Proceedings of the Specialist Meeting on
Selected Containment Severe Accident Management Strategies

Organised by OECD Nuclear Energy Agency
in Collaboration with the Swedish Nuclear Power Inspectorate
and
Hosted by the Swedish Nuclear Power Inspectorate


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Compiled by Wiktor Frid and Susanne Carlberg, SKI

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The conclusions and viewpoints presented in this report are those of the authors
and do not necessarily coincide with those of the OECD or SKI.
In preparing these proceedings, the original manuscripts, as submitted
by the authors, were used without further editorial changes.
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OECD SPECIALIST MEETING ON
SELECTED CONTAINMENT SEVERE ACCIDENT MANAGEMENT STRATEGIES

An OECD Specialist Meeting on Selected Containment Severe Accident Management Strategies was held in Stockholm, Sweden from 13th to 15th June 1994. It was sponsored by the NEA Committee on the Safety of Nuclear Installations (CSNI) and organised in collaboration with Statens Kärnkraftinspektion (SKI), the Swedish Nuclear Power Inspectorate.

Discussions in the Task Group on Containment Aspects of Severe Accident Management (CAM) of CSNI's Principal Working Group on Confinement of Accidental Radioactive Releases (PWG-4) had shown general interest in questions such as accident management strategies, long-term accident management (e.g., long-term cooling of core debris in the vessel, long-term cooling of the containment atmosphere, use of active systems, etc.), depressurization, bypass situations, timing (important for accident management strategies), hydrogen management techniques in containment, coolability of molten core debris in the containment, etc. Following recommendations made by the CAM and supported by PWG-4, the CSNI had decided to sponsor a Specialist Meeting on Selected Containment Severe Accident Management Strategies in relation to:

- Feasibility (including timing, time windows, criteria for action)
- Effectiveness (including its evaluation, e.g. through PSA techniques)
- Positive/Negative Aspects
- Long-Term Impact.

The scope of the meeting had been focused on water reactors, mainly on existing systems. The expected technical content was the following:

(i) Provision for the containment function
   - leakage detection
   - containment bypass and isolation failure detection and control
     (particularly at an early stage)

(ii) Mitigation of energetic events
   - hydrogen burn (including long-term radiolysis problems)
   - other events such as direct containment heating, steam explosions

(iii) In-vessel core melt and debris cooling
   - in-vessel cooling
   - external flooding of the vessel

(iv) Ex-vessel core melt and debris cooling
   - quenching in water pools
   - spreading and flooding (including protection of critical components
     such as penetrations, pillars, etc.)
   - mitigation of basemat melt-through consequences

(v) Containment over-pressure protection (short-term and long-term)
   - sprays
   - filtered venting systems
   - long-term cooling
Other long-term aspects of containment accident management
- leakages (including handling of radioactive wastes)
- system maintenance
- accessibility
- determination of the conditions inside the containment
- fire protection.

The objective of the Specialist Meeting was twofold:

- to exchange information on containment severe accident management strategies in place;

- to discuss developments and plans for containment severe accident management strategies to be implemented in the future or under active investigation.

This dual objective has been met by the meeting.

We thank very much the Swedish Nuclear Power Inspectorate for their kind and generous hospitality, and express our gratitude to the local organisers of the meeting, in particular Dr. Wiktor Frid (SKI) and Dr. Gustaf Lőwenhielm (Vattenfall AB). We are also grateful to Vattenfall AB for organising on 16th June a visit to the Forsmark Nuclear Power Plant.

Last but not least, we thank the members of the Technical Programme Committee and the Session Chairmen of the Specialist Meeting: the Chairman of the meeting, Dr. Gustaf Lőwenhielm (Vattenfall AB), Mr. Benoît De Boeck (AVN), Mr. Jacques Duco (IPSN), Mr. Jürgen Rohde (GRS), and Dr. Wiktor Frid (SKI). Their competence and their efficient work have been essential in the success of the meeting. Special thanks are due to Dr. Frid and his secretary who prepared the proceedings.

Jacques Royen
OECD(NEA)
SUMMARY AND CONCLUSIONS OF THE SPECIALIST MEETING

First CSNI Specialist Meeting on Selected Containment Severe Accident Management Strategies

The first specialist meeting on Selected Containment Severe Accident Management Strategies was held in Stockholm, Sweden, 13-15 June 1994. It was hosted by Statens kärnkraftinspektion (SKI). About 50 experts from 13 countries and one international organisation attended the meeting. Twenty papers were presented in four sessions, half of them dealing with accident management strategies implementation status, half of them with research aspects. It should be noted that about one third of the experts were from utilities; discussions were therefore very much oriented to practical issues. It also shows that this is a mature subject with direct impact on utilities.

The specialist meeting focused on accident management strategies in relation to:

- Feasibility (including timing, time windows, criteria for action)
- Effectiveness (including its evaluation, e.g. through PSA techniques)
- Positive/Negative aspects
- Long-term impact

Since the TMI accident the research concerning severe accidents have substantially expanded and developed in three phases:

- Severe accident phenomena research
- Implementation of mitigating measures
- Development of accident management (AM)

Research is still ongoing, although it is now more focused on specific topics, considered to be key contributors to loss of containment integrity. Actually some papers treated some unresolved research issues. Containment severe accident management programmes are in place in most OECD member countries, but implementation of mitigating measures and AM development have been performed differently. Some countries, which started early with implementation of mitigating measures, sought for robust solutions as the data base and PSA analysis were incomplete. Other countries have waited for a more complete picture concerning phenomena and have completed PSA analysis. This has led to an optimisation of mitigating measures and AM development.

Status of AM in response to various containment threats

From this meeting, the status from AM point of view for various threats to containment integrity is as follows:

Late overpressurization: This threat to the containment can be reasonably treated with filtered venting devices.

Mark I liner melt-through: This issue appears to have been resolved. Assuming that the vessel melt-through occurs with low pressure in the reactor coolant system, drywell flooding before vessel breach will probably prevent this containment failure mode.
Hydrogen combustion: This issue is on its way to resolution. Several possible solutions were presented and are being evaluated. Actually, installation of passive catalytic recombiners have been decided in principle by the Belgian utility and recommended by the German Reactor Safety Commission.

Direct Containment Heating: A reasonable solution adopted by most countries is depressurization of the primary system before vessel melt-through. It should also be pointed out that severe accident research has found that the probability of this event leading to containment impairment has been overestimated for most containment designs.

Leak tightness of the containment (short-term and long-term): The significance of this subject may be underestimated. It is imperative to detect and control leaks. Management related actions should be taken. However, if this is planned beforehand a reasonable solution can be found.

Coolability of core melt debris in the containment: There seems to be no immediate solution, which will cool the core melt, agreed upon by all experts. Preliminary core catcher work presented at the meeting was not encouraging, at least not for the geometry considered. However, it is agreed that in cavities with an area larger than about 30 m², it could be advantageous in most cases to use water. However, it has not been shown conclusively that the melt will be cooled. But radioactive aerosols generated from core-concrete interaction will be scrubbed by the overlying water pool.

A widely discussed topic is the question of adding water before or after vessel melt-through. From coolability point of view, adding water into the reactor cavity prior to vessel melt-through, is advantageous, as the initial heat transfer from the melt to the water is very high if the melt falls into water. On the other hand steam explosions have to be considered.

Another possibility is to cool the melt in the vessel, either with water inside or outside the vessel. In-vessel core Debris Cooling through Cavity Flooding (IDCCF) was discussed only briefly as the subject had been examined at the OECD Workshop on Large Molten Pool Heat Transfer held in March 1994. The value and the interest of the OECD RASPLAV Project were emphasised.

Steam explosions: This still seems to be an open issue. Main issue is today ex-vessel steam explosions in deep pools, which the melt can fall into after vessel melt-through. There is no way to guarantee that ex-vessel steam explosions can be suppressed but containment loadings can be limited by specific geometrical features of the containment. Careful attention should be paid to local steam explosions.

Conclusions

1 For plants with significant risk contribution from hydrogen combustion, this can be alleviated or even eliminated with currently available technique. A careful selection between available techniques may be needed.

2 Although direct containment heating leading to containment impairment may be less significant than previously thought, it is recommended that depressurization of the primary system is included in operator procedures either with secondary feed-and-bleed or primary
feed-and-bleed.

3 More emphasis on leak detection, control and management is recommended. The use of expert systems can be of great help to operators, especially during crisis situations. Severe accident source term evaluation remains a topic of great interest.

4 Further research is recommended for unresolved issues as melt coolability and ex-vessel steam explosions. On the latter point it is emphasised that small scale experiments performed with thermite may not give results representative of real reactor conditions. Strong interest was expressed in the results of the Japanese ALPHA Programme, in particular the experiments on the limitation of energetic interactions.

5 As it is impossible to cool the core melt ex-vessel in some plant geometries it should be emphasised that it is always advantageous to cool the melt in the vessel with water inside or outside the vessel. This type of research should be pursued.

6 Long term aspects of accident management have to be considered at an early stage. This is particularly true for containment aspects of long term accident management.

7 Consideration needs to be given to the question: "How and when do we know we have done enough?"
SESSION 1

CONTAINMENT ACCIDENT MANAGEMENT-STRATEGIES - GENERAL ASPECTS
Session Summary by J. Rohde, GRS (Chairman)

Seven presentations were given in this session. While six of them were concentrated on the general aspects of severe accident management strategies, selected for different reactor concepts in specific countries and containment designs to minimise the potential activity release from the containment, one paper tried to explain the physical boundary conditions preventing vessel melt-through in the TMI-event and what can be learned from it for accident management.

Most of the presentations, summarising the actual status of the implementation of containment related severe accident strategies were given from representatives of the utilities, for PWR and BWR plants.

In the first paper the French approach was described, explaining in detail the interrelation of procedures on French plants, but also the responsibilities of the control room operators, the safety engineers and the crisis team. The strategic options proposed to the team in charge of the situation regarding the safeguarding of the containment are determined pragmatically in the Severe Accident Intervention Guide, which includes the emergency procedures for containment monitoring and filtered containment venting (U2 and U5 procedures).

In parallel with the operating procedures, a guide devoted to the overall monitoring of the containment has been developed for the use by the crisis team.

First principles were given, how to include the knowledge base, gained by developing severe accident strategies for existing plants, into the design of future reactor concepts, specially for the EPR.

The next two papers described the provisions taken to cope with severe accidents in Swedish PWR’s and BWR’s.

Mainly the implementations such as filtered venting and external water sources at all Swedish PWR’s by 1988 were described together with the development of the related Emergency Operating Procedures. In the following time period the accident management system has been further improved and special attention was drawn on the long term aspects of a severe accident management. A new knowledge based handbook was written to support the plant leader and the members of the technical support centre including the specific issues of the Beyond Emergency Response Guidelines /BERG/ and the management of the long term severe accident situation. Both the BERG and the knowledge based handbook are updated periodically. The similar approach was described in the following paper for Swedish BWR’s. Additional information was given, concerning the interrelations between the management on the unit and the accident management centre.

In support of the ongoing periodic safety reviews in Switzerland an independent PSA was made, mainly concentrating on risk implications of severe accident management strategies. The
method used and some main findings were presented. It was mentioned that in terms of release frequencies the effects of severe accident management strategies showed a shift from filtered vented accidents to accidents with an intact containment. In some cases, the increased availability of the engineered safety features induced an increase in the frequency of the late containment failure. However, the consequences of these accidents which are already small in the base case analysis, are largely reduced by other mitigative effects. It was stated in general that large detrimental effects are not apparent by the adoption of any of the severe accident strategies which have been considered in the investigation.

The next two presentations gave an overview on first thoughts given by Japanese utilities for PWR and BWR plants severe accident management. Concerning BWR's different reactor concepts and containment designs were investigated like containments of type MARK I, II and the ALWR.

The basic philosophy of Japanese BWR's utilities in developing strategies is to fully utilise the existing plant resources due to the findings of PSA-work which has been carried out in the last years. Only minor plant modifications seem to be necessary. Special attention was drawn on the availability of instrumentation and information's needed to control the accident progression, development of guidelines and the training of the operators are in the planning status together with clarifying the organisation and the responsibilities in case of a severe accident.

Joint studies were done by Japanese PWR owners to fulfil the recommendations given by the Japanese NSC in May 1992 to develop severe accident strategies. Special strategies have been selected for further plant specific investigations on the basis of PSA work.

Severe accident strategies needs the preparation of specific procedures and limited plant modifications. This work will be performed in the near future.

The last paper gave some new interpretations of the situation in the lower head of the TMI reactor pressure vessel, after the relocation of molten materials into the lower plenum. Special emphasis was drawn to interpret the heat transfer mechanisms and the vessel wall situation together with the debris behaviour leading to a rapid cooling down, keeping the integrity of the vessel.

The author draws two main conclusions in case of an accident:

• water should be added to the RPV, regardless of the accident state to make use of the inherent in-vessel cooling mechanism, explained above and

• submerging the core debris with water in the RPV is sufficient to protect the vessel integrity and to cool down the debris.

In this session different approaches were presented from the different countries in developing severe accident management strategies:

• well organised strategic planning for the use of existing plant capabilities, if necessary with minor plant modifications;

• systematic plant modifications to enlarge the grace period of the containment before actions have to be taken. Special attention was drawn to keep the actions as simple and robust as possible as all detailed analysis was not finalised;
• based on detailed PSA work and extended experimental programs, the development of accident strategies against identified major threats to the containment integrity.

Some combinations of these three major approaches were demonstrated in the presentations.

Different technical aspects of the strategies selected were discussed but also more general questions were raised which could not be answered such as:

* what is the better way, manual actions or automatic systems

or

* how do we know that we have done enough?

In general it was agreed that some open questions still exist, mainly in connection with the short term strategic planning of actions, the long term accident management, and the duties and the training of teams being involved in the management of severe accidents.
The first paper, presented by GRS (FRG), deals with the use of catalytic recombiners as a passive mitigation technique, aimed at preventing hydrogen concentrations in the containment reaching a level threatening its integrity.

After reminding the principles of functioning, and the operating constraints linked to the environmental conditions in a reactor containment during a severe accident, the device itself is described as a thin, strongly adherent coating of porous metallic palladium or platinum on a stainless steel plate. The current status of the three varieties of recombiners developed in the FRG by GRS, Siemens and NIS Ingenieur GmbH is shortly described, as well as typical experimental results obtained in particular in the Battelle Model Containment in Frankfurt and in the HDR facility.

Future activities encompass additional, confirmatory testing of industrial catalytic modules, with particular attention to the following points:

a) their energy dissipation capability, even in hydrogen-rich atmospheres, to make sure they cannot exceed a temperature of approximately 600 °C (overheated support structures could then initiate mixture ignition) and

b) their resistance to poisoning and aging in a reactor containment environment.

The location of catalytic devices in new plants will be based on calculations of the hydrogen concentrations in the reactor containment for typical accident scenarios; their capacity has been specified in the FRG so as to recombine within 24 hours the hydrogen amount corresponding to 100% of zirconium oxidation; the combination of catalytic recombiners with other hydrogen mitigating measures may provide an additional efficiency to rapidly decrease the concentration in the early stages of a release; in-service inspection plans have also to be considered to make sure the long-term efficiency of the recombining devices is maintained.

Catalytic recombiners have demonstrated the capacity of reducing the hydrogen concentration under steam-inerted conditions and at very low hydrogen concentrations. In addition, they enhance containment atmosphere mixing by heat generation during operation, which may offer some flexibility regarding their location in the containment. No negative effect has been identified, as long as they are appropriately designed and fabricated to avoid excessive heat-up when functioning, adequately installed in the reactor containment and periodically inspected and tested.

In May 1994, the German Reactor Safety Commission recommended the installation of catalytic recombiners to protect the containment of PWR plants during beyond design events; this could be supplemented in the future by the consideration of igniters for the case of rapid, large hydrogen releases; nevertheless, the safety of such igniters is not currently established under all severe accident conditions.
The second paper addresses the issue of hydrogen management in CANDU reactors and was presented by AECL Research (Canada). In single-unit CANDU reactors, the assumptions made regarding the amount of hydrogen generated, as well as the natural and engineered mixing mechanisms inherent to the system, are expected to keep hydrogen concentration to non-flammable concentrations in the containment. Multi-unit CANDU stations similarly have engineered mixing systems, expected to appropriately homogenize hydrogen concentration in the containment, and are equipped with ignition systems (glow-plugs, automatically actuated), the number and distribution of which may be optimized by further accident analyses and research. As, in addition, one peculiarity of the multi-unit stations is to allow for combustion venting to large adjacent volumes. Research is oriented to deflagration behavior in stratified mixtures and complex geometries like room chains; flame accelerations and deflagration to detonation transitions are investigated, as well as standing flames. Catalytic recombining systems adapted to CANDU severe accident environmental conditions have also been studied and a high capacity hydrogen removal appears to be readily achievable.

The third paper, presented by Belgatom, describes calculations of the active surface area of catalytic recombiners that could prevent hydrogen concentrations high enough for jeopardizing containment integrity if combustion takes place. A post-processor, named CARE, has been developed at Belgatom, which determines for a single volume the instantaneous gas concentrations from thermal-hydraulic conditions and gas production rates calculated with MARCH3. CARE also calculates the containment gas concentrations for given active surface area and recombination rate of the catalytic device, and, ultimately, assesses the containment pressure resulting from an isochoric, adiabatic combustion. Attention is paid to a "SEY" sequence (a small break LOCA, with an early core melt and the availability of the containment cooling system), which corresponds to a high, rapid release rate of hydrogen, on the Doel 3 and Tihange 2 power plants, differing mainly from each other by the containment volume and the concrete composition of the basement. Conservative calculations show that a 250 m² catalyst area should be sufficient for preventing containment failure; it is worth to be observed that such catalytic devices improve containment atmosphere homogenization. The Belgian utility has made the principle decision to install such catalytic recombiners in all units and will make a detailed plant-specific proposal to the regulatory body.

The fourth paper, presented by Paul Scherrer Institute in Switzerland is related to the prevention of ignition in the LWR containment atmosphere by carbon dioxide inertization during the early stages of a severe accident. The author claims that large quantities (100 tons) of carbon dioxide can be stored under pressure (20 bar) on a nuclear plant site and delivered into the containment without energy supply during the short time window between the onset of a severe accident and the mass release of hydrogen due to zirconium oxidation. The paper describes an experimental program, carried out in a 34 liter, heated autoclave, to determine the ignition limits of four-component gas mixtures constituted of hydrogen, air, steam and carbon dioxide, under conditions typical of the early stages of a severe accident. Starting from a base load of one bar air pressure at 35 °C, hydrogen, carbon dioxide and water (in the wet gas mixture tests only) were added; for representativity, pressures and temperatures were varied between 1.6-5.0 bar and 35-160 °C, respectively. The author concludes that carbon dioxide inertization, reducing oxygen concentration to 6.4% in volume (30.5 vol. % air), should prevent ignition under the range of boundary conditions considered, and that an incomplete inertization, expected to result into a mild combustion, does not appear as a reliable option.
The fifth paper, presented by TNO Environmental and Energy Technology (the Netherlands), describes the decision analysis approach used to handle the complex issue of the best appropriate hydrogen mitigation technique to be chosen for the Borssele Nuclear Power Plant. The approach comprises nine steps. The first four steps (problem identification, definition of alternatives, selection criteria, definition of value ranges for the criteria) are basically desk work carried out on TNO premises. Step five consists in giving scores to each alternative in regard of each criterion: this task was achieved by thirty-two mitigation technique experts, nuclear safety experts, and nuclear technology experts, convened to a one-and-a-half-day workshop. Although the solicitation process of experts might always raise the issue of subjectivity of the advises, the procedure is generally deemed appropriate for the present case. After the valuation of the scores, the next critical step is to assign weighting factors to the criteria, which is the matter of a second workshop to be organized with the advisers of the Dutch Nuclear Safety Inspectorate. A calculation will ensue, in order to obtain the total score of each alternative, followed by a sensitivity analysis to have an insight on the stability of the best alternative(s) when scores and weighting factors are slightly altered.

In summary, various candidates exist for hydrogen management in a reactor containment during a severe accident, for the case the containment failure cannot be excluded as a result of hydrogen combustion. Some of these potential solutions have been extensively studied with significant success; one of them, the installation of passive catalytic recombiners in the containment, has recently been decided in principle by the Belgian utility and recommended by the German Reactor Safety Commission. Research efforts should be pursued to optimize the various possible solutions, which could combine several mitigation principles, for the different types of reactors currently existing or to be installed in the future.
The session contained five papers. Three of the papers addressed accident management and phenomenological aspects of ex-vessel melt-water interactions, one paper discussed the strategy adopted in the VVER-440 Loviisa plant in Finland to prevent slow, long-term overpressurization of the ice-condenser containment, and in one paper the critical review of the potential strategies to control iodine in the containment was presented.

The external spray cooling of the Loviisa containment was described in a paper presented by IVO International Ltd (Finland). The external containment spray plays an important role in the overall severe accident management strategy at the Loviisa plant. The paper presented the thermal-hydraulic basis of the external spray design, including calculations and experiments, the design principles of the system, as well as some operational aspects of the system. The presentation clearly demonstrated an obvious but important fact that accident mitigation strategies are plant specific and that combination of systematic experimental and analytical approach is necessary to resolve complex severe accident problems.

Initially, filtered venting of the containment was considered as a measure to keep the containment pressure under control. However, accident analyses for the Loviisa conditions showed that venting had several disadvantages for the Loviisa plant design, the main drawback being the poor subpressure performance of the Loviisa thin steel shell containment. Accident studies showed that spraying of the outer surface on the containment dome in order to condense steam was an attractive alternative to venting. It has to be stressed that specific features of the Loviisa plant, such as the steel shell containment and a relatively low steaming rate, make the external spray cooling feasible.

Discussion of the paper was concentrated around some technical details of the system and some aspects of the experiments. One important issue which was briefly addressed during discussion was selection and relevance of accident scenarios used in the process of designing the system. This aspect of the problem was only very briefly addressed in this interesting paper.

The second paper was presented by the Forsmark power plant in Sweden. It discussed the basis for installing a core-catcher in Forsmark 1 and 2. It was an interesting presentation since it demonstrated the kind of problems we are facing (in this particular case the regular yearly maintenance at the plant) with regard to the efforts aimed at mitigating the consequences of postulated ex-vessel melt-water-structures interactions. The long-term coolability of core material in the flooded lower drywell is one of the crucial issues in the accident management strategy. During the initial phase of the project it was concluded that uncertainties concerning basic phenomena and accident scenarios which would govern the design and function of the core-catcher were too large to allow design specifications. The main uncertainties were connected with the following issues: breakup of the melt stream in a deep pool of subcooled water, the mode of reactor vessel failure (local or global) and, finally, the likelihood and consequences (in terms of generation of small particles and dynamic loads) of steam explosions.
The paper concluded that it is important to learn more about the severe accident scenarios and phenomena before modifications are introduced in the plant. It is especially important as the installation of the core-catcher would interfere with regular yearly maintenance activities at the plant. Another conclusion was that emergency procedures must be developed in parallel to the new knowledge that is generated by ongoing research activities. The work in Forsmark is now focusing on the evaluation of alternative accident management strategies.

During the discussion of the paper, the problem of maintenance activities in the lower drywell in the presence of a core-catcher was addressed. It was acknowledged that to make the investigated core-catcher an acceptable solution, the maintenance personnel must understand and accept the safety benefits of this device.

The next two papers addressed ex-vessel coolability and steam explosions. The JAERI paper entitled "Accident management measures on steam explosion and debris coolability for Light Water Reactors" described experiments performed in the ALPHA facility in Japan. The main objective of the experiments had been to assess the effectiveness of possible accident management measures on steam explosions and the core debris coolability in the containment. This is potentially important considering that virtually all melt-water interaction studies focus on improving our understanding of the physics of steam explosions rather than looking for practical ways of preventing or mitigating energetic melt-water interactions. Two series of experiments have been conducted at JAERI, namely spontaneously triggered steam explosions using up to 20 kg of melt (which was dropped into water) generated by the thermite reaction between iron oxide and aluminum and melt coolability experiments in which water was poured onto the melt.

In melt-drop, steam explosion experiments the effects of melt mass, ambient pressure, water temperature and melt dispersing conditions were investigated. In general, spontaneous steam explosions occurred in all experiments when the melt was dropped into a pool of subcooled water at the atmospheric pressure without the use of a dispersion device. However, steam explosions did not occur when the ambient pressure was increased from 0.1 to 1.6 MPa, which by large is in agreement with the theory. Concerning the effect of the dispersing device, it was concluded that it may reduce the probability of steam explosions. Further investigations are apparently needed in this area.

In melt coolability experiments the heat transfer between the melt and overlying water was examined. It was concluded that adding water to the containment is, in general, an effective accident management strategy, with appropriate conditions. Thus, low water subcooling and high ambient pressure would decrease the probability of steam explosions. Unfortunately, these conditions are seldom satisfied in the containment. The dispersion device could also be a possible accident mitigation measure. During discussion of the paper two issues were addressed, namely how representative of the real reactor conditions is thermite melt, and the possible mitigating effect of moderately elevated containment pressure (0.2 to 0.3 MPa). It was proposed to perform experiments in the ALPHA facility at these pressure levels.

The second paper on ex-vessel steam explosions, entitled "A study of ex-vessel steam explosions in Swedish BWRs", was presented by ABB Atom. It described computer calculations of premixing and explosion phases of postulated steam explosions in a deep, subcooled water pool. Two computer codes were used for the analysis: PM-ALPHA for premixing calculations and ESPROSE.m for explosion calculations. The purpose of the study was to provide quantitative insights on the steam explosion for ex-vessel situations, mainly pressure distribution in the pool. This information is needed to assess dynamic loads on
pedestal structures. A number of cases were calculated in which parameters such as melt flow rate, particle size, water subcooling and melt superheat were varied in order to assess the sensitivity of results to assumed model parameters and accident scenarios. The presented analysis is one of a very few steam explosion studies for deep water pools for ex-vessel situations and probably the most comprehensive one. Calculations revealed a number of potentially important mitigative mechanisms (in moderating loads on nearby structures), such as "explosion venting" and "penetration cutoff". It was noted that structural response was not calculated. The discussion of the paper addressed some aspects of the model and accident scenarios. The opinion was expressed that a parametric study on the magnitude of the applied explosion trigger would be useful in distinguishing between propagating explosion and merely amplification of the trigger.

The last presentation in this session, entitled "Potential strategies to control iodine released into the containment in the case of a severe reactor accident" was given by CIEMAT and CSN (Spain). A comprehensive and critical review, from the accident management point of view, was presented of various "classical" strategies to control iodine released into the containment under severe accident conditions. Also, other potential measures to control iodine in containment based on the current state of knowledge of iodine chemistry were discussed. Discussion of iodine behaviour was structured with regard to boundary conditions (pH, dose rates etc.), global behaviour of iodine in the containment (iodine evolution and speciation), engineered safety features (sprays, suppression pool etc.) and pH control. A rather strong statement was made concerning the effect of pH. Thus it was concluded that pH is a crucial factor for iodine volatility in the containment and, consequently, if pH is controlled during an accident the important uncertainties are limited to the phenomena of iodine-surface interactions or revolatilization. If pH is uncontrolled then the issue of pH evolution becomes very important. Iodine volatility is also strongly affected by radiation. During the discussion, the importance of pH in sumps was emphasized.

In summary, the session provided interesting insights into some phenomena crucial for accident management and containment integrity. Uncertainties with regard to the ex-vessel melt-water interactions in deep, subcooled water pools should be further reduced, which motivates continued research in this area. The influence of chemistry on fission product behaviour, both in short- and long-term, is another issue where it seems desirable to further reduce uncertainties.
SESSION 3

SURVEILLANCE AND PROTECTION OF CONTAINMENT FUNCTION
Session Summary by B. De Boeck, AVN (Chairman)

In this session, three papers were presented that covered the monitoring of the containment leaktightness and the protection of the containment integrity by means of a filtered venting system.

The Swedish Kärnkraftteknik presentation "Selection of scrubber as a filtered venting device" describes the considerations that have led to the choice of the technology for the filtered containment venting systems for all Swedish nuclear power plants. Barsebäck was the first plant in the world being equipped with a containment pressure relief system designed for severe accident conditions. The design basis event was a large LOCA with total black-out and impaired pressure suppression system. No operator actions were assumed for the first 24 hours. Based on the knowledge in severe accident phenomenology available in the early eighties, a large passive condenser consisting of a 10,000 m³ building filled with gravel stones was selected as a filter.

Considering the increased knowledge about radioactive aerosols behaviour and transport, it was possible in 1986 to reflect on many filter alternatives for the other Swedish plants. Dry and wet filter technologies were considered. The dry filters considered included wire mesh filters, sand bed filters, gravel beds and cyclones. For reasons of efficiency, volume and cost, the wet filter scrubber technology was selected.

The evaluation of the two tenders received led to the choice of the multiple venturi scrubber system (MVSS). The MVSS contains over 800 venturi tubes connected in parallel and submerged in a circular shaped container. When the MVSS is put in operation after a severe accident, gas is forced by the containment pressure to pass through the tubes. The venturi accelerates the gas to high speed (100 m/s), and the consequent underpressure sucks water from the surrounding pool through holes in the tube. Due to the combination of high differential speed and fine sized water droplets, the particles carried by the gas are easily collected on the droplets, and the decontamination factor is high. An analytical and experimental programme has shown that the efficiency was higher than required.

The influence of the choice of the analytical tools on the design of the filter system was discussed. This is still an open question for the definition of any severe accident management strategy. The answer requires either the use of several independent codes, or the implementation of a solution which is not too dependent on the results of analytical calculations.

It was also stated that if the internal containment spray is operating, venting may not even be required.

The presentation from IPSN, France, "Containment by-pass and isolation failure detection with the expert system ALIBABA" describes an expert system which is part of the tools available at the emergency technical centre (CTC) to assess the status of barriers (fuel clad, reactor coolant system, containment building) and the related safety functions (subcriticality, primary system coolant...
ALIBABA allows to detect the occurrence of containment isolation faults. The information available at the CTC includes the indication of the containment isolation valves position, the global activity measurements in the ventilation ducts and at the stack, and the local activity measurements in the auxiliary buildings (measurements above sumps or near pipes). An expert system is valuable to obtain early indication. The knowledge base of ALIBABA includes information on the penetrations (potential leakage sources) and on systems (instrumentation and various equipment).

The assessment consists of four stages. It starts with the checking of the availability of equipment and sensors. It then proceeds to the upstream approach by checking the state of the containment isolation valves. It also performs a downstream search based on the activity measurements. The last step consists of a qualitative balance which is the synthesis of the two former tasks. Each penetration is given a coefficient according to the selection path (valve position, local activity, ventilation duct activity). This allows the identification of the most probable sources of a leakage. When selected, a penetration is shown on the computer screen in its environment.

Examples of use were given during the presentation. ALIBABA has been tested during emergency drills with positive results. It is presently being improved to take into account the experience feedback. The author pointed out that the development of the system benefited from the standardisation of the French nuclear programme.

The Italian ENEL presentation 'Containment leaktightness after severe accident' discusses some ideas about methods to inform the plant operators of proper containment isolation system performance. First the present knowledge about severe accident source terms in containment is reviewed with the aim to select an appropriate radionuclide release spectrum. The following phases were considered: gap activity release, early in-vessel release, ex-vessel release, and late in-vessel release. Considering the free volume of a typical containment, a table was produced which gives the activity concentration in the containment atmosphere for the various accidental phases. To detect a possible containment leak in accident conditions, it was then necessary to find discriminating elements. From the previous table, radionuclides were identified which could play the role of tracers. The radionuclide spectrum has been referred only to the phases "gap" and "early in-vessel" because, for evolutionary and passive reactors, the accident scenarios associated with "ex-vessel" and "late in-vessel" phases appear not credible.

In a second step, the containment penetrations of a typical pressurised ALWR and the related isolation system were analysed. All the containment penetrations have been subdivided into five categories according to their characteristics. This allows to define remedy actions in case of leaks.

Finally, a containment monitoring function is foreseen in the workstations available in the control room. One of the several displays that are accessible to the operators to perform the monitoring functions is related to the containment isolation system. This screen is dynamic, in the sense that it provides real time updated information and is capable to accept commands by the operator. The information provided on this screen includes the penetration number, the isolation valve status, the main plant parameters and the system related to the various penetrations. If after an accident, a given penetration leaks, high activity level will result in both the penetration piping and in the room of the faulty penetration system. This occurrence, on the basis of the source spectrum defined in the first step, will generate a tightness failure alarm.
The two last papers were complementary in the sense that the first one presented a tool for the central emergency team, related to existing plants, and the second one, a tool for the operators in the control room, related to future plants. Both papers draw the attention on the very important topic of ensuring the containment leaktightness after an accident. Severe accident management is made more difficult when there is a leak in the containment. It is therefore essential that the containment leaktightness is ensured during normal operation, that it is monitored after an accident has taken place, and that possible corrective actions be defined in advance. If this work is performed before the design of the plant is completed, it can be used to improve the design to lower the probability of isolation failures, to ease the detection of leaks, and to facilitate remedial action.
Session 1:

Containment accident management strategies (general aspects).

Chairman: J. Rohde
CONTAINMENT MANAGEMENT IN THE EVENT OF
A SEVERE ACCIDENT IN FRENCH POWER PLANTS
(X. Grimaldi; J.P. Berger EDF-SEPTEN ;
S. Renier; P. Lemaitre EDF-EPN)

ABSTRACT

Monitoring of the quality of the containment is a preoccupation which is present at all stages of construction and operation of French PWR plant units.

In case of accident, the emergency operating procedures give priority to preventing core uncovering. Monitoring of the containment then aims to detect any abnormal feature and to limit any risk of release which might not be strictly justified by a safeguard action. In the event of a severe accident, the priority is clearly given to safeguarding the containment. The strategic options proposed to the team in charge of the situation are determined pragmatically in the Severe Accident Intervention Guide which includes the emergency procedures for containment monitoring (U2) and decompression-filtration (U5).

In all cases, maintaining the containment is based on the earliest possible detection of the leaktightness defect or of bypass. For this reason, in parallel with the operating procedures, a guide devoted to the overall monitoring of the containment has been developed for the use of crisis teams.

All the monitoring techniques and strategies chosen aim, in the event of a severe accident, to bring the plant back as quickly as possible to a situation in which any radiological releases are delayed, limited and filtered in order to enable the population protection plan to be applied.

On future plant units, it is possible to include installation progress at design stage, both in order to reduce the risk of severe accident and to limit the impact that it would have. Special attention is given to preventing energy accidents within the containment, and to ensuring the removal of power outside the containment in the long term.
1 - INTRODUCTION

On French existing plants, improvements for better mitigation of a severe accident are implemented on a pragmatic basis within the limits of their actual design and within the current process of periodic reassessment of the plants.

In these fields, regular improvements have been implemented both to lower severe accident risk and to limit the radiological consequences of an accident with severe core degradation.

2 - PREVENTION OF CORE DEGRADATION

Some years ago, EDF performed the SPI-U1 procedure, SPI as a safety function monitoring procedure and a ultimate procedure U1 which should be used by the operator when the situation is quite serious.

Now, a particular attention has been paid to the severe accident prevention by the mean of the state oriented approach. As it seemed difficult for event-oriented procedures to deal with combination of event, compounded by material or human errors.

Moreover, this physical oriented approach is currently developed as a standard practice. This involves the change of all the event-oriented accidental procedures. These new procedures already are operational on a few 1300 MWe units and will be back fitted on all the previous units. They currently are extended for the accidents which could happen during shutdown states.

This approach is completed by the implementation of a set of procedures called U procedures (U for Ultimate).

One ultimate procedure (called U3) is devoted to cope with loss of containment heat removal system during the long term phase of an accident by the mean of mobile pump and heat exchanger.

3 - MAIN FEATURES AND PROCEDURES FOR CONTAINMENT MANAGEMENT UP TO SEVERE ACCIDENT

3.1. Generalities

Some procedures came from the analysis of the WASH 1400 report adapted to our type of reactor at the beginning of the eighties and after TMI accident analysis. In particular, the fact that French PWR have large dry containments help to select the three modes of containment failure which should be considered and which have led to some specific devices.

The more recent investigations, with the beginning of the Level 2 PSA have not put in evidence significant changings for the reassessment of the existing plants.

3.2. Penetration faults

3.2.1. U2 procedure

To deal with the random failure of a containment penetration during an accident, or a failure of the safeguard circuit carrying highly contaminated water outside the containment, a special procedure (U2) has been developed with the aim of pinpointing and sealing off the leak, and subsequently to provide devices for re-injection of the
recovered contaminated water back towards the reactor building. The associated thresholds result from accident with cladding rupture studies.

3.2.2. Prevention of the penetration fault risk

It must be emphasized that any penetration fault have to be detected as early as possible. The reduction of that kind of risk is managed both on design plan and on the operating plan during normal operation up to the accident operating procedures.

Each penetration is designed following the concept of double isolated penetration. In addition to the containment periodic leak tightness tests up to the containment design pressure, the operating rules insure that the manual penetration are in closed state (a system called SEXTEN installed on each plant is intended to detect the major problems on penetrations during normal operation). Periodic inspections guarantee correct operation of the automatic isolation and the integrity of the whole confinement.

These measures are furthered by the structure of the ventilations and filtration systems, by guaranteeing the limitation of the unfiltered releases risk and accessibility to the operating rooms up to severe accident conditions. A recent and systematic review on the different plants, in a beyond design context allowed to identify "radiological risk associated penetrations". The principle followed was to consider that "radiological risk penetration" are those which can lead (in case of double fault) to an unfiltered release of containment atmosphere or a release of hot contaminated water.

3.2.3. Linkage of containment management with NSSS management in operating procedures (see figure 1)

The recent evolution of U2 procedure has been implemented with the following principles:

- firstly, monitoring of correct operation of the automatic isolating devices (in case of Safety Injection or Containment Spray system actuation, in case of primary activity detection). It allows the better possibility of detection and makes the repair easier.

- secondly: it is necessary to favor NSSS emergency operating actions as long as severe core degradation has not occurred and in the event of a conflict with respect to containment actions (example: use of Safety Injection with a leak; use of radioactive Steam Generator as an ultimate secondary heat sink). In this context, in an earlier stage of the U2 procedure, the monitoring procedure defines a list of penetrations that must be isolated if the radiological situation in the containment becomes abnormal. These penetrations should not be opened later as they are not strictly necessary for the emergency procedures during a serious situation.

- thirdly: if any containment fault is detected by the control room staff, the local crisis team must be advised to fix the isolation while the control room staff must keep concentrated on the NSSS management.

3.3. Containment basemat perforation

The risk of basemat perforation remains compatible with the organisation of emergency plans as it has been shown that the basemat perforation cannot occur during the first 24 hours (average delay: 3 days) and that the releases should be filtered. Only for one particular site (CRUAS) due to parasismic design, a U4 procedure has been implemented: flooding of the cavity under reactor building, soda injection in the containment, complementary venting to insure low containment pressure if the basemat rupture occurs.
3.4. Venting system

Particular attention is paid to the system of venting and filtering in containment management in case of severe accident (see next paragraph).

4. VENTING AND FILTERING SYSTEM - US PROCEDURE

4.1. Generalities

The venting system is devoted to manage the risk of containment integrity loss and high activity release in case of a slow pressure increase following a severe accident.

This filtration system using a sand bed filter has been implemented with a double objective:

- to limit the containment pressure close to its design value,
- to reduce the radioactive releases to a level leading to radiological consequences compatible with the emergency plans.

4.2. Basic requirements

Studies of accidental scenarios issued from the WASH 1400 report, have led to the main functional characteristics of the system:

- the time allowed before the opening of the system at containment design pressure is at least 24 hours after the beginning of the accident,
- the main characteristics of the reference fluid to be filtered are:
  * pressure: containment design pressure (0.5 MPa),
  * temperature: 140°C
  * flowrate: 3.5 kg/s sufficient to ensure that after the opening of the system, containment pressure will not overstep the design pressure,
  * standard composition (in weight): air 33%, steam 29%, CO2 33%, CO 5%,
  * aerosols: maximum concentration: 0.1 g/m3 (at 0.5MPa), expected aerodynamic mass median diameter = 5 microns
- minimum filter efficiency: 10 related to aerosols.

The last phase of tests of the research program have been realized with a complete system on an industrial scale. The efficiency has been measured during 25 hours and the retention rate obtained for caesium is higher than 100. This efficiency measured with regards to aerosols can be applied to iodine under CsI form.

Molecular iodine which is probably present in the containment under severe accidental conditions is easily trapped by many materials. The retention of gaseous molecular iodine was measured and the lowest retention rate value was 9.

Two complementary requirements have been the subject of additional studies and improvements of the system. They are discussed in paragraph 4.3.

- hydrogen risk at the opening of the system,
4.3. Description of containment filtering and venting system

The system is presented in figure 2. All the French PWR's are now fitted out with the complete system corresponding to the basic requirements:

- a containment penetration with double isolation made by two manual valves in series located outside the containment as close as possible to it,
- an orifice plate located just downstream. It ensures the gas expansion near to the atmospheric pressure,
- a sand bed filter made of stainless steel, the filtering area is about 41 m²,
- a device for measuring radioactivity released downstream the filter. The measurement method based on a gamma-spectrometer allows to measure separately iodine and caesium activity released.
- an independent evacuation duct located inside the normal effluent stack of the plant.

Complementary devices:

- Hydrogen risk: provision is provided against the occurrence of combustion of the mixture in the pipe work and filter as condensation could lead to hydrogen enrichment.

The measures consist in a preheating system using an electric heater downstream the conditioning fan.

- Direct and indirect dose rate values (sky shine) induced by the limit values corresponding to the activity trapped in the filter for the most pessimistic hypothesis, have been reduced in order to keep a situation that can be managed on site (approximately to one order of magnitude) by implementing a solution based on a prefilter in the containment (this coarse filter uses metallic media and is designed to stop 90% of the aerosols activity).

4.4. Operating conditions

It seems worthwhile recalling that safety depends on containment and this principles should not be brought into question by the existence of a venting device. It would be preferable to postpone as long as possible and if possible to avoid putting the device in operation. That is why complementary studies are currently performed to be sure of the ability of global containment (included penetrations and recirculation loops) to withstand a pressure higher than the design. (in a 0.5 to 0.75 MPa range).

The decision to put the system into service is the responsibility of the plant manager in conjunction with the local and national level of the crisis organisation. The closing action supposes that a mean for residual heat removal, different from venting system has been recovered.

5 - SEVERE ACCIDENT INTERVENTION GUIDE

5.1. Goals of the guide

When EDF and The Safety Authority have at the beginning of the eighties to implement ultimate procedures and to build complementary devices, the need of a coordination tool was felt.
At the same time, the French Atomic Energy Commission (CEA), technical support of The Safety Authority had gathered a great amount of knowledge through its severe accident research program, and this became an objective to make it operational. Furthermore, the feedback experience of Crisis Teams shows how difficult it is to obtain a fast agreement among specialists to advise operators.

Lastly, accident management procedures, which were designed for fairly serious accidents, can't cover severe accident, because their main aim - to keep the core safe is then rendered obsolete.

When the first and the second barrier have obviously faulted (cladding and reactor coolant system), the very first objective is to maintain as long as possible the containment safety:

- to gain time for the implementation of emergency plans,
- to avoid or minimize radioactive releases by atmospheric pathway (leak or failure of containment) and by hydrological pathway (after basement melt-through by the corium)

5.2. The compulsions and the limits of the guide

The guide must:

- operate immediately and without having to adapt materials,
- integrate the ultimate means already implemented (U2 and U5),
- be consistent, as possible, with EOP's,
- give a recommendation on each major choice based on the best estimate of the current understanding of phenomena,
- and of course, avoid actions that may worsen the situation.

The guide is not fitted to cope with early failure of the containment due to hydrogen generalized detonation, and steam explosion, because their origins seems very unlikely with large and dry containment.

5.3. General method and instrumentation

Up to now, it has appeared impossible to implement actions derived as a result of the physical state examination for a main reason: the information collected from existing measurement devices can't be supposed reliable without new and specific studies.

So, except for the assumption of the entry criteria to decide the use of the guide, the required actions are as much as possible, only the result of the system availability diagnosis. This has led to a predefined matrix (system-actions) which is very simple to use and should remain efficient even in a high stress condition.

Nevertheless, current studies are performed to ensure that the availability of some minimum instrumentation should be verified during a severe accident scenario. The major points are related to entry criteria (core exit temperature and containment dose rate) and to US procedure (containment pressure).

5.4. Strategies

First, the containment structure must be protected by:

- maximum spray to keep pressure level low,
- if necessary, use the venting and filtering system according to U5 procedure.

Secondly, the releases must be reduced by:

- maximum spray to help iodine and aerosols deposition,
- systematic water plug on Steam Generators to limit the risk of by-pass in the event of tube rupture,
- continuing to apply U2 procedure to localise and repair (if it is possible) the leaks of the penetrations.

Thirdly, because any power release in the containment would induce a pressure increase, and potential use of venting, remove decay heat out of the containment by:

- maximum cooling with the steam generators, only if they have not been detected radioactive previously,
- try to cool with cold exchangers into reactor building, only after 1 day.

Fourthly, the concrete basemat melthrough is delayed by maximum water injection to the vessel in order to cool, at least partly, the corium.

Fifthly, avoid worsening the situation by an extreme care of risks of energetic events:

The pressurizer relief valves are kept opened in order to maintain the vessel at low pressure and avoid the Direct Containment Heating.

In case of late restauration of safety injection, the flow is increased slowly if possible.

At last, borated water is used preferably to clear water to avoid reactivity problems.

5.5. Guide evolutions

This guide, which is voluntarily rough, gives the state of the art procedure for operation.

Some subjects are currently under study: use and monitoring of steam generators, strategy with hydrogen concentration monitors, long term control, detection of vessel melt through.

6 - CRISIS TEAMS ORGANIZATION FOR CONTAINMENT MANAGEMENT

6.1. Principles

At last, for a real operational efficiency of containment management (see paragraph 3.2.3), it has been judged to be positive to implement a guide for the crisis team: the applicable procedures for the National and the local crisis teams for monitoring and restoring the confinement as soon as an accident situation has occurred, should the ultimate procedures be applied or not.

This guide formalizes a set of monitoring and diagnosis resources drawn up for containment defects. It is based on the capability of the national and local crisis confinement specialists with the aid of some analysis tools to perform diagnosis based on a great number of radiological informations.
When it is obvious that the situation will go wrong, the capability of prognosis can be used to decide some early isolation actions and also to manage some decision conflicts between NSSS operating actions and containment, more accurately in case of severe core degradation.

Currently on experimental phase during crisis drills, this guide will be systematically applied by the crisis teams as soon as a level 2 On Site Emergency Response Plan is activated on the site, i.e., as soon as the operating team uses an accident procedure for controlling a design basis accident.

At operational level, a large part of the crisis organisation set up by EDF following an accident is directly concerned with carrying out the actions deriving from application of the Containment management guide for Crisis Teams.

6.2. On site Emergency response plan

Implementation of the On-Site Emergency Response Plan involves the setting up of a decision-making Center, 3 operational Control Centres, and an analysis and reflection center.

The operational control centers comprise:

- the Local Control Center (LCC).
- the Health Physics Control Center (HPCC).
- the Logistic Control Center (LGCC).

- The Local Control Center (LCC), which is responsible for the operations and safety functions on the damaged units as well as initial first aid for the injured. The head of this control center is based beside the control room, and the actions of the control room shift are under his control: it is composed of two engineers, giving orders to the shift team and working closely with the safety engineer.

- The Health Physics Control Center (HPCC), responsible for the centralization and interpretation of radiological and meteorological measurements, in order to evaluate the radiological consequences.

- The Logistics Control Center (LGCC), responsible for checking the movement of personnel, coordinating use of vehicles, and in general ensuring internal logistics.

The personnel of these 3 control centers work according to the instructions and action sheets laid down in the On-Site Emergency Plan, and provide the information and assistance required by the 4th CC: the Management Control Center (MCC).

This Management Control Center (MCC) is the decision-making centre, and is run by the plant manager or his representative. It is solely responsible for the decisions to be taken to ensure the safety of installations, the protection of personnel, and controls the supplementary actions to be carried out by the LCC, the HPCC, and the LGCC.
At the local level, this organization is supplemented by an analysis and reflection center: The local crisis team (LCT) which is located in an appropriate room, called the Technical Support Center (TSC).

