will not exceed flammability limit. | - Provision of catalytic recombiners will be made, if required, on the basis of Hydrogen transport assessment. |
| 3. Steam explosions in core and in the containment | The possibility of occurance of steam explosion is eliminated by design provision. | |
| 4. Core concrete interaction in the containment | Core melting is not expected because of the presence of a large quantity of water in the calandria, calandria vault and in the reactor cavity. If at all there is core melt it will be arrested by the end shield, and the reactor cavity water below the core. | |
| 5. Recriticality | Recriticality is not possible in AHWR because of appropriate design provision. | |
| 6. Pressure buildup in the containment | Pressure reduction is achieved by providing large containment volume, suppression pool and continuous cooling of the containment by Passive Containment Cooling System (PCCS). | |
| 7. Containment bypass or loss of long term heat removal | Containment bypass is not possible due to passive containment isolation | Long term heat removal from the containment is achieved by the PCCS |

[a] *The AHWR design is still under development so modifications may be introduced.*

- reliable emergency core cooling system with redundancy
    - advanced accumulators
    - gravity driven water pool (to provide core cooling)
- availability of large quantity of water around the core (to facilitate prolonged heat removal and submergence of reactor core in water)
- passive containment isolation (to confine radioactivity)

Passive features provided for the mitigation of severe accidents:

1. Double containment and passive containment isolation are provided for fission product retention within the containment
2. Passive containment cooling system for long term containment heat removal
3. Large volume of primary containment of minimize peak pressure
4. Primary containment filtration and pump back system to reduce iodine inventory
5. End shield and reactor cavity water for corium management if there is any core melt down
6. Provision of passive catalytic recombiners will be made, if required, on the basis of assessment of hydrogen transport within containment.

The AHWR strategy for coping with seven accident phenomena and challenges are summarized in Table 12.

## 1.2. Observed commonality in basic design philosophy and design approach for advanced water cooled reactors

### *1.2.1. Basic design philosophy*

- For design basis accidents, the established design criteria, and therefore the design philosophy, is unchanged. Severe accidents are treated as design extension conditions[1] or in the safety margin basis[2]. Features provided to address severe accidents are not designed to meet the same stringent design criteria and requirements (redundancy, diversity and conservative analysis and acceptance criteria) as are applied to features provided to cope with design basis accidents: however, design features for coping with severe accidents are engineered to have a high reliability in the expected environment.
- All available resources on site would be used to cope with severe accidents.
- Designers of next generation water cooled reactors generally have the objective to enhance safety in such a way that significantly reduced emergency planning requirements could be technically justified.

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[1] Design extension conditions are described in the European Utility Requirements as a specific set of accident sequences that go beyond design basis accidents, to be selected on deterministic and probabilistic basis and including: complex sequences and severe accidents. Appropriate design rules and criteria are set for design extension conditions, in general from those for design basis accidents.

[2] This term is described in the EPRI Advanced Light Water Reactor Utility Requirement Document (Vol. 1-ALWR Policy and Summary of Top Tier Requirements). The ALWR design will include both safety design and safety margin requirements. Safety design requirements (referred to as the Licensing Design Basis) are necessary to meet the NRC's regulations with conservative, licensing-based methods. Safety margin requirements (referred to as the safety margin basis) are plant owner-initiated features which address investment protection and severe accident prevention and mitigation on a best estimate basis.

*1.2.2. Design approach*

- The design approach is guided by insights from a combination of deterministic analyses, probabilistic considerations, operational experience and engineering judgement to reduce vulnerability of design features. Evaluations of cost effectiveness are also used to guide the design approach.
- There is a balanced approach for prevention and mitigation. For severe accident sequences for which mitigation is difficult to implement and/or ensure, prevention is given priority.
- Accident management will remain necessary. Accident management procedures should be simple and clear.
- Early containment failure is avoided by system design.
- Design features for in-vessel retention of the molten core will be incorporated into those designs for which measures for in-vessel retention can be proven effective and without adverse consequences. This may reduce the role of mitigative measures to maintain the integrity of the containment.
- Plant status information systems are incorporated to provide guidance to the operator in taking action to arrest the progress of and to contain the consequences of severe accidents.
- Measures for mitigation are generally passive due to concerns about station blackout and possibly limited accessibility of components for recovery action.
- To increase the reliability of preventive measures, diversification of emergency power sources, simplicity, and ample margins are generally incorporated into the designs.
- Explicit or implicit use of cut-off probability is generally made when reasonable. Examples are the applications in eliminating consideration of energetic in-vessel steam explosion and in eliminating consideration of high pressure melt ejection when reliable depressurization is put into the design.

## 2. STATUS OF KNOWLEDGE REGARDING SEVERE ACCIDENT PHENOMENA

This section presents the main highlights and the most relevant conclusions from the discussions held by the members of the Working Group on the subject of Status of Knowledge Regarding Severe Accident Phenomena.