6.3. EDF National Organization

The national organization, which is based at EDF headquarters in Paris, comprises: a decision-making center and a center for analysis and reflection.

- The Management CC, led by the head of Nuclear Power Plant Operations Division or his representative, which provides a central decision-making level in permanent contact with the plant manager. It provides links with EDF General Management.

- The National Crisis Team (NTc) responsible for supplementing the information from the Management CC, and for reporting its opinions and recommendations as required.

The national crisis team is in close contact with the local crisis team which provides it with information and analysis. It is also in contact with the team of the Safety Authority.

6.4. Roles of operational centers and crisis teams in containment management

* The operators, applying their procedures, are responsible for:

  - detecting any containment isolation defects after confirmation of automatic orders and informing the safety engineer and the Local Control Center (LCC).

  - carrying out preventive or corrective actions requested by the safety engineer or the LCC and which can be performed from the control room (or exceptionally locally if this does not hinder the application of procedures and does not require any special facilities).

* The safety engineer, within the context of the application of his procedure, monitors:

  - activity of the primary circuit

  - activity at the plant stack

  - the dose rate within the containment and in the sumps of the peripheral buildings.

As the containment dose rate exceeds 0.02gy/h, he requests the operators to take the following preventive measures called "early containment isolation":

  - move all ventilations onto iodine filters,

  - confirm that all containment penetrations which are not mandatory to the NSSS control strategies in progress are isolated (list drawn up beforehand in keeping with control procedures).

In close collaboration with the LCC, the safety engineer requests the application of the U2 procedure when the following thresholds are exceeded: primary circuit activity, measured stack activity, containment dose rate, dose rate in auxiliary buildings sumps. In addition to confirming previous actions, this procedure makes it possible to reinject the radioactive waste into the reactor building if necessary.
Local and national crisis teams (LCT, NCT) applying the Crisis Team Guide, with the support of the operational PC's:

- carry out monitoring similar to that performed by the safety engineer (human redundancy).
- analyse any defects detected by the control team in order to initiate corrective action in the correct order.
- carry out special monitoring of radiological risk penetrations which they ask to be equipped with mobile detectors as soon as the accident begins and follow the changes in the measurements of activity and dose rate performed on the plant and in the environment by the HPCC.
- ensure a potential leak diagnosis by the mean of systematic analysis of this measurements (collected leaks, potential bypass),
- manage any action which might cause conflict between safeguarding the core and restoring the containment (eg: analysis of the possibilities of isolating one train of the Safety Injection System or Containment Heat Removal System, in recirculation configuration, and affected by an external leak). Some prognosis calculations aids can be used for this purpose.

The operational PC's are in charge of monitoring operations and preventive or corrective actions requested by the crisis teams and which need to be carried out locally:

- monitoring:
  - fitting and monitoring extra mobile activity detectors, on the radiological risk penetrations or potential pathways leading to rapid bypass of the containment.
  - measurements performed in the environment.
- for preventive actions: confirmation that the doors of rooms likely to be contaminated have been closed.
- for corrective actions: attempts to isolate actuators locally or to transfer isolation onto other actuators.

7 - FUTURE PLANTS

7.1. Principles

A significant improvement at the design stage of the safety of the next generation of nuclear power plants has been asked by safety authorities. If the search for improvement is a permanent concern in the field of safety, the necessity of a significant step at the design stage clearly derives from better consideration of the problems related to severe accidents, not only in the short term but also in the long term, due to the potential contamination of large areas.

The EPR is a PWR which will be based on the experiences gained by EDF, the German Utilities, Framatome and Siemens.

The most significant safety improvements of the EPR compared to existing plants are:
7.2. Overall approach

Despite the uncertainties with respect to the requirements actually necessary to ensure a convenient retention, the overall approach to limit the external source term is to maintain the Containment function. This approach aims at:

- avoiding early Containment failure or bypass,
- cooling of the corium in the Containment and retention of fission products both by water covering,
- preservation of the Containment functions (low leakage towards the environment, reliable isolation of the Containment on demand, prevention of the basemat meltthrough, ultimate pressure resistance to cope with the more likely energetic events),
- pressure reduction of the Containment by means of heat removal,
- collection of unavoidable leakages into the atmosphere of the Annulus and release to the stack after filtration.

The investigations concerning core meltdown accidents are based on findings of national and international reactor safety research projects. Various accident sequences are considered that could lead to core meltdown. They can be classified as follows:

1) Sequences leading to a core melt with the primary system at low pressure.
2) Sequences leading to a core melt with the primary system at high pressure.
3) Sequences causing a severe core damage and a containment bypass.
4) Sequences with severe core damage and an independent containment loss or preexisting leaks.

The sequences of type 1 are considered for the third barrier design.

The sequences of type 2 will be reduced to such a low probability that their consequences need not be considered in the definition of containment design loads. This can be achieved by provision of specific means for prevention of the high pressure core melt scenarios.

Similarly, the sequences of type 3 and type 4 will be addressed in the EPR design, and in particular in the design of containment isolation provisions and in the main and auxiliary fluid systems design, e.g. those foreseen for mitigation of Steam Generator Tube Rupture accident scenarios. As regards the type 4 sequences, specific attention is paid to the possible accident scenarios during shutdown and refuelling conditions.

7.3. Implementation

The EPR strategy includes both preventive measures aiming at practically eliminating the corresponding accident situations and mitigating features aiming at limiting the releases within the prescribed limits.
7.3.1. Preventive measures

The EPR design includes:

- The prevention of high pressure core melt situations, firstly by a high reliability of the decay heat removal systems, complemented by depressurization means (pressurizer relief valves). This depressurization at the same time eliminates the danger of direct containment heating. The consequences of an instantaneous break of the RPV with full cross section at a pressure of about 20 bar are nevertheless taken into account for the layout.

- The prevention of hydrogen combustion with high loads (high turbulent global deflagration/DDT/detonation) by reducing the hydrogen-concentration in the containment at an early stage by catalytic H₂-recombiners and, if necessary, by selectively arranged igniters. The prevention of molten core-concrete interaction contributes in reducing the amount of hydrogen generated. The potential effects resulting from the deflagration phenomena are considered in the design of the containment and of the internal structures.

- The prevention of ex-vessel steam explosion endangering the containment integrity by minimizing the amount of water where the corium is spread.

- The prevention of the molten core-concrete interaction by spreading the corium in a dedicated spreading chamber. This original EPR feature consists in a large area (about 150m²) outside the reactor pit.

The reactor pit and the spreading compartment are connected via a melt discharge channel which has a slope to the spreading area and is closed by a steel plate. This steel plate (possibly covered with refractory material) resists meltthrough for a certain time in order to accumulate the melt in the pit.

The spreading compartment is connected with the In-Containment Refuelling Water Storage Tank (IRWST) with pipes for water flooding after spreading; these pipes are closed during normal operation and accident conditions.

7.3.2. Mitigating features

- The limitation of the containment pressure increase by a dedicated containment heat removal system which consists in a spray system, with a possibility, in the long term, to subcool the water and therefore to decrease the containment pressure down to the atmospheric pressure.

Even if all other systems are unavailable, the containment design pressure (= 7,5 bar) grants a grace period of about 12 to 24 hours after the accident before having the necessity to use the spray system. This design pressure provides a comfortable margin above the maximum pressure expected for the PCC design basis accidents (LOCA or Steam Line Break), which is in the range of 5.5 bar.

- Finally, the overall confinement functions are ensured by a double wall containment.

That double wall containment consists of:

- an inner wall in prestressed concrete without liner,
- an outer wall in reinforced concrete,
- a basemat in reinforced concrete.
In this concept the structures and systems contributing to the containment function comprise:

- the inner and outer containment and the space between them, which is designated as the annulus. This annulus is maintained at a subatmospheric pressure in order to collect all possible leaks through the inner wall and filter them before release to the stack,

- systems required for isolation and for retention and control of leakages,

- systems required to maintain pressure and temperature conditions inside the containment within limits compatible with leaktightness and structural integrity of the containment.

All system penetrations through the containment inner wall are provided with isolation valves, either locked closed or automatically closing in case of an accident. Leakages from or through such fluid systems are collected in peripheral buildings and can then be routed, via filtration units, to the plant stack.

Leakage collection is in addition foreseen for openings for equipment and personnel access, equipped with hatches or locks, as well as for ventilation system penetrations, which are open to the containment atmosphere.

8 - CONCLUSION

The evolutions undertaken from both the design and the operational standpoint, on current French nuclear power units, mean that it is now possible to conceive of a pragmatic management capacity for the containment in case of a severe accident, consistent with the population protection plans.

On future plants it is possible to include improvements to reduce the risk of severe accident and for limit their impact. Special attention is given to the prevention of energy accidents in the containment and to provide energy removal outside the containment in the long term.

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CONTROL ROOM OPERATORS  SAFETY ENGINEER  CRISIS TEAM

SITUATIONS

Incidents

Accident

Accident with multiple failures

NSSS EOPs

Early containment isolation

Systems survey

Permanent monitoring of safety functions

Severe accident intervention guide

Containment management Guide

Permanent monitoring of safety functions

Severe accident intervention guide

NSSS management

Containment devoted actions and monitoring

LINKAGE OF PROCEDURES ON FRENCH PLANTS
CONTAINMENT VENTING AND FILTERING SYSTEM

Figure 2
A KNOWLEDGE BASED
SEVERE ACCIDENT HANDBOOK FOR PWR

by

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A KNOWLEDGE BASED SEVERE ACCIDENT HANDBOOK FOR PWR

ABSTRACT

During the last decade the level of knowledge about severe accident phenomena has increased dramatically. The improved understanding has been achieved by extensive research but also from feedback of experience from actual incidents/accidents such as Three Mile Island and Chernobyl.

In Sweden mitigating measures such as filtered venting and external water source were implemented at all nuclear power plants by 1988. In parallel the Emergency Operating Procedures (at Ringhals called Emergency Response Guidelines, ERG, and Beyond ERG, BERG) were developed to include these new features. However, the accident management system has since then been further improved and one important aspect is the long-term accident management. The new information obtained has been one of the basis for a new knowledge based handbook to support the unit leader and the Technical Support Center. The handbook contains information concerning specific issues in the BERG and advice how the organization can manage a long-term severe accident situation.
1. **BACKGROUND**

At the time of ordering and the construction of the Swedish PWR's the knowledge of severe accident phenomenology was poor. The plants were constructed to operate at 100% power with yearly short outages. After the TMI accident it became obvious that much more attention must be directed towards severe accident phenomena and accident management.

At the time of the TMI accident Emergency Operating Procedures (EOP) were event oriented. However, the TMI accident showed that these EOP's needed improvement and within the Westinghouse Owner's Group new symptom based procedures were developed. These were called Emergency Response Guidelines (ERG) (ref 1) and were implemented at nuclear plants of Westinghouse design, including Ringhals 2-4. The ERG's are relevant for the operators up to the time of vessel melt-through.

In 1986 the Swedish government issued a requirement to the Swedish utilities to implement a program of severe accident mitigation measures before the end of 1988. The basic guidelines and criteria, which by the government decision shall apply to the severe accident mitigation measures of Swedish nuclear plants are:

- The same basic requirement regarding the maximal quantity of released radioactive substance shall apply to all reactors irrespective of site and power.
- Land contamination, which impedes the use of large areas for a long period, is to be prevented.
- Deaths in acute radiation disease shall not occur.
- Incidents of extremely low probability are not to be considered.

The requirements may be considered as fulfilled if a release is limited to 0.1% of core inventory of Cs-134 and Cs-137 in a 1800 MW thermal power reactor provided that the other nuclides of significance from the point of view of land use are retained to the same degree as Cs.

In order to meet the requirement an accident management strategy was defined aiming at a final stable state in which the core is cooled (preferably in the reactor pressure vessel but in the worst case on the containment floor) in an intact containment at low pressure. The plant modifications needed to implement this strategy consist mainly of adding a redundant diesel driven water injection system operating through the containment spray system and a filtered containment vent system. As a design basis case for the new systems total loss of all core cooling (including loss of all AC power) was chosen. In essence the accident management strategy means that when all attempts to terminate the accident and recover the core cooling in the pressure vessel have failed, water is injected into the containment until the bottom of the reactor vessel is covered by water and cooled. Containment venting is used to control the containment pressure and prevent overpressurization of the containment. The strategy is illustrated in Figure 1 for a PWR of Westinghouse design.

As these mitigating devices were developed, it became obvious that the EOP's must be further developed and also cover the time until containment failure, as the key function of the new devices is to protect the containment. These EOP's were developed by Westinghouse and were called Beyond ERG (BERG), see ref 2.
In 1989 a project for long-term accident management, the so called FRIPP project was started. With long-term is meant up to five years, in contrast to the earlier accident management, which usually deals with the time frame up to a few days. Important issues investigated were dose rates, chemistry, radiolysis, system leak tightness, etc. It seemed appropriate to develop a guidance for the organization according to the findings in the FRIPP project. Also important was the understanding that it is impossible to foresee all possible situations, in particular in a late stage of a severe accident. This lead to the conclusion that it was necessary to complement existing EOP's with a knowledge based handbook.

This paper describes the purpose of the Handbook, its development and a brief description of the content is given.

2. DEVELOPMENT OF THE HANDBOOK FOR UNIT LEADERS

It was recognized that the main purpose of the handbook was to support the unit leader and the Technical Support Center in two aspects:

- to support in specific issues in the BERG
- to support the long-term accident management.

In Figure 2 the relation is shown to other EOPs, i.e. ERG's and BERG's.

In the discussion of development of the handbook, certain aspects were found to be important. To obtain acceptance of this type of documentation it is important that plant operators and technical staff participate in the development of the handbook. This further reinforces one of the benefits of the development of mitigating measures described above, i.e. developing the mental preparedness of the plant people that a severe accident may occur. This is a safety enhancement in itself although it is difficult to evaluate.

Therefore the project was performed by three persons from the Ringhals plant and two persons from the main office. The group covered operational, severe accidents and human factors knowledge. The project was started in 1991 and was finished early 1993.

3. CONTENT OF THE HANDBOOK

In this chapter the title of each chapter in the handbook is given including a short description.

Chapter 1: User manual

This is a chapter defining the purpose of the handbook.

Chapter 2: Accident mitigating measures

This chapter is a complement to the BERG procedures.
Chapter 2a: Strategy for hydrogen control

In the BERG procedures much emphasis is put on the hydrogen control. In particular, if the hydrogen concentration is larger than 10 vol% in dry air, the BERG procedures do not allow containment spray before consulting the Technical Support Center. This chapter gives advise of possible measures in such a situation.

Chapter 2b: Measures to minimize leakage

This chapter gives advise how to avoid leakage from the containment. For example, it is recommended to use only one train in each system to minimize contamination of systems in the auxiliary building. Low pressure in the containment is also recommended.

Chapter 2c: Core Damage Assessment

This chapter refers to documents of how to assess the core damage with:

- radiation monitors
- process parameters
- hydrogen concentration
- Post Accident Sampling System

Chapter 2d: Reactor vessel failure

This chapter supports the BERG procedures for determining if vessel melt-through has occurred. Until this has been established the operator must try to prevent vessel melt-through in parallel of protecting the containment. Although the time is not crucial, it is important to determine vessel failure.

Chapter 2e: Strategy for injection of water into the containment

Here the background is given for the BERG procedures.

Chapter 2f: Strategy for venting

This is also a background chapter for the BERG procedures.

Chapter 2g: Measures to restore containment integrity

This chapter suggests immediate measures to stop possible leakage paths. It also prepares for long-term accident management.

Chapter 3: Long-term measures

This chapter is to a large extent based on the FRIPP project briefly described above.

Chapter 3a: Measures to keep the heat sink

This chapter describes possible heat sinks and the necessary heat sink capacity for the decay heat. The ultimate heat sink is the scrubber.
Chapter 3b: Measures to adjust the pH value

It was shown in the FRIPP project that the pH value is important. This chapter gives the measures to adjust the pH value in the containment sump.

Chapter 3c: Measures to minimize water leakage

This chapter deals with further arrangements how to keep the containment intact for an extended period of time.

Chapter 3d: Measures to increase the containment water level

An important task in the BERG procedures is to increase the water level in the containment. This chapter describes the possibilities to do this.

Chapter 3e: Measures to lower the containment water level

This chapter describes the different possibilities to lower the containment water level in the long-term.

Chapter 3f: Measuring the containment dose rates and chemistry

This chapter describes briefly the possibilities to measure dose rates and chemistry parameters. Mostly this refers to the Post Accident Sampling System (PASS) and relevant documents.

Chapter 3g: Strategy for long-term hydrogen control

This complements chapter 2a. For example in the long-term hydrogen control radiolysis has to be considered.

Chapter 3h: Strategy for long-term venting

Although undesirable, the strategy for long-term venting is given in this chapter as an ultimate possibility.

Chapter 4: Instrumentation

In a severe accident situation very adverse conditions prevail in the containment. In this chapter the instrumentation reliability is discussed. Also the necessary type of power is discussed. Some crucial instrumentation in a severe accident is backed up with batteries for 8 hours.

Chapter 5: Safety measures for the personnel

This chapter gives the relevant documentation for the behaviour at emergency situations. Important information or documentation about the radiation levels in the station is given.

Chapter 6: AC power

In this chapter instructions are given how the AC power should be started again after total blackout and DC power has been exhausted
Chapter 7: Communication

Communication between the control room and the Emergency Control Center is vital and this chapter describes the different possibilities for this.

Chapter 8: Mobile resources

This chapter gives information of how to find mobile pumps, hoses and other vital equipment in emergency situations.

Chapter 9: Principles for non-damaged units

At the Ringhals site there are four units. Some of the auxiliary systems are linked together in pairs, such as ventilation, fresh water, sea water intake, etc. In this chapter this is considered and principles for, e.g., ventilation to avoid contamination in the non-damaged units are discussed. Other important measures is information to the personnel and possible help to the damaged unit. All communication should be through the Emergency Control Center.

4. SPECIFIC ISSUES

At the Ringhals PWR the unit leader is allowed to make many decisions. There are, however, two important judgements the unit leader should consult the Technical Support Center for advice. These are:

- If vessel failure has occurred
- How to handle a hydrogen concentration larger than 10 vol% in dry air.

These are treated in chapter 2a and 2d, respectively, in the Handbook.

4.1 Judgement of vessel failure

The BERG's are initiated by a temperature larger than 650 °C in the Core Exit Thermocouple (CET). This means that the ERG's and the BERG's are used in parallel until vessel failure is confirmed. At this time the operators can leave the ERG's and devote their effort solely to the BERG. However, the time to confirm vessel failure has been found not to be crucial and the operators can together with experts take their time to identify this specific event.

The parameters studied to verify vessel melt-through are:

- Primary system pressure
- Containment pressure
- Containment temperature
- Hydrogen concentration in containment
- Containment water level
- Primary system Hot leg temperature
- CET temperature
- Containment radiation

The most difficult situation to judge vessel melt-through is after core degradation. The indications are then very similar for Surge Line rupture caused by creep rupture and vessel
melt-through. There is, however, a possibility to judge this from an existing hydrogen monitor, which includes a temperature monitor, in the cavity. This instrument will behave differently for these cases or stop functioning at vessel melt-through.

4.2 Strategy at high hydrogen concentration.

At hydrogen concentrations $<10$ vol% in dry air the operators should try to use the recombiners (they have, however, a rather low capacity) or start the containment spray to initiate a hydrogen burn. At these relatively low hydrogen concentrations a hydrogen burn cannot threaten containment integrity and will only rarely exceed rupture disc pressure leading to release to the scrubber.

When the hydrogen concentration $>10$ vol% in dry air the correct actions are very difficult to predict. It is not allowed to use the recombiners as these can initiate a hydrogen burn. The two remaining possibilities are then to vent the containment to the scrubber or to wait with a high, steam inerting content ($>55\%$). The second option gives the operator more time to evaluate the situation. This will also decrease the amount of radioactivity due to decay, which for noble gases is relatively important during the first 10 hours. The first option is the only method that can resolve the hydrogen problem. The release should be limited as much as possible and when a hydrogen concentration $<10$ vol% in dry air has been reached the venting can be stopped. If the containment has large leakages, venting must be performed as soon as possible to limit contamination in the auxiliary building.

5. CONCLUSIONS

In this paper the development and the content of a knowledge based handbook have been described. It is a complement to the event based and symptom based emergency procedures. The main reason to develop these was the understanding that it is impossible to foresee all possible situations, in particular at a late stage of a severe accident. Besides, important findings had been established in a project investigating the long-term severe accident issues. The development of the handbook was performed by persons with operational, severe accident and human factors knowledge. The resulting handbook contains advice about specific issues in the BERG as well as advice to the organization what to think of in the long term accident management if, unlikely, a severe accident would occur.

We believe that the work with mitigating measures has increased the mental preparedness that a severe accident may occur, which is a safety enhancement in itself, although hard to quantify. However, it is important to continuously work with this issue, follow up new findings and improve existing documents. We feel this handbook has contributed to this work.

6. REFERENCES


Figure 2: Relationship between Emergency Operating Procedures at a Swedish PWR
ACCIDENT MANAGEMENT STRATEGIES AND PROCEDURES
AT THE FORSMARK NUCLEAR POWER PLANTS

by

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ACCIDENT MANAGEMENT STRATEGIES AND PROCEDURES AT THE FORSMARK NUCLEAR POWER PLANTS

ABSTRACT

In Sweden all nuclear power plants are equipped with a filtered containment venting system and back-up sources to the containment spray system. These measures reduce the environmental consequences of a severe accident and were implemented by 1988.

Emergency Operating Procedures (EOP) have been developed at Forsmark including these new features. Two levels of EOP:s are used in the control room, one by the reactor operator and one by the shift supervisor.

A knowledge based handbook for severe accidents has also been developed. This will be used by the accident management centre in case of a severe accident. A first version of this handbook exists. Work has started to upgrade this version.

This paper is intended to give a summary of accident management strategies and procedures with emphasis on severe accidents at the Forsmark NPP:s.
1. INTRODUCTION

Forsmark is a site with three BWR plant of ABB Atom design with internal circulation pumps. Forsmark 1 and 2 are twins with a net electrical output of 1000 MW each. They started commercial operation in 1980 and 1981. Forsmark 3 has a net electrical output of 1200 MW and was taken in commercial operation in 1985.

By the end of 1988 a severe accident mitigation programme was completed for all nuclear power plants in Sweden. The strategy to handle a severe accident is to a high degree based on this programme. In parallel the documentation for accident management has been developed. From a user point of view we make a difference between documents in the control room and in the accident management centre. Emergency Operating Procedures have been developed for the staff in the control room. A knowledge based handbook has been written to be used in the accident management centre in case of a severe accident.

In a separate project the long term effects of a severe accident has been studied. The goal of the project was to achieve an increased knowledge about severe accident phenomena for a time period of up to five years after the beginning of an accident. Some of the conclusions from the project have been used in the development of the EOP:s and the handbook mentioned above.

2. ACCIDENT MANAGEMENT STRATEGIES

Based on the severe accident mitigation programme a strategy to cope with severe accidents has been developed. The development of this programme is described in ref. 1.

By installation of containment venting systems at the Forsmark NPP:s late overpressurisation after a severe accident has an extremely low probability. If the venting system is activated after a severe accident the release of Cs-134 and Cs-137 will be less than 0.06% of the core inventory of these isotopes.

A system has also been installed to protect against an early overpressurisation of the containment caused by a large LOCA combined with a reduced pressure suppression function. This system consists of a large pressure relief line with a rupture disc and valves for reclosure after activation.

In case of reactor vessel melt-through it is important to reduce the core-concrete interaction. Therefore the area below the reactor vessel is flooded before vessel failure using water from the condensation pool.

In a severe accident the containment spray is used for the following reasons:

- To reduce the pressure in the containment
- To wash out aerosols from the containment atmosphere
- To fill water in the containment up to a desired level
To increase the reliability of the containment spray a backup of the water supply to the system is available. At Forsmark, diesel-driven pumps can be connected to the fire water system. Thus the spray system will work also in case of a total black-out.

As mentioned earlier, a project has been run in order to study the long term effects after a severe accident. This is described in ref. 2. Conclusions from the project have been used in the development of the documentation concerning the following issues:

- Water level in the containment after vessel failure
- Chemistry in the containment
- Temperature in the containment

A water level slightly above the bottom of the reactor vessel is recommended. A higher level of water might cause an increased leakage of water through the containment.

To reduce the corrosion rate of steel in the containment an acidity in the range 10-10.5 and a temperature of less than 100 degrees C is recommended.

3. ACCIDENT MANAGEMENT PROCEDURES

Forsmark NPP:s have five levels of procedures to cover sequences from normal operation to severe accidents. This is shown in the lower part of figure 1. The procedures in the two lower levels are used in normal operating conditions. The three upper levels are procedures for incidents or accidents.

Emergency Operating Procedures

The EOP (Emergency Operating Procedure) for the reactor operator is a symptom based step by step procedure. The EOP for the shift supervisor on the other hand is a function based procedure.

Use of EOP is initiated by reactor scram or an event which should have caused a reactor scram. By use of the EOP the shift supervisor checks the four safety functions:

- Reactivity
- Core Cooling
- Heat Sink
- Radioactive Barriers

The first priority in an accident situation is to have the reactor subcritical and to keep the core cooled. However, if it is not possible to prevent core meltdown and failure of the reactor vessel the first priority will at some time shift to maintaining the integrity of the containment.
In a severe accident the EOP:s are of limited use for the crew in the control room. Instead the measures are taken by guidance from the accident management centre. An important document at this stage is the handbook indicated at the top of the lower picture in figure 1. This handbook will be described shortly in the next section.

**Handbook for Severe Accidents**

The handbook for severe accidents is knowledge based. It contains the most important results from severe accident analysis performed for the Forsmark plants. The handbook is subdivided in a number of chapters where also references to further reports are given. The most important headings in the handbook are the following:

- Use of mitigating systems for severe accidents
- Alternatives to restore AC power
- Actions to minimize radioactive leakage from the containment
- Estimation of the degree of core damage
- Integrity of the containment
- Strategy for hydrogen control in the containment
- Instrumentation in the containment
- Communication
- Principles for operation of neighbouring reactors
- Long term cooling of the containment

Each chapter in the handbook contains general information and recommendations to be used by the accident management centre as basis for decisions. The handbook is intended to cover both the short and the long time frame after the initiating event.

In many cases it is necessary to take long term effects into consideration early. An example is filling the containment with water up to a certain level after reactor vessel melt-through. It is simple to increase the water level but the opposite can be complicated due to the high activity of the water.

The organisation at normal operation is shown in figure 2. There are three levels of operation management. VHI (officer on duty) does not take part in any decisions during normal operation. However in an emergency case VHI substitutes operation management level 1-3 until the accident management centre is established.

The organisation in case of disturbance and accidents is shown in the upper part of figure 3. The accident management centre is manned mainly from the not concerned units.
The use of different documents in the three levels of the organisation is shown in the lower part of figure 3. The handbook for severe accidents (denoted as "Technical Handbook for AL" in the figure) is used by operation management level 1. The EOP is used by operation management level 2 and 3 and also in the control room by the shift engineer and the reactor operator.

4. FUTURE WORK

As mentioned above work has started to upgrade the handbook for the accident management centre.

In this work some chapters will be rewritten and new knowledge will be included. For some issues new chapters will be written.

A proposal is that new chapters with the following headings will be written and included in the handbook:

- Judgement of the status of the reactor vessel
- Measures to adjust the acidity in the containment after a severe accident
- Measurement of activity level and chemical conditions in the containment

An important activity is of course also to keep the presumtive users of the handbook informed about the content following a planned schedule.

5. REFERENCES


Figure 1

OPERATIONS PHILOSOPHY

- EFFORT
- MOTIVATION

PROCEDURES
- METHODS
- RULES

COMPETENT PERSONNEL
- TECHNOLOGY
- RESOURCES

DOCUMENTS, ACCIDENT MANAGEMENT

- HANDBOOK

- EOP FOR SHIFT SUPERVISOR
- EOP FOR REACTOR OPERATOR

- PLANT OPERATING PROCEDURES
- SYSTEM OPERATING PROCEDURES
ORGANISATION AT NORMAL OPERATION

Figure 2

COMPANY MANAGEMENT

MD

Appointed deputies

cFX

OPERATION MANAGEMENT

MD

Appointed deputies

(level 1)

UNIT-Manager

Appointed deputies

(level 2)

Operations Manager

Appointed deputies

(level 3)

OFFICER ON DUTY
SUBSTITUTE IN CHARGE
CONCERNING
OPERATIONAL ISSUES

VHI
Figure 3

**DOCUMENTATION**

**DISTURBANCE**
- Operations management, level 1:
- Operations management, level 2 and 3:
- Control Room: Shift eng., Reactor op.

**ALARM - Emergency organization**
- "Level 1 - File"
- Accident Management Procedures
- Teknical Handbok for AL
- Unit Management procedures
- EOP

**ACCIDENT MANAGEMENT CENTER**
- OL = MD
- Appointed deputies
- Center manned mainly from "not concerned units"
- See: "EMERGENCY PREPAREDNESS ORGANIZATION"

**COMPANY MANAGEMENT**
- MD
- Appointed deputies

**OPERATIONS AND ACCIDENT MANAGEMENT**
- cFX
- Unit Manager (level 2)
- Appointed deputies
- Operations Manager (level 3)
- Appointed deputies
1. Introduction

The Swiss Federal Nuclear Safety Inspectorate (HSK), in collaboration with Energy Research, Inc. (ERI), is in the process of conducting independent Probabilistic Safety Analyses (PSAs) of all Swiss nuclear power plants, in support of the on-going periodic safety reviews, and other regulatory requirements. These studies are full scope level 1/level 2 PSAs, including external events and uncertainties. The uncertainty analyses address both the so-called lack of knowledge uncertainties as well as stochastic uncertainties. These studies are being maintained as living PSA for use in regulatory decisionmaking.

An example level 2 study for a Pressurized Water Reactor (PWR), with large, dry containment is used for the present accident management study. In the analysis, the evaluation of accident progression does not credit automatic systemic recoveries (due to recovery of AC power, for instance), for lack of plant-specific data, and severe accident management strategies (SAMs), which are not yet implemented. The risk implications of SAMs are discussed in the present article. For all SAM strategies, with the exception of accident management involving SGTRs, all of the level 2 analyses are repeated.

2. Details of the Level 1 and 2 PSA

The regulatory level 1 analysis for the Swiss PWR under consideration uses a large fault tree/small event tree methodology. All frontline and support systems fault trees are quantified in detail, and basic events and initiator frequencies are for the most part taken from the utility PSA, after careful review. Where data is lacking, or inconsistent (especially regarding human factors and common causes), the data is recalculated from plant-specific and/or generic information. Parametric uncertainties are evaluated and propagated with standard Monte Carlo methods, using KAERI/MODULE [1] and STUK/SPSA [2].

The results of the level 1 PSA for the example plant show that approximately half of the total core damage frequency results from accidents initiated by loss of offsite power (LOSP), or by external events (seismic, fires, and internal floods), whose accident progression is similar to that of LOSPs. A large part of these accidents is accompanied by failure of containment isolation (this system is also included in the level 1 analysis). Failure of containment isolation induces a large leak to the environment prior to core damage, and in most cases isolation has been found to be unrecoverable prior to core melt. In addition, a small fraction of core damage frequency is comprised of steam generator tube ruptures, with secondary valves stuck open (SGTR). Thus, already in level 1 the containment is found to be ineffective for more than 10% of the total core damage frequency.

The level 2 analysis is truly separated from level 1, thus avoiding potential conflicts in crediting various
recovery actions. All of the core damage prevention measures are credited as part of level 1; while all of the post core damage recovery actions are credited as part of the level 2 study. Uncertainty issues for inclusion in the accident progression event trees are identified through sensitivity and screening analyses, while extensive plant-specific deterministic studies are used to quantify uncertainty distributions. These analyses include the use of the MELCOR code for containment performance and accident progression, CONTAIN for issues such as Direct Containment Heating (DCH), TEXAS and IFCI for steam explosion, SCDAP/RELAP5 for detailed core damage progression and natural circulation studies, and the ERPRA-BURN code for the evaluation of loads following hydrogen (and CO) combustion, and source term uncertainties.

Event trees have been developed and quantified for nine Plant Damage States (PDSs, or groups of accident sequences which are expected to have similar accident progression). The two PDSs (V and SGTR) which result in containment bypass are not analyzed with an event tree. The code EVNTR [3] is used to calculate the conditional probability of the end states (release classes) of the event trees, while uncertainties in phenomenological issues are propagated in the event trees (and in the evaluation of accident frequencies and source terms) with a stratified Monte Carlo code (LHS77, see Reference [4]). Thus an integrated process has been developed to consistently propagate uncertainties. Figure 1 shows the flow of data in this methodology.

In general, a large number of radiological source terms need to be evaluated in this process. In the present analysis, over $3 \times 10^5$ release bins were generated from the accident progression analysis. A fast, plant-specific code (ERPRA) has been developed to calculate source terms, based on a semi-parametric approach, with explicit models for natural deposition of aerosols, and for scrubbing due to containment sprays. Thus, the time dependency of aerosol transport and release is preserved, with appropriate thermo-hydraulic boundary conditions. The quantification of uncertain parameters in these processes is performed on the basis of the MELCOR plant-specific calculations, Surry and Zion STCP analyses, and the results of the QUASAR project [5].

The level 2 analysis shows that the PWR under consideration is sufficiently strong to withstand most pressure loads resulting from impulsive phenomena, such as steam explosions, DCH, and hydrogen combustion, which may occur until (or around) the time of vessel breach. Therefore, early containment failures are not a major concern. In the late stage of severe accidents, hydrogen (and CO) buildup in the containment may be sufficient to reach concentrations which, if a combustion were to occur, could lead to late containment failure. This is especially true if containment sprays are recovered, thus deinerting the atmosphere. Source terms from these accidents are, however, very small, thus the plant risk profile is not overly affected by this failure mode. If sprays are not operable (such as in the LOSP damage states, which contribute more than 50% of the total core damage frequency), the filtered venting system is shown to be very effective in controlling and mitigating releases to the environment.

Source terms are, overall, small or very small. Large releases are predicted for steam generator tube rupture accidents, event V, unmitigated (that is, if containment sprays are inoperable) early containment failures, and unmitigated releases from accidents with failure of containment isolation system. The total frequency of releases exceeding the HSK-defined reference source term for this plant is greater than 10% of the total core damage frequency (CDF).
Figure 1
Flow of Data in the Level 2 Regulatory Analysis

Probabilistic Safety Assessment Methodology

S.100 events

Initial

Event tree (probabilities)

Final

S.20 events

S.15 - 20 sequences

millions of sequences

Screening for low frequency PES's

Binning process

consideration of uncertainties in event tree (probability data and phenomena) and radiological release

Binning by similarity in accident history

very large sequences

10 - 20 sequences

5 - 20 sequences

millions of sequences

very large sets of data for parameterization, sensitivity analysis, and uncertainty analysis

Selected accident sequences

S.100 damage categories

Selected accident sequences

Probabilistic operational sequences

Probabilistic operational sequences

Conceptual format of damage categories

Conditional damage categories

Damage categories

Probabilistic safety assessment

Damage categories

Plant damage estimates

Accident progression

Plant damage estimates

Accident progression

Consequence data

Probabilistic safety assessment

Consequence data

Simulations

Risk integration

Risk integration
Several systems for accident management strategies have already been implemented at the plant, and more are under consideration, as a result of the regulatory analysis. In addition, systemic recoveries during accident progression, even though included in the event trees, have not been credited for lack of plant specific data. Therefore, a number of sensitivity studies have been conducted, using the level 2 model and framework described in this section, to show the (positive or negative) impact of accident management strategies on the risk profile of the plant. These studies will be described in the next section. For each sensitivity, with the exception of accident management involving SGTRs, a full level 2 reanalysis was performed (from accident progression to source terms).

### 3. Severe Accident Management Strategies

SAM strategies can be subdivided into two groups, namely: (1) in-vessel strategies (i.e., those strategies that can be relied on to arrest core damage and prevent vessel breach) and (2) ex-vessel (i.e., those strategies that can be relied on to arrest and/or mitigate damage progression and prevent containment failure and/or fission product releases to the environment).

The in-vessel SAM strategies selected for the present study include:

**In_V1 Addition of Water to a Degrading Core**

In damage states involving Loss of Off-site Power (LOSP), if AC power is recovered, ECCS injection could be restored fairly quickly. In general, provided ECCS water injection begins fairly quickly (within an hour or so, and prior to formation of a large in-core melt crucible) after the beginning of core degradation, there is a fairly high likelihood that core damage progression could be halted. However, if water is injected after gross degradation has already occurred (typically when large blockages are formed and crucible growth is well underway), the likelihood of arresting the melting process is very much reduced. In addition, water addition would enhance steam generation and hydrogen production thereby posing additional challenges to containment integrity. These trends are consistent with the SFD experiments, TMI-2 core examination data, and the OECD-LOFT experiments. Therefore, the present sensitivity study considers two time frames, ECCS recovery prior to, and after substantial core degradation, respectively. In the first case, a conditional probability of 50% is assigned to core damage arrest, while in the second case this probability is reduced to 10%. In addition, containment loads from hydrogen combustion are recalculated increasing zirconium oxidation by 10%, if water can be added to a degrading core.

More than four hours are available from the beginning of the accident, until complete melt and relocation of major portions of the core to the lower plenum (and shortly afterwards, vessel breach is shown to occur). A 50% conditional probability of recovery within the first phase of core degradation was estimated. On the other hand, following substantial core damage progression and melt relocation to the lower plenum, relatively less time is available for recovery actions; therefore, a conditional probability for power and ECCS recovery of about 10% is assigned in this time frame.

**In_V2 Manual Reactor Coolant System (RCS) Depressurization**

Depressurization of the RCS is achieved by opening of the pressurizer valves. For the example plant, the current Emergency Operating Procedures (EOPs) do not instruct the operators to depressurize the RCS in case of a LOSP accident. Since more than four hours are available for this action, a conditional probability of 0.99 is assigned to successful depressurization for all LOSP damage states, with the exception of a small fraction of seismically-induced accidents, where the operators are postulated to be out of action.
In_V 3 Isolation of Steam Generator Tube Rupture Accidents

For this study, it is assumed that 50% of the damage states involving SGTRs can be isolated, thus the corresponding frequency for this PDS is reapportioned to an SLOCA PDS (SEFI) which includes SGTRs with isolated steam generator (note that in this case there is some chance of arresting core damage, since injection or recirculation may still be available; this recovery, however, was not included in the present sensitivity study).

In_V 4 Addition of Firewater to the Secondary Side of the Damaged Steam Generator

The secondary side of the damaged steam generator can be filled with water, and level can be maintained up to the stuck-open valve. Accident frequency is unchanged, but releases to the environment are assumed to be reduced by a decontamination factor (DF) in the range of 10 to 100, which corresponds to that of deep water pools.

In_V 5 Recovery of含ainment Isolation Prior to Core Damage

It should be remembered that, isolation failure, in the vast majority of accidents for the example plant, is not a cause for core damage, but only an attribute for accident progression. For this study, it is assumed that containment isolation is always achieved prior to core damage, thus the corresponding PDSs are reanalyzed as normal large LOCAs and transients (mostly LOSPs). Small LOCAs are a minor contributor to these PDSs, and are grouped together with the transients.

The ex-vessel SAM strategies selected for the present study include:

Ex_V 1 Addition of Fire Water to the Pedestal/Cavity

In the present plant configuration, the fire water system can be aligned to flood the pedestal/cavity. In all PDSs where the ECCS has not been operating, thus leaving the cavity dry, it is assumed that the operators have a 50% conditional probability of aligning the systems and thus flooding the pedestal with water after relocation of core debris to the cavity. Accident progression is not expected to change, but releases, especially from core-concrete interaction, can be largely mitigated.

Ex_V 2 Manual Containment Venting

The containment venting system can operate passively (i.e., via actuation of the rupture disk at a pressure setpoint of 5 bars) or actively (i.e., via manual operator action which involves opening vent valves at pressures below 5 bars). It is currently proposed that operators open the valves to the venting tank when containment pressure reaches about 4 bars. The base case study only considered the automatic actuation of rupture disk at 5 bars. For this sensitivity study, activation of the venting system in the very late phase of accidents is assumed to occur at about 4 bars (via operator action), with a success probability of 0.99. Conditional probabilities of vent activation are then re-evaluated, on the basis of late containment loads.

Ex_V 3 Recovery of Containment Sprays and Fan Coolers

For the in-vessel SAM strategy "In_V 1" (recovery of ECCS), recovery of sprays and fan coolers has not been considered, in order to separate the effects of containment systems recoveries. For this sensitivity case, Engineered Safety Features (ESFs) are allowed to restart with a probability of 0.9, if AC power is restored. Recovery of AC power in the very late phase of accidents is assumed to have a conditional probability of about 0.5.
Ex_V Alignment of the Fire Water and Spray Systems

Given the ample time which is available, a conditional probability of 0.9 is assigned to the success for manual actuation of containment sprays, both in the early and the very late phases of accidents. Note that in this sensitivity analysis, since an additional system is brought on-line, sprays are assumed to begin operating also in cases where the ESFs had mechanically failed.

A final sensitivity study has also been performed, which combines several SAM strategies together. Note that this is possible without conflict, since the system most often used (fire water) can be aligned to different systems in accidents which are mutually exclusive (e.g., SGTRs and LOSPs).

4. Results

Several surrogate risk indices can be evaluated to determine the potential impact of the selected SAM strategies on the plant risk profile.

A first set of risk indices can be based on the results of accident progression. Release bins and their frequencies can be grouped into larger classes, based on the expected severity of resulting offsite consequences. Thus, relative changes in the frequency of early containment failures (ECF), containment bypass (BYP), late containment failures (LCF), filter-vented sequences (VNT), and accidents with intact containment (NCF), respectively, can be used as surrogates to measure the effectiveness and merits of the selected SAM strategies. Vented accidents include failures of the venting system as a result of hydrogen combustion within the filtration system. Table 1 summarizes the results of the sensitivity studies for these risk indexes. Note that, this simplification of the containment matrix is necessary, in order to keep the analyses manageable. In so doing, however, information on release characteristics which are relevant to mitigation of radiological source terms (spray actuation time, water in the containment) disappear.

Table 1 Simplified C-Matrix for the SAM Strategies Sensitivity Studies

<table>
<thead>
<tr>
<th></th>
<th>ECF</th>
<th>BYP</th>
<th>LCF</th>
<th>VNT</th>
<th>NCF</th>
</tr>
</thead>
<tbody>
<tr>
<td>Base</td>
<td>1.2x10^2</td>
<td>1.09x10^-1</td>
<td>3.7x10^2</td>
<td>6.88x10^-1</td>
<td>1.54x10^-1</td>
</tr>
<tr>
<td>In_V 1</td>
<td>1.0x10^2</td>
<td>1.09x10^-1</td>
<td>3.9x10^2</td>
<td>4.41x10^-1</td>
<td>4.01x10^-1</td>
</tr>
<tr>
<td>In_V 2</td>
<td>8.0x10^-3</td>
<td>1.07x10^-1</td>
<td>3.7x10^2</td>
<td>6.88x10^-1</td>
<td>1.54x10^-1</td>
</tr>
<tr>
<td>In_V 3</td>
<td>1.2x10^2</td>
<td>9.5x10^-2</td>
<td>3.7x10^2</td>
<td>6.88x10^-1</td>
<td>1.68x10^-1</td>
</tr>
<tr>
<td>In_V 4</td>
<td>1.2x10^2</td>
<td>1.09x10^-1</td>
<td>3.7x10^2</td>
<td>6.88x10^-1</td>
<td>1.54x10^-1</td>
</tr>
<tr>
<td>In_V 5</td>
<td>1.2x10^2</td>
<td>3.8x10^-2</td>
<td>3.7x10^2</td>
<td>7.55x10^-1</td>
<td>1.58x10^-1</td>
</tr>
<tr>
<td>Ex_V 1</td>
<td>1.2x10^2</td>
<td>1.09x10^-1</td>
<td>3.6x10^-2</td>
<td>6.88x10^-1</td>
<td>1.55x10^-1</td>
</tr>
<tr>
<td>Ex_V 2</td>
<td>1.2x10^2</td>
<td>1.09x10^-1</td>
<td>3.0x10^2</td>
<td>7.11x10^-1</td>
<td>1.38x10^-1</td>
</tr>
<tr>
<td>Ex_V 3</td>
<td>1.0x10^2</td>
<td>9.6x10^-2</td>
<td>4.1x10^-2</td>
<td>3.58x10^-1</td>
<td>4.95x10^-1</td>
</tr>
<tr>
<td>Ex_V 4</td>
<td>1.4x10^2</td>
<td>1.09x10^-1</td>
<td>9.4x10^-2</td>
<td>4.33x10^-1</td>
<td>3.50x10^-1</td>
</tr>
<tr>
<td>Comb.</td>
<td>1.5x10^2</td>
<td>9.7x10^-2</td>
<td>8.2x10^-2</td>
<td>4.94x10^-1</td>
<td>3.12x10^-1</td>
</tr>
</tbody>
</table>
For instance, Ex_V 4 (manual actuation of containment sprays via fire water alignment) produces a 17% increase in early containment failures (due to a slightly higher probability of deinerting the containment atmosphere around the time of vessel breach) when compared to the base case, while late containment failures almost triple (for the same reason). Accidents with intact containment, on the other hand, more than double in frequency, while vented sequences decrease in frequency. Therefore, it is not clear whether this measure, alone, can be detrimental (at least from the point of view of containment integrity, it should be regarded with caution).

In general, it becomes apparent that all SAM strategies would reduce the frequency of vented sequences and increase the frequency of accidents involving intact containment (no failure), or late failure. In all cases, however, the shift normally would be from small to even smaller releases, but a possible decrease in risk cannot be precisely quantified. In addition, the frequency of accidents which may lead to relatively large releases (early containment failures, and containment bypass events) is not appreciably affected by any of the SAM strategies which have been included in the present study.

A less ambiguous set of risk indices is based on the relative variation in radiological releases. For the present study, the changes for the cesium group (which has the largest significance for chronic health effects and long term land contamination) at a frequency of $10^5$ per reactor year has been chosen as an appropriate surrogate to measure the impact of the SAM strategies. Results for the iodine group (of largest significance for acute health effects) have been found to be similar to those of cesium, but, for the example plant, iodine releases are of less interest, since, given the core inventory, population distribution, and effective emergency protective measures (sheltering), early health effects are expected to be largely suppressed.

Figure 2 shows the results for the risk index based on the variation in release of the cesium group with respect to base case results, for all the SAM strategies considered in the present study. In addition, the result is shown for the sensitivity analysis which combines several mutually exclusive actions.

In_V 5 and Ex_V 4 strategies show the largest variation. The first strategy (which essentially leads to prevention of large, unmitigated releases from an unisolated containment) is under consideration. Moreover, this sensitivity study assumes that all isolation failures can be recovered, which may not be possible. The second strategy entails the realignment of the spray with the fire water system. This can already be achieved at the plant. In addition, In_V 1, In_V 4, and Ex_V 3 strategies appear to provide some mitigation for cesium releases, while all other actions, when considered one-at-a-time, produce only marginal effects.

As a whole, the strategies which have been classified as 'In-vessel' appear slightly more effective than the 'Ex-vessel' strategies. For the latter, in some cases (i.e., Ex_V 1, addition of water to the cavity after vessel breach), no mitigative benefits is apparent as far as the cesium release is concerned. This is because most of the cesium is released during the in-vessel phase (as well as due to revaporization from RCS surfaces following vessel breach), and thereby little benefit is expected from an ex-vessel strategy, other than perhaps the spray and/or the filter-vent system. These components cannot be mitigated by the measure under consideration. However, if on the other hand the risk index was to be redefined in terms of a refractory fission product group (for instance, the barium-strontium group) then a large reduction would be observed for this strategy, since most of the strontium releases are due to core-concrete interactions, and these would occur under a deep water pool in the Ex-V SAM 1 scenario, thus would be subjected to a considerable scrubbing effect (a reduction of approximately a factor of three, see Figure 3). Releases of refractory elements are, however, of much less significance than those of cesium, both because of magnitude of release, and their expected consequences.
5. **Conclusions**

In general, all SAM strategies investigated here were shown to be beneficial in reducing radiological releases to the environment. The most effective strategy appears to be connected with the use of the fire water system, either to refill the secondary side of a damaged steam generator, or to drive the containment sprays.

The frequency of classes with expected large releases (early containment failures and bypasses) is not appreciably affected by any of the strategies. However, the consequences of these classes are appreciably reduced by adopting one or all of the selected measures.

In terms of frequency of releases, the effect of the SAM strategies is a shift from filtered-vented accidents, to accidents with intact containment. In some cases, the increased availability of the ESFs induces an increase in the frequency of late containment failures. However, the consequences of these accidents, which are already small in the base case analysis, are largely reduced by other mitigative effects (e.g., scrubbing by sprays). Therefore, large detrimental effects are not apparent by the adoption of any of the SAM strategies which have been considered.

Finally, the implementation of the combined effects of certain SAM strategies, is shown generally to reduce radiological source terms by approximately a factor of five.

**References**

Figure 2  Relative Changes in Cesium Release due to the Severe Accident Management Strategies, with respect to Base Case Results

Figure 3  Frequency of Exceedance of Barium-Strontium Release: Comparison of Mean and 95-th Percentile with Base Case Results for Ex_V 1 SAM Strategy
Development of Accident Management Strategies for BWR plants in Japan

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1. Introduction

Although Japanese traditional safety philosophy has set priority on preventing abnormal occurrences and accidents by assuring quality in design, fabrication, construction, maintenance and operation, it is also important to prepare the mitigative strategies that supports containment function in severe accident conditions. After implementing the symptom-oriented emergency operating procedure to prevent core damage, Japanese BWR utilities started a joint study on the further development of accident management strategies including prevention and mitigation of severe accident, and guidelines for their implementation. Issuance of a sort of generic letter by the Japanese nuclear regulatory bodies in 1993 encouraging Utilities to develop accident management strategies and to implement them boosted this activity of Utilities self-regulation. All the Japanese Utilities reported in the end of March 1994 to the regulatory body IPE and accident management strategies for respective plant, which will be described in detail in this report.

Japanese BWR utilities considers the important elements of accident management program as follows and plant modification is not necessarily a major concern;
(a) individual plant PSA (level 1.5PSA),
(b) strategy development with solid technical basis,
(c) studies on the availability of instrumentation & information,
(d) guideline development & training,
(e) organization and delineation of responsibilities.

Basic philosophy in developing strategies is to fully utilize the existing plant resources, because PSA has shown that plant capability in prevention & mitigation of severe accident can be enhanced significantly by use of existing plant resources with minor plant modification.

2. Strategy development for BWRs

Strategies described in this paper are standardized strategies for four reference plants representing BWR plants in Japan;
(a) BWR-3 with Mark-I containment,
(b) BWR-4 with Mark-I containment,
(c) BWR-5 with Mark-II containment and
(d) ABWR.

PSA (level-1.5) has been instrumental in identifying plant vulnerabilities & coping strategies and developing technical basis for the strategies. IPE has been useful in determining plant specific strategies.

2.1 Strategies for Mark-I containment

Table-1-a-2-b are the summary of PSA results and selected strategy for BWR-3,4 plant with Mark I containment. PSA for BWR-3 plant has shown that LOCA followed by ECCS failure is dominant in core damage frequency, alternative water injection to reactor vessel has bee selected using following pumps;
(a) Containment cooling spray pump installed separately from ECCS,  
(b) Diesel-driven fire protection pump, and  
(c) Make-up-water pump.

Since transient followed by loss of high pressure injection and failure of manual depressurization (TQUX) is dominant for BWR-4, it was determined to make the depressurization automatic.

Regarding containment failure probability (CFP) (Table-1-b), leakage due to over temperature and containment shell attack are the dominant mode leading to failure of Mark-I containment. Leakage will develop at flange gasket and electrical penetration which are considered relatively sensitive to overheating.

Since Mark-I containment has an opening in pedestal, molten corium dropped onto pedestal floor may spread out to dry-well floor and then transfer thermal energy directory to the containment boundary. To cope with this thermal attack and to quench molten corium on drywell concrete floor, water injection to the floor using all possible means (including the above a)b)c) was selected as coping strategy to this threat. Although details of the procedure is yet to be determined and capability of the instrumentation to detect vessel melt-through, operator is expected to pour water onto the pedestal floor switching the destination of water from reactor vessel at an appropriate stage before or after the melt-through.

The objectives to inject water into containment are;
(a) to prevent containment shell attack (Mark-I containment),  
(b) to minimize non-condensable (core concrete reaction) gas and fission product aerosol generation,  
(c) to mitigate containment temperature increase and  
(d) to prevent basement melt-through.

Containment shell attack is a unique sequence to Mark-I containment. Scaled simulation experiments using molten stainless steel was carried out for in dry and wet floor respectively by Japanese BWR utilities and venders to confirm the effectiveness of water injection to mitigate debris spreading

Some other containment threats such as over pressure in ATWS sequence can be eliminated by reducing the likelihood of the occurrence by preventative strategies as shown in Table-1-a ~ 2-b.

### Table-1-a Major contributors to CDF and Strategies for BWR-3 with Mark-I containment

<table>
<thead>
<tr>
<th>Core damage sequence</th>
<th>Contribution to total CDF</th>
<th>Selected preventive strategy</th>
</tr>
</thead>
<tbody>
<tr>
<td>LOCA followed by failure of ECCS (AES1E52E)</td>
<td>66 %</td>
<td>(1) Alternative water injection using existing pumps</td>
</tr>
<tr>
<td>Transient followed by failure of ECCS (TQUV)</td>
<td>23 %</td>
<td>(1) Alternative water injection using existing pumps</td>
</tr>
<tr>
<td>ATWS (TC)</td>
<td>7 %</td>
<td>(2) Recirculation pump trip and alternative rod insertion</td>
</tr>
<tr>
<td>Station black out (TB)</td>
<td>3 %</td>
<td>(3) Emergency power utilization from adjacent unit (4) Recovery</td>
</tr>
</tbody>
</table>

### Table-1-b Major contributors to CFP and Strategies for BWR-3 with Mark-I containment

<table>
<thead>
<tr>
<th>Containment failure sequence</th>
<th>Contribution to total CFP</th>
<th>Selected mitigative strategy</th>
</tr>
</thead>
<tbody>
<tr>
<td>Leakage due to over temperature</td>
<td>51 %</td>
<td>(1) Dry-well spray using normal pumps</td>
</tr>
<tr>
<td>Containment shell attack</td>
<td>24 %</td>
<td>(2) Dry-well spray using normal pumps</td>
</tr>
<tr>
<td>Over pressure after ATWS</td>
<td>15 %</td>
<td>( same as preventive strategy (2) in Table-1-a)</td>
</tr>
<tr>
<td>DCH due to high pressure melt ejection after SBO</td>
<td>4 %</td>
<td>( same as preventive strategy (3)-(4) in Table-1-a)</td>
</tr>
</tbody>
</table>
### Table 2-a Major contributors to CDF and Strategies for BWR-4 with Mark-I containment

<table>
<thead>
<tr>
<th>Core damage sequence</th>
<th>Contribution to total CDF</th>
<th>Selected preventive strategy</th>
</tr>
</thead>
<tbody>
<tr>
<td>Transient followed by failure of high press. ECCS and manual depressurization (TQUX)</td>
<td>48%</td>
<td>(1) Automatic depressurization</td>
</tr>
<tr>
<td>Station black out (TB)</td>
<td>17%</td>
<td>(2) Emergency power utilization from adjacent unit (3) Recovery</td>
</tr>
<tr>
<td>LOCA followed by failure of ECCS (AE,S1E,S2E)</td>
<td>17%</td>
<td>(4) Alternative water injection using normal pumps</td>
</tr>
<tr>
<td>Transient followed by failure of ECCS (TQUV)</td>
<td>13%</td>
<td>(5) Alternative water injection using normal pumps</td>
</tr>
<tr>
<td>ATWS (TC)</td>
<td>3%</td>
<td>(6) Recirculation pump trip and alternative rod insertion</td>
</tr>
<tr>
<td>Transient followed by failure of decay heat removal system (TW)</td>
<td>2%</td>
<td>(7) Alternative heat removal using existing equipment (8) Recovery (9) Hardened wet-well venting</td>
</tr>
</tbody>
</table>

### Table 2-b Major contributors to CFP and Strategies for BWR-4 with Mark-I containment

<table>
<thead>
<tr>
<th>Containment failure sequence</th>
<th>Contribution to total CFP</th>
<th>Selected mitigative strategy</th>
</tr>
</thead>
<tbody>
<tr>
<td>Leakage due to over temperature</td>
<td>50%</td>
<td>(1) Dry-well spray using normal pumps</td>
</tr>
<tr>
<td>DCH due to high pressure melt ejection after SBO</td>
<td>18%</td>
<td>(same as preventive strategy (2)-(3) in Table 2-a)</td>
</tr>
<tr>
<td>Containment shell attack</td>
<td>15%</td>
<td>(2) Dry-well spray using normal pumps</td>
</tr>
<tr>
<td>Over pressure due to loss of decay heat removal</td>
<td>8%</td>
<td>(same as preventive strategy (7)-(9) in Table 2-a)</td>
</tr>
<tr>
<td>Over pressure after ATWS</td>
<td>8%</td>
<td>(same as preventive strategy (6) in Table 2-a)</td>
</tr>
</tbody>
</table>

### Figure 1 Typical Containment Types in Japan

![Mark-I containment](image1)

![Mark-II containment](image2)

![ABWR containment](image3)

**Mark-I containment**

**Mark-II containment**

**ABWR containment**

### Figure 2 Schematic View of Alternative injection and Hardened wet-well venting

![Schematic View](image4)

- **HVAC**: added part
- **SGTS**: added part
- **RHR sys.**: added part
- **Fire Prptection sys.**: added part
- **Make-Up-Water sys.**: added part
2.2 Strategies for Mark-II containment

Major contributors to CDF and selected strategies for them are listed in Table-3-a. TQUX is a dominant sequence for this type of plant, thus automatic depressurization was selected as a coping strategy for this sequence. For loss of decay heat removal (DHR) sequence which is a secondary contributor, three strategies were selected as shown in Table-3-a. Strategy-(2) in Table-3-a is to utilize existing containment fan cooler and/or other non-essential heat exchangers. Strategy-(4) is so called 'hardened containment venting' intended to control containment pressure in case containment over pressure failure is imminent.

As for the containment strategy, major contributors to CFP and strategies are listed in Table-3-b. The dominant mode leading to containment failure are identified as over-pressure due to loss of DHR, steam explosion at wet-well and over pressure in ATWS for this type of containment. Steam explosion in wet-well water volume in case corium slumped through pedestal floor might be a unique threat to Mark-II type containment and its reality is highly uncertain. Water injection into pedestal area to quench debris before its penetration into wet-well water volume is expected to make this explosion highly unlikely. Alternative water injection using the above b)c) in 2.1 means will serve this purpose as well thus piping will be routed to inject water directly to pedestal area. Besides pedestal injection purpose, these pumps will serve for reactor injection and dry-well spray as shown in Figure-2.

For other containment failure sequence, such as over-pressure due to loss of DHR, over-pressure in ATWS and direct containment heating (DCH), it is showed that their probability can effectively be reduced when preventive accident management (Table-3-a) comes into effect. For hardened wet-well venting, although it is mainly considered as preventive strategy, scrubbing effect using suppression chamber water volume was confirmed by tests and computer code to predict DF based on empirical correlation was established(2).

### Table-3-a Major contributors to CDF and Strategies for BWR-5 with Mark II containment

<table>
<thead>
<tr>
<th>Core damage sequence</th>
<th>Contribution to total CDF</th>
<th>Selected preventive strategy</th>
</tr>
</thead>
<tbody>
<tr>
<td>Transient followed by failure of high press. ECCS and manual depressurization (TQUX)</td>
<td>41 %</td>
<td>(1) Automatic depressurization</td>
</tr>
<tr>
<td>Transient followed by failure of decay heat removal system (TW)</td>
<td>31 %</td>
<td>(2) Alternative heat removal using existing equipment (3) Recovery (4) Hardened wet-well venting</td>
</tr>
<tr>
<td>Station black out (TB)</td>
<td>12 %</td>
<td>(5) Emergency power utilization from adjacent unit (6) Recovery</td>
</tr>
<tr>
<td>Transient followed by failure of ECCS (TQUV)</td>
<td>6 %</td>
<td>(7) Alternative water injection using normal pumps</td>
</tr>
<tr>
<td>ATWS (TC)</td>
<td>5 %</td>
<td>(8) Recirculation pump trip and alternative rod insertion</td>
</tr>
<tr>
<td>LOCA followed by failure of ECCS (AE,S1E,S2E)</td>
<td>5 %</td>
<td>(9) Alternative water injection using normal pumps</td>
</tr>
</tbody>
</table>

### Table-3-b Major contributors to CFP and Strategies for BWR-5 with Mark II containment

<table>
<thead>
<tr>
<th>Containment failure sequence</th>
<th>Contribution to total CFP</th>
<th>Selected mitigative strategy</th>
</tr>
</thead>
<tbody>
<tr>
<td>Over pressure due to loss of decay heat removal</td>
<td>76 %</td>
<td>(same as preventive strategy (2)-(4) in Table-3-a)</td>
</tr>
<tr>
<td>Steam explosion at wet-well</td>
<td>12 %</td>
<td>(1) Water injection to pedestal floor using normal pumps</td>
</tr>
<tr>
<td>Over pressure after ATWS</td>
<td>10 %</td>
<td>(same as preventive strategy (8) in Table-3-a)</td>
</tr>
<tr>
<td>DCH due to high pressure melt ejection after SBO</td>
<td>1 %</td>
<td>(same as preventive strategy (5)-(6) in Table-3-a)</td>
</tr>
<tr>
<td>Leakage due to over temperature</td>
<td>1 %</td>
<td>(2) Water injection into pedestal floor (3) Dry-well spray using normal pumps</td>
</tr>
</tbody>
</table>

80
2.4 Strategy for ABWR containment

Since ABWR was designed with PSA as one of the guiding principle in the stage of design, core damage frequency is naturally low almost to the point no further preventative strategies are necessary. Its RCCV Containment design reflects latest technology for Japanese BWRs. PSA indicates that major threats to the containment integrity are DCH due to High Pressure Melt Ejection in SBO and leakage at flange gasket or electrical penetration caused by gradual temperature rise in the containment. Trying to reestablish AC or DC power by emergency power supply from adjacent plant power system eventually terminate this sequence and will serve to depressurize reactor pressure to avoid this DCH threat. ABWR other containment strategies are the same as those for BWR-5 with Mark-II containment as shown in Table-4-a, 4-b. Fusible valve to introduce water from suppression pool to the lower pedestal region was not preferred because routing of the piping from alternative water sources will provide flexibility to pour water before or after the reactor vessel breach.

Table-4-a Major contributors to CDF and Strategies for ABWR

<table>
<thead>
<tr>
<th>Core damage sequence</th>
<th>Contribution to total CDF</th>
<th>Selected preventive strategy</th>
</tr>
</thead>
<tbody>
<tr>
<td>Station black out (TB)</td>
<td>54 %</td>
<td>(1) Emergency power utilization from adjacent unit (2) Recovery</td>
</tr>
<tr>
<td>Transient followed by failure of high press. ECCS and manual depressurization (TQUX)</td>
<td>32 %</td>
<td>-</td>
</tr>
<tr>
<td>Transient followed by failure of ECCS (TQUV)</td>
<td>8 %</td>
<td>(3) Alternative water injection using normal pumps</td>
</tr>
<tr>
<td>Transient followed by failure of decay heat removal system (TW)</td>
<td>5 %</td>
<td>(4) Alternative heat removal using existing equipment (5) Recovery (6) Hardened wet-well venting</td>
</tr>
</tbody>
</table>

Table-4-b Major contributors to CFP and Strategies for ABWR

<table>
<thead>
<tr>
<th>Containment failure sequence</th>
<th>Contribution to total CFP</th>
<th>Selected mitigative strategy</th>
</tr>
</thead>
<tbody>
<tr>
<td>DCH due to high pressure melt ejection after SBO</td>
<td>46 %</td>
<td>(same as preventive strategy (1)–(2) in Table-4-a)</td>
</tr>
<tr>
<td>Leakage due to over temperature</td>
<td>42 %</td>
<td>(1) Water injection into lower dry-well</td>
</tr>
<tr>
<td>Over pressure due to loss of decay heat removal</td>
<td>12 %</td>
<td>(same as preventive strategy (4)–(6) in Table-4-a)</td>
</tr>
</tbody>
</table>

3. Guideline development

To implement containment severe accident management strategy effectively, two kinds of guideline are being considered. One is a guideline to assess plant condition ("assessment guideline"); another is a guideline to prioritize action to be taken ("action guideline"). Assessment guideline is to provide operators or technical support staff with methods to obtain information on critical containment parameters from instrumentation system and to analytically estimate those parameter values not directly measurable but important to assess the plant damage status. Action guideline is to provide operators or technical support staff with candidate strategies that may be at particular accident status together with action priority, information on the availability of relevant equipments and water sources and information on predicted the plant response if certain action is not taken.
4. Reference

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(2) I. Kaneko (Toshiba), K. Miyata (TEPCo), et.al. "Experimental Study on Aerosol Removal Effect by Pool Scrubbing" 22nd DOE/NRC Nuclear Air Cleaning and Treatment Conference (1992)
Containment Severe Accident Management Strategies for PWR Plants in Japan

The Kansai Electric Power Co., Inc.  
Takao Nakamura

Presented at the OECD Specialist Meeting on Selected Containment Severe Accident Management Strategies  
13th-15th June 1994 Stockholm, Sweden
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Abstract: Five Japanese PWR utilities have conducted the joint studies on containment severe accident management for PWR plants in Japan. The utilities have discussed the severe accident management strategies for 4 typical PWR plants from the viewpoint of in-house safety activities. Knowledge obtained from the results of PSA and severe accident studies has been systematically investigated, and the event sequences leading to core damage or containment vessel (CV) failure were classified into several categories. Then, the accident management strategies against the dominant sequences were specified. The dominant physical events leading to CV failure were classified into 9 categories. Four candidates for accident management measures against those events were selected, and their effectiveness, feasibility and influences on existing safety functions have been identified.

- CV cooling by natural convection
- Water injection into CV
- Forced depressurization of primary system
- Controlled burning of hydrogen (applied only for the plants with ice condenser CV)

Introduction
This paper introduces the containment accident management strategies for PWR plants in Japan. The accident management strategies have been positively addressed in Japan, as severe accident studies progress. The Kansai Electric Power Co., as a part of accident management program, has been improving operation management such as preparation of procedures providing with safety knowledge and implementation of education and training. We intend to prevent core damage by means of full utilization of existing facilities as an in-house safety assurance activity, even if an accident beyond the design basis should occur.

In these circumstances, the Nuclear Safety Commission (NSC) addressed the policy in May, 1992 which strongly recommends that the utilities should voluntarily develop severe accident management strategies. After that, in July 1992, Ministry of International Trade and Industry (MITI) required utilities to implement Probabilistic Safety Assessment (PSA) to understand safety characteristics of their nuclear power plants and to further develop accident management strategy as an in-house safety assurance measure. In this March, we submitted MITI the accident management strategy report concerning our nuclear power plants. MITI is now conducting technical investigation of the report.

Understanding of severe accident phenomena and plant characteristics
Five Japanese PWR utilities (Kansai, Hokkaido, Shikoku, Kyushu, JAPCO) have collaborated on developing the accident management strategies. First of all, 4 typical plants; 2-loop type, 3-loop type, 4-loop type with dry CV and 4-loop type with ice condenser CV were selected. Severe accident phenomena were systematically investigated based on the results of PSA, existing safety analysis and severe accident studies. At the same time, the event sequences leading to core damage or CV failure were classified into several categories, and the accident management strategies against dominant sequences were investigated with taking plant safety characteristics from PSA into consideration.
The dominant physical events to CV failure were classified into 9 categories. Portion of each category was also calculated based on the results of level 2 PSA. (Figure 1) As a result, CV overpressure failure by decay heat, CV bypass, and base mat melt through were dominant. In addition, hydrogen deflagration dominated for the plant with ice condenser CV.

The PSA including CV failure probability assessment was conducted on all the 23 PWR plants which are in operation and under construction in Japan to understand safety features of each plant. In addition, it has been confirmed that those features are incorporated into the results of studies concerning typical plants.

Selection of candidates for accident management

When discussing containment accident management measures, safety functions required for preventing CV failure as well as causes leading to loss of these safety functions were specified and then countermeasures against these causes were developed as the accident management strategies. 4 accident management candidates, "CV cooling by natural convection", "water injection into CV", "forced depressurization of primary system", and "controlled burning of hydrogen" were selected. As for each strategy, the effectiveness, feasibility and influences on the existing safety functions have been identified.

The followings are the basic concepts for prevention of CV failure;

- Prevent base mat melt through after reactor vessel failure by quenching molten core in the cavity with "water injection into CV".
- Convert core decay heat into steam by quenching the molten core to generating saturated CV atmosphere, and prevent CV overpressure failure by decay heat conducting "CV cooling by natural convection" or "water injection into CV". Accordingly, it is not necessary to adopt CV venting.
- "Forced depressurization of the primary system" prevents reactor vessel failure under high pressure leading to direct containment heating (DCH).
- "Controlled burning of hydrogen" prevents hydrogen deflagration/detonation at the plants with ice condenser CV.
- Improve the preventive measures against core damage concerning CV bypass events such as the steam generator tube rupture and interface LOCA.

CV cooling by natural convection

- Purpose
In the event of loss of emergency CV cooling system function, pressurization of CV is controlled by steam condensation due to natural convection running water through the normal CV cooling system or the spray system outside CV.
(It is also useful to prevent core damage due to CV failure on loss of recirculating water cooling system function.) (Figure 2)

- Applicable category
CV overpressure by decay heat

- Effectiveness
The heat removal rate of normal CV cooling system under natural convection at high temperature steam atmosphere was calculated on the typical 4-loop type dry CV plant with SAMIT code.\(^{1,2}\) The result showed that the heat removal capacity of the normal CV cooling system was large enough to remove core decay heat. (Figure 3)

To assess the CV cooling performance, the severest case in which the emergency CV cooling system failed during large break LOCA was analyzed with MAAP code. As a result, it was confirmed that the normal CV cooling system under natural convection could suppress CV pressure. (Figure 4)

In addition, the similar analysis was made on the plants with the spray system outside CV and its result showed that operation of the spray system outside CV could also suppress CV pressure. (Figure 5)
Water Injection into CV

· Purpose
To inject water into CV to quench molten core in the cavity through CV spray line with use of the fire water system or other equipment. This strategy has another purpose to depressurize CV by condensing steam.

· Applicable category
CV overpressure by decay heat and base mat melt through

· Effectiveness
This accident management strategy was reviewed with MAAP code on the severest case, in which ECCS or the emergency CV cooling system failed during large break LOCA. The result showed that water injection into CV could quench the molten core and generate saturated CV atmosphere. In spite of uncertainty concerning the ability to cool the molten core, it is expected that converting decay heat into steam can mitigate core-concrete interaction.

Water injection into CV is conducted after detecting the core damage condition by the area radiation monitor. The molten debris abrades the concrete floor with increasing its weight as time passes. A time lag in water injection may cause increase in the amount of steam generated during core quenching operation with leaving significant influence on CV. Therefore, water injection in the early stage is favorable. On the other hand, if the molten core drops into the cavity filled with water, it is possible to cause steam explosion. However, it is expected that the explosion has little influence on the entire CV in spite of a large local pressure wave generated.

To confirm the suppression effect of CV pressure by condensing steam with water injection into CV, an analysis with MAAP code was made on a case in which this accident management strategy was adopted, when ECCS and the emergency CV cooling system failed during large break LOCA. The result showed that the time up to the CV overpressure failure could be prolonged by utilizing the CV spray injection with use of the fire water system or other equipment to store core decay heat in the liquid phase.

Forced Depressurization of Primary System

· Purpose
To prevent direct containment heating (DCH) by depressurizing the primary system, in the event of the core melt while the reactor under high pressure.

· Applicable category
DCH and direct contact of the molten core with CV

· Effectiveness
When the core melts and the reactor vessel fails under high pressure, the molten core may scatter and cause DCH. The primary system pressure under which the molten core will not scatter depends on the configuration of the reactor cavity. Considering the uncertainty about that pressure value, the index of 20 kg/cm² (or 2 MPa) is adopted as our index referring the analytical results of the overseas plants. Forced depressurization of the primary system is to be conducted according with this index value.

An analysis with MAAP code was made on the case in which the primary system depressurization was conducted with use of the pressurizer relief valve, when the ECCS high pressure injection failed during loss of all feed water while the primary system under high pressure. The result showed that reactor vessel failure could be controlled or its timing could be delayed by forced depressurization of the primary system after detecting core damage, and that scattering of the molten core could be controlled in case of reactor vessel failure.

The past experimental results indicate that there is little possibility of a large scale steam explosion inside the
reactor vessel during forced depressurization of the primary system,\textsuperscript{10,11} which may cause CV failure. In addition, it is expected that the protective structures such as the missile shield installed at the upper portion of the reactor vessel can hold back the influence on CV in case of steam explosion.

**Controlled Burning of Hydrogen**

- **Purpose**
  To control CV pressure increase by burning hydrogen generated from zircaloy-water reaction at low concentration.

- **Applicable category**
  Hydrogen deflagration/detonation in the plant with ice condenser CV

- **Effectiveness**
  An analysis was made with MAAP code on the core damage accident progress at the plant with ice condenser CV. The result showed about 40\% of the zircaloy-water reaction on the severest case in which ECCS injection failed during small break LOCA, when considering the maximum generated amount and maximum released rate of hydrogen.

  Furthermore, based on the analytical result, another analysis was made with MAPHY-BURN\textsuperscript{12} code on the hydrogen concentration distribution inside CV, assuming the zircaloy-water reaction at more conservative value of 75\%. The result showed that the flow path from the hydrogen release zone to the upper compartment via the ice condenser was dominant, (Figure 6) and that the hydrogen concentration was transiently high at the hydrogen release zone and its adjacent area, while its distribution inside CV homogenized in a short period.

  To confirm the pressure suppression effect, an analysis was made with MAPHY-BURN code on the CV pressure increase in case of the controlled burning of hydrogen at the inlet and outlet of the ice condenser. The result showed that the peak pressure was lower than the maximum design pressure even if hydrogen was ignited at concentration of 8\% with regarding 100\% of the burning ratio inside the zone at the most conservative value. Therefore, the containment integrity can be maintained. (Figure 7)

  The core damage accident progress analysis was also made with MAAP code on a PWR plant with dry CV. The result showed that the peak pressure does not reach the value of CV failure even if all the amount of the generated hydrogen is burned completely. Therefore, it is expected that there is little possibility of CV failure due to hydrogen burning.

**Conclusion**

We have selected the containment accident management strategies for the PWR plants in Japan based on the knowledge on severe accident and the plant characteristics obtained from PSA, and confirmed their effectiveness, feasibility and influences on existing safety functions.

We intend to further develop the accident management strategies by preparing specific procedures and proceeding modification of facilities. At the same time, we will improve the existing implementation system and education/training method concerning accident management for further safety assurance.
References

1. SARJ-93, "A study of the heat transfer characteristics of Containment Recirculation Coolers during severe accident conditions"
5. JAERI-M 92-035
12. K.Takumi, A.Nonaka and J.Ogata, "Proving Test on the Reliability for Reactor Containment Vessel Hydrogen Mixing and Distribution Test at NUPEC", SARJ-92, Nov.4-6, Tokyo, Japan
Figure 1 C/V Failure Frequency by AM Category

Typical Two Loop Plant
- Failure of C/V Isolation
- Base Mat Melt
- C/V Bypass
- C/V Reaction
- Overpressure by Decay Heat

Typical Three Loop Plant
- Failure of C/V Isolation
- Base Mat Melt
- C/V Bypass
- C/V Reaction
- Overpressure by Decay Heat

Typical Dry C/V Four Loop Plant
- Debris Direct Contact to C/V

Typical Ice Condenser C/V Four Loop Plant
- Debris Direct Contact to C/V

Figure 2 C/V Cooling by Natural Convection

Spray Ring
- External Spray
- Mihama Unit 1 & 2 only

Service Water Storage Tank

Containment

C/V Recirculation Fan Cooler

Additional line

CCW

C/V Spray Pump

RHR Cooler

Low Pressure Injection Pump

High Pressure Injection Pump
Figure 3: Heat Removal Capacity of Alternative C/V Cooling (Typical 4-Loop Plant)

Figure 4: C/V Ambient Pressure using Alternative C/V Cooling (Typical 4-Loop Plant)

Figure 5: C/V Ambient Pressure with C/V External Spray (Plants which have External Spray)
Figure 6 Arrangement of Igniters

upper compartment (~14)

exits of ice condenser (~10)

ice condenser

entrances of ice condenser (~10)
(lower compartment)

Arrangement of Igniters
- Igniter
--- Flow path of mixture gas
----- Flow path of air return fans

Figure 7 MAPHY-BURN Results
(Containment Vessel Pressure)
Abstract

In the TMI-2 accident, approximately 20 tonnes of molten core material drained into the RPV lower plenum at about 3 hours and 47 minutes into the accident. The TMI-2 Vessel Investigation Project determined that a significant part of the reactor vessel lower head (an elliptical region ~ 1 m by 0.8 m) was substantially overheated. Specifically, one part of the steel wall is estimated to have reached a temperature of at least 1100°C for ~ 30 mins. and then experienced a rapid cooldown at a rate in excess of 10°C per min. The cause and nature of this rapid cooling are of substantial importance since the TMI-2 vessel was at a pressure of ~ 11 MPa (1600 psi) during this time. With this internal pressure, the reactor vessel wall would have undergone significant creep and eventual rupture if sustained at 1100°C for an extensive interval. Consequently, this rapid cooling of the reactor vessel at some time after four hours into the accident was responsible for maintaining the RPV integrity.

What caused this rapid cooling of the vessel in the TMI-2 event? Was this due to some external action or was it an inherent mechanism that could be considered, counted on, and utilized as part of accident management strategies? These are the fundamental questions that should be considered and evaluated in support of accident management evaluations.

Current accident management guidance and strategies consider that molten core material could drain into the RPV lower plenum. With concerns of debris coolability and the debris configuration in the lower plenum, (continuous or particulated) accident management evaluations consider that debris may not be coolable and that the ramifications on the RPV lower head could be significant overheating of the reactor vessel wall. This is considered as a possibility even if there is sufficient water in the reactor vessel to submerge the debris. Much of this is due to insights gained from investigating the TMI-2 accident.

As a result of the above considerations, there has been extensive consideration of ex-vessel cooling of the RPV lower head. While this is certainly an important, and for some designs, a convenient means of maintaining the RPV structural integrity, many containments do
not provide for easy access of water to the external surface of the RPV lower head. Therefore, inherent mechanisms which could provide for cooling of core material within the reactor pressure vessel and maintain the structural integrity of the reactor vessel lower head are of major importance in accident management evaluations. Specifically, an inherent cooling mechanism that would protect the RPV wall once water has been accumulated within the reactor pressure vessel would have a major impact on accident management actions.

This paper discusses an inherent cooling mechanism related to the following specific conditions.

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

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

The focus of the mechanism presented here is that minimal localized damage of the vessel wall (strain) can be expected to result in cooling of the RPV lower head. This is compared to the experience in the TMI-2 accident, both with respect to the wall temperatures and the observed final characterization of the RPV lower head. The results of the proposed inherent cooling mechanism are consistent with these observations from the TMI-2 Vessel Investigation Project.

The impact of the proposed inherent cooling mechanism on the technical support staff training philosophy and severe accident management strategies is also discussed.

1.0 INTRODUCTION - Why is the issue important?

The TMI-2 accident indicated many substantive issues that must be addressed as part of accident management. These include:

- substantial core damage can occur in a few tens of minutes after the core has been more than half uncovered,

- considerable hydrogen can evolve from a severe accident and result in combustion in the containment,

- melting of core materials and downward relocation can result in a relatively tightly packed geometry with limited coolability,

- core material relocation, including drainage into the lower plenum, can occur even though the material is completely submerged in water,
drainage of material into the lower plenum can cause substantial overheating of the RPV wall, and

- the debris which overheats the RPV wall may experience a relatively rapid cooling after substantial wall overheating has occurred.

Of the above points, all but the last are typically observed by severe accident analyses to a greater or lesser extent. However, the evaluations of the TMI-2 accident, including the internationally sponsored TMI-2 Vessel Inspection Program, which identified this overheating and rapid cooling, with impressive detailed forensic research have not presented a mechanistic description as to why this last phenomenon occurred.

Why is this so important? If an inherent mechanism cannot be identified to limit the potential for challenges to the reactor vessel integrity when water is added to the reactor vessel, then the accident analyses available in the scientific community would suggest that the RPV may subsequently fail in the absence of external cooling to the reactor pressure vessel lower head. Many containments were not designed to provide water around the reactor vessel and thus assuring external cooling of the vessel under accident conditions would be difficult. Moreover, providing such modifications to a plant would be extremely expensive, and if an inherent mechanism can be identified, these expenditures would certainly not be justified. Hence, the focus on the behavior of the RPV lower plenum for TMI-2 like conditions, with debris in the lower plenum submerged in water, is of crucial importance in the accident management evaluations.

In the evaluations discussed herein, a mechanism is identified for such an inherent cooling mechanism which relates to limited material creep of the RPV wall. We will first outline this mechanism and then provide a general perspective of how much creep is sufficient to assure long term cooling. Next, this approach is compared to the behavior in the TMI-2 accident to give a perspective of how the general phenomenological evaluations compare with the reported observations. Lastly, this behavior is applied to accident management evaluations, with particular emphasis on how this would be viewed for plants operated by Northeast Utilities.

2.0 INHERENT COOLING MECHANISM

For a configuration like that shown in Figure 1, high temperature molten core material draining onto the RPV lower head would result in substantial overheating of the reactor vessel wall. If the molten material entered the lower plenum at a time when there was a significant water inventory, EPRI-sponsored experiments (Hammersley, et al., 1993) and the TMI-2 experience suggest that a significant contact interface resistance would be established as the debris submerges water-filled crevices in the RPV wall. Vaporization of this trapped water (Figure 2) would separate the wall and the debris causing a contact resistance. This has been observed on repeated occasions, and not only controls the energy transfer from the melt to the wall, but also the adherence of the debris to the RPV wall.
Figure 1  
Accumulation of core debris on the RPV lower head.

Pipe is initially water-filled.

<table>
<thead>
<tr>
<th>Interface Temperature on Contact is:</th>
</tr>
</thead>
<tbody>
<tr>
<td>• Much greater than the critical</td>
</tr>
<tr>
<td>temperature for water.</td>
</tr>
<tr>
<td>• Sufficient to freeze the</td>
</tr>
<tr>
<td>molten debris.</td>
</tr>
<tr>
<td>• Insufficient to melt the</td>
</tr>
<tr>
<td>steel wall.</td>
</tr>
</tbody>
</table>

Figure 2  
Possible mechanism for developing a thin steam layer (contact resistance).
It is interesting to note that many steam explosion experiments, such as those reported by Long (1957), observe that contact between the high temperature melt and an unpainted wall initiated explosive interactions. When the wall was painted, no such interactions occurred. This is an indirect but important indication that water trapped in surface cavities (unpainted surface) vaporizes quickly once contact occurs. Furthermore, the fact that painted surfaces prevent such interactions indicates that when wall cavities are not available, this mechanism is eliminated. For the stainless steel clad surfaces in an RPV, wall cavities are always available. While Elevated pressures have been shown to prevent steam explosions (Henry and Fauske, 1979), vaporization of water in the surface cavities would still occur immediately following contact of the high temperature debris. Hence, these additional observations are entirely consistent with the formation of a contact resistance at the wall.

As a result of stresses imposed by the internal pressure and the dead weight of the material, the RPV lower head may experience creep at elevated temperatures. As the material creeps, the brittle oxidic core material is not subject to the same pressure stresses, hence it would not experience the same creep and would tend to separate from the RPV surface. Consequently, if such a creep mechanism were established, the lack of adherence of the debris to the RPV wall and the differential structural response would result in a stretching of the RPV wall with respect to the core material as illustrated in Figure 3. (The extent of creep is greatly exaggerated in Figure 3 to clearly illustrate the mechanism.) This would then create paths for water to ingress between the debris and the RPV wall as illustrated in Figure 4. This relative growth of the RPV wall compared to the core debris, while quite small (gap dimension of the order 100 \( \mu \text{m} \)), would be extremely important in terms of the thermal response of the RPV wall.

It is interesting to evaluate the growth (creep) of the RPV wall which would be necessary to cool the outer surface of the debris and the inner surface of the RPV wall such that the wall does not increase further in temperature, and in fact, cools down. The maximum heat flux that can be removed due to boiling in narrow gaps has been experimentally investigated (Monde, et al., 1982). The data for maximum heat flux in a narrow gap can be correlated by

\[
\frac{q}{A}_{\text{gap}} = C \frac{h_f}{\sqrt{g}} \sqrt{\frac{\rho}{g} \left( \frac{\rho_f}{\rho_g} - \frac{\rho_f}{\rho_g} \right)}
\]

where the constant \( C \) is expressed as

\[
C = 0.16 \left[ 1 + 6.7 \left( 10^{-4} \right) \left( \frac{\rho_f}{\rho_g} \right)^{0.6} \left( \frac{\ell}{\delta} \right) \right]^{-1}
\]

The variables in these equations have their standard definitions with \( \delta \) being the gap thickness and \( \ell \) the length of the gap. This results in heat fluxes of 0.1 MW/m\(^2\) to 0.7 MW/m\(^2\) for gap dimensions of a few hundred microns. Such heat fluxes are more than sufficient to cool the RPV wall and preserve the vessel integrity.
Figure 3  Schematic of relative growth due to creep.

Figure 4  Possible cooling paths created by differential growth between the core debris and steel wall.
In this regard, it should also be noted that the water required to provide the cooling rate observed in the TMI-2 accident, i.e. ~ 10°C per minute, only requires a water flow rate 60 g/sec-m². Thus for an area of 4 m² of the RPV wall only 0.24 kg/sec of water would be required to "percolate" in the gap between the wall and the debris to cool the wall. As discussed above, this can be achieved through a very small gap between debris and the wall, i.e. typically a few hundred microns.

Next, let us consider how fast the RPV wall could creep at elevated temperatures, i.e. how long would it take to develop a sufficient gap to cool the wall. Figure 5 shows creep rupture data as represented by the Larson-Miller (1952) parameter and Figure 6 illustrates the changing strength of RPV steel with temperature. If the internal pressure is 10 MPa, the stress in the lower head would be approximately 78 MPa (11,000 psi). Using Figure 5, this corresponds to a Larson-Miller parameter of about 51 (extrapolating the available data). If the temperature reached 1100°C (2470°F) as was the case in the TMI-2 accident, the Larson-Miller representation suggests a time to creep failure of 4.3 hours. Assuming a strain of 0.2 at failure and a linear rate, the material creep would cause a sufficient gap (~ 400 μm) about 15 secs. Of course, these numbers are only approximate but they clearly indicate that only a small gap is required and a sufficient separation can develop quickly when the temperatures reach values of about 1100°C. It is also interesting to note that the temperature would not get much above this value with an internal pressure of 10 MPa. Thus, the fact that the temperature reached this value in the TMI-2 accident, in and of itself, is strongly indicative of a creep related mechanism.

3.0 APPLICATION TO THE TMI-2 ACCIDENT

To investigate the relevance of this to the TMI-2 accident, let us consider a continuous segment of core material on the bottom of the RPV head with a diameter of about 1 m and a thickness of 0.2 m. Approximating this volume as a pancake, this corresponds to ~ 1600 kg of core material or about 1.6% of the total fuel. On a percentage basis, the energy generated in this portion of the core debris would be about 0.5 MW. Conservatively assuming that all of this power needed to be extracted in the gap between the debris and the RPV wall, the heat flux necessary to sustain cooling would be ~ 0.6 MW/m². Considering the primary system pressure to be 10 MPa, the above expressions would yield this heat flux when the gap is ~ 240 microns, i.e. a value requiring very little strain of the RPV wall. (This strain would correspond to about 0.05% which would not be detectable in the post-accident vessel inspection program.) However, the major point of this exercise is not that the strain is 200 microns, rather that it is very limited. In this regard, let us consider the potential cooling if the gap were 1 mm for the same conditions listed above. In this case, the heat flux that could be removed from the gap is three times greater than that conservatively estimated above. Hence, the inherent cooling mechanism is such that the strain would continue until there was sufficient water ingestion to cool the RPV wall and stop the local creep of the steel wall. Even in the most conservative assessment of the cooling conditions required, the strain (gap) necessary for effective cooling is very small compared to the radius of the continuous debris layer accumulated on the lower head. Hence, this relative growth of the RPV wall compared to the core debris, while quite small, would be extremely important in terms of the thermal response of the RPV wall.
Figure 5  Master creep rupture curve for 316 stainless steel. Note T is in degrees Rankine and t, is in hours.

Figure 6  Temperature dependent yield and tensile strength for RPV wall steel. (Source: Rempe, 1990 and Combustion Engineering, 1982.)
With the development and implementation of this model, the MAAP4 code provides a consistent representation of how the debris was finally cooled in the TMI-2 accident and how the RPV integrity was maintained. It should be noted that in order for this strain to occur, the RPV lower head had to achieve temperatures in excess of 1000°C. This is certainly in agreement with the temperatures determined by metallographic examinations during the TMI-2 Vessel Inspection Program (Wolf and Rempe, 1993).

MAAP4, which as been developed for accident management evaluations, includes models for molten debris draining into the lower plenum, non-wetting of the RPV wall (contact resistance), debris crust formation, heat transfer between the debris crust and the RPV wall, material creep of the vessel wall and water ingestion into the RPV wall-debris crust gap as the gap dimension increases due to material creep. Figures 7 and 8 illustrate two of the key results of the MAAP4 benchmark calculation with the TMI-2 accident. First, the model predicts ~ 56,000 lbm (25.5 tonnes) to drain into the lower head at about 3.77 hours (226 mins.) into the accident (Figure 7). This is in agreement with the observations that about 19 tonnes drained into the lower plenum at 227 mins. Figure 8 illustrates the calculated RPV wall thermal transient, with TRV(1,1) being the inner surface temperature of the RPV wall at the bottom of the vessel and TRV(1,5) is the temperature of the vessel outer surface at this location. As shown, the inner wall temperature increases rapidly to a temperature of about 2150°F (1177°C) where the decreasing metal strength and stress redistribution results in material creep. Eventually this creep is sufficient to enable cooling of the RPV wall by water ingestion as evidenced by the reversal of the temperature profile in the vessel wall. This peak temperature is consistent with the metallographic estimates of 1100°C in the TMI-2 Vessel Investigation Project (Wolf and Rempe, 1993).

4.0 Application to Accident Management Evaluations

Current evaluations for accident management behaviors and strategies consider that molten core material could drain into the RPV lower plenum. With concerns of debris coolability and the debris configuration (continuous or particulated) current accident management evaluations consider that debris may not be coolable and that the ramifications on the RPV lower head could be significant overheating of the reactor vessel wall. This is considered as a possibility even if there is sufficient water in the reactor vessel to submerge the debris. Much of this is due to insights gained from investigating the TMI-2 accident.

As a consequence, the current severe accident management training program at Northeast Utilities (NU) emphasizes that molten core debris will likely result in RPV lower head failure. This is reinforced by the Individual Plant Examinations (IPE) for NU's nuclear units which assign little likelihood of arresting the accident progression this late into a severe accident.

As a result, there has been extensive consideration of ex-vessel cooling of the RPV lower head. While this is certainly an important, and for some designs, a convenient means of maintaining the RPV structural integrity, none of the NU nuclear units provide for easy access of water to the external surface of the RPV lower head. Thus, NU technical support staff could be left in a position of considering accident management strategies to achieve external cooling of the lower head which are not easily implemented. Specifically, if an accident were to occur.
Figure 7  Mass of core debris relocating to the lower head in the MAAP4 TMI-2 benchmark.

Figure 8  Calculated RPV wall thermal transient due to debris relocation for the MAAP4 TMI-2 benchmark.
and water was successfully added to the reactor vessel, other actions could be taken in an attempt to establish ex-vessel cooling. This would be difficult and time consuming for some containment configurations, for example the flood-up of the Millstone Unit 1 (MP1) Mark I containment. The time required to submerge the lower head at MP1 is estimated to be between 5 and 10 hours, depending on the number of available pumps. Furthermore, this would result in the elimination of the pressure suppression function for MP1 and could also jeopardize the wetwell venting strategy for maintaining pressure within the containment and scrubbing of fission products in the suppression pool. In addition, PWR designs like Connecticut Yankee and Millstone 2 and 3 would have substantial difficulties in submerging the RPV lower head. For these plants, the strategy of Reactor Coolant System depressurization has been emphasized in order to avoid high pressure melt ejection and the potential for direct containment heating. Furthermore, the IPEs for Connecticut Yankee and Millstone Unit 2 have concluded that RPV failure at low pressure is likely to inevitably result in containment basemat melt-through because of the small surface area for ex-vessel debris cooling, and difficulty of flooding the lower reactor cavity.

Therefore, inherent mechanisms which could provide for cooling of core material within the reactor pressure vessel and thereby maintain the structural integrity of the reactor vessel lower head are of major importance in accident management evaluations. Specifically, an inherent cooling mechanism that would protect the RPV wall once water has been accumulated within the reactor pressure vessel would have a major impact on accident management strategies and training programs.

5.0 CONCLUSIONS

There are two major conclusions to this evaluation of an inherent in-vessel cooling mechanism.

1. If an accident should occur, water should be added to the RPV, regardless of the accident state, at least until the level is above the original core boundary (Top of Active Fuel/TAF).

2. Submerging the core debris with water in the RPV is sufficient to protect the vessel integrity and eventually cool the debris.

In other words, if water can be added to the RPV, the China Syndrome does not exist. This knowledge is crucially important to accident management evaluations.

6.0 REFERENCES


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Session 2a:

Hydrogen management techniques.

Chairman: J. Duco
Introduction

Depending on the extent of metal-steam-reactions during core degradation and the reaction of molten core materials with concrete in a severe accident situation, a significant amount of hydrogen will be released into the containment atmosphere within several hours from the onset of the event. According to local gas concentrations, turbulence and structural configurations within the containment, the released hydrogen can reach the boundary of deflagration or under certain conditions cause local detonations.

As the containment acts as the last barrier against fission products release, careful planned actions are required to prevent hydrogen concentrations reaching a potential level to threat its integrity.

Existing solutions like deliberate ignition or thermal recombiners designed for LOCA can be problematic because of the huge release of hydrogen expected, due to adverse effects or risks involved (1). At GRS under the sponsorship of the German Ministry of Environmental, Nature Conservation and Nuclear Safety, thoughts have been given to find out an alternate passive device requiring no power, no action by the plant personnel and limited maintenance (2).

Principles of Functioning

The capability of this device is based on the catalytic removal of hydrogen by letting oxygen from the air and hydrogen react to form water. Depending on the design, such
devices can have additional advantages like promotion of convection and dissolution of stratified layers. Right after the onset of the reaction, this device will produce steam thereby will partially inertise the surrounding atmosphere.

This device should function in the presence of extensive amount of steam as well as in the presence of elements known to act as catalytic poisons. Moreover, as the catalytic reactions increases exponentially with temperature and as the reaction of hydrogen is exothermic, so at higher temperature the yield of transformation will increase. Also, due to large surface area provided by the catalysts, an equilibrium at higher temperature between the heat generated at the plate and heat transfer out of the plate, will cause a higher yield of transformation.

Device for the Catalytic Removal of Hydrogen

Metallic Palladium and Platinum are known to act as catalytic mediums to promote the reaction of hydrogen and oxygen forming water (3). However, this catalytic reaction can be retarded or inhibited by the presence of poisons like CO, I₂ or steam containing boric acid etc. In order to remove the negative influences on the catalytic reaction, porous but well bonded catalysts with tunnel type of interconnections between the pores as well as inherent removal capacity to remove CO have been developed and intensively tested.