Before the discussion started, the following list of questions and issues was agreed to be addressed by all the members of the group in order to ensure a methodical approach:

- Identification of relevant phenomena
- What information is available?
    - Results of experiments and analysis;
    - Relevance to reactor conditions;
    - Consensus on accurancy and relevance of data;
    - Is the existing information sufficient or is there a need for more.
- What further information is required?

This approach was adopted after recognizing that due to the complexity of most relevant phenomena related to severe accidents, significant uncertainties are likely to remain, regardless of the amount of R&D activities. It is, therefore, very important to clearly define design objectives for each challenge and evaluate whether available results are sufficient.

It was also recognized that, in some cases, even though existing data do allow resolution of the problems at the design level, further insights may be recommended in order to decrease uncertainties and related costs. However, considering the on-going R&D programmes it seems premature to define specific actions before completing evaluation of test results from current experimental activities. At the engineering level, the industry (i.e. vendors, utilities) have expressed their needs for validated computer codes accepted by the nuclear regulatory authorities, if possible, internationally. This would allow them to evaluate the impact of severe accidents in the design of new NPPs utilizing qualified computer codes.

For the identification of the relevant phenomena and challenges, Table I of the report of the TCM on "Identification of Severe Accidents for the Design of Future NPPs" (c.f. also Table 1. from the Summary) was taken by the group as the basis to start the discussions.

The different phenomena and challenges, and the associated candidate strategies as described in this table were reviewed by the group. A revised list of phenomena and challenges was proposed by the group as follows:

1. Reactivity Insertion
2. High Pressure Melt Ejection (HPME) and Direct Containment Heating (DCH)
3. Hydrogen Production and Removal
4. In-vessel Steam Explosion
5. Ex-vessel Steam Explosion (in containment)
6. Molten Corium-Concrete Interaction (MCCI)
7. Containment bypass or loss of long term heat removal
8. Fission Product Behaviour under Degraded Core Conditions.

The main differences between this list and the list developed during the TCM on Identification of Severe Accidents for the Design of Future Nuclear Power Plants (October, 1995, IAEA, Vienna) are the:

(a) Combination of "hydrogen production in vessel" with "hydrogen production and combustion in containment" into a single phenomenon: "hydrogen production and control".

The reason is that the main contribution to hydrogen production is the Zr/steel reaction with steam in the vessel. Once the core uncovers, the hydrogen production is very difficult to stop (or even reduce) unless some preventive measures (e.g., core recovery) are successfully implemented. On the other hand in-vessel hydrogen burns or explosions are considered very improbable, and therefore the main risk of hydrogen production is its concentration inside the containment above the detonation limit when there is a breach in the RCS.

(b) Reactivity insertion

The item (b) was already identified in the mentioned report. The reason to include it here is the probability of accident sequences leading to reactor conditions of prompt criticality (i.e., certain boron dilution sequences).

*2.1.1. Reactivity insertion*

Some particular accident sequences initiated by an inadvertent boron dilution (deviating from the DBA) might lead to reactor conditions of prompt criticality. The issue and mechanisms are well understood. It is an engineering related issue, and there are enough means to prevent this situation. Everything needed to deal with this issue is available.

*2.1.2. HPME and DCH*

Adequate level of knowledge of this phenomenon exists: PWR experiments have been performed for many years: for example at ANL (1/30 scale) and at SANDIA (1/6 and 1/10 scale). Models have been developed to scale up to prototypic size and conditions. Promising methods, such as ROAAM (Risk Oriented Accident Analysis Methodology) have been recently applied to this issue (NUREG/ER-6075).

Most of the new reactors designs include diverse and reliable features for RCS pressure reduction and/or specific cavity layout to prevent high pressure core melt, and therefore DCH. For CANDU reactors DCH is not an issue, because an unmitigated loss of all heat sinks at high pressures would lead to an in-core failure of a fuel channel with a high power density, and this would rapidly depressurize the heat transport system, and therefore no significant melt ejection at high pressure would occur.

*2.1.3. Hydrogen production and removal*

The process of hydrogen generation is well understood, but the timing and the amount of hydrogen generated is subject to uncertainties. Experimental data and conservative assumptions are used to derive hydrogen production curves.

The issue of hydrogen control can be handled at the design level via:

(i) Containment size and geometry

Dealing with hydrogen control there is general agreement that hydrogen detonation inside containment compartments or containment free volume has to be eliminated. However there are still large uncertainties in the definition of the threshold for detonation. There is general agreement that below 10% volume content of hydrogen (in dry air) detonation will not occur. Research has been performed at BNL, PSI, CALTECH, and FZK.

Even larger uncertainties exist for transition from deflagration to detonation (TDD). Research is underway to better understand the problem (RUT, Russia). However, it seems possible to eliminate the problem at a layout level as it was shown in many test programmes (AP600, TADOTSU, SANDIA).

(ii) Mitigating Devices: inerting containment atmosphere, igniters, and/or recombiners

The choice between igniters and recombiners seems to be controversial, although igniters are well accepted in some European countries (Sweden, Finland) and in the USA. There is also high confidence in recombiners, although their poisoning in very harsh and wet environments typical of severe accidents has to be further assessed.