Fig. 1 shows a coated plate on which the device, to be described later is based. Essentially it consists of a stainless steel plate of 1 mm thickness. On this cleaned and sand blasted plate, an alloy of 95% Pd - 4% Ni - 1% Cu of 2000 Å thickness has been injected and coated by plasma or flamm injection process. These injection processes create a strong bondage by high temperature interdiffusion between the plate and the coating and thereby eliminating any spalling of the catalyst. Spalling of the coated catalyst should be avoided, because a spalled piece of coated catalyst in the containment atmosphere will further support the exothermic reaction and being devoid of the heat transfer possibility will heat-up very soon to cause an ignition source for the hydrogen containing atmosphere.
Status of Development

The coated plate, as described previously, generally form the basic element with which a catalyst device is designed. In Figs. 2, 3 and 4 the special features of the concepts of the devices as designed by the vendor, utility and the GAS, are presented. As these figures are self-explanatory, so no further comments will be made about their features. However, some remarks should be made about their performances. Tests carried out with the devices of utility and vendor within large volumes of the Model Containment at Battelle, Frankfurt have indicated sufficient capacities (4, 5) for removal of hydrogen and generating good convections. While results of all those above mentioned devices tested within 10-12 m³ volume are comparable, the device designed by GAS (Fig. 4) is specially marked for the mutual help of the catalytic plates and faster removal of hydrogen due to the access of gas flow available from all sides.

A large number of experiments both in small and large volumes with catalysts have been performed by the above mentioned institutions in the Federal Republic of Germany. Table 1 illustrates the type of catalysts and the main objective of the experiments with their essential results. Numerous experiments revealed that the catalysts were capable of recombining hydrogen and oxygen in the presence of excessive amount of steam as well as with low hydrogen concentrations (6).

Even prolonged exposure at saturated steam and large number of experiments with the same specimen of the catalysts did not reduce the catalytic capability of the plasma or flame injected catalysts. Particular attentions have been paid to examine the influences of potential poisons on the catalytic capability of different kinds of recombiners. Results of the works with poisons by the three institutions have been reported in (7). A short summarized version of the performed tests has been listed in Table 2. All these specified tests and more under adverse conditions have been successfully carried out both by utility and the vendor. It is essential to mention that all the specified tests with poisons and repeated loads of different atmospheric compositions and ageing have also been undertaken by the GRS in cooperation with KFA-Jülich with a single plate type of plasma injected specimen.
Large scale tests by Siemens and NIS for specific devices (Fig. 2, 3) have been carried out successfully at the Battelle Model Containment at Frankfurt. For this purpose full scale device has been tested within large volumes of 210 up to 640 m³.

Typical Experimental Results

In order to demonstrate the catalytic capabilities in large volumes as an example the results of one HDR-Test within a spherical volume of 9.5 m³ is presented in Fig. 5. For this demonstration tests catalytic plates were prepared with 20 x 20 cm stainless steel plates of 1 mm thickness. On these plates 2000 A of an alloy consisting of Pd 95 %, Ni 4 %, Cu 1 % was deposited in vacuum of 10⁻⁷ Torr. 10 of these plates were connected with U-shaped rails to have a dimension of 0.40 x 1 m of an aggregate of plates. For this test 2 of these aggregates i.e. 1.6 m² of catalytic surfaces (both sides of the plates) were symmetrically located within the sphere.

Apart from demonstrating the repeated catalysis of the same plates, an additional purpose of these tests was to test such a device against the aerosol depositions during core melt accidents. For investigation of this influence, 3 plates of the aggregate were jacketed with 6 HEPA filters. Before starting these tests, vacuum deposition of 3000 A of Fe had been undertaken on those filters to be jacketed on both the sides of the plates. During the tests in steam atmosphere, the finely deposited iron would react to form voluminous Fe₂O₄ and partially block the accesses of the gases to the catalytic plates. For testing the behaviour of temperature generated at the plates, thermocouples were located between the filters and the plates.

Fig. 5 shows the temperature increase of a plate located at the bottom of the plate assembly as a function of time. The corresponding concentration changes of hydrogen with time at a position 40 cm from the plate is also presented. Following the stabilisation of steam and air atmosphere after a time lapse of 118 min, 7 Vol.-% of hydrogen was introduced with spreaded jets located at the lower section of the vessel.

Due to exothermic catalytic reaction the temperature of the plate rose from 100° to 410° C. After the fall of the temperature to 350° C and the hydrogen concentration to 3 % at a location 40 cm from the plate, the second thrust of 10 Vol.-% hydrogen was given at 139 min. The temperature rose within 1 min due to catalytic reaction to
520° C probably due to accumulative influence of introduced and left over hydrogen concentrations of the previous test. The following test with 5 Vol.-% hydrogen caused a temperature increase to 610° C. Due to the accumulation of hydrogen for this test the measured concentration went up to 10 Vol.-% to react with the catalytic plates.

Although the plates with filters and aerosol deposition showed the catalytic capability throughout the four tests, two uncovered plates limited the catalytic activity during the last two injections of hydrogen as the surfaces were partially covered with steam entrenched with rust from the vessel wall. Therefore, also the protection of the catalytic activity against poisoning effects during stand-by conditions of normal reactor operation as well against the surface covering during catalytic activity becomes an important subject of additional further research and qualification of special devices.

Fig. 6 illustrates the tested catalytic activities within a ternary diagram at various atmospheric compositions of hydrogen, steam and air. These tests were performed with the vacuum deposited palladium alloys. It is evident from this figure that the catalysts are capable to remove hydrogen even at very low hydrogen and at very high steam concentrations.

**Future Activities**

As it was indicated in the previous section, that the poisoning of the catalytic surface due to the adsorption of CO, S, H$_3$BO$_3$ and other gaseous impurities as well as the covering of the surfaces by lubricants and aerosols remain as problems against the secure catalytic activities and to be addressed in future. Both the designs of devices to safeguard the catalysts from surface adsorptions as well as from covering are available for future testing.

Due to the tremendous heat generation by the exothermic reaction of hydrogen and oxygen at higher H$_2$ concentration, only those devices are to be selected for testing having a high heat removal capacity to limit the surface temperature increase below 600° C. High convections within an enclosed volume where parallel catalysts supply heat to each other does not ensure high heat removal capacity, as the convection brings fresh hydrogen to develop further heat by reaction. For the use at high H$_2$ concentration the energy dissipation capability of the special device has to be optimized.
Catalyst coated on a plate or granulates should be able to expand and contract during repeated reactions harmoniously if they would be having nearly the same thermal expansion coefficients to avoid spalling. Formation of cracks in the coated catalyst will generate potential differences to form corrosion cells. These cells can grow to cover the surface inhibiting locally the catalytic capability. Also, the spalled coating of catalyst exposed to hydrogen containing atmosphere will further generate heat. As the heat removal capacity of the substrate will not exist so at higher temperature the spalled coating could function as an ignition source. Therefore a very good interdiffusion to create a strong bondage between the catalyst and carrier material is to be ensured by future activities.

Conceptions for Large Dry Containments

Location

Large scale experiments at the Battelle Model Containment within volumes from $210^{3}$ up to $640^{3}$ m$^3$ have demonstrated that the sole application of catalysts could be sufficient as a mitigation measure against hydrogen threats during most severe accident situations. Application of the catalyst devices at various locations and compartments within the containment will be based on the calculations of main convections and the timely release of hydrogen to the containment atmosphere for typical accident scenarios. Analytical models have been developed by GRS for both the plate and box type of catalysts and have been verified by numerous post-calculations of experimental results. These models have been implemented in the modern version of RALOC-Mod4 for whole containment calculations. After the selection of an optimum device, calculations for a real containment will reveal both the optimum number and locations of the catalytic devices.

Capacity

In the Federal Republic of Germany, the agreement after intensive discussion has been to design the capacity of the catalytic devices so as to be able to reduce the hydrogen mass, equivalent to 100% zirconium oxidation within 24 h. Venting requirement also foresees the longterm capacity of the catalysts to remove hydrogen. Capacity of the catalytic devices are not only determined by the capacity to remove
hydrogen but also is governed by the ability of such device for heat dissipation. Attention has to be paid to keep the temperature increase below 600° C.

Quality Control

The longterm surface adsorption of poisons by the catalysts causing a decrease of effectiveness can occur at any time during the residence time of the catalysts within the containment. Even the convection from room to room could contaminate or may even cover the surface. Therefore periodic inservice inspection at regular intervals are necessary. Designs of devices can be performed in such a way where the catalysts are not disturbed but specimens from the same production batch within the same device can be examined for catalytic activities. A periodic annealing of the catalysts at larger internal will not only ensure the retention of catalytic capability but will increase it.

Possibilities of Combination with other Measures

The merits of the DUAL-concept of simultaneously applying the catalysers and the igniters are presently under intensive investigation at the Federal Republic of Germany. Placement of igniters as supplementary measure has been proposed for critical accident situation if a high release rate of hydrogen would cause a local large accumulation within a short time and the catalysts would not be in the position to prevent the formation of local burnable mixture. Moreover, the application of igniters has also been contemplated as a backup solution in case of loss of effectivity of some local catalysers or in the case of their destruction by local loads resulting from a system blowdown.

In the area of main convection flow paths and in the vicinity of the outer steel shell, catalytic devices should be placed with priority while thoughts have been given to concentrate igniters in areas of possible \( \text{H}_2 \)-release from the system. The capacity of the joint application of catalytic devices and igniters should be designed such, that in general locally 10 Vol.-% of hydrogen will not be exceeded.

Dilution of the atmosphere by inert gas injection with the association of the catalytic removal of hydrogen has also been contemplated to be an effective measure. The
increased level of inert gas in the containment atmosphere reduces the potential of high turbulent combustion while the catalytics remove the hydrogen continuously.

Recent developments at GRS have led to design of catalytic devices with the association of some 2-5 cm of chemicals on one side of the plate to release more than 20 Vol.-% of CO$_2$ within a very short time with simultaneous effective reduction of hydrogen keeping the temperature of the catalytic plates well below the desired limit.

Conclusions and recommendations

General conclusions:

- Catalytic recombiners reduce the H$_2$-concentration at extremely steam rich and at very low H$_2$-concentrations
- During the reduction of H$_2$-concentration the composition of gas mixture is shifted towards the steam rich corner of ternary diagram
- Recombiners promote by generating heat during catalysation the convections and thereby dissolve stratified layers
- During a continuous flow of hydrogen an equilibrium occurs between the heat generation at the plate and transfer out of the plate
- Plates with filter jackets could protect them from the deposition of dirt lubricants
- For filtered venting less H$_2$ in the atmosphere due to continuous H$_2$-reduction by catalytic recombination
- Longtime effectiveness of the catalysts even during the radiolyses of sumpwater
- Continuous energy input without any burning to containment atmosphere possible
- No loads on the instrumentation and walls.

RSK Recommendation to the German Reactor Safety Authorities:

The following recommendation was decided by the German Reactor Safety Commission (May, 1994):
In order to avoid an early or late loss of integrity of the containment of a PWR-plant due to hydrogen deflagration during beyond design basis accidents, the Reactor Safety Commission (RSK) recommends the application of catalytic recombiners. They recombine hydrogen before it reaches the deflagration limit as well as from steam inerted gas mixtures. Thus, a safety related meaningful amount of released hydrogen can be recombined within some hours and thereby contributions towards assuring the integrity of the containment as well as for the reduction of risk can be made. Catalytic recombiners are to be designed to achieve the efficiency. The catalytic recombiner is an unequivocal safety related measure to control hydrogen during beyond design events.

For the catalytic recombiners, concepts of prototype are existing which are technically well developed and proven by tests. Recombiners are passive constructional elements. Neither do they require any service by the operators nor do they require any energy supply. The installation of these recombiners within the existing PWR-plants does not pose any safety related problems.

RSK is proposing to optimise the constructional details concerning the specific forthcoming application. Test samples from catalytic devices should be tested annually to demonstrate the catalytic activity.

Regarding the number and the location of catalytic recombiners, the release rate of hydrogen as well as the characteristic gas transport times within the containments are to be considered.

These recombiners are to be placed primarily in the neighbourhood of global convection flows of the containment, near the containment steel shell as well as in those compartments where hydrogen will be released. Complying the physical principles within the voluminous containment, large scale convection loops will be formed.

With numerical calculations and engineering judgement and on the basis of existing knowledge regarding the distribution of hydrogen, the required number and location of the recombiners can be determined with sufficient exactness.
For the installation of the catalytic recombiners, the RSK is expecting quick preparation and submission of concrete technical plans from the utilities and vendors. Hereby the enveloping course of events can be based on the accident scenarios described in the introduction (late recoverable severe accidents with up to 100 % of Zirconium oxidation, low pressure core smelt scenario with core concrete interaction).

As for further step the RSK will examine the requirements to supplement the catalytic recombiners through fixed early burning of hydrogen by the igniters having short path of flamme propagation or through post dilution of the containment atmosphere. For the application of igniters, the RSK thinks it as necessary to prove the transferability of the recently obtained results and also the results of the planned tests on hydrogen combustion on the real conditions of a PWR-plant.
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functionability of catalysts for the removal of hydrogen during a core-melt-
down accident
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Table 1  
Research Work on Catalytic Recombiners in the FRG

<table>
<thead>
<tr>
<th>Institution</th>
<th>Working since</th>
<th>Type of Catalyst</th>
<th>No. of Tests performed</th>
<th>Main Objectives and Essential Results</th>
</tr>
</thead>
</table>
| Gesellschaft für Anlagen- und Reaktorsicherheit   | 1984          | Plate Type of catalysts, coating of Pd-Ni-Cu alloy on stainless steel             | Scoping test nearly 400 Large scale 6 | Effective catalyst from very low H₂-content to extremely high steam content  
Quick reactions even in the presence of poisons like I, CH₃I, CO, H₃BO₃ and vapours of lubricants, tested effectively against aerosol depositions |
| Siemens                                          | 1986          | Plate type of catalysts, coating of Pt on stainless steel                          | over 100              | Catalyst functional at wide compositional range even in the presence of poisons like I, CH₃I, CO, H₃BO₃ and vapours of lubricants. Layout, testing and qualification of device                                                                                   |
| NIS Ingenieur GmbH                               | 1989          | Granulate type of industrial catalysts, Pd coating on Al₂O₃                     | over 70               | Tests under atmospheric simulations of accident conditions in containment. Tested against lubricants, cable burns, I and CO, layout and testing of device                                                                                               |
### Table 2  Functional Tests of Catalysts under Various Parameter

<table>
<thead>
<tr>
<th>Test Parameter</th>
<th>Initial Concentration</th>
<th>Initial Temp°C</th>
<th>Beg. of Reaction</th>
<th>AT</th>
</tr>
</thead>
<tbody>
<tr>
<td>Vapours of organic solvants</td>
<td>9,5 Vol.-% H2 90,5 Vol.-% air</td>
<td>100</td>
<td>spontan</td>
<td>110°C</td>
</tr>
<tr>
<td>Vapours of weld (1,5 h unprotected over a welding place)</td>
<td>7,8 Vol.-% H2 92,2 Vol.-% air</td>
<td>100</td>
<td>spontan</td>
<td>87°C</td>
</tr>
<tr>
<td>Charging with boric aid</td>
<td>9,1 Vol.-% H2 90,9 Vol.-% air</td>
<td>100</td>
<td>spontan</td>
<td>100°C</td>
</tr>
<tr>
<td>Charging with CO</td>
<td>9,9 Vol.-% H2 90,1 Vol.-% air</td>
<td>100</td>
<td>spontan</td>
<td>115°C</td>
</tr>
<tr>
<td>Functionability after a longterm exposure in a gas mixture of hydrogen and nitrogen</td>
<td>9,1 Vol.-% H2 90,9 Vol.-% air</td>
<td>100</td>
<td>spontan</td>
<td>102°C</td>
</tr>
<tr>
<td>Dipping in water</td>
<td>9,1 Vol.-% H2 90,9 Vol.-% air</td>
<td>100</td>
<td>spontan</td>
<td>105°C</td>
</tr>
<tr>
<td>Functional tests of catalyst after an oil fire</td>
<td>9,1 Vol.-% H2 90,9 Vol.-% air</td>
<td>100</td>
<td>spontan</td>
<td>97°C</td>
</tr>
<tr>
<td>Functional tests after a hydrogen burn</td>
<td>9,1 Vol.-% H2 90,9 Vol.-% air</td>
<td>100</td>
<td>20 Vol.-% H2 80 Vol.-% air</td>
<td>spontan</td>
</tr>
<tr>
<td>Functional tests after a longterm charging with hydrogen</td>
<td>in average 10 % H2</td>
<td></td>
<td>spontan</td>
<td></td>
</tr>
<tr>
<td>Functional tests with low H2-concentration</td>
<td>1 Vol.-% H2 99 Vol.-% air</td>
<td>100</td>
<td>spontan</td>
<td>12°C</td>
</tr>
<tr>
<td>Functional test of catalysts after 7 months aging in atmosphere</td>
<td>5 Vol.-% H2 47,5 Vol.-% air 47,5 Vol.-% steam</td>
<td>100</td>
<td>spontan</td>
<td>46°C</td>
</tr>
</tbody>
</table>
Set up of a Catalytic Plate

Plate made of Stainless Steel ~1 mm

Coated with
100 % Pd or
95 % Pd - 4 % Ni - 1 %
Cu ~2000 Å

Figure 1 Set up of a Catalytic Plate
CHARACTERISTICS:

- Catalyser consisting of Pt-coated plates
- Plates arranged with intermediate gaps for gas flow
- Construction for strengthening the gas flow by chimney effect

Figure 2  Features of Recombination Devices: SIEMENS
SPECIAL FEATURES OF THE CATALYTICAL RECOMBINERS DEVELOPED AND QUALIFIED IN THE FRG

Concept of Utilities

- Catalysers consisting of Pd-coated granulates
- Coated granulates stacked between two wire nets forming an element
- Elements stacked with intermediate channels forming a module
- Modular construction open only at top and bottom to develop chimney effect

Figure 3 Special Features of the Catalytic Recombiners Developed and Qualified in the FRG
SPECIAL FEATURES OF THE CATALYTIC RECOMBINERS DEVELOPED AND QUALIFIED IN THE FRG

Concept of GRS

- Catalyser made by plasma injected Pd-Ni-Cu alloy on stainless steel
- Plates stacked in an inerted box to avoid contamination
- Plates will be automatically unfolded and distributed for maximum catalysation
- Plates enveloped by filters against aerosol deposition

Figure 4 Special Features of the Catalytic Recombiners Developed and Qualified in the FRG
Figure 5  Illustrates the Behaviour During Successive Catalysis of Four Times
- Performed Ignitions with Catalytic and Spark Igniters
- Successful Tests with Catalytic Recombiners

Operation Area for the Use of Catalysts and Igniters

Figure 6  Operation Area for the Use of Catalysts and Igniters
HYDROGEN MANAGEMENT IN CANDU REACTORS

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HYDROGEN MANAGEMENT IN CANDU REACTORS

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1.0 INTRODUCTION

CANDU is distinct from most reactor designs with respect to defining of severe accidents, and in particular in the analysis of hydrogen source term. Firstly, the initiating sequence for worst case hydrogen generation, loss of cooling accident (LOCA) followed by loss of emergency cooling injection (LOECI) and subsequent Zr-steam reaction, is not considered a 'severe accident' but rather, is considered within the design basis. Analysis and supporting research of accident scenarios involving fast evolution of hydrogen has thus been underway for more than twenty years, as part of the design basis for CANDU. Second, the quantities of hydrogen produced tend to be limited to oxidation of fuel cladding. Further hydrogen production is limited because the more than 200 tonnes cool, low pressure moderator water surrounding the fuel channels and the additional cool water in the shield tanks act as heat sinks and contribute to a comparatively low probability of core melt (4.6 x 10^-6 per year) as determined by probabilistic risk assessment (PRA) studies [1,2]. Thus, mechanisms for producing hydrogen, associated with core melt (i.e., oxidation of structural materials and core-concrete interactions), are improbable. Nonetheless analysis of scenarios involving core melt and core disassembly have been carried out [3] as part of a commitment to completeness of understanding for all postulated sequences.

The CANDU reactor has two very different containment types, single unit containments designed and built by AECL, and the multi-unit containments owned and operated by Ontario Hydro (see Figures 1 and 2). While accident sequences and hydrogen release scenarios are similar, the different containment designs have resulted in rather different approaches to hydrogen management.

This paper describes the approaches to reactor safety hydrogen management in CANDU reactors, for the different CANDU containment types and with reference to current issues, analyses and supporting research activities.

2.0 HYDROGEN MANAGEMENT IN CANDU STATIONS

The CANDU concept is basically a large dry subatmospheric containment combined with a large capacity dousing system for pressure suppression after LOCA. Single-unit stations
(CANDU 600) have a free volume of 48,000 m³ and a design pressure of 124 kPa(g). For the single unit design there are two units operating in Canada and ten more operating or under construction in Argentina, Korea and Romania. Multi-unit containments [4] consist of several (up to eight) slightly subatmospheric, in-line reactor buildings, each connected by a large duct to a central vacuum building which contains the dousing and pressure suppression systems (see Figure 3). The multi-unit concept has been deployed in twenty reactors in Canada.

**Single Unit Stations**

In single unit CANDU reactors, natural and engineered mixing mechanisms are relied upon to dilute hydrogen to non-flammable concentrations with available containment air. Analysis indicates that the hydrogen released from complete oxidation of the zirconium fuel cladding in one entire loop in the primary system (i.e., one half the fuel channels) cannot exceed more than 4% of the containment volume. The quantities and release rates of hydrogen are calculated using CHAN II, a computer model which incorporates the various feedback effects on fuel temperatures in a CANDU fuel channel [5]. It calculates the temperature transients of the fuel, pressure tube and calandria tube at each bundle position along the fuel channel as a function of channel power and flow rate. The hydrogen-steam mixture is initially released into the vault, an approximately 2300 m³ enclosure housing the reactor core and the primary heat transport system. Rapid mixing with air in the vault and with the remaining 46000 m³ containment air is promoted by large vault cooler fans which recirculate the air volume of the vault about every minute. As well, the vault has an opening at the top and a return flow pass at the bottom to allow natural high volume air exchange between the vault and the remainder of containment. The mixing and distribution of hydrogen in the containment atmosphere have been predicted using H2MIX and PRESCON2 computer codes [6].

The H2MIX code uses a hydrogen mixing model based on hydrogen density differences between nodes, whereas PRESCON2 can predict hydrogen distribution due to density differences (thermal) and air-steam mixture differences between adjacent nodes but with no consideration of hydrogen density differences. The worst case peak transient concentrations of hydrogen in the vault predicted by H2MIX and PRESCON2 are 6.6% and 8.1% respectively (see Figure 4). A parametric evaluation of the potential combustion threat posed by the local transient flammable gas mixture has been carried out using VENT, a simple lumped parameter model that predicts the combustion pressure within a vented enclosure [7]. The model contains conservative adiabatic, complete combustion assumptions, and the vent area and burn rate are inputs. Burning velocities have been input at 12, 25 and 50 times the laminar velocity for the respective mixtures. The predicted combustion overpressures in the
vault (with venting to the boiler room) for the base case (6.6%, 50 x laminar burning) is 65 kPa (see Figure 5). This magnitude of pressure differential is not viewed as a threat to containment structures. For the sensitivity case, 8.1% H₂, an acceptable margin is still obtained but with 25X the laminar burning rate. Considering the self-limiting peak concentration of hydrogen, the short duration of the peak concentration and the improbability of ignition from a random source, intervention by intentional ignition has not been implemented. Catalytic recombiners are under consideration for back-fitting and are viewed as having potential to further improve the safety margin.

**Multi-Unit Stations**

In multi-unit CANDU stations, the hydrogen release rates are calculated using CHAN-II in essentially the same manner as for single unit stations [8]. The release scenarios differ for different stations depending on the size of the reactor (1600 MW to 2700 MW) and the number of channels served by a single cooling loop. The maximum quantity of hydrogen calculated to be released for a high power reactor is typically 106 kg-mole with a maximum peak release rate of 0.1 kg-mole/s [9].

The multi-unit containment system differs from the single unit containment system in the response to an accident which results in hydrogen formation [10]. Prior to hydrogen release, self-actuating pressure relief valves located near the vacuum building open to permit venting of the hot steam/air mixture. The combined effects of vacuum building suction and containment heat sinks are sufficient to reduce containment pressure to below atmospheric within about 60 seconds. Thereafter, containment pressure continues to decrease due to steam condensation until balanced and eventually overcome by the pressure increasing effects of structural air inleakage and compressed gas addition. Long-term pressure control is maintained through venting via a well defined, monitored and filtered pathway.

Although the total containment volume (approx. 10^5 m³) is large enough to ensure non-flammable hydrogen concentrations, the hydrogen release can occur at a time and a rate such that uniform mixing in the entire volume is not possible. Multi-unit stations have therefore implemented distributed ignition systems in the vault regions of each reactor to ensure flammable volumes are promptly ignited at conditions near the limits of flammability where the combustion consequences are lowest.

The deliberate ignition concept is realized by placing TAYCO glow plug ignitors at strategic locations at different elevations in the reactor vault and surrounding areas. The ignitors are
contained within specially designed protective housings and powered by non-interruptible Class II 120 VAC power. A total of 12 to 58 ignitors are used per unit, depending on the station. The ignitor system is automatically initiated very early in the accident on a combination of the containment high pressure and the containment high activity signal, depending on the station.

This mitigation concept is based on ignition occurring within a predefined band of mixture compositions (the assured ignition limits) just within the absolute limits of flammability. This approach is illustrated in Figure 6, which shows the flammability limits, the assured ignition limits, and a typical progression of mixture composition following an accident which results in hydrogen release [9]. In this progression, it is postulated that the hydrogen release starts at 300 seconds and proceeds at 0.025 kg-mole/s. After 1150 seconds, conditions are such that the assured ignition limits are reached and ignition occurs.

The combustion pressures associated with deliberate ignition are conservatively calculated using the VENT code assuming adiabatic complete combustion [11]. Pressure relief is available by venting to adjacent containment compartments via large openings. Analysis shows the pressure transient in the ignition volume for the worst case composition is insufficient to result in a breach of the containment envelope [9].

Note that this approach relies upon the assumption of uniform mixing in the containment volume in which combustion occurs. This condition is assured by the presence of large vault cooler fans which are capable of turning over the vault volume in a matter of minutes. Should these fans be unavailable, a combination of strong natural convection and a prolonged period of steam inverting ensures adequate mixing still occurs. Detailed analysis is currently underway using the GOTHIC code [12] to better understand the mixing pattern and the concentration gradients established in the presence/absence of forced convection. Figure 7 illustrates these calculations by comparing the predicted mixture composition to the flammability limit for a typical calculation.
3.0 ISSUES

Safety margins for all aspects of reactor functions are under continuous review in Canada. This is motivated by an interest in continuous improvements in design, evidenced in the construction of 25 CANDU stations in as many years, and by an iterative rather than prescriptive regulatory process. That is, the regulator can periodically request reassessment or additional analysis and support for existing safety margins.

Current issues with respect to hydrogen focus on potential for non-uniform mixing and distribution of hydrogen within containment and subsequent uncertainty in combustion behaviour. Where reliance is solely on dilution of hydrogen by containment air, there is a need to improve confidence that transient local accumulations of hydrogen do not occur such as to pose a combustion threat to essential equipment or structures if such volumes become ignited. Where ignitors are installed, there is continued interest in analysis of mixing and distribution in terms of refining the rationale for number and placement of ignitors. As well, since combustion is intended, issues arise over combustion behaviour in non-uniform mixtures (flame acceleration and DDT) as well as the potential for standing flames, scaling effects in vented deflagrations, and equipment survivability.

4.0 EXPERIMENTAL PROGRAM

In support of analysis of accident scenarios involving hydrogen, an integrated research program is underway at the Whiteshell Laboratories of AECL jointly funded by AECL and the Canadian utilities (Ontario Hydro, Hydro Quebec and New Brunswick Power). The research programs contribute fundamental understanding, physical data for input to models and validation of effects of scale. The experimental program is presently focussed along four lines: complex deflagration behaviour, standing flames, deflagration to detonation transition (DDT) and development of catalytic recombiners for hydrogen removal.

Deflagration Behaviour

The deflagration program examines combustion behaviour in concentration gradients and stratified mixtures, flame acceleration in the presence of obstacles, effects of multiple ignition sources and vented combustion in 6 m³ and 10 m³ vessel of the Containment Test Facility (CTF). One of the main aims of the deflagration program is to develop and validate an advanced deflagration model for prediction of complex effects in real geometries.
A new 120 m³ facility has been built to provide validation of predicted complex deflagration behaviour on a large scale with geometric similarity to actual containment subvolumes and vent ratios encompassing the relevant range of interest [13]. The facility also is intended to provide full-scale equipment survivability testing in simulated post-LOCA atmospheres. A cut-away drawing of the facility is shown in Figure 8.

**Standing Flames**

A specially-designed 70 m³ facility examines the stability, the shape and the thermal loads from standing flames resulting from a jet of H₂-steam issuing into an air-steam atmosphere. The experiment uniquely simulates the post-LOCA containment atmosphere which is a key factor influencing the stability, the shape and the thermal loading from standing jet diffusion flames.

**DDT**

If a detonation is to occur in containment, the only credible mechanism is by deflagration to detonation transition (DDT). The Canadian experimental program on DDT is directed towards establishing the criteria for DDT in terms of mixture sensitivity, shock strength of accelerated deflagrations and turbulence parameters [14].

**Catalytic Recombiners**

A special catalyst material has been engineered in AECL as part of developing advanced catalytic processes for manufacturing heavy water. This new material has extraordinary wet-proof properties as well as a high kinetic rate for the H₂-O₂ reaction and shows good promise for application to hydrogen removal in nuclear containments [15].

A test program has been completed, demonstrating the performance of a catalytic hydrogen recombiner using the advanced AECL catalysts [16]. Figure 9 shows typical hydrogen removal capacities at different hydrogen concentrations. The recombiner has particularly good self-start characteristics even in cold, wet or oxygen-depleted atmospheres, responding to less than 1% H₂. The catalyst material is resistant to radiation and foreseeable poisons, including carbon monoxide and iodine and operates safely at temperatures greater than 700ºC. Hydrogen removal capacity of 100 kg/h is readily back-fittable to a large containment.
5.0 SUMMARY

The CANDU reactor design is significantly different from most other power reactor designs with respect to hydrogen source term. Analyses of hydrogen release from LOCA/LOEIC are part of the design basis. However the scenario does not lead to a loss of core structural integrity unless accompanied by other system failures. Thus the source term for hydrogen release is essentially limited to oxidation of fuel cladding and does not escalate to include significant oxidation of structural materials, or core concrete interactions.

The basic approach to hydrogen management is rapid source term dilution, by natural and engineered mixing with the large volume of available containment air. In the transient phase of the hydrogen release, analysis has indicated a potential for flammable mixtures to occur in regions of the reactor vault. Where such accumulations are not self-limiting at acceptable levels, ignitors are combined with engineered mixing to remove hydrogen, independent of the timing or the magnitude of the release.

Analysis continues over issues related to the confidence in predictions of hydrogen distribution, the validity of simple combustion models and the potential for standing flames or local DDT to damage essential equipment. These analyses are supported by scientific and engineering research aimed at producing advanced, validated tools (models) to resolve remaining issues and confirm margins of safety.

6.0 REFERENCES


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ABSTRACT

Severe accident calculations for Belgian PWRs have confirmed that the production of hydrogen and carbon monoxide is a concern for the containment integrity. Therefore several preventive solutions were examined, among which the catalytic recombiners offer decisive advantages.

This paper briefly describes the calculation method developed to determine the required surface area. The release rates are provided by STCP/MARCH3 runs and are used as input data of a postprocessor developed in Belgatom. This program computes the modified atmosphere composition for a given catalyst surface area. In addition to this, it calculates the adiabatic combustion pressure with and without recombiners, which allows to assess the risk of containment failure in both situations.

Calculations were performed for two 900 MW plants with different concrete compositions. One of them is of the siliceous type which releases limited quantities of carbon monoxide, while the second one is made of limestone common sand and produces large quantities of that gas during the decomposition process.

Because the recombination process is governed by a time constant, the predimensioning was based on accident scenarios with the highest computed instantaneous release rates of flammable gases. Hence, attention was focused on scenarios initiated by a small break LOCA because they lead to high release rates of hydrogen at vessel breach. The calculation results show that a catalyst surface area of 250 m² prevents the deflagration pressure to exceed the ultimate containment capability.

1. Introduction

Severe accident calculations performed for the PSA studies of two Belgian 900 MW plants confirmed that the hydrogen issue can become a potential concern for the containment integrity. Different countermeasures were examined to prevent the combustion hazard. The comparison of those different solutions highlighted that the catalytic recombiners offer decisive advantages which have already been presented in numerous publications such as references 1,2,3. The main conclusions of that comparison can be found in reference 1; the present paper is mainly devoted to the modelling of the PARs (Passive Autocatalytic Recombiners).

The PARs are totally passive and operate from very low hydrogen concentrations to extremely high steam concentrations. The recombination rate increases with the catalyst temperature, i.e. with the hydrogen concentration, and qualification programmes have confirmed the resistance against various types of poisoning (H₂, CH₃F, CO, H₂BO₃, vapours of lubricants ...) (ref 4,5).

Therefore, it was decided to investigate the behaviour of the catalytic recombiners in various accident scenarios to evaluate the total recombiner area needed to prevent the risk of combustion. The analysis was performed for a single volume from thermalhydraulic conditions and gas production rates computed with MARCH3.

A postprocessor, named CARE (Catalytic Recombiners), was developed in Belgatom which:

a) determines the instantaneous gas concentrations from the MARCH3 output files;

b) computes the gas concentrations for a catalyst area and a given recombination rate;

c) assesses the pressure of an isochoric adiabatic combustion process with and without recombiners.

In the present paper, the models of the catalytic recombination and of the combustion process are first presented. The calculation procedure is then developed. Finally, sample calculations are discussed to draw conclusions on the dimensioning of catalytic recombiners.
2. Recombination process

The time variation of the hydrogen concentration $x$ can be described in function of the total number of moles $n_t$, the hydrogen production rate $n_{H2}$ and the recombination rate $\lambda$, with the equation:

$$\frac{d}{dt}(x_n) = n_{H2} \cdot \lambda \cdot n_t. \quad (1)$$

The recombination rate $\lambda$ is related to the hydrogen half-life $\tau_{1/2}$ through the relationship:

$$\lambda = \frac{\ln 2}{\tau_{1/2}}. \quad (2)$$

According to the tests performed by Siemens-KWU [ref. 4], that parameter can also be evaluated with the simple formula:

$$\lambda = C_k \frac{S}{V} \quad (3)$$

where $S$ and $V$ respectively denote the catalyst area and the control volume; $C_k$ is a constant characterizing the recombiner.

Over short time intervals, both the total number of moles $n_t$ and the molar production rate of hydrogen $n_{H2}$ can be assumed constant. Hence, equation (1) can be analytically integrated to get the relationship:

$$x(\Delta t) = x_0 \exp(-\lambda \Delta t) + \frac{n_t}{\lambda n_t} \left[1 - \exp(-\lambda \Delta t)\right] \quad (4)$$

which is successively applied to each time step of the MARCH3 calculation. The variation of the number of moles due to the recombination and to the hydrogen source are thus neglected within each time step. To minimize the error in the mass balance, the calculation were performed with an averaged number of moles. In this case, the instantaneous mass error never exceeded 0.2% and was usually 2 or 3 orders of magnitude lower.

3. Combustion process

When the time dependent atmosphere compositions are known with and without recombiners, the containment load due to a global deflagration can be assessed in both situations. The combustion process is modelled as adiabatic and isochoric, which represents an upper bounded limit for the values observed in deflagration experiments [ref. 2].

The energy balance of the process is directly derived from the first principle of thermodynamics and is expressed by:

$$\sum n_k u_k = \sum n_k u_k^* + n_{CO} \Delta E_{CO} + n_{H2} \Delta E_{H2} \quad (5)$$

where $n_k$ refers to the number of moles of species $k$, $u_k^*$ to the corresponding specific internal energy and $\Delta E_k$ to the heat of reaction at constant volume. The exponents - and + respectively identify the states before and after the combustion.

Dividing equation (5) by the total number of moles of the reactant, $n_1^+$, allows to introduce the molar fractions:

$$x_k^- = \frac{n_k^-}{n_1^-} \quad (6)$$

and the pseudo molar fractions:

$$x_k^+ = \frac{n_k^+}{n_1^+} \quad (7)$$

to get the final form:

$$\sum x_k^+ u_k^+ = \sum x_k^- u_k^* + x_{CO} \Delta E_{CO} + x_{H2} \Delta E_{H2} \quad (8)$$

Solving equation (8), requires:

a) to select equations of state for the different species;
b) to model the combustion process with the relevant heats of reaction.

The perfect gas law is adopted and the molar isobaric heat capacities are expressed with a general polynomial formula:

$$C_p(T) = A_1 + A_2 T + A_3 T^2 + A_4 T^3. \quad (9)$$

Hence the molar specific enthalpy $h$ can be expressed as:

$$h(T) = \left(1 - \frac{R}{T}\right) + A_2 \frac{T^2}{2} + A_3 \frac{T^3}{3} + A_4 \frac{T^4}{4} + B \quad (10)$$

and the molar internal energy $u$ as:

$$u(T) = \left(1 - \frac{R}{T}\right) + A_2 \frac{T^2}{2} + A_3 \frac{T^3}{3} + A_4 \frac{T^4}{4} + B \quad (11)$$

For all non condensible gases (CO, CO2, H2, N2 and O2), the constant $B$ is set 50 that the specific enthalpy is equal to zero at 273.15 K. For steam, the specific enthalpy at 273.15 K is equal to the molar latent heat of vaporization.

The combustion of hydrogen and carbon monoxide are described through the equations:

$$H_2 + \frac{1}{2} O_2 \rightarrow H_2O \quad \Delta H_{ref} = 241 800 \text{ kJ/kmole} \quad (12)$$

and

$$CO + \frac{1}{2} O_2 \rightarrow CO_2 \quad \Delta H_{ref} = 283 000 \text{ kJ/kmole} \quad (13)$$

where the heats of reaction are given at constant pressure and reference temperature (298.15 K).

The heats of reaction mentioned above need to be corrected for a constant volume process and arbitrary initial conditions. This is done using energy balances and equations (11), (12) and (15); the detailed calculations are not reproduced here.
The equation of the combustion process (6) can now be solved with respect to the flame temperature \( T^* \), which finally allows to compute the peak pressure \( p^* \) from the formula:
\[
p^* = p_0 \left( \frac{n_1}{n_1^*} \right) \left( \frac{T^*}{T} \right). \tag{14}
\]

### 4. Calculation method

The MARCH3 code is first run with the "no combustion" option to produce the thermohydraulic conditions in the containment and the release rates of the non condensable gases; those data are stored in the MACEPT and CONTAN files.

The calculation starts with the initial inventories of the different gases.

When no recombination takes place, the quantities of nitrogen and oxygen remain constant during the accident because the containment is assumed to be isothermal. On the contrary, the quantities of CO, CO\(_2\), and H\(_2\) increase due to the zirconium oxidation and to the core-concrete interaction.

At each time step, the build-up of CO, CO\(_2\), and H\(_2\) is computed by integrating the instantaneous release rates stored in the CONTAN file. The number of moles of each non condensable gas can then be derived from the definition:
\[
n_k = \frac{m_k}{\mu_k}. \tag{15}
\]

The total number of moles in the containment \( n_1 \) is given by the summation:
\[
n_1 = n_{\text{CO}} + n_{\text{CO}2} + n_{\text{H}2} + n_{\text{H}20} + n_{\text{N}2} + n_{\text{O}2}, \tag{16}
\]

in which \( n_{\text{H}20} \) is expressed in function of the steam concentration \( x_{\text{H}20} \) read in the CORRAL file:
\[
n_{\text{H}20} = x_{\text{H}20} n_1. \tag{17}
\]

to obtain the equation:
\[
n_1 = \frac{n_{\text{CO}} + n_{\text{CO}2} + n_{\text{H}2} + n_{\text{N}2} + n_{\text{O}2}}{1 - x_{\text{H}20}}. \tag{18}
\]

The method gives the atmosphere composition at any time step of the MARCH3 results. Nevertheless, it must be kept in mind that the time step can be temporarily reduced in some subroutines, leading to the production of extra results at intermediate time levels. Therefore, a specific reading procedure was developed to select consistent input data for the postprocessor code.

A second calculation is performed in parallel to simulate the operation of the catalytic recombiners.

Starting from the same initial conditions, equation (4) is iteratively applied between successive time steps to calculate the variation of the hydrogen concentration in the containment. After each time step, the mass and mole inventories are recomputed to account for the removal of hydrogen and oxygen, and for the production of steam. This procedure is applied till the end of the accident with the following assumptions:

- The recombination process starts when the hydrogen concentration exceeds a user specified threshold value, and proceeds as long as the oxygen concentration is higher than a minimum value fixed by the stoichiometric limit.

- The steam produced by the recombination process tends to inert the atmosphere, but also influences the condensation processes in the containment. This latter aspect cannot be properly taken into account as the calculation is performed a posteriori from the MARCH3 results. Therefore, a maximum saturation ratio \( S_{\text{max}} \) (\( \leq 1 \)) can be imposed by the user so that the steam inventory exceeding the value:
\[
n_{\text{H}20,\text{max}} = \frac{S_{\text{max}} P_0 S(T) V}{RT} \tag{19}
\]

condenses and accumulates in the sumps. Selecting a negative value for \( S_{\text{max}} \) keeps the saturation ratio equal to the current value in the MARCH3 output data. In this case, all steam produced by the recombination is removed from the atmosphere, which leads to a conservative assessment of the risk of burning. This option has always been exercised in the calculations presented in this paper.

- Because the catalytic recombination is exothermic, it creates an heat source in the containment which could increase the atmosphere temperature. This effect cannot be modelled in the present calculations which postprocesses output data from MARCH3. However, it must be pointed out that the temperature increase observed in tests [ref. 4] is rather limited and can be neglected. Therefore, the ambient temperature is kept unchanged and the calculation of the modified ambient pressure only accounts for the number of moles.

- For a long time, the carbon monoxide has been considered as a potential poison for the catalytic recombiners. Tests have shown that this issue can now be discarded [ref. 4, 5]. Moreover, some vendors have mentioned that the catalyst also recombines the carbon monoxide. Hence, a calculation option allows the user to simulate the recombination of CO at the same rate as H\(_2\).

- Finally, the postprocessor also computes the adiabatic isochoric combustion pressure with the approach developed in section 3. The method is applied to both situations with and without recombiners, provided enough oxygen is available in the containment. When the combustion process is oxygen limited, the fraction of flammable gases which is effectively burned, is accordingly reduced.
5. Calculation results

The program CARE has been applied to accident scenarios defined for the PSA studies of Doel 3 and Tihange 2. As the recombination process is governed by a time constant, attention has been paid to the accidents with the highest computed instantaneous release rates. Those latter were observed in accidents initiated by a SBLOCA. Under those conditions, up to 70% of the available zirconium can be oxidized in the vessel and the corresponding amount of hydrogen is mainly released at the time of vessel breach.

Four different accident scenarios can be defined within the SBLOCA family (= S), depending on the time of core melt (E = early; L = late) and on the availability of the containment cooling system (Y = yes; N = no). Although all those calculations were performed for each plant over the first 48 hours of the accident, the present paper focuses on the run SEY which leads to the highest containment loads. The reason is twofold.

On one hand, the early degradation of the core speeds up the hydrogen release. On the other hand, containment cooling reduces the partial steam pressure, hence the mass inventory of the inert gases; this leads to higher temperature and pressure during an adiabatic combustion process.

With regard to the present analysis, the two power plants differ mainly from each other by the containment volume and the concrete composition. The containment volume is 60 600 m$^3$ in Doel 3, and 69 640 m$^3$ in Tihange 2, while the concrete is highly siliceous in Doel 3 (76.2% SiO$_2$) and of the limestone common sand type in Tihange 2 (43.9% Ca CO$_3$).

The calculations were performed assuming no carbon monoxide recombination and using the gas release rates produced with MARCH3. Those data can be considered as conservative because the code enhances the in-venelre production of hydrogen and tends to maximize the basement penetration.

The catalytic recombiners start to operate between 1 and 2 VoI% of hydrogen; the threshold value was fixed to 2% in the present analysis. The catalyst area has been taken equal to 250 m$^2$.

The scenario SEY is initiated by a SBLOCA. The emergency core cooling system is not operational, which leads to an early core melt, while the containment atmosphere is cooled either by fan coolers (Doel 3) or by a spray system (Tihange 2).

The main events of the accident scenario are summarized in table 1. The results for Doel 3 will be commented first.

The loss of inventory leads to the core uncoverage and degradation. Hydrogen is produced and massively released at vessel breach (fig. 1, t = 9 098 s). Without recombiners the mass inventory grows on during the molten core concrete interaction (MCCI). The production of carbon monoxide starts when the zirconium is depleted, i.e. at 34 031 s (fig. 2).

The recombiners efficiently reduce the hydrogen concentration which does not exceed 4.7% during the accident (fig. 3). Although the CO inventory remains unchanged (fig. 2), the surprisingly higher concentration (fig. 4) is explained by the reduction of the total number of moles, which is due to the recombination process.

The equivalent hydrogen concentration ($YH_2 + 0.6 YCO$) reaches the maximum value of 6.4% at 43 760 s (fig. 5). This time corresponds also to the maximum combustion pressure which is reduced below the design pressure (0.45 MPa) when 250 m$^2$ catalytic recombiners are implemented in the containment (fig. 6).

The same analysis was carried out for the scenario SEY applied to the plant Tihange 2. The results are presented in table 1 and in figures 7 through 12. Although they are similar to those for Doel 3, particular features are due to the concrete composition and need to be commented.

The high CaCO$_3$ content results in very large releases of CO. This appears in figures 8, 10 and 11. As the PARs are assumed not to recombine CO, the gas accumulates in the containment which leads to higher combustion pressure than in Doel 3 (fig. 12), despite the larger containment volume (69 640 m$^3$ vs. 60 600 m$^3$). Moreover, in this case, the combustion process is oxygen limited because of the very large quantities of available flammable gases (fig. 13). This explains the leveling-off of the combustion pressure curves in figure 12.

Though the maximum deflagration pressure with recombiners is higher than the design value of the containment it still stays below the failure pressure (0.90 MPa).

For illustrative purposes, calculations were also performed for both plants assuming that the carbon monoxide is recombined at the same rate than the hydrogen; the maximum values obtained for the main parameters are summarized in Table 2.

6. Conclusions

- For 900 MW PWR plants with containment volumes between 60 000 and 70 000 m$^3$, 250 m$^2$ catalyst area can prevent high hydrogen concentrations which could lead to containment failure if deflagration takes place.

  Using the same catalyst area, the hydrogen concentrations after 24 h were computed for 100% Zr oxidized. The containment atmosphere was conservatively assumed to be steam free. In those conditions, the final concentrations are respectively equal to $4 \times 10^{-4}$% in Doel 3 and to $1.3 \times 10^{-3}$% in Tihange 2.

- When only hydrogen is recombined, the maximum combustion pressure mainly depends on the concrete composition rather than on the containment volume. The highest values were obtained with limestone common sand concrete and the largest containment volume. Furthermore, the maximum pressure could have been even higher if the combustion process was not oxygen limited.

- Questions are still raised concerning a possible poisoning of the catalyst by the carbon monoxide [ref. 2]. Test results tend to demonstrate that the problem is not relevant anymore [ref. 4, 5]. Moreover, vendors argue that carbon monoxide is recombined too. Illustrative calculations show that this particular recombination capability, if reconfirmed, could significantly increase for some plants the safety margin with respect to the containment failure criterion.
• The catalytic recombiners are fully passive and self-starting devices, and improve the atmosphere homogenisation. Because no feature has been identified which could worsen the situation during a severe accident, the Belgian Utilities have made the principle decision to implement catalytic recombiners in all units. This will require detailed implementation studies and final approval by the Regulatory Body [ref. 1].

**NOMENCLATURE**

- $C_p$: isobaric molar heat capacity (kJ/kmole K)
- $C_A$: recombination constant (m/s)
- $h$: specific molar enthalpy (kJ/kmole)
- $m$: mass (kg)
- $n$: number of kmoles (-)
- $m_r$: molar production rate (/s)
- $p$: pressure (Pa)
- $p_s(T)$: saturation pressure at temperature $T$ (Pa)
- $R$: molar gas constant (8.3143 kJ/kmole K)
- $S$: catalyst area (m$^2$)
- $t$: time (s)
- $T$: temperature (K)
- $u$: molar internal energy (kJ/kmole)
- $x$: gas fraction (-)
- $V$: control volume (m$^3$)
- $\Delta E$: heat of combustion at constant volume (kJ/kmole)
- $\Delta H$: heat of combustion at constant pressure (kJ/kmole)
- $\lambda$: recombination coefficient (/s)
- $\mu$: molar mass (kg)
- $\tau_{1/2}$: half life (s)

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### TABLE 1: MAIN EVENTS OF THE SEY SCENARIOS

<table>
<thead>
<tr>
<th>Event</th>
<th>Doel 3 Time(s)</th>
<th>Tihange 2 Time(s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core recovery</td>
<td>5.65</td>
<td>5.67</td>
</tr>
<tr>
<td>Start melt</td>
<td>7.73</td>
<td>7.43</td>
</tr>
<tr>
<td>Vessel breach</td>
<td>9.098</td>
<td>9.475</td>
</tr>
<tr>
<td>Start MCC1</td>
<td>19.280</td>
<td>19.739</td>
</tr>
<tr>
<td>Zr depleted</td>
<td>34.031</td>
<td>26.417</td>
</tr>
<tr>
<td>Basemat melt through</td>
<td>153.140</td>
<td>&gt; 172.800</td>
</tr>
</tbody>
</table>

### TABLE 2A: MAXIMUM VALUES - DOEL 3 SCENARIO SEY

<table>
<thead>
<tr>
<th>H2 (%)/Time(s)</th>
<th>CO (%)/Time(s)</th>
<th>H2_{eq} (%)/Time(s)</th>
<th>Burn pr. (MPa)/Time(s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>No recombination</td>
<td>20.3/153 140*</td>
<td>3.6/153 140*</td>
<td>22.5/153 140*</td>
</tr>
<tr>
<td>H2 combined</td>
<td>4.7/34 940</td>
<td>5.1/153 140*</td>
<td>6.4/43 760</td>
</tr>
<tr>
<td>H2 &amp; Co combined</td>
<td>4.7/34 940</td>
<td>2.4/39 859</td>
<td>6.1/39 859</td>
</tr>
</tbody>
</table>

### TABLE 2B: MAXIMUM VALUES - TIHANGE 2 SCENARIO SEY

<table>
<thead>
<tr>
<th>H2 (%)/Time(s)</th>
<th>CO (%)/Time(s)</th>
<th>H2_{eq} (%)/Time(s)</th>
<th>Burn pr. (MPa)/Time(s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>No recombination</td>
<td>13.6/62 099</td>
<td>10.3/75 179</td>
<td>19.7/62 099</td>
</tr>
<tr>
<td>H2 combined</td>
<td>5.3/11 224</td>
<td>12.8/75 179</td>
<td>9.3/62 159</td>
</tr>
<tr>
<td>H2 &amp; Co combined</td>
<td>5.3/11 224</td>
<td>3.5/34 199</td>
<td>5.6/32 219</td>
</tr>
</tbody>
</table>

* Basemat melt through
** Oxygen limited Combustion
Fig. 1 Hydrogen Mass

Fig. 2 Carbon Monoxide Mass
Fig. 3 Hydrogen Concentration

Fig. 4 Carbon Monoxide Concentration
Fig. 5 Equivalent Hydrogen Concentration

Fig. 6 Combustion Pressure
Fig. 11 Equivalent Hydrogen Concentration

Fig. 12 Combustion Pressure
Fig. 13 Fraction of Burning Gases

BELGATOM

TIHANGE2

SEY

Time ($10^3$ s)

Fraction

No recombines

With recombines
In the light water reactor (LWR) under severe accident conditions, the containment is the last barrier against radioactivity emission to the environment. To assure the containment integrity pressure control is of utmost importance. The containment atmosphere is typically composed of inert gases, e.g. steam (H₂O); oxidizing gases, e.g. oxygen (O₂) in air; reactive gases, e.g. hydrogen (H₂). Depending on composition, this gas mixture may be subject to uncontrolled combustion/explosion processes with accompanying pressure (P) / temperature (T) increases. One of the methods to exclude containment atmosphere ignition in a deterministic way is to flood the containment in the early stage of a severe accident by an inert gas, e.g. carbon dioxide. Therefore the purpose of this work was to determine the ignition limits of the 4-component gas mixture CO₂, steam, air, H₂ under conditions as they may prevail in the containment in the initial phase of a severe accident: P = 1.6 - 5.0 bar, T = 35 - 160°C.

The ignition tests were carried out in an autoclave of a large, semi-technical volume, equipped with a fan to assure instantaneous gas mixing and a high energy source. Starting with a base load of 1 bar air at 35°C, H₂, CO₂ and H₂O were added and the ignition limits, the pressure peak and the residual gas composition determined.
Introduction

In the light water reactor (LWR) core damage is initiated by an insufficient cooling of the reactor core. Above 1100°C the Zircalloy fuel rod cladding and above 1370°C the steel components react with water. These reactions lead to the formation of hydrogen (H₂) and are strongly exothermic. The liberation of hydrogen into the containment atmosphere leads to a pressure increase and eventually, without preventive measures, to the formation of an ignitable gas mixture, which may be subject to uncontrolled slow combustion, deflagration or detonation with an accompanying pressure (P) and temperature (T) peak, jeopardizing the containment integrity. One of the methods to exclude containment atmosphere ignition in a deterministic way is to flood the containment in the early stage of a severe accident by an inert gas, e.g. carbon dioxide (CO₂). Therefore the purpose of this work was to determine the ignition limits of the 4-component gas mixture hydrogen, air, steam, CO₂ under conditions typical for the early stage of a severe accident (in-vessel reactions, before reactor pressure vessel failure).

Published data e.g. [1] for a gas mixture consisting of a) reactive (H₂), b) oxidizing (air), c) inertizing (steam) components allow a rough estimate of the ignition limits at typical P (1.6 - 5.0 bar) and T (35 - 160°C) conditions:

- Reactive gas (H₂) > 5 % H₂
- Oxidizing agent (O₂ in air) > 22 % air; > 4.6 % O₂
- Inertization (H₂O + CO₂) < 60 % H₂O + CO₂

The purpose of this experimental work was to answer the following questions: a) ignition yes/no, - ignition limits of gas composition; b) gas composition before/after, - completeness of combustion; c) Pressure recording, - peak load during combustion.

1. Experimental Set-up

Figure 1 shows the experimental set-up. In order to avoid effects of geometry, a horizontal cylinder autoclave of 350 mm inner diameter and 350 mm inner length with a volume of 34 litres, made of austenitic steel, was selected. The heated autoclave was equipped with a magnetically coupled fan to assure homogeneous gas mixing before the test and turbulence during the test. On the cylinder belt line 6 ignition sources were installed at a distance of 30°. The ignition sources extended to the autoclave center to avoid effects of position. Nickelin (Nickel-copper alloy) exploding wire was used as a high energy ignition source, - setting free 4 Joules.

The temperature in the autoclave center was recorded by a thermocouple (1 mm steel sleeve, 50 ms response time). The pressure was recorded by a piezoelectric transducer mounted in the cylinder wall (1 ms response time). From the strip chart of the fast recorder the actual pressure peak and the apparent temperature peak during ignition were obtained. The gas composition before and after ignition was analyzed by mass spectrometry (MS). Since the MS worked with reduced gas pressure, the gas sample pressure was adjusted in an evacuated expansion vessel to 0.3 bar.
The tests in an autoclave with a large, semi-technical volume had the purpose to simulate the containment atmosphere at the initial phase of a severe accident without containment venting. Therefore the autoclave always contained 1 bar air at 35°C or the corresponding higher pressures at elevated temperatures. H₂, CO₂ and steam were added to the base load of air leading to varying initial pressures P₀. All pressures are indicated in absolute values (bar = bar absolute) at temperature.

The tests followed a standard procedure:

- The gases were filled into the heated, evacuated autoclave in the following sequence: Air, hydrogen, CO₂
- After a period of equilibration, this gas mixture was ignited by the first ignition source.
- When no ignition occurred, more hydrogen was added and the second ignition source activated.
- For tests with water in the gas mixture, water was injected into the bottom of the autoclave. After reaching equilibrium pressure, the other gases were introduced.

2. Test Data

2.1 Dry gas mixtures

The results of the ignition tests of dry gas mixtures (H₂, air, CO₂) are summarized in the following figures:

- Figure 2: T 30 - 40°C; Pₐir 1.0 - 1.2 bar; P₀ 1.6 - 4.4 bar
- Figure 3: T 95 - 105°C; Pₐir 1.1 - 1.3 bar; P₀ 2.1 - 3.9 bar
- Figure 4: T 155 - 165°C; Pₐir 1.4 - 1.6 bar; P₀ 3.4 - 5.0 bar

Each figure is divided into 3 sections:

- Section a: ignition yes/no - ignition limits as a function of the O₂(air)-concentration and the H₂-concentration in the gas mixture before ignition. The discrimination between Pₘₐₓ / P₀ above or below 2.0 gives an indication of the reaction propagation.
- Section b: Gas composition after ignition; the residual O₂ (air)-concentration and H₂-concentration after ignition shows the completeness of the reaction.
- Section c: The ratio Pₘₐₓ / P₀ shows the peak pressure load. P₀ is the initial pressure of the gas mixture at the test temperature indicated. Pₘₐₓ is the maximum pressure as recorded after ignition.
2.2 Wet gas mixtures

The results of the ignition tests of wet gas mixtures (H₂, air, CO₂, H₂O) are summarized in the following figures:

- **Figure 5**: 100°C H₂O-saturated gas mixture at T₀ = 100°C
  
  T 98 - 105°C; P_{air} 1.1 - 1.3 bar; P_{H₂O} 0.9 - 1.0 bar; P₀ 2.3 - 4.6 bar

- **Figure 6**: 120°C H₂O-saturated gas mixture at T₀ = 160°C
  
  T 155 - 165°C; P_{air} 1.4 - 1.6 bar; P_{H₂O} 1.8 - 2.0 bar; P₀ 3.3 - 4.6 bar

At 160°C the H₂O-saturation is approximately 6 bar; with 1.4 bar air in the gas mixture no ignition can be reached. Therefore the H₂O-concentration was fixed at 2.0 bar (32%) corresponding to a saturation temperature of 120°C. Because of this high H₂O-concentration the determination of the gas composition after ignition, - section b of figure 6 -, was not possible.

3. Results and Discussion

3.1 Ignition limits

**Figure 7** and **Table 1** summarize the ignition limits between 35°C and 160°C for dry and wet gas mixtures. Conservatively, the ignition limits were based on experiments where the H₂- or O₂-concentration were highest and just no ignition occurred.

For the dry gas mixture the data were fitted with the expression:

\[
[O₂] = a - b \log [H₂]
\]

For the wet gas mixture the following expression was used:

\[
[O₂] = \frac{a[H₂]}{b + [H₂]} + \frac{c[H₂]}{d + [H₂]}
\]
Table 1: Summary of ignition limits between 35°C and 160°C

- Dry gas mixture (H₂, air, CO₂); \( P_0 \) 1.6 - 5.0 bar
- Wet gas mixture (H₂, air, CO₂, H₂O); \( P_0 \) 2.3 - 4.6 bar

<table>
<thead>
<tr>
<th></th>
<th>35°C dry</th>
<th>35°C wet</th>
<th>100°C dry</th>
<th>100°C wet</th>
<th>160°C dry</th>
<th>160°C wet</th>
</tr>
</thead>
<tbody>
<tr>
<td>H₂-lower limit (%)</td>
<td>5.1</td>
<td>--</td>
<td>4.7</td>
<td>5.3</td>
<td>4.5</td>
<td>6.3</td>
</tr>
<tr>
<td>(excess oxygen)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>O₂-lower limit (%)</td>
<td>6.8</td>
<td>--</td>
<td>6.5</td>
<td>6.7</td>
<td>6.4</td>
<td>6.7</td>
</tr>
<tr>
<td>(excess hydrogen)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Table 1 allows the following conclusions with respect to the lower H₂- and O₂-ignition limits:

- At higher temperatures the ignition limits are shifted to lower values.
- Wet gas mixtures show higher ignition limits than dry gas mixtures; steam inertization.
- The lowest ignition limits were measured for dry gas mixtures at 160°C.
- The pressure dependence of the ignition limits was not investigated. It should be of interest if controlled containment venting is considered as a strategy. The indications are that the initial pressure has little influence on the ignition limits [2].

To exclude deterministically ignition of the containment atmosphere under all conditions prevailing during the initial phase of a severe accident, the CO₂-inertization should depress the O₂-concentration to a value below 6.4 % (30.5 % air). This experimental value is about 1/3 higher than initially estimated (> 4.6 % O₂) and means less CO₂-storage and less pressure buildup in the containment during inertization.

In Figures 2 - 6, section a, an arbitrary distinction between ignition with low peak load (\( P_{\text{max}} / P_0 < 2.0 \)) and ignition with high peak load (\( P_{\text{max}} / P_0 > 2.0 \)) is made. Ignition with low peak load was only measured at low H₂-concentrations. Further, low and high peak load points cannot be separated and may occur at apparently the same experimental conditions. This means that partial inertization to ignition with low peak load (mild combustion) is not practical [3].

3.2 Pressure buildup

Figures 2 - 6, section c, show the peak pressure during ignition expressed by the ratio \( P_{\text{max}} / P_0 \). Starting at low H₂-concentrations \( P_{\text{max}} / P_0 \) increases until the H₂- / O₂-concentration reaches stochiometric composition at about 14 % H₂ and 7 % O₂. Highest \( P_{\text{max}} / P_0 \)-values of 4 - 5 were measured for dry gas mixtures at low temperature (35°C). At higher temperatures, dry and wet gas mixtures reach a \( P_{\text{max}} / P_0 \)-value of about 3.
Together with the pressure peak the measured temperature of the gas mixture during ignition increases. This temperature increase however may be an artefact due to the measuring technique with the exception of mild combustion, the real flame temperature at deflagration is assumed to be constant.

3.3 Completeness of combustion

Figures 2 - 6, section b, show the residual gas concentration after ignition is shown. Despite the "ideal, controlled" experimental conditions, the H₂- / O₂-concentrations at apparently the same conditions, varies strongly:

- Starting with low H₂-concentrations between 5 and 12 %, after ignition a residual H₂-concentration of 1 to 9 % and an O₂-concentration of 13 to 6 % remains.

- Near the stochiometric H₂- / O₂-gas composition of 12 to 18 % H₂, after ignition the gases are nearly totally consumed. The residual H₂-concentration is 0 to 3 % and the O₂-concentration 0 to 2 %.

- At excess H₂-concentration above 18 % H₂, after ignition the O₂ is fully consumed and the overstoichiometric H₂ remains.

Conservatively, for low H₂-concentrations between 5 and 12 %, it has to be assumed, that after ignition only a few % (3 - 4 % H₂) are consumed and that the H₂-concentration remains high. The addition of a few % H₂ may lead again to an ignitable gas mixture.

Summary and Conclusions

The CO₂-inertization of the containment atmosphere was investigated in an autoclave of a large, semi-technical volume, equipped with a fan to assure instantaneous gas mixing and a high energy ignition source. Starting with a base load of 1 bar air at 35°C, H₂, CO₂ and H₂O were added. For realistic conditions, as they may prevail in the containment during the initial phase of a severe accident, P₀ = 1.6 - 5.0 bar, T₀ = 35 - 160°C, the ignition limits, the pressure peak and the residual gas composition were determined. The following conclusions were drawn:

1) For dry gas mixtures (H₂, air, CO₂) the maximum allowable O₂-concentration, where just no ignition occurred, was measured at 160°C / 5.0 bar: 6.4 % O₂; for wet gas mixtures (H₂, air, CO₂, H₂O) at 160°C / 4.6 bar: 6.7 % O₂. Therefore CO₂-inertization to 6.4 % O₂ should prevent ignition under all conditions.

2) The pressure peaks after ignition were recorded. The highest P_max / P₀-ratio of 4 - 5 was measured for dry gas mixtures at low temperatures.

3) Since the pressure peaks at apparently the same experimental conditions showed extreme variations, partial inertization to mild combustion is not considered a practical option.
4) At low H₂-concentrations between 5 and 12 % only a few % H₂ may be consumed after ignition. Since the residual H₂-concentration remains high, the addition of a few % H₂ may lead again to an ignitable gas mixture.

Acknowledgement

The authors express their gratitude to Dr. B. Covelli (TECOVA AG, CH-5610 Wohlen) who acted as a consultant for this work, to the Swiss Federal Nuclear Safety Inspectorate (HSK) for the assistance in technical questions and for the financial support and to the Swiss Association for Pressure Vessel Control (SVDB) for the use of their facilities in Dübendorf, CH-8600, Switzerland.

References


FIGURE 1: Experimental set-up
FIGURE 2: Dry gas mixture; Temperature 35 °C, 1.0-1.2 bar air, CO₂, H₂; P₀ 1.6-4.4 bar

a. ■ Pₘₙₐ/P₀ > 2.0; ○ Pₘₙₐ/P₀ < 2.0; X No ignition

b. After ignition ◇ H₂-concentration; ● O₂-concentration

c. ▽ Pₘₙₐ/P₀
FIGURE 3: Dry gas mixture; Temperature 100°C, 1.1-1.3 bar air, CO₂, H₂; P₀ 2.1-3.9 bar
Symbols as in Fig. 2
FIGURE 4: Dry gas mixture; Temperature 160°C
1.4-1.6 bar air, CO₂, H₂; P₀ 3.4-5.0 bar
Symbols as in Fig. 2
FIGURE 5: Wet gas mixture; Temperature 100°C,
1.1-1.3 bar air, 0.9-1.0 bar H₂O, CO₂, H₂; P₀ 2.3-4.6 bar
Symbols as in Fig. 2
FIGURE 6: Wet gas mixture; Temperature 160°C
1.4-1.6 bar air, 1.8-2.0 bar H₂O, CO₂, H₂; P₀ 3.3-4.6 bar
Symbols as in Fig. 2
Dry gas mixture: \( \text{H}_2, \text{air, CO}_2; \)  
Temperature 35-160°C; \( P_0 \) 1.6-5.0 bar

Wet gas mixture: \( \text{H}_2, \text{air, CO}_2, \text{H}_2\text{O}; \)  
Temperature 100-160°C; \( P_0 \) 2.3-4.6 bar
'The selection of a Hydrogen Mitigation Technique for a PWR Large Dry Containment Using a Mathematical Decision Methodology'

by

P. Bockholts of TNO Environmental and Energy Technology
W.P.M. Mercx of TNO Prins Maurits Laboratory
G.L.C.M. Vayssier of the Netherlands Nuclear Safety Inspectorate

to be presented at the

Specialists Meeting on
Selected Containment Severe Accident Management Strategies
13 - 15 June 1994, Stockholm - Sweden
Summary

In the Borssele Nuclear Power Plant a hydrogen management strategy for postulated severe accident during which hydrogen will be produced, has to be chosen. Several mitigation techniques are available for this purpose. These alternatives have very differing characteristics and are subject to strong opposing opinions among experts. A choice for the best alternative for the Borssele Nuclear Power Plant is therefore very complicated and is subject to discussion in the task group ‘Hydrogen Management’ of the Advisory Committee on Reactor Safety (ACRS) from the ‘Dutch Nuclear Safety Inspectorate’. The TNO department of Industrial Safety has been asked to investigate this problem by using a decision analysis approach.

The decision analysis approach is an approach to handle decision problems in an objective way in which pros and cons from various alternative solutions can be compared very effective. Since hydrogen mitigation is a very complex issue it was evitable to let several experts participate in the analysis. Workshops are very effective for such participation. Two workshops were scheduled in the analysis for this reason. The first workshop dealt with an inventory of the present knowledge and views of the problem, and resulted in scores on given criteria from the defined alternatives. The second workshop will put attention on weightfactors and valuefunctions for the criteria.

The first workshop has taken place, the second is scheduled for autumn this year.

The decision analysis approach consists of nine steps
The preparation of workshop 1 the first four steps have been dealt with: the problem has been identified and defined, alternatives and criteria have been defined and value ranges have been defined for each criterium. The workshop itself dealt with step five: the evaluation of alternatives.

A workshop attended by many experts and with the chosen set-up, has proven to be a successful and effective method for the collection of a great amount of controversial information. The use of a list of selected criteria helps with structuring this information. Afterwards the score results have been further analyzed. This analysis allows some first conclusions regarding the alternatives. The results show which alternatives are generally accepted and on which alternatives disagreement exists. In general the score results show that some consensus has been reached during the workshop.

The results of the workshop form a solid basis for the remaining four steps in the decision process: the valuation of the effect scores, the weighing of criteria, the comparison of the alternatives and the final sensitivity analysis. These steps are carried out in a follow-up project, involving a second workshop. This second workshop will conclude the decision analysis approach and will provide an advice on the choice to be made for the best available mitigation technique for the Borssele Nuclear Power Plant based on todays knowledge and careful evaluated uncertainties.
1 Introduction

The selection of a hydrogen management strategy for the Borssele Nuclear Power Plant in the Netherlands is subject to discussion in the task group 'Hydrogen Management' of the Advisory Committee on Reactor Safety (ACRS) from the 'Dutch Nuclear Safety Inspectorate'. It has been decided by the task group to approach the subject as a decision problem and to select an investigation method accordingly. The Industrial Safety Department from TNO proposed an approach with workshops in order to obtain the state of the art information and understanding of opposing opinions in a very short time. A description of the Borssele Plant is given in appendix 1.
2 Objectives

The objective of the project is to investigate the alternative mitigation techniques for a postulated severe accident in the Borssele type nuclear power plant in the case that hydrogen will be produced. The approach with workshops has been chosen, for this is an opportunity to bring experts together with knowledge of the various mitigation techniques. Through this it was expected to have detailed information on hand of these techniques and to be able to compare them with each other. The comparison is made by using a list of criteria.

The objective of the first workshop was to obtain a detailed understanding of pros and cons. It was and is by no means the intention to convince any of the participants or to change their opinions. It is also not the intention to use the workshop as a master class. All participants were masters themselves.

The participants in the first workshop were: representatives from mitigation techniques (position experts), representatives from reactor safety (panel experts) and representatives from nuclear technology (discussion experts). If the consensus of the scores turns out to be insufficient then more emphasis will be put on the opinion of the panel. The workshop was prepared and chaired by TNO in assignment by and cooperation with the Dutch Nuclear Safety Inspectorate.

The objective of the second workshop is to evaluate the results of the decision analyses. The sensitivity analysis, being the last part, will be of particular importance in this evaluation. The participants in the second workshop are the advisers from the Dutch Nuclear Safety Inspectorate.
3 Decision analysis approach

Essential for a decision analysis is a formal approach in nine steps. The workshop covered step five in this sequence which means that four steps proceeded and another four will follow. The understanding of the workshop's process requires an explication of all the steps. A short explication is given of the nine step approach.

Step 1 Identification of the problem for which the decision analysis approach is chosen
The subject is the hydrogen mitigation technique during severe accidents. The severe accident management with well defined accident scenario's form the boundaries of the problem.

Step 2 Define alternatives
A set of ten alternative mitigation techniques is defined and the solution will be one of these. These ten alternatives are give in appendix 2.

Step 3 Appoint criteria
The selection of the best alternative depends on a facility to compare the alternatives. The set of criteria is the tool to distinguish the alternatives from each other. The criteria therefore are most crucial in the decision analysis. The development of the set has been carried out with consultation of a number of experts. The criteria are given in appendix 3.

Step 4 Define value ranges for the criteria
Distinguishing of alternatives for a given criterium requires a range of values. These values may be qualitative (e.g. minor, moderate, major) and quantitative (e.g. 1, 2, 5, 10).

Step 5 Evaluation of alternatives (workshop 1)
The mechanism of giving an effect score for each alternative on each criterium, permits a comparison of alternatives. The workshop actually dealt with this step. All participants gave their scores after they got well prepared and informed about all the pro's and con's of each alternative.

Step 6 Valuation of effect scores
A value function is the relation between the effect scores on a criterium and the degree of appreciation of this effect score. The appreciation is (arbitrarily) measured on a scale between 0 and 1 or 0 and 100. The value function can be created in advance by valuing hypothetical effect scores between the minimum and maximum of the effect score scale. In constructing a value function, a reasonable first estimate is to consider the function linear in the relevant domain of the effect score, i.e. between the worst and best acceptable level. However, this initial shape can be adapted to meet the judgements of a decision-maker.

Step 7 Appoint weighting factors to criteria
Weighting factors are used to appoint the relative importance of the criteria. Criteria are seldom equally important. Weighting factors represent in percentages the contribution of each criterium to a total score of each alternative. This step will be subject of a second workshop. Judgement from a number of experts will be used to determine the weighting factors. The process for this determination will be the same as for the determination of the scores on the criteria. (Step 5.)

Step 8 Compare alternatives
A mathematical calculation can now be performed to calculate the total score of each alternative as a result of the score on individual criteria and the weighting factors of the criteria. The alternatives can then be ranked accordingly.
Step 9  Analyze ranking sensitivity (workshop 2)
The decision analysis will be completed with a sensitivity analysis. This analysis will
give the understanding of the dominating criteria, values and weighting factors. It also
gives inside in the stability of the best alternative towards alterations of scores and
weighting factors.
The process of a decision analysis in these nine steps is rather formal and also evident. Decision making includes these steps practically always. However it may be emphasized have that the steps 1 to 3 may look very tracial but are the most difficult ones, and therefore sufficient attention should be given.

In particular step 3, the appointment of criteria requires great care, because the criteria distinguish the alternative solutions.

TNO developed the DATUM software for all the mathematical part of the analysis. DATUM can handle an arbitrary number of alternatives, any hierarchical structure of criteria, any shape of value function and uses a graphical interface for the sensitivity analysis.
4 Workshop 1

4.1 Preparations

The workshop dominated this project of the first part of decision analysis approach. The preparation of the workshop, however, consisted of the first necessary steps of the decisions analysis method (step 1, 2 and 3).

The main activities were to define a set of alternative mitigation techniques and to develop a set of criteria that could be used for a clean and clear comparison of the alternatives. The hydrogen mitigation is subject to strong opposing opinions among the experts. However, a thorough investigation required to consider all the different positions. An accurate preparation and a careful approach was needed.

TNO prepared a manual for the workshop participants containing the objectives of the workshop, a description of the used decision analysis, a description of the Borssele Power Plant, a description of the mitigation techniques and the draft list of criteria that was developed for this particular purpose.

A number of participants was selected and invited to represent all the alternatives and they were asked to prepare a position paper according to the criteria list. These position experts have taken their obligation very consciously and produced valuable documents. These documents have been distributed to all participants in order to give everyone ample opportunity to prepare him/herself. Two other groups of participants were invited. One group was selected and invited to intensify the discussions and to oppose the positioners. The other group was selected and invited as panel members, because they were considered as being independent from sensitive situation. The members of the task group 'Hydrogen Management' of the ACRS were invited as observers.

All invited experts joined the workshop with the only exception of a representative of the Swiss Federal Nuclear Safety Inspectorate. One of the other members presented the Swiss position. The selection and invitation of experts was done by the Dutch Nuclear Safety Inspectorate.

4.2 Workshop structure

The program of the workshop consisted of three main parts.

The first part was dedicated to presentations of the position papers and round-table discussions on each of the alternatives.

The second part was used for giving scores to each criterium of each alternative. This task was executed by twenty-two participants. A multiple choice datasheet was prepared for this task consisting of a table with the alternatives along one axis and the criteria along the other. The scores of each criterium of each alternative could be given with a letter A, B, C or D. The multiple choice datasheets were collected and processed.

The third part of the workshop was dedicated to a panel discussion based on the results of the scores. Although a detailed analysis of the scores was not feasible during the workshop, it was possible to identify a number of scores that required further and more detailed discussion. The workshop was concluded with a general feeling that everything was brought on the table that should be to considered and is important in this major decision problem. The workshop was successful in this respect and fulfilled its goals.
4.3 Workshop Performance

As indicated in the decision analysis description the workshop dealt with step 6, the valuation of all alternatives on each criterium by giving scores. The program of the workshop consisted basically of three parts. Part one and two covered the first day and part three covered the second day. The workshop terminated after the lunch on the second day. The first part was used to review all the alternatives. The second part as used to give the scores of all alternatives on the given criteria. The third part was used to have a preliminary discussion on the results of the scores and to draw preliminary conclusions.

The attention in the first part was focused on the understanding of the different opinions. The formulae that was chosen for this part was to let experts explain their position and restrict the discussion to clarification only. No room was given to discussion on disagreements. These discussions restricted to clarification gave all participants ample room for further explication, insight in uncertainties and expectation from further research. All participants were briefed about this formulae in advance in the workshop manual.

The adjacent part two of the workshop that was used for giving scores took place at a time that all the participants were informed on all the relevant details. Each participant filled out the multiple choice matrix (10 alternatives x 27 criteria). An example of an empty score document with this matrix is given in Appendix 4. Part two concluded the workshop program of the first day. TNO summarized the individual scores for presentation on the second day during the evening.

The program on the second day covered the third part of the workshop being a panel discussion on the alternatives that showed the most controversial scores. This discussion required additional clarification from the position experts. All participants were invited during this panel discussion to reconsider the scores that they gave. This resulted in only a few changes. The panel discussion therefore did not influence the scores significant.

A final table-round showed that all participants were satisfied with what had been said and they agreed that no relevant information was left unattended. The workshop ended after one and a half day of hard working.
5 Results of workshop 1

The main goal of the workshop was to get insight in advantages and disadvantages of the ten proposed mitigation techniques which are available for a severe accident in the Borssele nuclear power plant in case hydrogen will be produced. This can be the basis for a well-founded decision. The weighing of these advantages and disadvantages was not part of this workshop. This will be part of a second workshop in which Dutch policy-oriented people will participate. During the workshop it was therefore not necessary to decide on or choose the best alternative. This choice is up to the owner of the Borssele power plant and has to be accepted by the Dutch Nuclear Safety Inspectorate. The second workshop will provide an advice in this matter.

A number of position experts was asked to present a position-paper on one or some of the alternatives. The presentations were all well prepared and gave a good impression of facts and opinions regarding the presented alternatives. Especially the discussion after the presentations was valuable in making a distinction between facts on the state of the art and expectations and opinions on future developments.

The discussions were well structured and showed a great discipline of all the participants. The critical reviews and questions of the participants also showed that an appropriate selection of participants had been made. The selected criteria appeared to be helpful in structuring the discussion. The criteria covered a wider range than only the aspects regarding effectivity but also practical aspects such as managing the technique. With the help of these criteria a complete picture of all topics which play a role in a decision process on hydrogen mitigation techniques could be obtained.

After all position-papers had been presented the participants were asked to fill in a score-document. In this score-document a score had to be given on each criterium for every alternative. An overview of the score result formed the basis for the discussion on the second day. This discussion and a further analysis of the score results allow some further conclusions.

In the discussion everyone agreed that recombiners formed an alternative with little negative side-effects. There is also a general confidence in the state of technology. With respect to these aspects this was supported by the score results. During the discussion it was also concluded that recombiners would not form a very effective alternative as a stand alone technique. The score results of recombiners on the effectivity criteria (criterium 1.1a) are not very high. For the effectivity on the long term however, catalytic recombiners gave the best score.

Deliberate ignition was subject of a lot of discussion. This discussion concentrated mainly on the experiments which have been performed with igniters. Differences of opinion and lack of confidence in the applicability of the experiments were also represented in the score results. The distribution of the scores was rather wide. The score results of deliberate ignition also influenced the scores on the combined alternatives which included deliberate ignition. The influence on the scores by deliberate ignition was much higher than the influence on the scores by catalytic recombiners. This is consistent with the general conclusion that catalytic recombiners have little or no negative side-effects.

Permanent inertisation was considered to be very effective (most effective alternative). The confidence in the state of technology was also very high (best scores). This alternative has however some negative consequences on other aspects such as operating characteristics and compatibility with the existing installation.

There were some doubts regarding the effectivity of post-inertisation. This will need some further research. The scores on the alternatives including post-inertisation show a strong correlation. The
post-inertisation with early venting alternative caused some discussion. A German participant stated that this alternative didn’t need any consideration since early venting was forbidden in Germany. In the Dutch situation this is not the case, so this consideration should not play a role in the discussion on this alternative. Analyzing the score results however, shows that a large group of German participants gave significantly lower scores on this alternative. Post-accident dilution is considered to be the least effective alternative. It scored relatively bad on other criteria as well.

The score results showed that a proper selection of the criteria had been made, in so far that the scores on the criteria make a distinction between alternatives possible: no criteria give exactly the same score for the alternatives. In distinction between accident scenarios for criterium 1.1 (expected effectivity of risk reduction) and criterium 1.2 (negative consequences on risk reduction) is however not necessary. For the first three scenarios the scores are more or less the same. A distinction in three different scenarios provides therefore no additional information. With regard to negative consequences the fourth scenario also gives the same scores as the other scenario’s. The expected effectivity with this scenario is very low for all alternatives and gives therefore no additional information for a comparison between alternatives.

In the follow-up of this project weighting factors for the criteria will be chosen. Analyzing these weighting factors will also supply information on the value of a criteria. If the ranking of the alternatives does not change if the weighting factor of a criterium is changed (ranging from zero to one), it means that this criterium does not provide a distinction between the alternatives.

The score results of participants were divided in three categories: scores of panel members, discussion participants and of position experts. Since the position experts presented their view on one or more of the alternatives it was expected that they would have more extreme opinions and they would have more differences of opinions than the more neutral panel and discussion members. The score results supported this expectation on several points. The distribution of the score results for this group was more wide than the distribution of the scores for the other groups. As a group they also gave slightly more negative scores. As representative of one or more alternatives they gave significantly higher scores.

Apart from these differences, the analysis of the score results shows that almost all participants more or less agree with each other. The scores from only four participants (two position experts and two discussion participants) deviated clearly from the other scores. This means that there is a certain consensus among the participants. This increases the value of the (score) results of the workshop.

With this workshop and the analysis of the score results on the alternatives the first five steps of the decision analysis approach have been completed. In the preparation of the workshop the first four steps have been dealt with: the problem has been identified, the boundaries of the problem have been set, alternatives and criteria have been defined and value ranges have been defined for each criterium. The workshop itself dealt with step five: the evaluation of alternatives. A workshop attended by this many experts and with the chosen set-up, has proven to be a successful and effective method for the collection of a great amount of controversial information. The use of a list of selected criteria helps with structuring this information.

The results of the workshop form a solid basis for the last four steps in the decision process: the valuation of effect scores, the weighing of criteria, the comparison of the alternatives and the final sensitivity analysis. These steps will be carried out in a follow-up project, involving the already mentioned second workshop.
6 Workshop 2

6.1 Preparations

The second workshop dominate the second part of the decision analysis approach. The preparations here are again an essential part (Step 6 and 7).

The main activities were to prepare a document to help define the weight factors of the criteria. Since the criteria were defined in two levels viz: primary criteria (1 to 5) and secondary criteria (1.1 to 1.3 etc.) these weight factors were also given on the two levels. A form was chosen to facilitate the participants in appointing these weight factors.

Similar to workshop 1, a number of experts were selected and invited to participate in the second workshop.

These participants were invited to appoint the weight factors and TNO processed the data accordingly.

The final preparative task was to select value function for each of the criteria. These value functions indicate the effect of a score. Eight different shapes of curves can be used to describe the relations between low, medium and high. For each criterium one of the available eight curves was selected.

With all this done it is possible to make the calculations and compare the alternatives. The result is a ranking of alternatives and detailed information on the influence of selected scores, weight factors and value functions.

The presentation of these results, the discussion on it and an extensive sensitivity analysis is the subject of workshop 2.

6.2 Parallel research

Mainly as results from workshop 1 is was felt necessary to put attention to four issues before an appropriate evaluation of mitigation techniques could be carried out. These issues are:

- more insight in the expectations of the ongoing combustion experiments and the confidence in the absence of large negative effects with the use of deliberate ignition.
- more insight in the expectations of post inertisation and the consequences of installation.
- more insight in early venting. This alternative is rather new.
- summarization of calculations of pressure increase and hydrogen production as functions of time for the postulated severe accidents.

6.3 Workshop structure

The programme of the workshop consists of three main parts. The first part includes an explanation of what was done in the preparation. The second part refers to a presentation of the results of the calculations leading to the ranking of the alternative mitigation techniques and the third part deals with the sensitivity analysis.
6.4 Workshop performance

Workshop 2 deals the final step of the decision analysis. It will lead to a ranking order of the ten alternative mitigation techniques and shows how this ranking is constructed. This insight is most important because it explains which criteria and weight factors are responsible for the final result. If the differences in the ranking are large then the conclusions for an alternative are evident. If the differences are little then a sensitivity analysis shows how scores and weight factors must be altered to change the ranking order. This insight is extremely helpful in getting conclusions.
7 Conclusions

Although the project has not been finished yet some conclusions may be drawn. The selection of a hydrogen mitigation technique is subject of many controversial aspects. The scenario's of severe accidents are being discussed, experiments show very scattered results, opinions of experts vary and legislation may exclude certain alternatives. The wish of the Dutch Nuclear Safety Inspectorate to study this problem and find a feasible path for the safety strategy in the Borssele Power Plant resulted in an approach making use of decision analysis techniques. This technique is rather simple in itself. It is the way it is used which makes the results applicable. The stepwise approach is therefore used consequently. The development of a set of criteria in step 3 was carried with greatest care, and relatively much time was given to this item. The workshop 1 was a very effective way to collect the state of the art on mitigation techniques and to have all the controversial views around the same table. This workshop gave the Dutch Nuclear Safety Inspectorate essential information for her further policy preparation. Four intermediate items are now under investigation as a result of this workshop. These are:
- estimation of expectation from ongoing research on combustion of hydrogen;
- insight in details of post inertisation;
- insight in details of early venting;
- summarization of calculations on pressure increase and hydrogen production for the postulated severe accidents.

In the mean time experts have been selected to participate in the second workshop and activities are going on with step 6 and 7. The workshop 2 is scheduled for autumn 1994 and shall lead to an alternative that is considered to be the best.
Appendix 1

Description of the Borssele Nuclear Power Plant

The Borssele NPP is a two-loop, 480 MWE (1365 MWth) Pressurized Water Reactor of the Kraftwerk Union design, and is in commercial operation since 1973.

Core heat removal system

The core consists of 121 fuel elements in a 15 x 15 array of fuel pins with a heated length of 2.65 m.
Under nominal conditions the primary system pressure is 15.6 MPa and the total core mass flow is 10,000 kg/s. Two U-tube steam generators remove the heat from the primary system. The total steam production is 735 kg/s steam at a pressure of 5.7 MPa. Plant schemes are given in figure 1, 2 and 3 on the pages 7, 8 and 9.

Table 1 gives some important system parameters at nominal conditions of the Borssele NPP.

<table>
<thead>
<tr>
<th>Table 1 Nominal operating conditions in the Borssele NPP</th>
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<tbody>
<tr>
<td>Thermal nuclear power</td>
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<tr>
<td>Primary system pressure</td>
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<tr>
<td>Total cor mass flow</td>
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<tr>
<td>Core inlet temperature</td>
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<tr>
<td>Number of fuel elements</td>
</tr>
<tr>
<td>Core average heat flux</td>
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<tr>
<td>Steam generator outlet pressure</td>
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<tr>
<td>Steam mass flow</td>
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</table>

The ECC-system of the plant was licensed under the USNRC criteria of that time. The ECC-system consists of four accumulators, four high pressure injection pumps and four low pressure injection pumps, connected to both the hot legs and the cold legs. It is envisaged to separate the components into two independent redundancies.
An independent, bunkered decay heat removal system has been added, to cope with area events and other challenges to the normal decay heat removal.

Containment layout

The Borssele NPP has a primary and a secondary containment. The secondary containment is capable of collecting leakages from the primary and release these through appropriate filters. It functions also as a shield building.

Primary containment

The primary containment is a sphere with a steel shell. The failure pressure is calculated to be about 0.9 MPa.
The primary system is housed in concrete compartments. Explosion hatches take care of connections between the different compartments in the case of a LOCA. The strength of the compartment wall
is designed to withstand large break LOCA. They have not been designed to withstand loads from explosions, as may arise from the hydrogen generated in a severe accident. Their design pressure is in the order of magnitude of 15 kPa. Their failure pressure is at present under investigation.

The reactor vessel compartment is a narrow enclosure of the reactor vessel with thick and strong walls. The floor surface is small, which has consequences for molten core-concrete interaction (MCCI). The concrete contains a considerable amount of CaCO₃.

The total volume of the compartments in which the primary system is housed amounts to 5,000 m³ and the volume of the surrounding space to approximately 32,000 m³. A containment spray system and a thermal hydrogen recombiner are available. The spent fuel pool is located inside the containment.

Secondary containment

The secondary containment/shield building is a cylinder with a spherical dome. The wall is of concrete. The space between the two spheres is 1.5 m. In the cylindrical part auxiliary and safety related equipment is installed.

Aspects of severe accidents and accident management

After the Tsjernobyl event it was decided that measures should be taken to mitigate the consequences of core melt accidents in Dutch NPP’s, where feasible. Several measures have been proposed and specified. Others however are still under investigation.

Containment venting

Severe-accident analyses have shown that a core melt, combined with MCCI, will lead to a gradual pressure increase. After approximately 5 days the shell failure pressure might be exceeded. Therefore, it was decided to install a filtered vent system. The vent system should also be activated if a containment bottom melt-through is expected to occur.

Primary system depressurization

If an accident occurs that is outside the design basis of the plant, it is generally beneficial to depressurize the primary system. This enables special cooling modes, such as 'bleed & feed'. It prevents also that, during a core melt accident, an eventual vessel melt-through will occur at high primary pressure, which may have unpredictable consequences (direct containment heating, vessel lift-off). The primary safety and relief valves are replaced in Borssele for this purpose. The new valves are qualified to transport steam, two-phase mixture, water and non-condensables. Elevated temperatures are considered for the design of their sub-components.

Mitigation of hydrogen

A total oxidation of cladding material will generate approximately 412 kg of hydrogen. This amount may lead to conditions in which the explosion limits are exceeded. The maximum amount, the rate and timing of the production depend on event sequence and eventual operator actions. For instance, an interrupted and restarted feed and bleed sequence may lead extra hydrogen spikes, which are not yet predictable. The timing and character of the hydrogen sources are comparable with those from the German Risk Study, Phase B. Of course, Borssele is a smaller plant than Biblis, hence the total amount of hydrogen generated is smaller. In addition, the Borssele MCCI (molten core - concrete...
interaction, after vessel melt-through) is expected to take place in a dry cavity, since water ingress from the sump is not expected to occur, due to the thick walls surrounding the vessel.

An additional problem is that the primary containment is compartmentalized, as indicated above, and, hence, inhomogeneous distributions may arise. The components inside the compartments act as obstacles so that an eventual ignition may result in flame acceleration. Also jet ignition to adjacent compartments may occur.

It should be mentioned that the volume of the containment and the cooling surfaces are relatively large compared to similar reactors. Therefore, the inertisation effect of steam may be smaller.

A mitigation technique is not yet selected (which is a major reason for this workshop). Borssele intends to follow the line to be set out in Germany in this respect. The vendor, Siemens/KWU, favours the so-called 'dual concept', a combination of catalytic recombiners with spark igniters. This concept is, therefore, an important part of the discussions in the workshop.
1. Reactor pressure vessel
2. Steam generator
3. Main coolant pump
4. Pressurizer (YP)
5. Quench tank (Y1)
6. ECCS Accumulator tanks (TJ)
7. Main steam lines (RA)
8. Main feedwater lines (RL)
9. Fuel pool
10. Personnel hatch
11. Material hatch
12. Emergency hatch
13. Biological shield
14. Cylindrical Protection wall
15. Primary containment
16. Secondary shield

*Figure 1* Reactor building (01/02)
1. Reactor pressure vessel
2. Steam generator
3. ECCS accumulator tank
4. Polar crane
5. Ladder
6. Cylindrical protection wall
7. Primary containment
8. Shield building
9. Refuelling machine

Figure 2  Reactor building
1. Reactor pressure vessel
2. Pressurizer (YP)
3. Quench tank (YP)
4. Refuelling machine
5. Fuel pool
6. Radioactive water storage tank (TR)
7. Main steam line (RA)
8. Stack
9. Secondary safety/relief valves
10. Personnel hatch

Figure 3  Reactor building and auxiliary building
Appendix 2

Description of the potential mitigation techniques

This note describes the most important basic candidate techniques (1 to 6) including some combinations of individual techniques (7 to 10). The purpose of this note is, in order to avoid misunderstandings, to use uniform names in the discussions.

1. Permanent inertisation

The containment is fully inerted during reactor operation. Personal access is impossible without respiration devices.

2. Post-inertisation

The containment is inerted after the accident by the addition of a large amount of inert gas, such as nitrogen or carbon-dioxide. The intention is to suppress the flammability completely. Without venting, the containment pressure will rise to about 0.3 MPa. The eventual pressure built up by steam is not considered. However, if steam is also present, less inert gas needs to be added.

3. Post-inertisation with early venting

As 2, but early in the accident a venting from the containment atmosphere is done in order to reduce the pressure rise of the containment. The venting should be done through the containment filter. However, the bulk of the early venting is intended to be done before a substantial amount of fission products is airborne.

4. Post-accident dilution of the containment atmosphere

Post-accident inert gas is added, but less than envisaged in the cases 2 and 3. Hence, the mixture remains flammable. It is expected that the dilution of the atmosphere will reduce the combustion loads, such as arise from flame acceleration or jet ignition in adjacent compartments.

5. Catalytic recombiners

These devices remove hydrogen by recombination with oxygen, stimulated by catalysts. They have large surfaces of catalytic material. They work passively. It takes a few hours to recombine the hydrogen with oxygen, if a sufficient number of these devices is available.

6. Deliberate ignition

This technique intends to ignite the hydrogen as soon as its concentration is flammable. Specially designed igniters are used, which can cope with the adverse conditions in the containment. There are spark plug igniters and glow plug types. The basic principle is that the combustion loads are expected to be within the load bearing capability of the containment, if the hydrogen is burnt at low concentrations.
7. **Post-accident dilution of the containment atmosphere with catalytic-recombiners**

This is a combination of techniques 4 and 5. The dilution itself does not remove hydrogen. Therefore, recombiners are added.

8. **Dual concept**

This technique combines 5 and 6. The basic hydrogen removal is expected to be done by the recombiners, whereas the igniters deal with those periods where the hydrogen concentration eventually is above the flammability limits.

9. **Deliberate ignition plus containment atmosphere dilution**

This technique combines 4 and 6. It accepts the hydrogen combustion, initiated by igniters, but it tries to keep the combustion loads low by added inert gases, reducing eventual flame accelerations or jet ignition of adjacent compartments.

10. **Dual concept plus containment atmosphere dilution**

This technique combines 4 and 8 (or 5 and 9). It reduces further eventually remaining risks, associated with the individual techniques.
Appendix 3

Description of the criteria/attributes

This note describes the criteria that will be used to compare the alternative mitigation techniques. The list is subdivided into groups of criteria. The value that will indicate the score of an alternatives is given for each of the criteria.

Criteria

1. Risk reduction/safety contribution

   All alternatives are focused on the prevention of explosions that may damage the containment. Risk reduction is therefore focused on consequence reduction and does not influence the probability of hydrogen production.

   The success of the technique may vary among the different scenarios and be influenced by other phenomena such as steam production, compartmentalization of the structure & local damage. Four scenarios for hydrogen production have been distinguished.
   - Large break LOCA's (e.g. through total or partial loss of ECCS).
   - Small break LOCA's (e.g. through total or partial loss of ECCS).
   - Transients (e.g. through failure of the ultimate heat sink).
   - Containment bypass scenarios.

   The emphasis is focused on the short term (up to one day) and the intermediate period (up to five days) after the beginning of the severe accident.

1.1a Indicate the expected effectivity rate that can be attached to the method for each of the given scenarios. The indication is based on today's understanding and available proof.

   Value: very effective, effective in a number of cases, ineffective.