Relevant experiments conducted so far or on-going:

- Phébus FP (CEA/Cadarache), EPRI and NRC sponsored (USA), "$H_2$" and "CONT" (EU).

### *2.1.4. In-vessel steam explosions*

Much experimental work has been done, and there is the consensus that the conditional (i.e., if there is a core melt) probability of containment failure is very low (<0.001). However uncertainties still exist and will probably remain.

Experiments have been performed or are being performed in: FZK, JAERI, KROTOS facilities, "RPV" and "RPVSA" projects (EU). Need for further R&D should be assessed once all the above experiments are finished, and the results are known.

Key phases in the understanding of steam explosions are: pre-mixing and fragmentation and propagation. These phenomena have recently been addressed by experiments funded by the US-DOE and performed in two facilities, MAGICO and SIGMA, which have been used to benchmark two computer codes PM-ALPHA (premixing) and ESPROSE.m (propagation). Application of these tools were integrated (under ROAAM) with the melt relocation physics and the structural response of the RPV lower head under impulsive loads to assess the likelihood of failure of RPVs partially submerged in a water filled cavity.

### *2.1.5. Ex-vessel steam explosions*

This issue is still very controversial and remains as a challenge. Although ex-vessel steam explosion research is performed at many laboratories (JAERI, UCSB, RIT, FZK, CEA, ISPRA, etc.), there are still many uncertainties in the description of this phenomenon:

- Melt jet diameter and mass flow rate
- Location of melt discharge
- Amount of water available
- Extent of fragmentation
- Strength of Trigger required
- Pressure pulse generated
- Containment fragility.

By definition, "steam explosion" is an unstable physical state, and therefore very difficult to reproduce and/or model. It is very premature to recommend additional R&D until the experiments underway are completed.

At engineering level there are different ways to prevent this phenomenon:

- Preventing vessel melt through by water filled cavity (AP600)
- Water injection into the primary system
- Eliminate the problem by means of dry cavities.

### *2.1.6. Molten corium concrete interaction (MCCI)*

The molten corium-concrete interaction under both dry and wet conditions has been extensively investigated. This phenomenon seems to be reasonably understood, in case of dry conditions. Some uncertainties still exist concerning assumptions on aspects such as:

- Corium spreading
- Corium composition
- Corium coolability (only in case of wet conditions).

Different types of containment basemat materials have been investigated (i.e. limestone, limestone-common sand, etc.).

The extent of MCCI and its effect on the containment are influenced by many factors, including the three mentioned above. In addition, others must be also considered: amount of core debris, type of concrete, heat transfer mechanisms, etc. Several of them can be controlled or optimized through the containment design;

- Provide for flooding of the reactor cavity
- Furnish a large unobstructed area for core debris to spread in the reactor cavity floor (Research CEA, FZK, RIT, SIEMPELKAMP, ISPRA & CEA, JAPAN)
- Select type of concrete that either decreases the amount of non-condensible gases or inhibits erosion
- Provide core melt retention devices (i.e. "core catchers").

Melt coolability is essential to prevent base-mat melt through and continued containment pressurization. Attention has to be given to the potential for continuous debris cooling in case of the formation of a crust at the surface of the corium pool.

The MACE project is investigating melt coolability by adding water on top of 2000 kg of a $UO_2$+$ZrO_2$ melt pool, interacting with concrete. A successful test has not been performed yet, but a new test is scheduled for December 1996. Additional research is being or has been done by the EU (Project "MCCI", Cluster "EXV"). Further tests are considered as necessary.

*2.1.7. Containment bypass/loss of long term heat removal*

This is an issue well understood and requires just engineering design improvements. Limited R&D if needed, should be dedicated for assessment of component capability.

*2.1.8. Fission product behaviour under core degradation conditions*

At engineering level the early phases of core degradation are well understood. Several experimental investigations on fission product release under core degradation conditions have been performed at several facilities: ORNL, KFK, IPPE, SANDIA, PHEBUS, etc. However, there is not sufficient information about the effects on fission product release of (1) high heating rates (~60°C/min), and (2) the dependance on inside containment conditions for some nuclides.

The main open issues concerning fission product behaviour are:

- Behaviour of low volatiles (i.e., chemical form of nuclides)
- FP re-suspension and re-vaporization

Numerical codes (MELCOR, SCDAP/RELAP) will have to be modified to include results of experiments underway.

For late phases of core degradation, and molten core/concrete interactions, there are still some uncertainties which will be addressed in future experiments.

Further discussions with regard to severe accident phenomena which should be investigated are presented in the OECD/NEA documents "Nuclear Safety Research in OECD

Countries: Capabilities and Facilities", OECD, Paris, 1997 and "Nuclear Safety Research in OECD Countries: Areas of Agreement, Areas for Further Action, Increasing Need for Collaboration", OECD, Paris, 1996. The reader is referred to these documents for additional material on this topic.

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