   Explanation:
   The method does not necessarily need to score well on all scenarios. It may be sufficient to cover only the more important scenarios. The objective is risk reduction, not risk annihilation. Effective operator actions, influencing the scenario, should not be assumed unless such actions are credible.
   The assessment of 'today's understanding and available proof' is addressed in section 2 of the criteria list.

   In this question the emphasis is on the short (up to one day) and intermediate period (from 1 to 5 days) after the beginning of the severe accident, i.e. on the periods where accident management cannot benefit rom outside support and eventual additional countermeasures from outside, such as the installation of thermal recombiners. The long term is addressed in question 1.1b.

1.1b To what extend will the proposed method also cover long term periods (from 5 days on)?

   Value: completely, reasonably, not.
Explanation:
On the long term, radiolysis in the sump may become important, which produces both oxygen and hydrogen.

1.2 Can the method have large negative consequences for certain scenarios (e.g. significant radioactive releases)? Indicate this for the scenarios addressed under item 1:
- large break LOCA’s,
- small break LOCA’s,
- transients,
- containment bypass scenarios.
Value: yes, in certain cases no.

Explanation:
Examples of processes with a potential for large negative consequences are: uncontrolled or unexpected explosions, explosions despite inertisation, explosions in a pre-pressurized (by steam or added inert gases) containment. In the last case the AICC - multiplied pressure may exceed the containment failure pressure, if the AICC factor is e.g. 3 or 4 and the initial pressure is around 3 bar, due to steam or added inert gases.
In the assessment process, such negative consequences could be acceptable, if they would only apply to a very small number of the core melt scenarios.

1.3 Indicate the potential improvement for each of the given severe accident scenarios.
Value: much, some, minor.

Explanation:
If today’s understanding and available proof is not sufficient for the method, then indicate the potential improvement for each of the given scenarios, after additional research has been completed.

2. Confidence in the concept/state of technology

2.1 To what depth is the proposed method supported by theoretical and representative experimental evidence?

Value: full scale experiments, medium scale experiments, small scale experiments, no experiments possible.

Explanation:
First principles include basic laws of physics and basic phenomena. Experiments are only to be credited if they simulate the problem properly, e.g. realistic hydrogen sources and realistic scaling.

2.2 What additional work (theoretical/experimental) needs to be done to reach an acceptable level of proof?

Value: none, some work, much work.
Explanation:
Theoretical work, such as computer codes, may need further development (e.g. 2D or 3D-effects) and/or further validation. Experimental work may need more realism in simulating reactor conditions and/or a proper upscaling to plant conditions. However, the level of proof required depends on the level of risk, i.e. the smaller the risk is, the smaller the level of proof may be. In other words, the more important scenarios require better proof.

2.3 How much time is needed before the method will be established and adequately supported by an appropriate research and development programme?

Value: approximately 2, 5, 10 years.

Explanation:
The estimate here should be realistic, i.e. based on a realistic allocation of funds and people. Progress made in the past on items of similar complexity can give here a good guidance.
3. Compatibility with the existing installation

Incorporation of hydrogen mitigation equipment affects the present installation in various ways. This may apply to the normal daily operating and also to emergency situations.

3.1 Is the proposed method compatible with the general safety concept of the power plant? I.e. the method should not disturb the safety actions, initiated either automatically or manually after an accident. It should also be compatible with the basis lay-out and configuration of the plant.

Value: yes, partially, not.

Explanation:
The value of this question could also be extended in a more principle direction; do we accept a method that opposes the defense-in-depth principle or the multi-barrier concept of a nuclear power plant? E.g. an early containment venting (i.e. within approximately 1 hour after the beginning of the accident) could be seen as opposing the general safety concept of isolating the containment after an accident. The question here, however, is meant with its main emphasis on the technological issues.

3.2 Does the method introduce risks for safety grade structures, systems and components (e.g. safety/relief valves, the containment spray system), either by functioning as designed or by improper operation?

Value: no, some, potentially important.

Explanation:
An example is an intended or accepted ignition that, through pressure waves or high temperatures, damages safety features, such as safety/relief valves. It should be noted that the S/R valves have a function to intentionally depressurize the primary system, in order to prevent a high pressure melt-through.

3.3 Would the proposed method be qualified for the severe accident conditions foreseen (pressure, temperature, radiation, aerosols, ev. jetforces from breaks).

Value: yes, not fully, no.

Explanation:
The proposed method should also survive the severe accident conditions it is intended to mitigate; it should be indicated here whether this is the case or whether collapse of the device(s) might be possible.

3.4 Will the proposed method potentially affect other severe accident management features foreseen (such as provisions for bleed & feed, the containment vent system)?

Value: positive, not, negative.

Explanation:
If the hydrogen is not removed by the method then the containment vent system may be challenged by explosions during operation.
The capacity of the venting system must be taken into account.
3.5 Does the proposed method introduce risk for plant personnel, either by functioning as designed or by improper operation (including inadvertent actuation)?

Value: none, some, much.

Examination:
An example is an unintended actuation of an inert-gas supply system, suffocating plant personnel in the containment.

4. Technical characteristics

The technical characteristics of the alternatives are part of the comparison.

4.1 Is the equipment for hydrogen mitigation sensitive to fires, explosions or mechanical impact?

Value: resistant, little, sensitive.

4.2 Are external technical provisions required such as energy, cooling water, etc.

Value: none, some, many.

4.3 What are the expected costs of installation?

Value: low, reasonable/moderate, high.

4.4 What is the effort for maintenance?

Value: none, minor, frequent.

4.5 What are test possibilities during plant operation or during maintenance?

Value: full, partially, none.

5. Operating characteristics

Incorporation of hydrogen mitigation technique may affect operating tasks.

5.1 Does the method require special training for the operator?

Value: no, minor, moderate, intensive.

5.2 Which actuation does the method require?

Value: automatic, manual.

5.3 Can the method be de-activated if necessary?

Value: automatic, manual, not.
5.4 How much time does the method require to become effective?

Value: seconds, minutes, quarters, hours.
Session 2b:

Other containment accident management strategies.

Chairman: W. Frid
External Spray Cooling of the Loviisa Containment

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1. Introduction

A basic approach to manage severe accidents in nuclear power plants is to minimize the radioactive releases to the environment. If the prevention of core damage was not successful, mitigation of consequences should concentrate on all possible efforts to maintain the integrity of the containment.

The Loviisa NPP in Finland comprises two Russian VVER-440 reactors furnished with ice condenser containments. The first requirements concerning severe accident mitigation of the Loviisa plant were given by the Finnish Regulatory Authority, STUK, in the aftermath of the Chernobyl accident. Already before that, general design regulation YVL1.0 (1981) required that severe accidents have to be accounted for in the design of new nuclear power plants.

At that time, assessment of the severe accident loadings of the Loviisa containment had not been progressed very far. Therefore, studies were started to find out importance of various loadings and influence of eventual mitigation methods. The studies focused first on prevention of slow overpressurization of the containment. An initial idea was to introduce a filtered venting concept. However, more detailed studies of possible means of the overpressure protection revealed several significant disadvantages in the venting concept [Tuomisto, 1987]. The main drawback of the filtered venting concept comes from a poor subpressure performance of the thin steel shell containment (see features of the Loviisa plant in Table 1). On the other hand, an external spray system appeared as an attractive alternative. Since the basemat concrete at Loviisa does not contain any CO₂, generation of noncondensible gases (except hydrogen) would be practically zero even in case of sustained core-concrete interaction. This fact made it feasible to limit overpressure protection to condensing of the produced steam by spraying the outer surface of the containment dome. Later, the concept of in-vessel retention of corium, initially proposed by [Theofanous, 1988], was pursued as a cornerstone of severe accident management strategy. Thus, the core-concrete interaction is not even expected to occur. The advantage of external spray system comparing to the filtered venting is that no deliberate venting of noble gases takes place. The external spray systems were constructed and commissioned for the Loviisa containments during 1990-1991.

At the same time, the strategy of Severe Accident Assessment and Management (SAM) was developed for the Loviisa plant. This strategy integrates probabilistic safety studies (PSA) to ensuring the containment integrity, as discussed by Tuomisto and Theofanous [1991]. The external containment spray system has an important role in the overall SAM strategy.

The thermal-hydraulic basis of the external spray design including calculations and experiments is discussed in Section 2. The design principles and solutions of the system are presented in
Section 3, and finally, Section 4 discusses some operational aspects of the new system.

2. Design calculations and supporting experiments

In the Loviisa containments, slow overpressurization during a severe accident can be caused only by continuous steaming. Plant specific features such as the steel shell containment and a relatively low steaming rate make the external spray cooling feasible.

2.1. PREDEC calculations

It was found necessary to calculate a great number of slow, long-term transients and to study sensitivity of pressurization to several parameters. For these studies, a fast-running and easy-to-use calculational tool, PREDEC, (PRESSurization by DECay heat) was developed. The PREDEC model consists of a single volume with uniform pressure and temperature. The heat sinks are approximated roughly by dividing them into four groups: the internal concrete structures, the inner and outer containment, the internal steel structures and the sump water inventory. The internal concrete structures can be further divided into groups by their thickness. The volume of heat sinks taken into consideration can be easily varied. Both convective and condensing heat transfer from the air-steam atmosphere to the internal concrete structures and to the containment wall are modelled with the heat transfer coefficients for each time step. The heat transfer coefficients are calculated according to the methods proposed by Covelli et al [1982] and Corradini [1983]. A description on, how the heat transfer coefficients are applied in the PREDEC program, can be found in [Tuomisto et al., 1990]. On the outer surface, it is assumed that all the containment dome area is wetted by the spray system. A relaxation method has been employed to calculate the temperature profile of the concrete structures, while the steel structures are assumed to follow the ambient temperature without delay. In the containment wall temperature calculations, the radiation and convection between the inner and outer containments are taken into account, too.

Several calculations were made to find out the importance of the following parameters, see [Pekkarinen et al., 1988]:

- melting time of ice condensers
- decay heat level
- the amount of sump water
- the amount of concrete and steel structures acting as heat sinks
- heat losses through the containment to the open air
- the heat transfer models
- the outside spray flow rate and temperature
- filtered venting.

The mass of the heat sinks, mainly concrete structures and sump water, was found to have a significant influence on the pressurization rate as illustrated by Fig. 1. Since PREDEC is a single-volume model with just one pressure and temperature, it was necessary to vary the amount of the structures acting as heat sinks.

In the very long term the structures have a minor role, when the temperatures reach a saturated state. Heat transfer models were varied to see, if the heat transfer is adequate. The long-term
pressure behavior of external spray cooling with various spray temperatures and flow rates with the two heat transfer models can be seen in Fig. 2. These heat transfer models were obtained from small-scale facilities.

After the PREDEC studies, two problematic areas were identified, where more experimental evidence were needed to justify the external spray cooling. These unresolved aspects were

- to show that the applied convective and condensation heat transfer models give sufficiently reliable results in the full geometry and that heat transfer is thus sufficiently effective, and
- to show that the external spray system is capable of wetting the whole containment dome area.

Because large-scale, experimental data was not existing, it was proposed to examine the phenomena in the German HDR facility. The experiments with the external spray were then carried out during the experimental series E11 [Tuomisto et al., 1990]. Figure 3 demonstrates how the upper compartment of the HDR containment gives a fairly good simulation of the Loviisa containments in scale 1:2.

2.2. HDR experiments

The external spray system was constructed on the containment steel dome of the HDR facility under cooperation agreement between IVO and the HDR-Project. The spray system consisted of the inlet system with the pump supplying spray water on the steel dome through 48 spray nozzles, and of the outlet system.

The measuring system consisted of 700 temperature sensors, 45 gas concentration sampling units (each consisting of a hydrogen and helium measurement, a humidity sensor and a temperature sensor), and flow velocity measurements.

The spray system was operated in the experiments E11.2 and E11.4. These experiments were aimed at studying hydrogen distribution during stratified conditions inside the containment. The spray system was started during the cooling down period, when the containment pressure reached 1.7 bar. The initial spray flow of 6 kg/s in the HDR corresponds to 30 kg/s at the Loviisa containments when scaled according to the ratio of heat transfer area of the domes, 628 m² and 3100 m², respectively.

Heat transfer coefficients in the dome area were defined on the basis of the measurements using two methods: The average heat transfer coefficient was defined from the heating rate of spray water. Local heat transfer coefficients were defined from the temperature measurements as a result of solution to the inverse heat conduction problem.

In addition to the primary interest, i.e. heat transfer in the dome area, the measurements were performed for the concrete heat transfer blocks to clarify the behaviour of the major heat sinks in the containment.

The wetting properties of the spray nozzles were visually observed during trial runs of the external spray system. The visual observations and indications from the temperature measurements implied that the desired wetting properties were achieved.
2.3. Conclusions from the design calculations and HDR experiments

The PREDEC calculations with varying fraction of the concrete structures of HDR experiments E11.2 are shown in Figure 4 and compared to the experimental pressure data. In these calculations the corrections of the boundary conditions, e.g. of injected steam flow rates and heat losses were introduced as proposed and discussed by Valencia [1992] and Karwat [1993]. The pressure behavior is predicted rather well in case that only half of the structures are heated up. This can be explained with the stratified temperature conditions during experiment E11.2. The influence of the external spray operation as predicted with PREDEC is illustrated in Figure 5.

The main result from the HDR experiments was that the PREDEC program can be used for the design calculations of the external spray system of the Loviisa containment.

2.4. VICTORIA experiments

The influence of the external spray system has been also tested in the VICTORIA facility. VICTORIA represents a 1/15-scale model of the Loviisa ice condenser containment [Hongisto and Tuomisto, 1991]. The main aim of the VICTORIA experimental program is to study the hydrogen distribution in the containment and to demonstrate efficiency of the proposed hydrogen management strategy.

3. External spray system

3.1. Design basis

The external spray system was designed to prevent slow overpressurization of the containment due to continuous steaming. Such situation might take place after the melting of the ice condensers, when other means of decay heat removal from the containment are not operable during severe accidents. The long time delay allows to use an active system for this purpose.

The dimensioning criterion is to condense the steam on the inner surface of the containment dome at containment pressure of 1.7 bar (abs). The required spray flow is 30 kg/s at temperature 30 °C. The wetted dome area is 3,000 m² and the required heat transfer capacity is 3 MW.

The system is started and controlled manually from a separate, remote control panel. Autonomous operation of the system independently from the plant emergency diesels is ensured with dedicated local diesel generators. The single failure criterion is applied. The active parts of the system are independent from all other containment decay heat removal systems. There are no active parts of the system inside the containment.

3.2. The system description

The both units Loviisa 1 and 2 have their own external spraying circuits and spray water storage tanks. The cooling circuit of the spraying system and the dedicated diesel generators are common for both units.
The external spray system consists of

- water storage tank 120 m³
- recirculation buffer tank 5.5 m³
- two parallel spray pumps of the spray circuit (2x100%)
- cooling water circuit with two parallel cooling pumps and heat exchanger.

The ultimate heat sink of the system is seawater (2x100%, common for two units). The spray water is delivered to the containment dome from the pumps through the heat exchanger and through 80 spray nozzles located in four nozzle rings. The water is gathered from the containment dome back to the recirculation buffer tank, and from there, further to the spray pump suction. An additional connection has been provided to allow spray water replenishment with fire trucks. The instrumentation and control principles of the system have been discussed by Tuomisto et al. [1992]. Figure 6 illustrates the principle scheme of the whole system.

3.3. System operation

The external spray system will be started manually, when the containment pressure reaches the design pressure of 1.7 bar. During severe accidents the pressure increase may begin in the containment after the ice condensers have melted. According to the performed analyses, pressure would reach the design pressure of 1.7 bar after 18 – 36 hours after the beginning of the accident.

The capacity of the water storage tank is 120 m³, which covers all foreseen losses of water during spraying. To cover any unforeseen water losses the provisions can be taken to supply water to the tank after three days from the beginning of the accident.

In case that the pressure would increase faster than predicted, e.g. in case of partial bypass of the ice condensers, it is possible that containment pressure has increased in maximum to the estimated ultimate failure pressure of 3.25 bar. Spraying of the steel dome under these conditions causes steam production and temporary pressure increase in the annular space between the outer concrete building and the steel containment. The pressure control of the annular space is provided with flaps to cope with these pressure transients.

4. Concluding remarks

External spray systems were constructed on the Loviisa containments in order to prevent slow overpressurization due to continuous steaming during severe accidents. The external spray system replaces a filtered venting that was originally planned for Loviisa after the Chernobyl accident. The associated long time delays allow utilization of an active system. An extra benefit is that deliberate release of noble gases can be avoided.

The design calculations were verified with almost full-scale and real condition experiments in the German HDR containment.
REFERENCES


2. B. Covelli et al. (1982), Simulation of containment cooling with outside spray after a core meltdown, Nuclear Engineering and Design 69 (1982) 127-137


TABLE I Features of the Loviisa ice condenser containment

<table>
<thead>
<tr>
<th>Feature</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Free volume, total</td>
<td>57,000 m³</td>
</tr>
<tr>
<td>- lower compartment</td>
<td>6,000 m³</td>
</tr>
<tr>
<td>- upper compartment</td>
<td>47,000 m³</td>
</tr>
<tr>
<td>Design pressure, absolute</td>
<td>1.7 bar</td>
</tr>
<tr>
<td>Thickness of the steel shell</td>
<td>15 - 26 mm</td>
</tr>
<tr>
<td>Ice mass, design minimum</td>
<td>835 Mg</td>
</tr>
<tr>
<td>Volume of internal concrete structures</td>
<td>7,900 m³</td>
</tr>
<tr>
<td>Minimum permissible subpressure</td>
<td>3.5 kPa</td>
</tr>
<tr>
<td>Bypass from the lower to upper compartment</td>
<td>0.5 m²</td>
</tr>
<tr>
<td>Air return fans</td>
<td>no</td>
</tr>
<tr>
<td>Internal spray system</td>
<td></td>
</tr>
<tr>
<td>- redundancy</td>
<td>4 trains</td>
</tr>
<tr>
<td>- maximum capacity</td>
<td>2 x 250 kg/s</td>
</tr>
<tr>
<td>External spray system</td>
<td></td>
</tr>
<tr>
<td>- redundancy</td>
<td>2 trains</td>
</tr>
<tr>
<td>- capacity</td>
<td>2 x 35 kg/s</td>
</tr>
<tr>
<td>- heat removal capacity</td>
<td>minimum 3 MW</td>
</tr>
</tbody>
</table>

Figure 1. Long-term containment pressure with various heat sinks and ice depletion times
Figure 2. The influence of spray parameters and heat transfer correlation

Figure 3. The scale comparison between the HDR and Loviisa containments
Figure 4. Comparison of PREDEC calculations with the experiment data

Figure 5. Influence of external spray in experiment E11.2: PREDEC calculations
Figure 6. Principal diagram of the Loviisa external spray system
OECD Specialist Meeting on Selected Severe Accident Management Strategies


by Göran Hultqvist- Forsmark Technical Department

The Basis for installation of a Core Catcher in Forsmark Nuclear Power Plant

According to my abstract for this paper I should present the knowledge received from a study about the possibilities to install a Core-Catcher in Forsmark 1/2. In this report should the problem encountered in finding a optimised solution for a core debris coolability purposes without giving the maintenance personal too much problem in their activities inside containment and not to harm any other equipment. This study has not been done because problem received when we should establish the demands on a Core-Catcher. Therefore this paper will present the difficulties we encountered when we should establish the demands for a Core-Catcher with the knowledge of to day.

By Demands from the Swedish authorities Forsmark as well as all other Swedish Nuclear Plant has installed a Filtered Ventilation system to handle severe accidents without damaging the Containment and only release small amount of activities outside the plant. This modification was finished 1.1 1989. One of the unsolved questions in the summary report to the inspectorate was the coolability of the corium in containment. This problem should be solved as soon as possible with a yearly presentation of the status to the authorities.

By following and supporting research work in the field of severe accident and also following modern design of nuclear power plant Forsmark has have the opportunity to find out when modification work could be initiated concerning solving the coolability question for Forsmark 1/2 how has a too small floorarea in the lower containment to be able to receive a passive coolable coriumbed in that area.

Experiences from ABB-Atom bid to Finnish utility- the 5th-reactor

By Forsmark strong policies to implement modifications as soon as the knowledge are such that it is clear that the modification will increase the safety of the plant a decision was taken too install a Core-Catcher in the plants. This decision was partly bases on the knowledge received from the work that was done within ABB-Atom in offering a fifth reactor to Finland. The Finnish Authorities had put the demand on the new reactor that it would resist a core-meltdown-scenarios and that this scenario should end up with melted core that was cooled and that no harm or risk for penetrating the containment existed. Because of these demands ABB-Atom in their bid to the Finnish utility presented a Core-catcher.
This core-catcher was calculated to withstand the mechanical forces that was received by falling steam of fragmented coriumparticals and from falling controlrod-drive mechanism. The core-catcher would also resist the total weight of the core and extra materials falling down during a severe accident scenario. The basis for this core-catcher was that the fragmented particles that fell into it should each one be in a cooled status. When the total amount of core-debris has reached the core-catcher the decay heat could heat the inner part of the corium bed to melt temperature. A crusta would be surrounding the inner melted part of the coriumbed on the core catcher. The design of the core catcher was such that it forced water to circulate under it. The heat from the corium would be the drive-mechanism for the circulation of the water in the lower containment. Special plates are installed to control the flow of the water so that the coldest water will reach the bottom-surface of the core-catcher. The flow of water under the core-catcher will cool the surface of the corium in the core-catcher for as long as it is needed. The inner molten part of the debris will sometimes build up so much energy that it will penetrate the crusta of the corium and part of the corium will penetrate the corecatcher and fell down to the floor of the containment. This amount will be limited to some fraction of the total corium and will not destroy the effect of the core catcher. There should be no risk for penetration of the containment.

**Developing of a demand list for a core-catcher**

With the knowledge above Forsmark started a project to specify the demands on a core-catcher. Based on this specification a core-catcher design for Forsmark 1/2 should be developed and verified concerning the strength of the material.

The basic demand was clear. The core-catcher should

- be the solution for the coolability question rested form our Filtered Ventilation project-(FRISK).

- it should be constructed to interfere with regular-yearly maintenance activities as little as possible.

It was quit clear that a core catcher will give disturbances to the regular maintenance activities. This fact makes it important to sell the basis for the core-catcher to the maintenance personal. A core catcher that is not based on rigid bases will not be able so sell to the maintenance personal.

To evaluate the demands on the core catcher we learned from specialists on severe accidents and we evaluated the latest work and knowledge in this area, partly by visiting a seminar arranged by the APRI-project in ARE-Sweden spring-94.

The project collected the information and analysed it to present a specification for core-catcher design for Forsmark. The result of this work was that the description of
design event scenarios has changes all lot the last years and that in certain areas there do not exist a common understanding of the development of the scenarios.

It is important for the core-catcher function and design that clear understanding exist concerning the development of the scenarios in the following areas.

- a) will the melted core come down to the core -catcher as cooled fragmented particles or will it come down as a continuos stream of melted corium?

- b) will the penetration of the reactor vessel occur through a small hole in the bottom of the reactor vessel in one of the existing penetrations or will it occur through a circumferencial crack around the lower head of the reactor vessel?

- c) will steam explosion occur or not?

If the corium reaches the core-catcher as a continuos stream of uncooled corium no core-catcher will be able to withstand these loads and its heat. The core-catcher will be penetrated and most of the corium will be built up on the bottom floor of the containment.

If the corium causes the complete bottomhead of the reactor vessel to fell down in to the lower containment it is clear that a core-catcher will not improve the situation.

If steam explosion will occur it will be difficult to demonstrate or calculate that a core-catcher together with its plates that controls the flow of the cooling water will unaffected by these forces.

As you know there exists no consensus concerning the question above a)- c). We have therefore come to the conclusion that it is very difficult to convince the people at the plant, especially the maintenance personal, that the negative effects of a core-catcher will be able to motivate. So far it is only clear that a core-catcher will be effective a portion of thinkable scenarios of all the severe accident scenario that are known today.

**Other possibilities to receive cooled corium in severe accident scenarios**

Looking for scenarios to develop the specification of the core-catcher other knowledge was received concerning the possibilities of development of severe accident scenarios.

The knowledge concerning possibilities to cool the corium inside the reactor vessel so that vessel penetration will not occur is very interesting for a utility that want to develop simple emergency procedures.
If it will be possible to install extra cooling system or capacity to cool the vessel from inside or outside it will a very attractive solution for the safety of the plant. As we understand it the time for melting inside the reactor vessel and the transport of corium from the core to the lower head of the vessel is important for evaluating of this question. Today there are a spectrum of times presented concerning when the lower head will be penetrated by the corium 2 hours up to 10 hours are presented in different studies.

Forsmark will during the next half year learn more about the possibilities to solve the coolability-problem by cooling the reactor vessel from inside or outside by new cooling system or capacities.

Summary

From our work to introduce a core-catcher in Forsmark 1/2 we have reached the conclusion that

-the knowledge today concerning severe accident scenarios has not reached the needed level of consensus so that a specification for a corecatcher can be set up.

-other solutions my be the introduced to solve the coolability question for Forsmark 1/2.

-it is important to learn more about severe accident scenarios until modifications are introduced in the plant

-emergency procedures must be developed parallel to the new knowledge that are coming from the research activities

At Forsmark we will evaluate during the next month the evaluate the coverage, concerning solving the coolability problem, by different modifications as
- core-catcher,
- cooling outside reactor vessel,
- cooling inside reactor vessel
- other possibilities

By this coming studie Forsmark will be better prepared for coming demands on modifications and development of emergency procedures.
Two series of experiments to investigate FCIs for LWRs have been performed as a part of ALPHA program at JAERI. One of the major objectives of the experiments is to assess the effectiveness of the possible accident management measures on steam explosions and debris coolability in a RCV. In the melt drop steam explosion experiments, a melt simulating a molten core was dropped into a pool of water. Effects of pre-dispersion of the melt on FCI phenomena were examined based on the evaluation of void fraction in a coarse mixing region, melt settling velocity in a water pool, propagation velocity, expansion velocity, energy conversion ratio and debris characteristics. It was found that the probability of the occurrence of spontaneous steam explosions could be reduced by pre-dispersing the melt. However, the results showed that pre-dispersion of the melt could cause a more energetic steam explosion. Knowledge of the parametric effects of melt mass, ambient pressure and water temperature on the occurrence of spontaneous steam explosion was extended and confirmed under a melt drop geometry. In the melt coolability experiments, water was poured onto a simulated core melt. Heat transfer characteristics between the melt and the overlying water were investigated. Berenson’s correlation for a flat plate film boiling could be applicable for the prediction of the heat transfer from the melt to the overlying water. An explosive interaction, which might have been resulted from melt eruptions into the overlying water, occurred in one out of seven experiments. The melt eruptions were suppressed when water was near the saturation temperature or supplied through a spray nozzle. The explosive interaction in a stratified geometry was found to be much smaller in magnitude than steam explosions in a melt drop configuration. Water addition onto the melt was thus found to be generally effective.

1. INTRODUCTION

The reactor containment vessel (RCV) of a light water reactor (LWR) plays an important role to minimize accidental release of fission products to the environment as the last physical barrier[1]. Maintaining the integrity of the RCV is, therefore, crucial to minimize the consequence of a severe accident. It has been known that the RCV integrity could be challenged under extreme thermal and mechanical loads potentially generated in a severe accident[2]. It is considered that one of such extreme loads may be caused by "fuel-coolant interaction" (FCI) when the molten core consisting of fuel and core component materials contacts with the coolant. In particular, an energetic FCI which is referred to as "steam explosion" is considered to be one of threats to the RCV integrity due to its destructive force[3][4]. Another threat is a long term load due to the release of decay heat and noncondensable gases into the RCV atmosphere during molten core concrete interaction (MCCI).

In order to prevent the occurrence of a severe accident and to mitigate the consequences of the severe accident, various measures have been proposed as "accident management". Intentional supply of water into the RCV is considered as one of such accident management measures for LWRs in order to cool the molten core and the RCV atmosphere. The use of thermally unablated basemat of a reactor cavity is one of the possible ways to avoid the MCCI. Adding water into the RCV increases opportunities for contact between the molten core and the water, which might result in an energetic steam explosion. As shown in Fig. 1, various contact configurations between the molten core and the water could be developed when the accident management measure is activated during a severe accident; the molten core penetration into a deep or a shallow water pool in a reactor cavity, water accumulation onto the molten core overlying concrete or unablated basemat of the reactor cavity.
Concerning the contact configuration (a) illustrated in Fig. 1, large or semi-large scale experiments such as FARO[5] and KROTOS[6] at Joint Research Center at Ispra, and FITS[7][8] at SNL (Sandia National Laboratories) have been performed with high temperature molten materials. Valuable phenomenological aspects were obtained from these experiments to develop and validate analytical models. It was also confirmed in these experiments that higher ambient pressure and the elevated water temperature could suppress the occurrence of spontaneous steam explosions. Several experiments were conducted in FITS series under the configuration that the melt penetrated into a shallow water pool (configuration (b) in Fig. 1).

Recent studies with computer models predict that the initial mixing condition of the melt, water and steam strongly affect the occurrence of steam explosions[9][10][11]. If enough water is not available for the heat transfer from the melt to the water, spontaneous steam explosions will be suppressed. Such "water depletion" condition may be satisfied in a reactor vessel, which implies that a large scale in-vessel steam explosion is unlikely[12]. It is also known that enforcement of the melt dispersion using a grid-shaped device suppresses spontaneous steam explosions[13]. This can be understood as such artificial melt dispersion develops the water depletion condition. However, the possibility still remains that the artificial melt dispersion results in a more energetic steam explosion by realizing the suitable mixing condition.

To investigate quench behavior of the molten core interacting with the concrete basemat (configuration (c) in Fig. 1), SWISS[14] and WETCOR[15] experiments at SNL, and MACE experiments[16] at Argonne National Laboratory have been performed. In most of these experiments, progression of the MCCI was not terminated after water was poured due to the formation of the stable crust at the top of the molten material, which acted as a thermal resistance. Since the behavior of the crust formation is considered to depend strongly on the scale, experiments with larger scale should be required to apply the experimental results into the realistic reactor situation.
Experimental research on FCIs in a stratified geometry (configuration (d) in Fig. 1) with the realistic molten core simulants is quite limited. Two phenomenological experiments were conducted at SNL[17] using the alumina/iron thermite melt. Accumulation of additional findings based on such experimental study with detailed instrumentations is of a great importance.

In order to clarify the phenomena which could threaten the RCV integrity and to establish accident management measures, ALPHA (Assessment of Loads and Performance of Containment in a Hypothetical Accident) program was initiated at JAERI (Japan Atomic Energy Research Institute) in 1990[18]. As one of four test items, the melt coolant interaction tests have been performed to quantify the phenomena observed during FCIs and to evaluate the coolability of a molten core[19]. Two series of experiments are included in the melt coolant interaction tests; melt drop steam explosion experiments and melt coolability experiments. A simulated molten core was dropped into a pool of water (configuration (a) in Fig. 1) in the melt drop steam explosion experiments in order to qualify and quantify the FCI phenomena and to study the effectiveness of the possible accident management measures to suppress spontaneous steam explosions. Water was poured onto a simulated core melt (configuration (d) in Fig. 1) in the melt coolability experiments to investigate the heat transfer characteristics between the melt on an unablated basemat and the overlying water together with steam explosion characteristics in a stratified geometry. Some findings from the early experiments were reported in the previous papers[20][21][22]. Recent activities in the melt drop steam explosion experiments have focused on the evaluation of volume fractions of the melt, water and steam in the mixing region in cases when spontaneous steam explosions occurred and the melt was artificially dispersed. We have also tried to quantify the effects of melt dispersion on the FCI characteristics such as settling velocity of melt in a water pool, propagation and expansion velocities, conversion ratio and debris size distribution. All of the major findings from the melt coolant interaction tests regarding accident management measures in the RCV are summarized in the present paper.

2. DESCRIPTIONS OF EXPERIMENTS

2.1 Melt Drop Steam Explosion Experiments

2.1.1 Experimental apparatus

The conceptual diagram of the melt drop steam explosion experiments is shown in Fig. 2. Melt (molten core simulant) was dropped into the water pool located inside the model containment vessel (MCV). The MCV has an inner diameter of 3.9 m, a height of 5.7 m and an inner volume of 50 m³. Several viewing windows are provided in the MCV for visual observations.

The melt was generated in the melt generator by the thermite reaction of iron oxides (FeO and Fe₂O₃) with aluminum. The melt generator was placed on the top of the MCV. The maximum melt generating capability is 100 kg. An orifice of 200 mm diameter was located at the bottom of a crucible in the melt generator. Two types of melt dropping mechanisms were applied. In the early experiments (from STX001 through STX015), two layers of the thermite were installed in the melt generator; the upper layer of larger mass and the lower layer of 1 kg of thermite which formed a melt dropping device. Magnesium oxide (MgO) powder of 4 kg weight was inserted between the two layers as a thermal insulator. The two layers of thermite and MgO powder were placed on a glass plate which plugged the orifice at the bottom of the crucible. The lower layer of thermite was electrically ignited after all of the thermite in the upper layer had melted. High temperature produced by exothermic thermite reaction in the lower layer broke the glass plate.
Since the MgO powder may influence the break-up behavior of the melt in the water pool, the powder was replaced with two layers of 3 mm thick carbon steel plates in experiments after STX016 through STX021. Masses of the thermite powder were approximately 16 kg and 4 kg, above and below the carbon steel plates, respectively.

Two kinds of water pool were employed as shown in Figs. 3 and 4. One was the cylindrical water pool made of carbon steel with an inner diameter of 1 m and a height of 1.2 m, and another was box-shaped transparent water pool made of acrylic panels. In the latter, three sizes of water pool were used. The large water pool has a cross section of 0.88 m × 0.88 m with a height of 1.2 m, and the medium water pool has a cross section of 0.63 m × 0.63 m with a height of 1.12 m. The smallest one has a cross section of 0.45 m × 0.45 m with a height of 1.38 m.

In some of the experiments, the dispersion device as shown in Fig. 4 was placed in order to force the melt to disperse during the free fall into the water pool. The essential part of this device was a grid of 0.63 m × 0.63 m which was fixed at the center of the steel frame. The grid was made of steel wire of 2 mm diameter with a pitch of 25 mm. A smaller dispersion device was employed when the experiments were conducted with a medium size acrylic water pool. The dispersion device was placed at the upper part of the water pool so that the grid plane was located horizontally at 10 cm above the water surface. In STX021, the grid plane was set at 10 cm below the water surface.

Several instrumentations were provided for measurement of thermal-hydraulic characteristics in the melt drop steam explosion experiments. In earlier experiments, pressure history in the water and the MCV atmosphere were measured using strain gauge type pressure transducers. In recent experiments after STX010, piezoelectric pressure transducers were employed, some of which were suspended in the water pool for measurement of pressure history in the water phase. Four thermocouples of 0.25 mm diameter were used to measure the rapid temperature increase in the MCV atmosphere along with 1.6 mm diameter thermocouples in the experiments of STX016 through STX021. Measured pressure histories were recorded by a high speed recording system with a sampling rate of 500 kHz and other data were recorded with a low speed recording system with a sampling rate of 100 Hz. In the experiments from STX018 through STX021, liquid level detectors were used to measure a water level swell after the melt dropped into the water pool. Floats were employed to aid the visual observations of the water surface.
A high speed 16 mm camera and a high speed video were used to observe the steam explosion phenomena in experiments prior to STX016. Another high speed camera were added at a 90 degree different angle to the original one in STX016 through STX021. The high speed photographs were usually taken at approximately 4000 frames/s.

2.1.2 Experimental conditions

The experimental conditions of the melt drop steam explosion experiments are summarized in Table 1. The experiments using air as the surrounding atmosphere in which 20 kg of melt was dropped into subcooled water at the atmospheric pressure were chosen as the base case and repeated three times in STX002, STX003 and STX005. The atmosphere in the MCV was replaced by nitrogen in STX009. The above base case experiments were repeated using a small size acrylic water pool in STX016, STX017 and STX018. The MCV was pressurized up to 1.6 MPa in STX008 and STX012 and to 1.0 MPa in STX015 with nitrogen. Melt with a mass of 10 kg was used in STX001, STX010 and STX013. The melt was dropped into a pool of saturated water at atmospheric pressure in STX014. The falling melt was dispersed by the dispersion device in STX006, STX011 and STX020 with a large acrylic water pool. The dispersion device was installed in a medium size acrylic pool in STX019. In STX021, the dispersion device was placed 10 cm below the water surface of a large acrylic water pool.

Table 1 Summary of melt drop steam explosion experiments

<table>
<thead>
<tr>
<th>Run No.</th>
<th>Melt Mass (kg)</th>
<th>Pressure (MPa)</th>
<th>Atmosphere</th>
<th>Water Temp (K)</th>
<th>Water Pool (Yes/No)</th>
<th>Water Depth (cm)</th>
<th>Water Mass (kg)</th>
<th>Comment</th>
</tr>
</thead>
<tbody>
<tr>
<td>STX002</td>
<td>20</td>
<td>0.1</td>
<td>Air</td>
<td>289</td>
<td>Steel Yes</td>
<td>100</td>
<td>785</td>
<td>Nitrogen Atmosphere</td>
</tr>
<tr>
<td>STX003</td>
<td>20</td>
<td>0.1</td>
<td>Air</td>
<td>292</td>
<td>Steel Yes</td>
<td>100</td>
<td>785</td>
<td></td>
</tr>
<tr>
<td>STX005</td>
<td>20</td>
<td>0.1</td>
<td>Air</td>
<td>300</td>
<td>Acryl(L) Yes</td>
<td>100</td>
<td>774</td>
<td></td>
</tr>
<tr>
<td>STX009</td>
<td>20</td>
<td>0.1</td>
<td>N₂</td>
<td>289</td>
<td>Acryl(L) Yes</td>
<td>100</td>
<td>774</td>
<td></td>
</tr>
<tr>
<td>STX016</td>
<td>20</td>
<td>0.1</td>
<td>Air</td>
<td>295</td>
<td>Acryl(S) Yes</td>
<td>90</td>
<td>174</td>
<td></td>
</tr>
<tr>
<td>STX017</td>
<td>20</td>
<td>0.1</td>
<td>Air</td>
<td>286</td>
<td>Acryl(S) Yes</td>
<td>90</td>
<td>174</td>
<td></td>
</tr>
<tr>
<td>STX018</td>
<td>20</td>
<td>0.1</td>
<td>Air</td>
<td>283</td>
<td>Acryl(S) Yes</td>
<td>90</td>
<td>174</td>
<td></td>
</tr>
<tr>
<td>STX001</td>
<td>10</td>
<td>0.1</td>
<td>Air</td>
<td>293</td>
<td>Acryl(L) No</td>
<td>100</td>
<td>785</td>
<td></td>
</tr>
<tr>
<td>STX010</td>
<td>10</td>
<td>0.1</td>
<td>Air</td>
<td>297</td>
<td>Acryl(L) Yes</td>
<td>100</td>
<td>774</td>
<td></td>
</tr>
<tr>
<td>STX013</td>
<td>10</td>
<td>0.1</td>
<td>Air</td>
<td>284</td>
<td>Acryl(L) No</td>
<td>100</td>
<td>774</td>
<td></td>
</tr>
<tr>
<td>STX014</td>
<td>20</td>
<td>0.1</td>
<td>Air</td>
<td>372</td>
<td>Steel No</td>
<td>100</td>
<td>816</td>
<td>Saturated Water</td>
</tr>
<tr>
<td>STX008</td>
<td>20</td>
<td>1.6</td>
<td>N₂</td>
<td>288</td>
<td>Acryl(L) No</td>
<td>100</td>
<td>774</td>
<td>High Pressure</td>
</tr>
<tr>
<td>STX012</td>
<td>20</td>
<td>1.6</td>
<td>N₂</td>
<td>290</td>
<td>Acryl(L) No</td>
<td>100</td>
<td>774</td>
<td>High Pressure</td>
</tr>
<tr>
<td>STX015</td>
<td>20</td>
<td>1.0</td>
<td>N₂</td>
<td>282</td>
<td>Acryl(L) No</td>
<td>100</td>
<td>774</td>
<td>High Pressure</td>
</tr>
<tr>
<td>STX006</td>
<td>20</td>
<td>0.1</td>
<td>Air</td>
<td>298</td>
<td>Acryl(L) No</td>
<td>100</td>
<td>774</td>
<td>Dispersion Device</td>
</tr>
<tr>
<td>STX011</td>
<td>20</td>
<td>0.1</td>
<td>Air</td>
<td>290</td>
<td>Acryl(L) Yes (Mild)</td>
<td>100</td>
<td>774</td>
<td>Dispersion Device</td>
</tr>
<tr>
<td>STX019</td>
<td>20</td>
<td>0.1</td>
<td>Air</td>
<td>281</td>
<td>Acryl(L) Yes</td>
<td>90</td>
<td>392</td>
<td>Dispersion Device</td>
</tr>
<tr>
<td>STX020</td>
<td>20</td>
<td>0.1</td>
<td>Air</td>
<td>281</td>
<td>Acryl(L) No</td>
<td>100</td>
<td>774</td>
<td>Dispersion Device</td>
</tr>
<tr>
<td>STX021</td>
<td>20</td>
<td>0.1</td>
<td>Air</td>
<td>281</td>
<td>Acryl(L) Yes</td>
<td>90</td>
<td>697</td>
<td>Dispersion Device</td>
</tr>
</tbody>
</table>

The melt temperature was measured with a pyrometer in independently conducted separate tests[23]. The measured temperature at the melt surface was about 2700 K when the thermite reaction was completed. The melt travels between the melt generator and the water surface for about 3.5 m.

2.1.3 Data reduction

In addition to thermal-hydraulic measurements during the experiments, a lot of information was derived from the visual observations of the phenomena using high speed cameras. Settling velocity of the melt in the water and expansion velocity of the exploding region were evaluated from the visual data. Triggering location and propagation velocity of a shock front were obtained from high speed films by assuming that the color of mixing region changed from bright white to dark brown or black.
due to the rapid cooling of the melt when a shock wave passed the region.

Volume fractions of the melt, water and steam in the mixing region were evaluated based on the visual observations with the high speed cameras. "Apparent increase" in water volume, which is defined as increase in water level multiplied by cross sectional surface area of the water pool, corresponds to the total volume of the melt and steam. Steam volume was evaluated when the collected mass of the debris after the experiments was quantified. The mixing region contains the melt, water and steam. Thus, the difference between the volume of the mixing region and the apparent increase in the water volume gives the volume of water in the mixing region. Therefore, if both the apparent increase in water volume and volume of the mixing region are estimated at any instance of the experiments, average volume fractions of the melt, water and steam in the mixing region can be evaluated.

The level swell of water was measured with liquid level detectors and estimated by the visual observations. Estimation of the mixing region volume was made from the visual observations. It should be noted, however, the photographs only provide a two dimensional image of the mixing region, and two high speed cameras with 90 degree different angles are not sufficient to reconstruct the three dimensional structure of the mixing region. The method chosen for the evaluation of the mixing region volume is shown in Fig. 5. The mixing region was sliced with horizontal planes with every two centimeters in vertical direction. The horizontal length of the upper and lower edges of the sliced portion of the mixing region was measured from the front and the side views. Each sliced portion was approximated by an oblique frustum with ellipsoidal top and bottom surfaces. The line of apsides and minor axis of the ellipses are assumed to be the measured horizontal length of the upper and lower edges of the sliced portion. The total volume of the mixing region was calculated as the sum of volumes of the oblique frusta.

The major uncertainty in the present evaluation of volume fractions could result from evaluation of the volume of the mixing region with the assumption that the upper and lower surfaces of the sliced portion are ellipses. The maximum area for both surfaces can be obtained when they are assumed to be rectangles with measured horizontal length as two sides. Postulating that both surfaces form rhombi with the measured horizontal length as two axes, the surface area can be approximately evaluated to be minimum. Since the measured horizontal length was not much different between the upper and lower edges of the sliced portion, uncertainty of the volume in the mixing region could be related to surface area ratios of rhombus/ellipse and rectangle/ellipse, i.e. $2/\pi$ and $4/\pi$, which roughly corresponds to $\pm 30\%$ at most.

2.2 Melt coolability experiments

2.2.1 Experimental apparatus

The conceptual diagram of the melt coolability experiments is shown in Fig. 6. These experiments were also performed in the MCV of the ALPHA facility. Thermite used in the melt drop steam explosion experiments was employed to simulate a molten core. The melt was generated and held in an interaction vessel. Typical structure of the interaction vessel is illustrated in Fig. 7. A crucible made of MgO was placed in a carbon steel cover, and the gap between them was filled with a thermal insulator of MgO powder. The steel water pool was used in order to hold overflow water from the interaction vessel. Water was poured onto the melt surface through a water supply system with a pipe.
nozzle or a spray nozzle. Several thermocouples were employed to measure temperatures of the crucible wall, the overlying water pool and the MCV atmosphere.

2.2.2 Experimental conditions

The experimental conditions of the melt coolability experiments are summarized in Table 2. Melt depth was set at about 8 cm. For ACM002, ACM003, ACM005 and ACM007, a 10 kg of thermite was ignited in the interaction vessels with an inner diameter of about 0.2 m. Smaller or larger interaction vessels were provided for ACM004, ACM006 and ACM008 in order to investigate the effect of contact surface area between the melt and the water. Water was poured onto the melt surface through a pipe nozzle in all the experiments except ACM003, in which a spray nozzle was employed.

The nozzle exit was located at about 0.3 m above the center part of the melt surface. For cases in which the pipe nozzle was used, water fell down from the nozzle at the initial superficial velocity ranged between 0.46 m/s and 0.66 m/s at the exit of the nozzle. In the case of the spray nozzle, somewater flowed down the side wall of the crucible while some directly fell on the surface of the melt as droplets. Saturated water was supplied in ACM005 and subcooled water was used for other experiments. Water supply was initiated at approximately 30 seconds after the ignition of thermite. Water mass flux poured onto the melt, which was calculated by dividing water mass flow rate with the surface area of the crucible, is shown in Fig. 8. In all the experiments except ACM004 and ACM005, water was stepwise added onto the melt.

Temperature of the melt surface at the initiation of water addition was estimated from the results of two separate tests using 3 kg of thermite [23]. The diameter of crucibles used in the separate tests was determined in order that the depth of the molten thermite was the same as the depth of the melt in the melt coolability experiments. The surface temperature of the melt was measured using a pyrometer. Duration of the thermite reaction was also measured in the separate tests by detecting failure of a type-K thermocouple located at the bottom of the thermite. The surface temperature of the melt was approximately 2500 K at about 30 seconds after the thermite ignition. However, it must be mentioned that the bulk temperature of the melt should be higher than the surface temperature because of a large temperature gradient at the surface of the melt due to heat transfer to the environment. It is supposed that the bulk temperature of the termite melt was close to the temperature at the end of the thermite reaction (2700 K).

Duration of the thermite reaction in the melt coolability experiments was evaluated from the temperature measurements at the bottom wall of the MgO crucibles. The evaluated thermite reaction time is also listed in Table 2. It was slightly longer than the measured value in the separate tests, which ranged from 10 to 15 s. As shown in Table 2, the thermite reaction completed before the water addition onto
the melt. Due to the large difference in density between the immiscible products of the thermite reaction (i.e. aluminum oxide and iron), it was anticipated that a stratified state of the two melt components must have been quickly established after the thermite reaction completed.

Table 2: Summary of major experimental conditions of melt coolability experiments

<table>
<thead>
<tr>
<th>Run No.</th>
<th>Melt mass (kg)</th>
<th>Melt surface diameter (m)</th>
<th>Water pouring mode</th>
<th>Water surface pouring diameter (mm)</th>
<th>Inlet water temperature (K)</th>
<th>Duration of thermite reaction (s)</th>
<th>Water pouring timing(s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>ACM002</td>
<td>10.0</td>
<td>0.2</td>
<td>Pipe Nozzle</td>
<td>16.7</td>
<td>288</td>
<td>17</td>
<td>36</td>
</tr>
<tr>
<td>ACM003</td>
<td>10.0</td>
<td>0.2</td>
<td>Spray Nozzle</td>
<td>-</td>
<td>283</td>
<td>19</td>
<td>26</td>
</tr>
<tr>
<td>ACM004</td>
<td>2.5</td>
<td>0.1</td>
<td>Pipe Nozzle</td>
<td>8.0</td>
<td>293</td>
<td>10-15</td>
<td>34</td>
</tr>
<tr>
<td>ACM005</td>
<td>10.0</td>
<td>0.2</td>
<td>Pipe Nozzle</td>
<td>16.7</td>
<td>373</td>
<td>19</td>
<td>28</td>
</tr>
<tr>
<td>ACM006</td>
<td>31.5</td>
<td>0.355</td>
<td>Pipe Nozzle</td>
<td>32.9</td>
<td>297</td>
<td>23</td>
<td>31</td>
</tr>
<tr>
<td>ACM007</td>
<td>10.0</td>
<td>0.196</td>
<td>Pipe Nozzle</td>
<td>16.7</td>
<td>296</td>
<td>14</td>
<td>32</td>
</tr>
<tr>
<td>ACM008</td>
<td>30.0</td>
<td>0.333</td>
<td>Pipe Nozzle</td>
<td>16.7x3</td>
<td>299</td>
<td>17</td>
<td>35</td>
</tr>
</tbody>
</table>

1) Evaluated from measured temperature in the bottom wall of the MgO crucibles
2) Estimated from the separate tests

3. EXPERIMENTAL RESULTS AND DISCUSSIONS

3.1 Melt Drop Steam Explosion Experiments

3.1.1 Reproducibility of spontaneous steam explosion

As shown in Table 1, spontaneous steam explosions occurred in all of the seven experiments when 20 kg of the melt was dropped into the subcooled water without the dispersion device under the atmospheric pressure condition. A spontaneous steam explosion did not occur when mass of the melt was 10 kg in STX001 and STX013, while it occurred in STX010 which was performed under the almost same condition. It is reported that there existed a minimum critical mass for the occurrence of spontaneous steam explosions in the experiments with several to several tens kg of melt[24]. Although such minimum critical mass could depend on material properties and the geometry, it is revealed that the critical mass is less than 10 kg according to the present experiments under the atmospheric pressure and low water temperature conditions. It may be understood that the difference of the reproducibility of the spontaneous explosion for the experiments with 20 kg and 10 kg of the melt indicates that the occurrence of steam explosions tends to be more uncertain when the melt mass decreases to the minimum critical mass.

In the experiments of STX006 and STX020, spontaneous steam explosions were suppressed by using the dispersion device above the water surface in a large acrylic water pool. However, spontaneous steam explosions occurred in STX011, STX019 and STX021 which were performed under almost the same condition as STX006 and STX020 except that a medium acrylic water pool was used in STX019 and that the dispersion device was set at 10 cm below the water surface in STX021.

3.1.2 Characteristics of spontaneous steam explosion

(1) Melt settling velocity in water pool
Settling velocity of the melt in water are summarized in Table 3 together with propagation velocity of a shock front and expansion velocity of the exploding region for STX017 through STX021. The

Fig. 8 Water mass flux poured onto melt in melt coolability experiments
settling velocities listed in Table 3 are averaged values for 0.02 to 0.1 seconds prior to the spontaneous triggering of steam explosions. For STX020, where a steam explosion did not occur with the dispersion device, the averaged settling velocity was evaluated for 0.15 seconds after the melt reached about 30 cm below the original water surface. It was observed in each experiment that the settling velocity was slightly unsteady probably due to the transformation of the melt configuration in a water pool. When the dispersion device was used with the large acrylic water pool in STX020 and STX021, the settling velocity was much slower than that without the dispersion device. On the other hand, the settling velocity in STX019, where the dispersion device was prepared for a medium acrylic water pool, was close to the values for STX017 and STX018 without the dispersion device. This observation suggests that the melt was less dispersed in STX019 compared with STX020 and STX021. Cause of less melt dispersion in STX019 was not well understood. Local failure of the dispersion device, however, would be one of the possible reasons.

Table 3 Major results obtained from melt drop steam explosion experiments

<table>
<thead>
<tr>
<th>Run No.</th>
<th>Settling Velocity (m/s)</th>
<th>Propagation Velocity (m/s)</th>
<th>Expansion Velocity (m/s)</th>
<th>Av. Void Fraction (-)</th>
<th>Modified Void Fraction (-)</th>
<th>Conversion Ratio (Press.) (-)</th>
<th>Conversion Ratio (Temp.) (-)</th>
</tr>
</thead>
<tbody>
<tr>
<td>STX016</td>
<td>NA</td>
<td>NA</td>
<td>NA</td>
<td>NA</td>
<td>NA</td>
<td>0.86</td>
<td>0.96</td>
</tr>
<tr>
<td>STX017</td>
<td>5.17</td>
<td>395</td>
<td>43</td>
<td>0.203</td>
<td>0.182</td>
<td>3.66</td>
<td>4.76</td>
</tr>
<tr>
<td>STX018</td>
<td>5.50</td>
<td>320</td>
<td>66</td>
<td>-0.252</td>
<td>-0.231</td>
<td>3.33</td>
<td>4.76</td>
</tr>
<tr>
<td>STX019</td>
<td>4.88</td>
<td>523</td>
<td>99</td>
<td>0.419</td>
<td>0.405</td>
<td>5.67</td>
<td>19.45</td>
</tr>
<tr>
<td>STX020</td>
<td>2.09</td>
<td>NA</td>
<td>NA</td>
<td>-0.497</td>
<td>-0.483</td>
<td>NA</td>
<td>NA</td>
</tr>
<tr>
<td>STX021</td>
<td>3.19</td>
<td>380</td>
<td>56</td>
<td>0.630</td>
<td>0.607</td>
<td>4.05</td>
<td>4.95</td>
</tr>
</tbody>
</table>

(2) Propagation velocity
The propagation velocity of the shock front was estimated to be in the order of 300 to 600 m/s from the change of the color in the mixing region. The estimated propagation velocity was in same range as FITS experiments (250 to 560 m/s)[7], where 1 to 5 kg of alumina/iron thermite melt was dropped in a water pool. It was difficult to quantify the unsteady behavior of the propagation velocity since it was so fast and large spacial distribution was observed. The maximum propagation velocity was obtained in STX019.

(3) Expansion velocity
It was estimated that the expansion velocity of the high pressure mixture region consisting of the melt, water and steam ranged from 40 to 100 m/s as average values during 0.5 to 4.5 ms in the expansion phase. Although some uncertainties should be included in the evaluation of the expansion velocity because of two dimensional treatment of the visual data, the expansion velocity was generally higher in the earlier than in the later phase of the expansion. As same results with the evaluation of the propagation velocity, the largest expansion velocity was observed in STX019.

(4) Conversion ratio
It is known that the conversion ratio, which is defined as ratio of the mechanical energy produced during a steam explosion to the initial thermal energy of the melt, is one of the index showing the magnitude of the steam explosion. The mechanical energy consists of kinetic energy of the materials surrounding the expanding region (water and debris missiles) and compression work imposed onto the atmosphere. In the present study, the ratio of the compression work to the initial thermal energy of the melt was approximately evaluated from both the temperature and pressure increases in the MCV atmosphere according to the methods proposed by Farawila and Abdel-Khalik[25]. Typical pressure and temperature histories in the MCV atmosphere are shown in Figs. 9 and 10. The first peak in the temperature history would indicate compression of the atmosphere. The estimated values for STX016 to STX021 are presented in Table 3.
The following assumptions were used for the present evaluation of the compression work from the temperature history; 1) the atmosphere was adiabatically compressed, 2) mixing did not occur between the gases of the MeV atmosphere and steam generated by steam explosions, and 3) deformation of the MeV could be ignored. The average values of the temperature increase measured by all thermocouples of 0.25 mm diameter installed in the MeV were used in the evaluation. The evaluated conversion ratio based on the temperature increase for the experiments prior to STX016 was excluded from the table since no thermocouple of 0.25 mm diameter was installed. Application of the temperature increase to the evaluation of the conversion ratio includes the uncertainties due to temperature inhomogeneity in the MeV and the response capability of thermocouples. It should be noted that the temperature in the upper dome area was generally higher than that in the cylindrical part.

It is anticipated that the more accurate evaluation of the compression work is needed from the pressure rise in the MeV. Unfortunately, pressure measurement in the atmosphere of the model containment vessel was not reliable for the earlier experiments, because of the slower response of the strain gauge type pressure transducers used in these experiments. Since fast response piezoelectric type pressure transducers were installed to measure the atmospheric pressure, the conversion ratio was evaluated from the pressure history in the MeV for the experiments preceding STX016.

The conversion ratio ranged from 0.6 to 5.7 % for the pressure based evaluation, and from 0.9 to 19.5 % for the temperature based evaluation. Higher values were obtained for both conversion ratios in STX019. Localized temperature increases exceeding 300 K measured in the upper dome area of the MeV resulted in the much higher conversion ratio in STX019. This large temperature increase was detected by all (two) thermocouples installed in the upper dome of the MeV. These thermocouples were protected with a steal plate from direct contact of slug consisting of the melt and water. Similar temperature trends in the MeV atmosphere were also observed in other experiments. It is considered that a pressure wave generated by a steam explosion could easily travel upward since no obstacle such as the acrylic wall of the water pool was present in the vertical direction, resulting in the higher temperature increase. Differences between the minimum and maximum conversion ratios based on the temperature increase in the MeV atmosphere were 0.4 ~1.59 %, 2.05 ~5.26 %, 2.41 ~7.91 %, 3.02 ~>37.9 % and 1.05 ~8.85 % for STX016, STX017, STX018, STX019 and STX021, respectively.

(5) Volume fraction of steam and water in mixing region
Averaged volume fractions of the melt, water and steam in the mixing region were evaluated from the water level swell and the volume of the mixing region. Results for the volume fraction of steam (void fraction) are presented in Table 3. The averaged void fraction was estimated to be between 0.2 and 0.8 for STX018 through STX021. These results showed that more steam was generated when the dispersion device was used especially with a large acrylic water pool. It is considered that this was caused by the area enhancement of the melt available for heat transfer due to the artificial melt dispersion.
It was not clear that the melt was coarsely fragmented in the water in the cases the spontaneous steam explosion occurred such as in STX005. However, it was clearly observed that the melt was well dispersed in the wide range of the water pool in STX006 and STX020 in which the dispersion device was used and no spontaneous steam explosion occurred.

It was assumed in the evaluation of the averaged volume fraction that the mixing ratio among the melt, water and steam was uniform in the mixing region. However it was found from the visual observations that the color of the melt changed only in a part of the mixing region when a explosion was triggered. This suggests that the heat transfer which contributed to the explosion was limited in that part (referred to as propagation region) and the melt was not uniformly dispersed in the mixing region. It is considered that the volume fraction of water and steam in the region of melt existence is essential to understand the mechanism of steam explosion. The volume fraction of steam in the propagation region was calculated by assuming that all of the melt existed in the propagation region and volume ratio of steam to water was uniform throughout the whole mixing region. Results of reevaluation are also presented in Table 3 as modified void fraction. As observed in the averaged void fraction, the larger modified void fraction was derived when the melt was dispersed above or below the water surface.

(6) Debris characteristics
Debris formed by steam explosions was collected and size distribution was measured. The major debris characteristics including mass collected after experiments and mass median diameter are summarized in Table 4. Because of inclusion of 4 kg of MgO powder in the melt in the earlier experiments, the debris characteristics are listed in the table only for STX016 through STX021. The debris particle size varied from about 10 microns to several millimeters when the steam explosion took place. The debris particles had much smoother surface when the steam explosion occurred, indicating that the fragmentation process occurred when the melt temperature was still high. In the course of the evaluation of the debris characteristics, an assumption was made that the debris particles larger than 2 mm could not contribute to the fragmentation process during steam explosions. Therefore, mass median diameter of the debris was quantified for the debris particle less than 2 mm.

It is clear from Table 4 that almost all of the melt was fragmented into debris less than 2 mm diameter and smaller mass median diameters were obtained in STX019 and STX021. This characteristic feature of the debris size in addition to the results for the propagation velocity, the expansion velocity and the conversion ratio obtained in STX019 indicates that the magnitude of the steam explosion observed in STX019 was the largest among all the experiments. The reason why the smallest debris size was obtained in STX021 is not clear. However, it is believed that mixing region volume and the void fraction affected the consequences of the steam explosion. In addition to the largest void fraction as shown in Table 3, the largest mixing region volume was established in STX021, which was 1.5 times greater than that in STX019. This suggests that the melt was well dispersed within the mixing region, resulting in the lower heat transfer rate although the melt was finely fragmented by a shock wave. Furthermore, after the passage of the shock wave, lower pressure was thought to be developed in the relatively large mixing region.

3.1.3 Suppression of spontaneous steam explosion

(1) Dispersion device
Effects of pre-dispersion of melt on the occurrence of a spontaneous steam explosion were first investigated by Long[13] with aluminum melt. It was found in his experiments that spontaneous steam explosions were less likely to occur when the melt was coarsely fragmented before it contacted with water. Influence of the melt dispersion was further investigated in a large scale geometry with
high temperature thermite melt in the ALPHA program.

As mentioned earlier, the spontaneous steam explosion was suppressed by using the dispersion device above the water surface in the large acrylic water pool. On the other hand, the spontaneous steam explosion occurred in STX011 and STX019. Considering good reproducibility for the occurrence of the steam explosion in the experiments where 20 kg of the melt was used under atmospheric pressure condition without the dispersion device, it can be mentioned that the artificial dispersion of the melt reduces the probability of the occurrence of spontaneous steam explosion.

Observations with the high speed cameras showed that the melt was not completely dispersed in STX011. On the other hand, in STX006 and STX020, the melt was widely dispersed in the whole region of the water pool. According to the post-test observation, the dispersion device was locally broken in STX011 while it was less damaged in STX006 and STX020. This implies that the initial mixing conditions in the water for STX011 were different from those for STX006 and STX020.

Debris size distribution is compared among STX005 (steam explosion), STX008 (no steam explosion) and STX011 (steam explosion with the dispersion device) in Fig. 11. It must be noted that the MgO powder was used in the melt generator as a thermal insulator, and the initial particle size of the MgO powder was distributed between 0.5 mm and 2 mm. When the steam explosion did not occur as in STX008, the debris size was mostly over millimeter in range. On the other hand, once the steam explosion occurred it was in the order of several tens or hundreds of microns. It is also shown that the debris had larger size distribution in STX011 than in STX005, which indicates that the steam explosion was less energetic when the dispersion device was used in STX011.

However, the most energetic steam explosion was observed in STX019. The important initial conditions such as the settling velocity of the melt in the water pool and the void fraction in the mixing region in STX019 were different from those in STX020. The important finding from this result is that the development of the optimized coarse mixing condition consisting of the melt, water and steam to cause a more energetic steam explosion may be possible even though the melt was artificially dispersed.

One possible explanation for the suppression of the steam explosion by the melt dispersion could be that the dispersion device enhanced mixing of the noncondensable gases into the mixing region, and the dispersed melt in the water was wrapped with steam including noncondensable gases. It is known that noncondensable gas content in the vapor film significantly suppress the occurrence of steam explosions[24]. In order to confirm this, the experiment was performed with the dispersion device below the water surface of a large acrylic water pool in STX021. A spontaneous steam explosion was induced and triggered at the lower edge of the melt when the melt approached the bottom of the water pool. The settling velocity of the melt was low in STX021 as shown in Table 3, and the timing of the occurrence of the steam explosion was delayed compared with other experiments. Therefore, it is considered that the vapor film surrounding the melt was relatively unstable because of the decrease of the melt temperature and mixing of smaller amount of the noncondensable gases in the vapor film. The results from STX021 suggests that the inclusion of the noncondensable gases in the vapor film surrounding the melt may play an important role to suppress spontaneous steam explosions.

(2) Ambient pressure
A steam explosion was suppressed by the high ambient pressures of 1.0 MPa and 1.6 MPa. Effect of ambient pressure on the occurrence of the spontaneous steam explosion was reported with similar
experiments[7]. However the data was limited to 5.4 kg of melt and to a pressure of 1.09 MPa. The present experiment confirmed that the extended evidence was obtained using 20 kg of the melt up to 1.6 MPa. It is considered that the high ambient pressure affects the premixing behaviors, since the film boiling heat transfer is strongly influenced by the ambient pressure. However the detailed mechanism of this effect is not fully understood and further investigation is needed.

(3) Water temperature
No spontaneous steam explosion was observed in STX014 when the melt was dropped into the pool of the saturated water. Other conditions of STX014 other than the water temperature were almost the same as STX002, STX003, STX005, STX009, STX016, STX017 and STX018, in which the spontaneous steam explosion was observed. Suppression of a spontaneous steam explosion was also found in KROTOS experiments[6] when 1.5 kg of molten alumina was dropped into 10 K subcooled water. It is anticipated that the occurrence of steam explosion was suppressed by the formation of thicker and more stable vapor film surrounding the melt.

3.2 Melt Coolability Experiments

3.2.1 Heat Transfer Characteristics

The temperature history detected by the lowest thermocouples for the overlying water is shown in Fig. 12. Before water was accumulated to the level of these thermocouples, the temperature exceeded the saturation temperature of water. This is because the thermocouples detected the high temperature of the gaseous products from the thermite reaction and the radiation from the melt. The temperature of the overlying pool water then decreased gradually with time, starting with the saturation temperature, as the water was continuously poured. It should be noted that the large temperature distribution in the overlying water pool was not observed.

The heat flux at the top surface of the melt was evaluated with the measured temperature of the overlying water assuming that the overlying water was well mixed and that the vaporization of water was negligible. The energy balance for the overlying water pool can be written as,

\[
q = \frac{\rho C_p}{A_s} \left[ w(T_p - T_s) + V_p \frac{dT_p}{dt} \right],
\]

where

- \( A_s \) : Top surface area of the melt (m²),
- \( C_p \) : Specific heat of water (J/(kg K)),
- \( T_s \) : Inlet water temperature (K),
- \( T_p \) : Pool water temperature (K),
- \( V_p \) : Water pool volume (m³),
- \( q \) : Heat flux (W/m²),
- \( t \) : Time (s),
- \( w \) : Volumetric flow rate (m³/s),
- \( \rho \) : Density of water (kg/m³).
The evaluated heat flux at the top surface of the melt is shown in Fig. 13 for all the experiments except ACM005. Plots in Figure 13 was started after the lowest thermocouple over the melt detected water accumulation to its level. The heat flux decreased with time, reducing the surface temperature of the melt. Since the heat transfer rate at the melt surface is influenced by many parameters such as the melt temperature, the water temperature in the overlying pool, surface roughness, detailed analyses considering convection and phase change in the melt and heat loss through the crucible wall should be required to interpret the observations in the experiments.

The heat flux at the melt surface can be expressed as,

\[ q = h(T_m - T_s) \]

where
- \( T_m \): Melt surface temperature (K),
- \( T_s \): Water saturation temperature (K),
- \( h \): Heat transfer coefficient (W/(m\(^2\) K)).

(2)

The heat transfer coefficient depending on the melt surface temperature gives valuable information for a general discussion on the coolability of a molten core. Unfortunately, the melt surface temperature could not be measured during the quenching phase in the experiments. However, assuming that a stratified state was quickly established in the melt consisting of aluminum oxide in the upper layer and iron in the lower layer, and that the solidification temperature of aluminum oxide (2320 K) could be applied as the melt surface temperature within the initial short period of the quenching phase, the heat transfer coefficient in the corresponding period can be evaluated from the heat flux. It can be seen in Fig. 13 that the heat flux ranged between 900 and 1400 kW/m\(^2\) at the beginning of the quenching phase. Using this heat flux range, the heat transfer coefficient ranging from 460 to 720 W/(m\(^2\) K) is obtained.

Based on heat conduction in the melt, the calculation was made to roughly estimate how much time should be required to cool down the melt surface to the solidification temperature of aluminum oxide. Boundary condition with the constant heat transfer coefficient by film boiling at the melt surface (melt surface was assumed to be 2400 K) and material properties of aluminum oxide[26] were used in the calculation. The calculation suggested that the melt surface temperature decreased to 2320 K at about 5 to 15 s, depending on the material properties used in the calculation, after water addition when the bulk temperature of the melt was assumed to be constant at 2700 K. This time period is shorter than that to accumulate water to the level of the lowest thermocouple, in the overlying water pool (about 30 s). Considering the occurrence of melt eruptions as described later, however, it was expected that thick and stable crust was not formed at the surface of the melt during the first 30 s in the quenching phase. This must be an evidence that the temperature of the melt surface was not much different from the solidification point of aluminum oxide in the initial quenching phase.

Since the pool water temperature in the early cooling phase was near the saturation temperature as shown in Fig. 12, Berenson's correlation for flat plate film boiling[27] including the effect of the radiation heat transfer can be applicable. The heat transfer coefficient calculated with the Berenson's correlation was 530 W/(m\(^2\) K) at the melt surface temperature of 2320 K with an emissivity of aluminum oxide of 0.4. Although the Berenson's correlation was developed for the surface temperature condition near the minimum film boiling point, it seemed to be applicable under the high temperature condition of the melt surface encountered in the present study.

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3.2.2 Characteristics of explosive interaction

An explosive interaction between the melt and the overlying water was observed only in ACM002. Such explosive interaction was also observed in the steam explosion experiments with a stratified geometry performed at SNL[17]. In the SNL experiments, the steam explosion was observed when the water was fed onto the molten thermite just after the completion of the thermite reaction and no steam explosion was observed when the water was poured at approximately 5 seconds after the thermite reaction had completed. In ACM002, on the other hand, the water was poured at approximately 19 seconds after the thermite reaction had completed and the explosive event occurred at about 9 seconds after the initiation of water injection. It is estimated that 3.6 cm depth of water was accumulated over the stable interfacial vapor film when the explosive event occurred. Water flow rate in the SNL experiment was more than two times greater than that in ACM002.

Immediately before the occurrence of the explosive interaction, consecutive eruptions of the melt took place. Similar melt eruption followed by an explosive interaction was also observed in the SNL experiments. The eruption behavior is believed to be an important phenomena to establish an adequate coarse mixing configuration and to trigger an energetic event in a stratified geometry. A well-mixed molten tin/water/steam mixture behind a shock wave was developed in the experiments by Frost et al.[28] for a stratified geometry with molten tin of 973 to 1073 K and water. This mixture supported propagation of the subsequent more energetic interaction. Unfortunately, since visual observations at the interface between the thermite melt and the water were not possible in the present study, it was not clear whether a shock wave propagated or not at the interface to produce melt eruptions before the explosive interaction occurred. In the experiments performed by Greene et al.[29], liquid-liquid contacts were observed when water was poured onto such liquid metals as lead, bismuth and Wood's metal under surface superheat conditions up to about 600 K. It was also visually confirmed in the Greene's experiments that liquid metal jets penetrated into the overlying water layer to establish a mixing layer consisting of the liquid metal, water and steam. These phenomena observed by Frost et al. and Greene et al., which require collapse of the interfacial vapor film resulting in the direct contact between stratified two fluids, would be possible mechanisms for the occurrence of the melt eruptions.

Bang and Corradini[3] observed two types of vapor film destabilization from the high-speed films in their vapor explosion experiments of a stratified geometry; (a) vapor film collapse due to minimum film boiling dynamics and (b) vapor film destabilization due to an internal pressure source near the interface. It is considered that the latter type of vapor film destabilization induced the contact between thermite melt and water. Localized destabilization of the interfacial vapor film could be caused by the mixing of the subcooled water in the overlying water pool in the present study.

Debris formed by the explosive interaction in ACM002 was collected and the size distribution was measured. It should be mentioned that the explosive interaction was limited at the vicinity of the melt surface. The total mass of collected debris was about 1.5 kg, which corresponded to the depth of the melt of 1.5 cm. The result of the size distribution measurement is shown in Fig. 14 along with data obtained from the melt drop steam explosion experiments (STX005). It can be seen that less fraction was occupied by smaller size particles in ACM002 than in STX005. This indicates that the explosive interaction which occurred in ACM002 was less energetic than the steam explosion observed in STX005.

![Debris size distribution comparison](image)

**Fig. 14 Comparison of debris size distribution between ACM002 and STX005**
3.2.3 Suppression of melt eruption

Several eruptions were observed in the other melt coolability experiments. The eruption timing for all the experiments is summarized in Table 5. In addition to ACM002, multiple eruptions were identified in ACM004, ACM007 and ACM008. However, from the visual observations, scale of these eruptions were clearly much smaller than those observed in ACM002. Visual observation was not possible in ACM006 due to the scattering of the external light source by vigorous generation of aerosols during the thermite reaction. However a lot of particles with smooth surfaces were found on the solidified melt in ACM006 and it suggested that there might have been melt eruptions. The melt eruptions were observed only in the case when the subcooled water was poured onto the melt through a pipe nozzle.

Table 5 Time difference between water pouring and eruptions

<table>
<thead>
<tr>
<th>Run No.</th>
<th>Time difference between water pouring and eruptions (s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>ACM002</td>
<td>9 (Double)</td>
</tr>
<tr>
<td>ACM003</td>
<td>No eruptions</td>
</tr>
<tr>
<td>ACM004</td>
<td>2, 8</td>
</tr>
<tr>
<td>ACM005</td>
<td>No eruptions</td>
</tr>
<tr>
<td>ACM006</td>
<td>Unsuccessful visual observation</td>
</tr>
<tr>
<td>ACM007</td>
<td>31, 33, 35</td>
</tr>
<tr>
<td>ACM008</td>
<td>14, 17, 20</td>
</tr>
</tbody>
</table>

Although the melt eruptions were observed in all experiments in cases when subcooled water was poured through a pipe nozzle, only one explosive interaction occurred. This poor reproducibility for the occurrence of explosive interactions would partly result from the uncertainty of the initial conditions of the thermite melt. Characteristics of the thermite reaction could depend on geometry, location of thermite ignition, and so on.

No eruption was observed in ACM003 and ACM005, in which water was poured through a spray nozzle or at the saturated temperature, respectively. In ACM003, stable crust must have been quickly formed at the melt surface because of the effective cooling by water droplets. From the visual observation in ACM003, a large amount of steam was generated immediately after water addition, indicating that the melt surface was effectively cooled. The vapor film between the melt and the overlying water was thought to be thickened and stabilized, resulting in the suppression of the contact of water with the melt, since poured water was heated up to nearly saturation temperature in ACM005.

4. CONCLUSIONS

Two series of experiments, the melt drop steam explosion experiments and the melt coolability experiments to investigate melt coolant interactions for LWRs have been performed as a part of the ALPHA program at JAERI. One of the major objectives of the experiments is to assess the effectiveness of possible accident management measures on steam explosions and debris coolability in the RCV. The following findings were obtained from the experiments;

(1) The mixing condition prior to a steam explosion was quantified by evaluating volume fractions of the melt, water and steam in the mixing region. Other characteristics of melt coolant interactions were evaluated such as settling velocity of melt in water, propagation velocity, expansion velocity, energy conversion ratio and debris size distribution.

(2) It is considered that the artificial dispersion of the melt generally reduces the possibility of the occurrence of spontaneous steam explosions. However, it should be noted that the optimized coarse mixing condition could be possibly realized, resulting in a spontaneous and more energetic steam explosion in an extreme case.

(3) Evaluation of void fraction in the mixing region clearly showed that a melt dispersion device enhanced steam generation. The increase in the steam generation does not necessarily result in the suppression of the occurrence of spontaneous steam explosions.

(4) Comparison of the experiments in which the dispersion device was located above or below the water surface implied that mixing of noncondensable gases in the steam phase of the mixing region during the melt dispersion process might play an important role in the suppression of spontaneous steam explosions.

(5) Knowledge of the parametric effects of melt mass, ambient pressure and water temperature on the occurrence of spontaneous steam explosion was extended and confirmed under a melt drop geometry in the present study. Especially, it was found that steam explosions were
suppressed when the ambient pressure was over 1.0 MPa and the water temperature was elevated up to near the saturation temperature.

(6) Heat transfer characteristics between the melt and the overlying water were examined. Adding water with a spray nozzle was effective to rapidly cool the melt surface during the initial quenching phase. The heat transfer coefficient during the initial period in the quenching phase was evaluated and compared with Berenson's correlation for a horizontal flat plate film boiling. The Berenson's correlation was found to be applicable even under the higher melt temperature conditions.

(7) In the melt coolability experiments with a stratified geometry, a delayed explosive interaction following a large scale melt eruption was observed in one out of seven experiments. It is considered that the large scale eruption of the melt is a necessary condition for the explosive interaction by establishing an appropriate coarse mixing condition. The melt eruptions were found when the subcooled water was supplied through a pipe nozzle. No such melt eruption was observed when the water was poured through a spray nozzle or at the saturated temperature.

(8) From the comparison of the debris particle size distribution, it is considered that the explosive interaction in a stratified geometry was far less energetic than the steam explosion in a melt drop configuration.

Findings from the present experiments suggest that adding water in the RCV with appropriate conditions is generally an effective accident management measure. Especially, water addition under higher temperature and higher pressure conditions would be desirable to suppress the occurrence of a spontaneous steam explosion. From the view point of reducing the frequency of spontaneous steam explosions, the present experimental results support that the use of the melt dispersion device is a candidate of the possible accident management measure. However, further experimental and analytical efforts are necessary to quantify the detailed mechanisms of the steam explosion process. In order to evaluate the debris coolability in the realistic configurations of existing LWRs, experimental investigations with the simulation of the continuous release of decay heat and the phenomena during the MCCI are also needed.

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A STUDY OF EX-VESSEL STEAM EXPLOSIONS IN SWEDISH BWRs

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ABSTRACT

In Swedish BWRs, ex-vessel coolability, and stabilization, is provided by means of a deep water pool under the reactor vessel. This mitigation philosophy raises issues about steam explosion loadings on the underwater structures. While increasing the water depth enhances debris coolability potential, the larger inertia constraint would be expected to increase the damage potential from such explosions as well. The purpose of this study is to provide quantitative insights on the steam explosion aspects of this problem. Of particular interest are melt penetration into the pool, as moderated by freezing, and the explosion wave dynamics responsible for the structural loads. The calculations were carried out with the PM-ALPHA and ESPROSE.m codes; they covered a range of relevant conditions; and they reveal a number of important mitigative mechanisms, especially that of “explosion venting.”

1. INTRODUCTION

Because of certain key developments in the past one to two years it is now possible to imagine the viability of a computational approach to the consideration of steam explosions in safety assessments (Theofanous, 1993a). This is especially important because the current emphasis shift from in-vessel to ex-vessel explosions has created the need for a much more detailed understanding of the explosion process itself, as well as of its interaction with the surrounding media. This need arises because the evaluation of potential damage to surrounding
structures requires knowledge of the pressure-time history on them, and it arises even in certain new in-vessel applications in which the integrity of the lower head is at issue. This is to be contrasted with past approaches, which, aimed at the so-called alpha-mode failure, found it sufficient to consider the overall energy yield—through a conversion ratio and the quantity of fuel involved in the premixture (Theofanous et al., 1987; Turland et al., 1994, Theofanous and Yuen, 1994a).

The key developments mentioned above can be summarized as follows:

(a) The two leading computational tools (CHYMES and PM-ALPHA) for premixing were favorably compared with each other (Theofanous and Yuen, 1994a), and with the first experiments (the MIXA and MAGICO) designed specifically to address the water depletion phenomenon originally proposed by Henry and Fauske (1981) (Angelini et al., 1992; Angelini et al., 1994; Denham et al., 1992; Fletcher and Denham, 1994).

(b) The concept of microinteractions was introduced (Yuen et al., 1992; Yuen and Theofanous, 1994; Theofanous and Yuen, 1994a), a first attempt was made at the constitutive laws that govern its behavior, and the code ESPROSE.m was created. For the first time, rapid escalation to highly supercritical detonations was computed (Yuen and Theofanous, 1994), while simultaneously the first highly supercritical detonation, under more-or-less well-defined experimental conditions, was observed in the KROTOS facility in Ispra (Hohmann et al. 1994). Moreover, the ESPROSE.m model correctly (quantitatively) distinguished between this experiment, which was run with aluminum oxide melt, and a previous experiment run with molten tin (1000°C) that produced only a very mild propagation (Theofanous and Yuen 1994b).

(c) The concept of “explosion venting” was introduced, and it was shown to be very important in moderating the loads on nearby structures (Yuen and Theofanous, 1994; Theofanous and Yuen 1994b; Theofanous et al., 1994) from ex-vessel explosions. This can be considered to be a basic behavior that can always be counted on, and its significance is similar to that of the water-depletion phenomenon in in-vessel explosions.

(d) A couple of potentially very important compensating factors on steam explosion energetics were identified (Theofanous, 1993a). The first one, called “penetration cutoff,” has to do with the spontaneous triggering of high velocity melts as they enter highly subcooled water pools. This situation can arise in ex-vessel situations, and instead of deeply (in the pool) placed large scale premixtures, it would be expected to yield a series of shallow, small-scale explosions. If confirmed, this basic behavior is expected to take on a significance similar to that accorded to the well-known “pressure cutoff” phenomena in in-vessel explosions. Even
though not fundamentally understood yet the “pressure cutoff” is well enough supported by empirical evidence (Fletcher, 1994) that it has been properly accounted for in the assessment of alpha failure (Turland et al., 1994). The other compensating effect is between particle size during premixing and explosion propagation; namely, small particles, while offering extensive interfacial area for fragmentation during propagation, thus enhancing the escalation process and the completeness of energy conversion, are favorable to producing voided premixtures and extensive melt freezing. Both voiding and freezing are strong mitigators of energetics. The freezing effect is obvious—it leaves less melt to interact. The voiding effect has been discussed by means of code calculations (Theofanous and Yuen, 1994a; Fletcher, 1994). Conversely, large particles remain molten longer and produce less or even non-voided (under highly subcooled conditions) premixtures, both being very favorable to rapid escalation to very high pressures. However, this is moderated by the reduced interfacial area such particles offer for fragmentation and microinteractions. This sort of compensating effect can be handled very well by the combined use of PM-ALPHA and ESPROSE.m.

As seen by the references cited, the computational approach used in this study was developed in conjunction with the basic ideas and concepts described above. A few more ideas need to be mentioned to make this description complete. First, both premixing and propagation are recognized as being dominantly 2D phenomena. Thus both PM-ALPHA and ESPROSE.m have 2D capabilities. While 3D capability is easy to achieve, and probably will be at some future time, it is expected not to materially alter the assessments obtained from 2D results. The second idea originates from recognizing that breakup during premixing is so dominated by phenomenological bifurcations that their deterministic treatment should be considered only as a long term goal. In all likelihood even if such a goal were achieved (in a credible modelling approach), it is doubtful it could ever be verified to the degree necessary for use in safety (conservative) assessments. However, based on the compensating effect discussed above, it seems that such a deterministic (predictive) approach is not really necessary. Rather, the behavior can be bounded with respect to the degree of particulation by consistent calculations with respect to melt interfacial area, premixture voiding, melt freezing, and fragmentation kinetics. Carefully done breakup calculations, on the other hand, can also be used to provide additional perspectives, primarily by creating additional degrees of freedom in examining the compensating effects noted. Finally, the third idea is that triggering is also such an instability-dominated phenomenon that only in special cases (as in the “penetration cutoff” example discussed above) can one ever expect to categorically predict its occurrence and escalation to energetic explosions. Thus, the sound approach is to provide reasonable bounds on the effect by considering a whole spectrum of trigger timing and positions. This approach is possible, again, because of compensating effects.
For example in relatively shallow pools the deeper the penetration before triggering the higher the resulting energetics, while explosion venting still provides significant relief. For very deep pools, on the other hand, significant freezing occurs well before the melt reaches the bottom and thus imposes a restriction on the search for an energetic envelope.

We have entered an era where this approach can be meaningfully implemented only very recently, and this work constitutes the third such application. In the first (Theofanous and Yuen, 1994b), a generic ex-vessel geometry (no reference to a particular reactor) and a couple of generally relevant melt pours in 1 and 3 meter deep pools were considered. The purpose was to demonstrate the explosion venting concept under the strong energetic conditions, as calculated by ESPROSE.m. In the second application (Theofanous et al., 1994), the study was extended to intermediate depths (up to 5 meters) and highly subcooled water pools. The purpose was to study in more detail the role of internal voids and the effect of melt freezing. In the present and third study we consider very deep pools and the effects of particle size and melt pour rates.

The specific geometry and conditions considered are relevant to Swedish BWR, as discussed in section 2 (see also Sienicki et al., 1994). However, it is emphasized that this is a study, its purpose being to help develop, in the perspective of the detailed results provided, an understanding of the key physical processes and of the overall technical approach. Although these are the necessary first ingredients toward a safety assessment, such an assessment would require further analysis.

The verification status of the PM-ALPHA and ESPROSE.m codes has been summarized recently by Theofanous et al. (1994). Verification work is expected to reach a major milestone on September 30, 1994, at which point both codes will have been fully documented and will be available for release (under DOE’s Applied Technology Program). The latest reference to their formulation is by Theofanous and Yuen (1994b). This paper is focused on the results of the calculations and their discussion.

2. SPECIFICATION OF GEOMETRY AND EXPLOSION SCENARIOS

As noted above, these specifications are relevant to Swedish BWR and were meant to be inspired from previous work on core melt progression and lower head failure in the product BWR90 and the operating Forsmark 3 (F3) designs. The diameters of the below-vessel regions are 10 and 8.4 meters, and the water-pool depths are 10 and 7 meters, respectively. We chose a common diameter of 10 meters, such as to make more crisp the effect of water pool depth.

The ambient pressure was specified at 0.13 to 0.15 MPa, and the water pool temperature was chosen such as to reflect a very high (77 K) or a moderate (32 K) subcooling.
The melt was taken to be oxidic, at a temperature of \( \sim 2800 \) K, with either a low (50 K) or high (200 K) superheat. For premixing, three different pour rates, 900, 1,700, and 2,630 kg/s were examined, in combination with three particle sizes—0.3, 0.4, and 1.0 cm. Such pour rates can be considered high compared to \( \sim 500 \) kg/s, which is considered, in general, appropriate for BWRs (Theofanous et al., 1994).

Four representative premixing cases were pursued on to the explosion phase by providing a strong trigger, at the leading edge of the premixing zone, 0.5 seconds into the transient. This corresponds to melt penetration depths of \( \sim 3 \) meters into the pool. In addition, one of the premixing cases was triggered at 0.8 seconds: once at the leading edge that had now penetrated down to \( \sim 5 \) meters, and once around the middle of the premixing zone, at a depth of \( \sim 2 \) meters.

3. PREMIXING RESULTS AND DISCUSSION

Starting with the first set of calculations, as described in section 1, we have adopted a run identification (ID) code, for easy recognition of the key parameters. In sets I and II we varied water depth, water subcooling, and melt pour rate, so the ID was written as:

PreMixing (Set #)—Water Depth/Water Subcooling/Pour Rate Factor

The pour rate factor represents the pour rate on a rough multiple (rounded to the first decimal) of 1,000 kg/s, the typical order of pour rates of interest. In the present set III the particle size and melt superheat were varied also to explore the outer boundaries of the premixing zone due to freezing, in such deep pools. Thus, two new parameters were introduced to the run ID, while the water depth was deleted, as both depths considered are large enough already. Accordingly the present ID is:

PreMixing (Set #) — Particle Size/Water Subcooling (Melt Superheat)/Pour Rate Factor

The complete definition of the four premixing runs in Set III together with the respective codes are listed in Table 1. The melt velocities were obtained from the initial velocities specified and the free fall acceleration to the entry of the computational domain of PM-ALPHA. The melt pour rates, the initial velocities and diameter of the melt release opening, were chosen from prior Swedish work on core melt progression and lower head failure (Sienicki et al., 1994). In particular the melt stream diameter was taken equal to a failed penetration (5 cm) and it ablated to 25 cm by the end of the discharge, while the assumed 2.25 meter of melt gravity-head in the lower head produced an initial (exit) velocity of \( \sim 12 \) m/s. The total quantity of melt was conservatively assumed to be 213,000 kg or 100,000 kg.
Based on the above specifications and by consideration of the free-fall velocity, intervening structures, etc., we chose the effective pour diameters of the entry of the computational domain, as shown in Table 1. From these, and the respective pour rates and inlet velocities, the inlet melt volume fractions were obtained. The values shown in Table 1 were derived by redistributing the above average values such as to reflect on inner denser "core" and an outer more dispersed flow regime, that can be qualitatively expected.

These PM-ALPHA computations were actually carried out with a computational domain of only 4 m in diameter. According to test calculations this convenience does not introduce any perceptible difference in the results.

In examining the premixing results, of primary interest are the development of voids, the amount of melt accumulation obtained due to deceleration in the pool, and the degree of freezing due to heat losses to water. Within the space limitations here, we present sample results, primarily in a comparative manner among runs, such as to make evident the key behaviors and trends.

The important interplay between particle size and water subcooling in determining the degree of voids in the premixture can be easily seen by comparing the results of all four runs. In particular, with 1 cm particles even moderate subcooling can effectively suppress void formation. This is because the melt momentum induces a circulatory pool motion that continuously bring in fresh (cold) water to be entrained in to the melt stream as it enters the water pool, as illustrated in Figure 1. For the same reason the first voids develop, as they have to, deep within the pool, as illustrated in Figure 2. With the very high subcooling in case PM(III)-0.3/77(200)/2.7, even small particles and high fuel concentrations cannot induce voids, except in very localized regions, as illustrated in Figure 3, but in moderate subcoolings this balance is tipped in reverse as illustrated in Figure 4. In fact, in this latter case very significant void development is observed (Figure 5), which impacts on the coolant flow regime (Figure 6) and through it melt distribution (the separated pattern seen in Figure 5). It is clear from these results that they effectively span a wide range of premixing behavior, especially with respect to the compensating effect of particle size on energetics discussed in section 1.

The degree of melt accumulation can be judged by looking at the melt volume fractions found in the core of the premixing zone in relation to those at the entry. These are found to approximately double from initial values of about 15% and 7% to about 30% and 12% for run PM(III)-0.3/77(200)/2.7 and all the other three runs respectively.

As expected, freezing increases with the exposure time to water, so that eventually a maximum melt penetration is obtained. An integrated perspective on all four premixing runs is shown in Figures 7 and 8. From Figure 7 we can see that the maximum melt quantity is asymptoti-
cally limited due to freezing, while in Figure 8 we see the respective limitation in penetration depth. The idea for Figure 8 is that in the presence of a significant (taken at 20% here) solid fraction the particles cannot participate efficiently in an explosion. This effectively eliminates an outer envelope of the premixing zone as being unable to participate in a propagating event. It appears that these limitations warrant a significantly more in-depth study to delineate the ranges of uncertainty, as it would be necessary for example in a safety assessment (see remarks in section 1).

Certain other more salient features in the premixing results are worth noting. These include the water surface “depression” in the region of melt entry, and the “head-like” region in the inverted-mushroom-shape of the melt volume fraction distributions. They are both related to the melt-water momentum coupling, and they are currently examined in more detail with the help of the MAGICO-2000 experiment. Such formations are found in entry type problems where the receiving medium is pushed out of the way, thus acquiring a radial velocity component which in turn induces a radial spreading of the particles. Typically, the “head” is a region of low voids, even for cases that favor highly voided premixtures (i.e. Figure 5), and it can be considered quite favorable for triggering (spontaneous collapse of vapor blankets) and propagation.

4. PROPAGATION RESULTS AND DISCUSSION

The explosion runs are identified in the same manner as the premixing runs, except for changing the prefix to E.m(III), and adding the timing (in seconds) and trigger depth (in meters). For example run E.m(III)-0.3/77(200)/2.7/0.5(3.8) was triggered after 0.5 seconds of premixing at a depth of 3.8 meters. In this case, we see that the trigger was applied at the leading edge of the premixture. The complete listing of all ESPROSE.m runs performed is given in Table 2.

In all these calculations the entrainment factor in the microinteractions model (see Theofanous and Yuen, 1994b) was kept constant, at 4, for the case of 1 cm particles, and 8, for premixtures involving 0.3 or 0.4 cm particles. In fact, we expect this treatment to be conservative because the entrainment is seen (from SIGMA experiments) to increase with time elapsed from the inception of the interaction, and the increase is rather nonlinear. Note that in this formulation the water entrainment rate into the microinteraction zone is proportional to the melt fragmentation rate, and in contrast to the above-described expected behavior, the entrainment rate in the calculation returns to zero after all the fuel has been fragmented. The effect of the increased entrainment is to quench the tail-end of the explosion, and this was confirmed by test calculations in which we specified that the water entrainment rate is no less than its values in all prior history of the explosion. However, the impact on the wall loading history is not very significant, and this matter was not pursued any further.
In all but the last two runs the frozen quantities were not allowed to participate in the fragmentation process. In the last two runs any fuel with solid fraction less than 15% was included. The premixture configurations, that is fuel (liquid fuel for the first four runs and fuel with solid fraction less than 15% for the last two runs) and void distributions, at the time of triggering for all 6 explosion calculations are depicted in Figures 9 through 13.

The trigger was applied by releasing, suddenly, the pressure from one computational cell assumed to be steam filled at 100 bar. On the basis of the "penetration cutoff" hypothesis explained in section 1, these delayed triggers, which allowed penetrations close to half the pool depth, would represent a conservative view of "how much fuel is possible to explode at once." Also, by allowing such deep penetrations, the effectiveness of explosion venting is reduced, which is also conservative. Again, it is emphasized that these decisions were made mainly for developing an understanding through perspective, rather than to claim conservative results applicable to safety assessments.

Wall reflections are very important in the overall pool dynamics, hence the full geometry was employed in the calculations. The pool depth is not contained in the run ID, but it can be looked up from Tables 1 and 2 for each case. An interesting new dimension in the present set III of calculations, in relation to the previous two sets, is the much more complicated pressure wave reflection patterns and associated pool dynamics. This arises because of the large pool depths, and with the explosion zone extending only in the upper half, the lower pool dynamics creates a second peak on the wall loading. For the discretization we used a 20 x 20 cm grid. Sensitivity to node size was checked in one run with a 10 x 10 cm grid and found to be adequately small.

In examining the propagation results, of primary interest are the development of the explosions and radiated waves, reflections and the role of venting off the free surface, and the resulting pressure pulses at the pool boundaries. These are the primary output of this work, and they are all one needs to assess the potential for structural damage. These pressure pulses are summarized in Figures 14 through 19 for all 6 cases considered.

The most interesting feature, not seen in any of the previous two sets of calculations, is a double-pulse at the upper half of the side wall. This is evident in all six runs, but depending on the amount of voiding in the premixture, the first pulse may be more-or-less suppressed, while it exhibits a secondary structure. For example, the suppression due to high void in the premixture can be seen in Figure 16. In Figure 14, on the other hand, a small amount of void in the premixture is seen to initially impose a similar behavior, which, however, quickly gives way to a major "first" pulse, following void collapse as in the non-voided cases. The amplitude of this pulse is seen to be similar to those other cases (Figures 15, 17, 18, 19), but
the suppressing effect of void is in fact active here too in reducing the pulse-width. A salient feature of this slightly voided case is that the velocities on the propagating front overtake the voided region, they become very high pushing up the fragmentation rate, and then yield short pressure pulses well above those found in the non-voided cases. However, this appears to be a very local/temporary phenomenon, with pressures “venting” into the void and the main character remaining pretty much as in the other cases. Generally, we observe peak wall pressures in the 100 to 300 bar range, and pulse-widths of ~2 to 4 ms. The second pulse, if present, is somewhat more spread out in time and has generally less amplitude. The impulses in each case are given in the respective figure captions. The general trend is that internal voids strongly mitigate the explosion energetics and this is very much in line with findings in the previous two sets of calculations. On the other hand, in Figures 18 and 19, as compared to Figure 17, we see that a deeper penetration can produce somewhat higher loadings, even if the total quantity of melt involved is somewhat less (see Figures 7, 8, 12 and 13). In the set II runs we found that for the same penetration a reduction of the melt quantity of 38% produced a load reduction (peak pressure and impulse at the wall) by about a factor of two. It appears, therefore, that the depth of penetration is somewhat more important than the quantity of melt involved.

With regard to wave dynamics, it is interesting to note the venting directly from the explosion zone to the gas region above (i.e. see Figure 20) and the subsequent venting of the propagating pulse in Figure 21. This behavior is evident in all six cases, and the decay includes, of course, the effect of radial divergence. In these detailed results we can also see the origin of the double pulse at the upper half of the wall – a downward moving pulse, reflecting off the pool bottom and venting upwards in a very complicated axisymmetric front, as for example, in Figure 22. This behavior is even more clear in the non-voided cases, as illustrated in Figure 23.

The timing of these pulses can be determined also, quite simply, from the pool acoustics. For example, in a pool with an aspect ratio of less than 1 (an explosion depth of 4 meters in a 5 meter radius pool) the first pulse should be contained between the wave arrival time (equal to the acoustic time in the horizontal direction) and the incremental time required for relief in the vertical direction (equal to the acoustic-time in that direction). The acoustic times are 2.6 and 3.3 ms respectively, which gives that the primary pulse should be contained in the most part between 2.5 and 5.9 ms. This is in excellent agreement with the calculated results (see for example Figure 15, elevation 5.5 m, and Figure 17, elevation 3.7 m). The secondary pulse, on the other hand, should begin at about 2 acoustic times, between the vertical position of the main explosion pulse and the pool bottom. For the case of run E.m(III)-1/77(200)/0.9/0.6(3.8), for example, this distance is ~6 meters, and for 2 acoustic times we have 8 ms, which is in excellent agreement with the ESPROSE.m results in elevation 3.7 m of Figure 15. The termination of the
secondary pulse should follow by another 2 to 3 ms, and we see in the same figure that indeed is the case.

5. CONCLUDING REMARKS

This study extends the work of the previous two sets of calculations to very deep pools, which are of interest to the Swedish BWR. Taken together, these three sets of calculations provide unprecedented insights on the behavior of large scale steam explosion, and on ex-vessel situation in particular. For the first time, all key ingredients (two-dimensionality, localized debris-water mixing, whole-pool wave dynamics interactive with the explosion zone) are present in the calculation, and the viability of such a computational approach for safety assessments is made evident. This is because in addition to the above-mentioned key ingredients, at this time we have: (a) experimental results on premixing that exhibit well the key features of PM-ALPHA, and new experiments underway to provide an extension of this to higher particle temperatures and to subcooled water pools, (b) experimental results that show clearly the supercritical detonation predicted by ESPROSE.m under certain conditions (KROTOS), and experimental data from which the constitutive laws for microinteraction under the relevant propagation conditions (SIGMA) can be obtained [more data in both facilities is presently being generated]. Moreover, we have developed an approach that identifies, accounts for, and demonstrates by actual calculations several compensating effects. On the basis of these effects, robust safety assessments can be made, thus bypassing certain modelling (and verification) difficulties because of the inherently chaotic/unpredictable (with bifurcation) phenomena of breakup (during premixing) and triggering.

More specifically, the results of the calculations indicate that:

- There is a tendency for small melt particles to be quenched or solidified in the highly subcooled water pool. This imposes a limit for the amount of molten fuel available to participate in an explosion. This also imposes a limit for the penetration depth of the melt.

- When a premixed fuel-coolant field is triggered the rapid early escalation can result in very high local pressures (> 100 Mpa) within the fuel-rich and low void region. However, these pressure waves are found to be effectively damped in regions of medium or high void fraction (in the premixture).

- Further damping occurs due to divergence of the flow field and wave reflections off the free pool surface, which mitigates the magnitude and duration of the pressure pulses on the side wall.
Finally, of major significance to Swedish BWRs, is the proposed new concept of “penetration cutoff.” Once confirmed, this will allow us to place upper limits on the depth of penetration of the premixing zone, and to demonstrate that these limits lead to rather shallow explosions, which together with the participation of “explosion venting” demonstrated in this study, will further buttress the overall case against major failures from ex-vessel explosions.

ACKNOWLEDGEMENTS

This work was sponsored by ABB Atom and the APRI Project. The APRI participants are: Swedish Nuclear Power Inspectorate; Vattenfall AB; OKG Aktiebolag; and Sydkraft.

REFERENCES


### Table 1. Key to the PM-ALPHA Runs Performed in the Set III Calculations

<table>
<thead>
<tr>
<th>Pool Depth (m)</th>
<th>Particle Size (cm)</th>
<th>Water Subcool (K)</th>
<th>Melt Superheat (K)</th>
<th>Pour Rate (kg/s)</th>
<th>Inlet Velocity (m/s)</th>
<th>Pour Diameter (m)</th>
<th>Melt Volume Fraction</th>
<th>Inlet/Outer*</th>
<th>Run I.D.</th>
</tr>
</thead>
<tbody>
<tr>
<td>10</td>
<td>0.3</td>
<td>77</td>
<td>200</td>
<td>2690</td>
<td>18</td>
<td>0.5</td>
<td>0.15/0.08</td>
<td>PM(III) - 0.3/77(200)/2.7</td>
<td></td>
</tr>
<tr>
<td>10</td>
<td>1</td>
<td>77</td>
<td>200</td>
<td>900</td>
<td>16</td>
<td>0.5</td>
<td>0.06/0.03</td>
<td>PM(III) - 1/77(200)/0.9</td>
<td></td>
</tr>
<tr>
<td>7</td>
<td>0.4</td>
<td>32</td>
<td>50</td>
<td>1700</td>
<td>11.4</td>
<td>0.4</td>
<td>0.07/0.03</td>
<td>PM(III) - 0.4/32(50)/1.7</td>
<td></td>
</tr>
<tr>
<td>7</td>
<td>1</td>
<td>32</td>
<td>50</td>
<td>1700</td>
<td>11.4</td>
<td>0.4</td>
<td>0.07/0.03</td>
<td>PM(III) - 1/32(50)/1.7</td>
<td></td>
</tr>
</tbody>
</table>

"Inner" is the central half-diameter pour area, and "outer" is the rest of the pour area.

### Table 2. Key to the ESPROSE.m Runs Performed in the Set III Calculations

<table>
<thead>
<tr>
<th>PM-ALPHA I.D.</th>
<th>Trigger Time (s)/Depth (m)</th>
<th>ESPROSE.m I.D.</th>
</tr>
</thead>
<tbody>
<tr>
<td>PM(III) - 0.3/77(200)/2.7</td>
<td>0.5/3.8</td>
<td>E.m(III) - 0.3/77(200)/2.7/0.5 (3.8)</td>
</tr>
<tr>
<td>PM(III) - 1.0/77(200)/0.9</td>
<td>0.6/3.8</td>
<td>E.m(III) - 1.0/77(200)/0.9/0.6 (3.8)</td>
</tr>
<tr>
<td>PM(III) - 0.4/32(50)/1.7</td>
<td>0.5/3</td>
<td>E.m(III) - 0.4/32(50)/1.7/0.5(3)</td>
</tr>
<tr>
<td>PM(III) - 1.0/32(50)/1.7</td>
<td>0.5/3</td>
<td>E.m(III) - 1.0/32(50)/1.7/0.5(3)</td>
</tr>
<tr>
<td>PM(III) - 1.0/32(50)/1.7</td>
<td>0.8/2.2</td>
<td>E.m(III) - 1.0/32(50)/1.7/0.8(2.2)</td>
</tr>
<tr>
<td>PM(III) - 1.0/32(50)/1.7</td>
<td>0.8/4.4</td>
<td>E.m(III) - 1.0/32(50)/1.7/0.8(4.4)</td>
</tr>
</tbody>
</table>
Figure 1. Illustration of the Induced pool motion that helps suppress voiding. Run PM(III)-1/32(50)/1.7 at 0.5 seconds. (The bottom line of the figure corresponds to 3 meters above the pool bottom.)
Figure 2. Illustration of the first appearance of voids deeply into the pool, in the pretense of significant subcooling. Run PM(III)-1/32(50)/1.7 at 0.9 seconds. (The bottom line of the figure corresponds to 1 meter above the pool bottom.)

liquid volume flux/steam volume fraction

PM(III)-1.0/32(50)/1.7

t = 0.9 s

Fmax = 687.5 cm/s
Figure 3. Localized void formation in run PM(III)-0.3/77(200)/2.7 at 0.7 seconds. (The bottom line of the figure corresponds to 5 meters above the pool bottom.)
Figure 4. Significant internal voiding with small particles, high melt concentrations, and moderate subcoolings. Run PM(III)-0.4/32(50)/1.7 at 0.7 seconds. (The bottom line of the figure corresponds to 2 meters above the pool bottom.)
Figure 5. Extensively voided premixture, and the impact on melt distribution. Run PM(III)-0.4/32(50)/1.7 at 1 second. (The bottom line of the figure corresponds to 2 meters above the pool bottom.)
Figure 6. The coolant flow pattern in the presence of significant voids. Run PM(III)-0.4/32(50)/1.7 at 1 second. (The bottom line of the figure corresponds to 2-meters above the pool bottom.)
Figure 7. Total fuel and liquid fuel mass in the pool as a function of time, for all 4 premixing runs.
Figure 8. The leading edge penetration of the total fuel and fuel with liquid fraction higher than 80%, for all 4 premixing runs.
Figure 9. Premixture configuration at the time of triggering for run E.m(III)-0.3/77(200)/2.7/0.5(3.8). (The bottom line of the figure corresponds to 5 meters above the pool bottom.)
Figure 10. Premixture configuration at the time of triggering for run E.m(III)-1.0/77(200)/0.9/0.6(3.8). (The bottom line of the figure corresponds to 5 meters above the pool bottom.)
Figure 11. Premixture configuration at the time of triggering for run E.m(III)-0.4/32(50)/1.7/0.5(3). (The bottom line of the figure corresponds to 3 meters above the pool bottom.)
Figure 12. Premixture configuration at the time of triggering for run E.m(III)-1.0/32(50)/1.7/0.5(3). (The bottom line of the figure corresponds to 3 meters above the pool bottom.)
Figure 13. Premixture configuration at the time of triggering for runs E.m(III)-1.0/32(50)/1.7/0.8(2.2) and E.m(III)-1.0/32(50)/1.7/0.8(4.4). (The bottom line of the figure corresponds to 2 meters above the pool bottom.)
Figure 14. The pressure-time history of various elevations on the side wall for run E.m(III)-0.3/77(200)/2.7/0.5(3.8). The total wall-average impulse is 48.1 kPa-s.
Figure 15. The pressure-time history of various elevations on the side wall for run E.m(III)-1.0/77(200)/0.9/0.6(3.8). The total wall-average impulse is 73.8 kPa-s.
Figure 16. The pressure-time history of various elevations on the side wall for run E.m(III)-0.4/32(50)/1.7/0.5(3). The total wall-average impulse is 12.14 kPa-s.
Figure 17. The pressure-time history of various elevations on the side wall for run E.m(III)-1.0/32(50)/1.7/0.5(3). The total wall-average impulse is 79.0 kPa-s.
Figure 18. The pressure-time history of various elevations on the side wall for run E.m(III)-1.0/32(50)/1.7/0.8(2.2). The total wall-average impulse is 95.2 kPa-s.
Figure 19. The pressure-time history of various elevations on the side wall for run E.m(III)-1.0/32(50)/1.7/0.8(4.4). The total wall-average impulse is 96.3 kPa-s.
Figure 20. Transient pressure distribution illustrating the direct venting of the explosion zone.
Figure 21. Transient pressure distribution illustrating the continuous venting of the propagating pressure pulse.
Figure 22. Transient pressure distribution illustrating the complex venting and reflection which leads to the double pressure pulse observed on the wall for a voided premixture case.
Figure 23. Transient pressure distribution illustrating the complex venting and reflection which leads to the double pressure pulse observed on the wall for a non-voided premixture case.
Figure 23. (Cont.)
POTENTIAL STRATEGIES TO CONTROL IODINE RELEASED INTO THE
CONTAINMENT IN THE CASE OF A SEVERE REACTOR ACCIDENT

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1. Introduction.

In the latest years a great research effort on iodine chemistry within containment during
hypothetical severe accidents in nuclear power plants has been carried out. This interest has
been demonstrated by extensive international participation in projects such as ACE
(Advanced Containment Experiments), whose phase B addressed the chemical interactions
of iodine within containment. ACE iodine studies along with others performed by individual
laboratories, have shown that the assumptions set forth in Regulatory Guides 1.3 and 1.4
concerning iodine behaviour are unrepresentative of what is expected in case of a severe
reactor accident.

The Regulatory Guides 1.3 [1] and 1.4 [2] assumed that 50% of the total iodine inventory
of the reactor core was released to the primary containment, 25% of which was
instantaneously available for leakage. The composition of this 25% was stated to be: 91%
elemental iodine (I₂), 5% particulate iodide and the remaining 4% organic iodides (OrgI).
These assumptions have significantly affected the design of engineering safety features,
largely optimized to remove I₂.

Almost 20 years later, NUREG-1465 [3] describes a complete different scenario for iodine.
Iodine would come into the containment progressively along the accident. Most of iodine
would enter during the first hours after the initiation of the fission product release from the
core, the so called early in-vessel and ex-vessel phases. Moreover, even though there are still
some uncertainties, it is accepted that the source of iodine into the containment would be
composed by particles of caesium iodide (CsI), with only a few percentage being gaseous,
either as elemental iodine (I₂) or hydrogen iodide (HI).

This paper is aimed at analyzing classical strategies to control iodine released into the
containment in the case of a severe accident, as well as at pointing out other potential
measures based on the current state of knowledge on iodine chemistry. It is organized as
follows. Firstly, the major boundary conditions influencing iodine behaviour within
containment are presented. Secondly, a general overview of iodine within containment in
case of a severe accident is given; specific points in which accident management strategies
could have a beneficial impact for iodine control are remarked. Thirdly, the impact of recent
research findings on engineering safety features performance is discussed. And, finally, a review of aspects specifically related to pH is carried out due to the exceptional importance of this variable on iodine behaviour.

Spanish Nuclear Regulatory Body requested CIEMAT to undertake a review of the existing literature on iodine behaviour within containment in case of a hypothetical severe accident from the accident management point of view. This work summarizes the major findings drawn to date.

2. Boundary conditions.

Iodine chemistry in containment in case of a severe accident is greatly influenced by prevailing conditions, such as pH, dose rate, temperature, etc. A comprehensive description of all of them may be found in ref. [4]. Next, ranges and significant aspects of the most relevant ones from the iodine control point of view are given.

2.1. pH.

pH is a key condition for iodine evolution in containment. A pH above 8 favours iodine ionic species (non-volatile), as long as a low pH increases the presence of volatile species in the system.

Acidity or basicity is a consequence of materials present in water pools. In systems where no pH-control chemicals exist, these materials are essentially: fission products (i.e. caesium compounds), core concrete aerosols (i.e. metal oxides and hydroxides) and compounds produced by radiation (i.e. nitric acid). Table I [5] lists the materials that determine pH in containment water pools under severe accident conditions. The most important acids in containment are nitric acid (HNO₃) and hydrochloric acid (HCl). The most important bases are caesium hydroxide (CsOH), caesium borate (CsBO₃) and in some plants pH additives, such as sodium hydroxide (NaOH) and sodium phosphate (Na₃PO₄).

TABLE I
Materials determining pH in a severe accident

<table>
<thead>
<tr>
<th>Material</th>
<th>Type</th>
</tr>
</thead>
<tbody>
<tr>
<td>Boron oxides</td>
<td>acid</td>
</tr>
<tr>
<td>Basic fission product compounds (as CsOH)</td>
<td>base</td>
</tr>
<tr>
<td>Iodine as HI</td>
<td>acid</td>
</tr>
<tr>
<td>Carbon dioxide and nitric acid</td>
<td>acid</td>
</tr>
<tr>
<td>Core-concrete aerosols</td>
<td>base</td>
</tr>
<tr>
<td>Hydrochloric acid from cable insulation</td>
<td>acid</td>
</tr>
<tr>
<td>pH additives</td>
<td>base</td>
</tr>
</tbody>
</table>

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The pH range for accident conditions is bounded between 4 and 11 [6]. Initially pH is supposed to be alkaline because of caesium compounds entering water pools. Once fission products come into pools, the associated $\gamma$ field causes formation of $\text{HNO}_3$ which tends to neutralize the pH rise provoked by basic solutes. Taking into account these offsetting effects, the maximum duration that a basic pH may be maintained in the absence of pH control additives was estimated to be between 60 and 100 h for seven accident sequences [5]. However a pH less than 7 may be attained in less than 24 h if $\text{H}_2\text{BO}_3$ or $\text{HCl}$ become a component of the water pool.

2.2. Dose rates.

Recent assessments of dose rates [7] for several sequences found that PWRs exhibit dose rates considerably higher than BWRs do. The presence of extremely large water volumes seems to be a distinct advantage for BWRs in this regard. Table II relates dose rates estimated with specific characteristics of plants studied. These calculations are based on the mass of each group of nuclides entering in containment (from STCP accident analyses) and the energy deposition rate per unit mass of each one.

TABLE II

Dose rates as a function of plants and sequences

<table>
<thead>
<tr>
<th>Plant</th>
<th>Sequence</th>
<th>Volume (m$^3$)</th>
<th>Temperature (°C)</th>
<th>Dose rate (Mrad/h)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Liquid</td>
<td>Gas</td>
<td></td>
</tr>
<tr>
<td>Grand</td>
<td>TC$\gamma$</td>
<td>4550</td>
<td>39650</td>
<td>102</td>
</tr>
<tr>
<td>Gulf</td>
<td>TQUV$\gamma$</td>
<td>5171</td>
<td>39650</td>
<td>60</td>
</tr>
<tr>
<td>Peach</td>
<td>AE$\gamma$</td>
<td>4000</td>
<td>7873</td>
<td>54</td>
</tr>
<tr>
<td>Bottom</td>
<td>TC2</td>
<td>4500</td>
<td>7873</td>
<td>118</td>
</tr>
<tr>
<td>Sequoyah</td>
<td>TBA</td>
<td>1465</td>
<td>36404</td>
<td>112</td>
</tr>
<tr>
<td>Surry</td>
<td>TMLB$^-$</td>
<td>115</td>
<td>51000</td>
<td>93</td>
</tr>
<tr>
<td></td>
<td>AB</td>
<td>172</td>
<td>51000</td>
<td>112</td>
</tr>
</tbody>
</table>

In agreement with the data previously presented and others existing in literature, dose rates can be bounded between 0.1 Mrad/h (0.278 Gy/s) and 10 Mrad/h (27.8 Gy/s) [8].

2.3. Organics.

The lower limit of organics concentration is that of their content in air [9]: $0.5 \times 10^{-7} - 10^{-6}$ M. However, there exist in containment a great variety of materials which can act as sources of organics: cables, seals, paints, etc. All these materials can decompose by pyrolysis and radiolysis and produce an organic concentration in gas phase even two orders of magnitude above the characteristic one in air.
2.4. Impurities.

Even though it has been studied the influence of several impurities on iodine behaviour (i.e. Fe and Cu ions), silver (Ag) has a singular importance as impurity due to its capability as iodine sink. It is present in large quantities in PWR cores (~2250 kg).

3. Global scenario of iodine within containment.

Iodine enters into containment essentially as CsI particles. Its total concentration ranges from $10^{-6}$ to $10^{-4}$ M [10]. Iodine reactivity distributes its total inventory in different species, whose nature determine, along with boundary conditions, system volatility.

System volatility could be expressed as the inverse of the total iodine partition coefficient, which is defined as:

$$H(I) = \frac{\sum [I]_I}{\sum [I]_g}$$  \hspace{1cm} (1)

where concentrations are given in at-g mole. Namely, system volatility is a measure of the amount of iodine in containment atmosphere. Therefore, the objective of iodine control under accident circumstances would be to reduce system volatility (i.e. to maximize $H(I)$) to levels as low as possible.

3.1. Initial species.

After assessing several accident sequences [7], it was concluded that a maximum of 5% of iodine would be present as elemental iodine and the remaining 95% would be as CsI. Likewise, the fraction of I or HI should not be lower than 1%. This distribution is largely different from that of Regulatory Guides 1.3 and 1.4.

There exist, however, uncertainties concerning RCS speciation which are recognized to be able to affect the in-containment source term of iodine [11]. Some of the most relevant ones refer to the reaction of cesium hydroxyde (CsOH) with oxide materials, the reaction of CsOH and CsI with boron compounds and the reactions of control rod materials (essentially cadmium, indium and silver) with iodine species. Although such interactions are known to occur, their extent is poorly characterized. This fact and the little influence of initial iodine speciation on long term iodine distribution in containment [12], make the aforementioned figures to be seen as reliable.

3.2. Iodine evolution.

Most of iodine, in the form of CsI particles, undergoes aerosol removal processes so that it is depleted from atmosphere into the aqueous sump. During such a depletion gaseous iodine fraction can interact with surfaces of both particles and structures. Once in the sump, Iodine forming CsI dissolves and many different species are generated as a result of iodine aqueous chemistry ($I^-$, $IO_3^-$, $I_2$, HOI, OrgI, ...). Some of them are volatile, being partially transferred to atmosphere (i.e. $I_2$ and OrgI). Those which have an ionic nature establish strong interactions within solution and their transfer is considered null. In the atmosphere, gaseous iodine species may undergo physical and chemical processes which can alter both their nature.
and the gaseous iodine inventory. Figure 1 is a simple sketch of the global iodine scenario described.

Iodine behaviour can be roughly classified in three time categories [7]. In the first (from initial release up to 17 to 20 hours) airborne aerosol reduction and physico-chemical interactions of entering gaseous species with aerosols would take place. $I_2$ formation in the sump would start in this phase and would extend along next phase (up to 2 to 3 weeks), which is identified with the aqueous chemistry of iodine and the transfer of no charged species into the atmosphere. Transformation from inorganic into organic iodine would occur during the second and third phases (long term phase). Uncertainties in iodine source term into containment only would effect the first of these three periods, but not its final distribution provided that an early containment failure does not happen.

### 3.3. Iodine speciation.

The scope of this section is restricted to a pool of fundamental chemical reactions closely related to iodine distribution between gaseous and liquid phases. A comprehensive review of most of potential reactions of iodine in containment may be found in ref. [4]. It must be underlined that HCl presence effects have not been taken into account.

In the presence of radiation water decomposes as follows:

$$4 \cdot 9 H_2 O \rightarrow 2 \cdot 7 e^- + 2 \cdot 7 OH + 3 \cdot 4 H^+ + 0 \cdot 7 OH^- + 0 \cdot 45 H_2 + 0 \cdot 75 H_2 O_2 + 0 \cdot 6 H$$ (2)

All these products interact each other or with other molecules in the pool such as oxygen ($O_2$) to yield an extent set of final products whose importance lies on their oxidizing or reducing capability. Among the most relevant are: $OH$ (oxidant), $e^-$ and $O_2^-$ (reductors). A key role is attributed to hydrogen peroxide ($H_2O_2$), whose behaviour depends on pH.

Once in the sump the global result of water radiolysis products reaction with $I$ can be expressed by an oversimplified phenomenological equation:

$$2I^- + hv \rightarrow I_2$$ (3)

This reaction, known as radiolytic oxidation of $I$, is effected by several factors: pH, iodide concentration, temperature, presence of oxygen and impurities, dose rate and dose. Among all of them, pH is particularly relevant because of the large sensitivity of Eq. 3 extent to this parameter variation (Fig. 2 [7]).

Molecular iodine ($I_2$) may undergo a fast hydrolysis reaction,

$$I_2 + H_2 O = I^- + HOI + H^+$$ (4)

whose extension is mainly a function of pH and temperature. The higher the temperature, the greater the equilibrium constant, attaining at 100 °C a value of 1.13 $10^{-10}$ M$^2$ [13]. Namely, only at very acidic pH ($\sim$ 4) most of iodine would be in the form of $I_2$, being the conversion from $I_2$ to $I$ and HOI practically complete at neutral pH (pH = 7).

Hypoiodous acid (HOI) dismutation is a slow process quite dependant on pH and
temperature:

\[ 3\text{HOI} = 2\text{I}^- + \text{IO}_3^- + 3\text{H}^+ \]  

(5)

Numerous uncertainties exist on HOI existence and some authors simply refer to an iodine species of oxidation state +1, without specifying its chemical form.

Eqs. 3 and 4 are often expressed as the total hydrolysis equation:

\[ 3\text{I}_2 + 3\text{H}_2\text{O} = 3\text{I}^- + \text{IO}_3^- + 6\text{H}^+ \]  

(6)

This reaction is strongly dependant on pH. Its equilibrium constant is of the order of \(10^{-40}\) M\(^0\) at 100 °C [13].

Iodate (\(\text{IO}_3^-\)) can be reduced by the products of water radiolysis to form \(\text{I}_2\):

\[ 2\text{IO}_3^- + h\nu \rightarrow \text{I}_2 + 3\text{O}_2 \]  

(7)

As in Eq. 3, this is a phenomenological expression that shows the overall effects of the presence of \(\text{IO}_3^-\) in an irradiated solution. It must be pointed out that relevant uncertainties exist around the chemistry of \(\text{IO}_3^-\) in irradiated solutions [14].

The volatile forms of iodine (i.e. molecular species) in solution partitionates between liquid and gas phases according to the general equilibrium equation:

\[ X_{\text{liq}} = X_{\text{gas}} \]  

(8)

being \(X\) any of the volatile iodine species. Although mass transfer is another area currently under study, it is well known that it is a process dependant on species and greatly influenced by temperature. Thus, at 19 °C the partition coefficient of \(\text{I}_2\) was measured to be 108 as long as at 92 °C this value decreases to 10.9 [15], illustrating the enhancement of \(\text{I}_2\) volatility with temperature.

Gaseous \(\text{I}_2\) can undergo an oxidation process by the ozone (\(\text{O}_3\)) resultant from air radiolysis:

\[ \text{I}_2 + 2\text{O}_3 \rightarrow (\text{I}_2\text{O}_4) \rightarrow 2\text{IO}_3^- \]  

(9)

The reaction product (\(\text{IO}_3^-\)) must be in condensed phase and, therefore, it will be submitted to aerosol phenomenology. In addition to \(\text{O}_3\), nitrogen oxides (\(\text{NO}_x\)) could act as well as iodine oxidants in the containment atmosphere.

There are large uncertainties on organic iodides (\(\text{OrgI}\)). It is still uncertain which is the major source of \(\text{OrgI}\): painted surfaces, organics in the sump or organics in the containment atmosphere. However, its formation has been shown to happen in either of these substrates: pools [16], surfaces [17] and gas phase [18]. Different reaction mechanisms have been proposed in each media. The importance of \(\text{OrgI}\) lies on their volatility (even higher than \(\text{I}_2\) for low molecular weight \(\text{OrgI}\)); namely, wherever they are formed, a substantial fraction will be eventually located in the atmosphere. The ACE experiments performed in the Radioiodine Test Facility (RTF) showed that in the presence of radiation and organic painted
surfaces (typical of reactor containments), organic iodides can dominate the gas phase [12].

In addition to its contribution as a substrate for organic iodine formation, surface can behave as a substantial sink for iodine. Reversible or irreversible nature of surface adsorption is largely dependant on surface type [19,20,21]. Besides structural surfaces, aerosols provide an additional area of the same order of magnitude available for iodine deposition; recent experiments demonstrated that incoming gaseous iodine into containment (i.e. I₂ and HI) were depleted associated to aerosols [22]. Gaseous iodine-aerosols interaction appears to be quite dependant on particles basicity.

Another important aspect concerning iodine-aerosols relation comes from the potential consequences of H₂ deflagrations, either controlled or uncontrolled, in terms of CsI oxidation to I₂. Even though it was shown the feasibility of this process [23], its phenomenology is not understood yet and evidences point out the protective effect of small amounts of steam (10%) [24].

Figure 1 summarizes the major physico-chemical processes in containment. As can be seen, unlike ionic species, molecular species are connected between gas and liquid phases by means of double arrows to remark their volatile nature. Of course, the actual iodine containment picture includes more species than mentioned, but attention is focused into the major ones.

3.4. Analysis.

From the previous section it is obvious that key points to control iodine in containment under severe accident circumstances are: the formation of volatile species in the sump, the retention of iodine within the sump and, once volatile species reach containment atmosphere, the removal of iodine from gas phase. Here are commented some aspects which could be somehow controlled or enhanced by mitigative measures to be taken under accident conditions.

In order to reduce iodine inventory available to be transformed into volatile species by aqueous chemistry, chemical substances capable to trap soluble iodine and convert it into non-soluble precipitates could be added into the sump. A possible choice could be a solution of silver nitrate (AgNO₃) which would yield silver iodide (AgI). Ag-I interaction can be explained by the following equilibria [4]:

\[ 2\text{Ag} + \text{I}_2 \rightleftharpoons 2\text{AgI (s)} \]  \hspace{1cm} (10)

\[ \text{H}_2\text{O}_2 + 2\text{Ag} + 2\text{H}^+ + 2\text{I}^- \rightleftharpoons 2\text{AgI (s)} + 2\text{H}_2\text{O} \]  \hspace{1cm} (11)

These processes have been shown to provide a means for fixation iodine in the sump, preventing the formation of both I₂ and CH₃I [25]. Therefore, the addition of adequate amounts of a soluble silver salt to the water phase before the dissolution of iodine can decrease the formation rate of volatile iodine by producing AgI precipitate (stable under irradiation).

Ozone formed by air radiolysis interacts with I₂ (Eq. 9) to produce solid oxides, I₄O₉ and
I₂O₅. Both compounds are hygroscopic and in contact with steam yield other substances which, after dissolved in water, generate IO₃⁻ anions [26,27]. The importance of this reaction is its capability to turn volatile species into condensed phases (more easily depleted). Concentration of about 0.1 to 1 µl/l would significantly attenuate I₂ inventory in atmosphere. This suggests that feasibility of O₃ (or other oxidants) introduction into the containment atmosphere during severe accidents could deserve be considered as a way to reduce gaseous iodine inventory.

4. Engineering safety features.

Engineering safety features (ESF) design was based on the assumptions taken in Regulatory Guides 1.3 and 1.4. It means that, as to their removal fission product function, they have been largely optimized to remove elemental iodine. Current view of iodine scenario in containment is certainly different from those assumptions. In this section performance of some engineering safety features related to iodine evolution within containment is assessed.

4.1. Spray system.

Containment spray system can be used to remove airborne fission products from the containment atmosphere (primarily iodine). This function may be enhanced by chemical additions to the spray for improved iodine removal. Even though spray performance depends on many factors such as drop size, height, flow rate, etc., attention here is drawn to those aspects of its effectiveness involved in iodine chemistry.

The spray system may consist of two subsystems: the spray subsystem and the additive one (absent in some designs). The latter may be attributed two functions: enhancement of fission product removal and control of sump chemistry. In both cases additive selection is a key point. In this regard the most used chemicals are (Table III [28]): dilute boric acid (H₃BO₃) buffered with NaOH, dilute H₃BO₃ with sodium thiosulfate (Na₂S₂O₃) buffered with sodium hydroxide and dilute H₃BO₃ with hydrazine (N₂H₄) buffered with tri (or di)-sodium phosphate (Na₃PO₄, Na₂HPO₄). Each of these solutions meets the functions mentioned by increasing pH, promoting high partition coefficients. Regardless spray additives, OrgI partition coefficients were initially given as a function of temperature [28]; thus, for temperatures less than 38 C H is considered to be 2.0 as long as at temperatures above 132 C it is taken H=1.0.

<table>
<thead>
<tr>
<th>Additive</th>
<th>Partition Coefficient</th>
</tr>
</thead>
<tbody>
<tr>
<td>NaOH</td>
<td>50 (pH ≤ 6.5)</td>
</tr>
<tr>
<td></td>
<td>5000 (pH ≥ 8.5)</td>
</tr>
<tr>
<td>N₂H₄ (50 ppm or greater)</td>
<td>5000</td>
</tr>
<tr>
<td>Na₂S₂O₃ (1.0 wt. %, pH &gt; 8.5)</td>
<td>100000</td>
</tr>
<tr>
<td>H₃BO₃</td>
<td>50</td>
</tr>
<tr>
<td>No additives</td>
<td>100</td>
</tr>
</tbody>
</table>
The design criteria for American Light Water Reactors (LWRs) [28] took into account elemental, organic and particulate iodine removal to assess the spray performance. Therefore, most of key species were considered. However the fact of accepting the "in-containment source term" defined in Regulatory Guides 1.3 and 1.4 overestimated the role of elemental iodine removal, treating organic and particulate iodine as species of much minor importance. As a consequence, buffer solutions behave as a very efficient sink for I₂, but not for OrgI. See H values given above for I₂ and CH₃I.

Among factors affecting the spray effectiveness removing iodine from containment atmosphere are: additive solution composition and performance mode.

Once in the chemical phase of iodine behaviour in containment (i.e. iodine in the gas phase dominated by volatile species), the spray effectiveness will be closely related to the additive solution composition and to the specific species in the atmosphere (essentially, I₂ and OrgI).

The removal of each form of iodine from the atmosphere by containment spray can be described by a factor λ defined as,

$$\lambda = \frac{FHE}{V}$$  \hspace{2cm} (12)

being F the spray volumetric flow rate; H, the instantaneous partition coefficient; V, the free volume sprayed and E, the absorption efficiency. Absorption efficiency depends on chemical species and buffer solutions; in the case of I₂ and CH₃I it is given by [28],

$$E = 1 - \exp \left[ \frac{-dK_{g}r_{s}}{d(H+K_{g}/K_{s})} \right] \quad \text{for } I_2$$  \hspace{2cm} (13)

$$E = \frac{K}{(F/V_{r}+K)} \quad \text{for } CH_3I$$  \hspace{2cm} (14)

where it is noteworthy that E is a function, among other variables, of partition coefficient (H) and reaction rate coefficient (K). Other approaches to model E have been proposed up to date [29].

Concerning molecular iodine experimental programmes carried out so far have demonstrated that iodine removal can be splitted into two phases [30]: a rapid one during which concentration goes down to 1% of its initial value and a slow one in which concentration reduction proceeds to about 0.1% at a much slower rate (Fig. 3). In addition, it was observed that solutions buffered at pH 9.5 are about three times more effective than neutral water removing I₂ [31] (Fig. 4).

Concerning methyl iodide, caustic borate and boric acid spray solutions showed a removal rate of about 0.058 h⁻¹; however, an addition of 1wt% of Na₂S₂O₃ increased the rate to 0.5 h⁻¹ [29]. This evidence can be explained by the reducing effect of Na₂S₂O₃, which in the presence of radiation fields stabilizes a triplet excited state of CH₃I, which is highly reactive and accelerates the rate of pickup of CH₃I [32]. Presence of I₂ appeared to be a favourable
factor for this process. Figure 5 shows a simple diagram of fundamental and excited states of CH₃I.

Once spray system solution is exhausted, recirculation mode begins. This change in spray operation has two implications. By one side, the liquid-gas interface between sump and atmosphere is increased, so that transfer processes are enhanced. By the other, during this operation mode it is particularly important to keep a high pH; if pH goes down to acidic values, radiative formation of volatile species becomes more effective. This latter effect becomes significantly more relevant in this operation mode due to the exchange area increase, which can eventually provoke an iodine revolatilization, increasing the iodine inventory in the containment atmosphere.

Summarizing, from the accident management point of view a couple of points should be looked at concerning iodine removal. Firstly, uncertainties may arise as to OrgI removal capability; in this regard, an additive solution buffered with Na₂S₂O₃ seems to be more convenient than other possible choices. However, Na₂S₂O₃ use has shown complications, so that hydrazine (N₂H₄) in basic solution seems to be more advisable [33]. Secondly, even though molecular iodine removal appears to be ensured, particular attention should be paid to the operation mode of sprays. Whenever an alkaline pH can be maintained no concern should exist; however, if pH decreases progressively to reach acidic values, allowance of spray recirculation should be questioned.

4.2. BWR Suppression Pool.

BWR suppression pools were not initially designed as a cleanup system; however, their capability to retain fission products such as iodine is areal and positive safety factor [34]. An important point to remark is that pool suppression design requirements do not include any measures for pH control.

Most of materials entering BWR suppression pools are expected to be in the form of aerosols, being vapours a small fraction of the total incoming amount. This is applicable for iodine, so that iodine vapour scrubbing is expected to be a less important phenomenon than in the case of sprays.

A large number of studies have been addressed to demonstrate the efficiency of suppression pools for particle removal and, in particular, for CsI retention in ponds [35]. In this regard, pools behave as an effective cleaning system where, once scrubbed, iodine undergoes the chemical processes already presented in section 3.3. As a general observation it can be quoted that mass transfer between iodine in the input gas and the pool is very fast and equilibrium or quasi-equilibrium was attained in most circumstances [36]. Table IV [37] shows that under typical conditions of accidents, DFs (i.e. ratio between incoming and leaving mass in the pool) greater than 10² should be generally expected.

Few experiments have studied fission product vapour retention in suppression pools up to date and most of them were essentially focused on I₂. Elemental iodine retention was shown to be dependent on chemical reactivity in the pool, increasing pool decontamination factors at iodine concentrations below 10⁻⁴ M and in the presence of additives such as Na₂S₂O₃. Methyl iodide retention was observed to may be affected by considerable mass transfer resistance in the liquid phase which could limit its decontamination factor. In general, it can
be stated that penetration of I₂ and CH₃I in pools at high concentrations (>10⁻⁵ M) is governed by their partition coefficients, as long as lower concentrations enhance their elimination from gas flowing through pools [38]. In Figure 6 it may be seen that I₂ DFs were found to be higher than CH₃I DFs for the same conditions.

**TABLE IV**

Experimental decontamination factors under accident conditions

<table>
<thead>
<tr>
<th>Test</th>
<th>Mix flow (cm³/s)</th>
<th>Steam Fraction</th>
<th>AMMD (μm)</th>
<th>α</th>
<th>Discharge Regime</th>
<th>DF</th>
</tr>
</thead>
<tbody>
<tr>
<td>RT-SB-12/13</td>
<td>420</td>
<td>0.11</td>
<td>3.0</td>
<td>0.14</td>
<td>Bubble</td>
<td>444-702</td>
</tr>
<tr>
<td>RT-SB-08/09</td>
<td>382</td>
<td>0.38</td>
<td>0.55/3.5*</td>
<td>0.43</td>
<td>Bubble</td>
<td>16-20</td>
</tr>
<tr>
<td>RT-SB-04/05</td>
<td>388</td>
<td>0.58</td>
<td>3.4</td>
<td>0.21</td>
<td>Bubble</td>
<td>168-169</td>
</tr>
<tr>
<td>RT-SB-00/01</td>
<td>385</td>
<td>0.90</td>
<td>3.4</td>
<td>0.16</td>
<td>Bubble</td>
<td>129-254</td>
</tr>
<tr>
<td>RT-SB-14/15</td>
<td>384</td>
<td>0.15</td>
<td>5.8</td>
<td>0.17</td>
<td>Bubble</td>
<td>52-53</td>
</tr>
<tr>
<td>RT-SB-10/11</td>
<td>411</td>
<td>0.35</td>
<td>7.2</td>
<td>0.02</td>
<td>Bubble</td>
<td>677</td>
</tr>
<tr>
<td>RT-SB-06/07</td>
<td>376</td>
<td>0.56</td>
<td>4.2</td>
<td>0.02</td>
<td>Bubble</td>
<td>419-858</td>
</tr>
<tr>
<td>RT-SB-02/03</td>
<td>453</td>
<td>0.87</td>
<td>5.0</td>
<td>0.08</td>
<td>Bubble</td>
<td>567-922</td>
</tr>
<tr>
<td>RT-SC-0/02</td>
<td>2095</td>
<td>0.10</td>
<td>1.7</td>
<td>0.25</td>
<td>Jet</td>
<td>116-128</td>
</tr>
<tr>
<td>RT-SC-P/01</td>
<td>2031</td>
<td>0.11</td>
<td>5.6</td>
<td>0.18</td>
<td>Jet</td>
<td>491-526</td>
</tr>
<tr>
<td>RT-MB-01/02**</td>
<td>3024</td>
<td>0.11</td>
<td>4.1</td>
<td>0.06</td>
<td>Bubble</td>
<td>1273-2913</td>
</tr>
</tbody>
</table>

α, particle fraction < 1μm
* bimodal distribution
** Multiorifice test

One of the major uncertainties as to suppression pool performance as an iodine sink is the potential for iodine re-evolution [3]. In case of failure or controlled venting of a containment, sumps can undergo an intense evaporation and, in extreme cases, can become saturated and boil to dryness. Experiments carried out so far [39,40] have demonstrated that iodine revolatilization is influenced by pH, radiation and impurities; Table V shows that at low pH and in the presence of radiation, iodine evaporation can be substantial.

Volatilization of iodine occurs as well during fast depressurizations (typical of late containment failures). LACE-LA6 experiment [41] showed that reentrainment of material in the pool should be expected under these circumstances. Iodine dissolved in droplets could, as in the case of sprays operating in the recirculation mode, behave as a source of volatile iodine whose intensity is primarily dependant on pH of drop solution. This suggests that iodine trapping in the pool as an unsoluble compound (as suggested in section 3.4) could make more difficult iodine revolatilization in any circumstance.
TABLE V

Iodine volatility during evaporation to dryness of 10^4 M CsI solutions

<table>
<thead>
<tr>
<th>Test Condition</th>
<th>Percent Volatile</th>
<th>Test Condition</th>
<th>Percent Volatile</th>
</tr>
</thead>
<tbody>
<tr>
<td>Initial pH</td>
<td>Borate (mg/l as H$_2$BO$_3$)</td>
<td>No radiation</td>
<td>Dose 2.1 Mrad</td>
</tr>
<tr>
<td>4.4</td>
<td>2000</td>
<td>7.2</td>
<td>&gt;99</td>
</tr>
<tr>
<td>4.4</td>
<td>None</td>
<td>2.0</td>
<td>3.6</td>
</tr>
<tr>
<td>6.0</td>
<td>2000</td>
<td>6.8</td>
<td>55</td>
</tr>
<tr>
<td>7.0</td>
<td>2000</td>
<td>32</td>
<td>3.0</td>
</tr>
<tr>
<td>Pure water</td>
<td>None</td>
<td>1.6</td>
<td>21</td>
</tr>
<tr>
<td>9.0</td>
<td>2000</td>
<td>1.8</td>
<td>22</td>
</tr>
</tbody>
</table>

Another important uncertainty concerning suppression pools is related to their behaviour at long term. Even though pools have been demonstrated to be an effective method of decontamination, absence of pH control requirement in their design prevent their consideration as a final sink. Namely, it is possible that after scrubbing significant amounts of iodine and other fission products during early phases of an accident, radiolysis is able to lower the pH to acidic values, enhancing iodine volatilization. Therefore, credit should not be given for long-term retention of iodine in suppression pools, unless pH above 7 can be ensured.

In summary, even though iodine is effectively removed when it passes through suppression pools, they should be seen as a potential source of volatile iodine during late phases of an accident, unless measures are devised in case of an accident to maintain an alkaline pH in the solution. In addition, if in the course of an accident special actions such as containment depressurization are to be adopted, it should be borne in mind the potential iodine reevolution from pools, increasing iodine inventory available to escape to the environment.

4.3. Ice Condensers.

Ice condenser is accepted as a clean-up system for elemental iodine removal if the ice contains a quantity of sodium hydroxide, sodium tetraborate or other proposed chemical additive sufficient to assure that the water solution from ice melting has a pH of at least 9.0 but not more than 11.0 [42]. If the ice, after melting but prior to any dilution, meets the pH requirements the system is considered effective for elemental iodine removal. An efficiency of 30% per pass for elemental iodine is assigned to systems similar to those of Sequoyah plant. On the contrary, the system is considered ineffective for organic and particulate iodine removal.

Experiments carried out demonstrated that ice surfaces could be seen as a means of elemental iodine removal [43]. Even at acid pH 80% of iodine injected was removed from pure steam
and figures even greater than 95% were found as an alkaline additive was used. A substantial difference between acid and basic runs were that as long as iodine trapped in acid solutions could be available for stripping by air flow, in basic melts it was fixed completely. This was explained as a consequence of the complete hydrolysis reactions that iodine undergoes under alkaline conditions. The figures given above become smaller when air accompanies steam flow. Figures 7 and 8 show the influence in I$_2$ trapping of pH and ice loading; there is a break in the dependence with increasing loading, but while acidic ice levels off at 49% recovery, alkaline ice reaches 68% at the break and then follows with a gentle slope up to reaching an efficiency of 78% at 48 in.

In the same pool of experiments 4 tests were made with CH$_3$I. In the absence of air flow it was found a recovery of 5.7% in acid ice and of 6.6% in alkaline ice. These recoveries dropped to values below 1% for air flow runs. This indicates that CH$_3$I hydrolysis is a slow process unless redox agents are present; a recovery of 10.4% was measured when sodium sulfite was added.

Particles (CsI) suspended in the gas flow through ice condensers are scrubbed rather efficiently by steam condensation and, if particles are large, by gravity settling. Therefore, iodine particulate phase (i.e. the initial phase in which iodine is supposed to enter containment as aerosols) would be shortened, making CsI aerosol reach containment sumps more rapidly than under natural depletion phenomena. Once in the containment sump, iodine behaviour follows aqueous iodine chemistry.

Volatile species decontamination factors have been shown to obey the following relationship [44],

\[
DF = 1 + \frac{V_d A_d}{G} + \frac{L H}{G}
\]  

where $V_d$ is the mass transfer coefficient, $A_d$ the deposition area available, $G$ the exit volumetric flow rate of gas, $L$ the exit volumetric flow rate of liquid and $H$ the partition coefficient. This equation holds for both I$_2$ and CH$_3$I, but $V_d$ and $H$ are different in each case. This expression was used in a theoretical study carried out to assess the efficiency of ice condensers removing iodine during a TML sequence in Sequoyah plant [45]. It was found that during a typical accident sequence when ice is present, DFs range from 1 to 2 for CH$_3$I, from 50 to 600 for particulate CsI, and even greater than 10$^5$ for I$_2$ (if meltwater is alkaline).

Current understanding of iodine chemistry makes CH$_3$I removal inefficiency in ice condensers an important point. If ice condensers operate as expected, at the long term stages of an accident in ice condenser plants gaseous iodine inventory may be dominated by organic species. A possibility of enhancing OrgI elimination from the gas flowing through ice baskets would be to use reducing agents as chemical additives dissolved in alkaline solutions.

4.4. Air filtration and adsorption units.

The design of air filtration and adsorption units are based on the specifications given by Regulatory Guide 1.52 [46], which in turn is based on the source term assumptions taken in Regulatory Guides 1.3 and 1.4. Then, in view of new source terms [3], attention should be
paid to the adsorption units performance to retain Orgl and to the large amount of iodine in the form of particles that filtration units are supposed to retain.

The ESF atmospheric cleanup systems consist essentially of some or all of the following components: moisture separators, heaters, prefilters, HEPA (High Efficiency Particle) and iodine adsorption units. The purpose of moisture separators and heaters is to remove most of water from the gas stream, reducing the relative humidity to prevent moisture trapping by the filters and adsorbers. Prefilters and filters are installed to remove particulates; prefilters retain large particles preventing the excessive loading of HEPA filters, as long as HEPA filters remove fine particles which avoids fouling of adsorbers. Adsorbers remove gaseous radioiodine from the air stream.

Charcoal beds of adsorbent units are usually impregnated with triethylenediamine (TEDA) and/or potassium iodides (KI). Charcoal adsorber beds can be design to remove from 90 to 99% of I\(_2\) and from 30 to 99% of the Orgl, depending upon the specific train design [46].

During normal reactor operation, charcoal filters undergo a slow but continuous degradation due to their exposure to moisture and various air contaminants. Furthermore, conditions expected during hypothetical severe accidents (temperatures, humidity, radiation fields, etc.) were seen as potentially threatening for filters performance [47]. As a consequence a great deal of theoretical and experimental work was carried out to demonstrate filter capabilities under accident circumstances and to find the most effective material for iodine retention.

Several experiments were performed under different conditions [48]. Temperature and relative humidity were shown to cause a reduction in the efficiency both of TEDA and KI impregnated charcoal filters, being relative humidity influence the most relevant (Fig. 9). Unlike these effect, radiation was demonstrated to improve slightly weathered charcoal filters efficiency (Fig. 10 [49]); however, the local temperature rise resultant from iodine decay heat can cause a volatilization of impregnant and a subsequent reduction of removal efficiency of the charcoal by about an order of magnitude [48].

Despite the degrading effects commented above, DF's measured for charcoal beds impregnated with TEDA or KI were observed to be large (10\(^4\) to 10\(^6\)) and independent of temperature and relative humidity for bed depths of 20 cm [48].

From previous discussions it seems that gaseous forms of iodine are expected to be retained in air filtration units quite effectively. Nonetheless, there still remain some uncertainties concerning their performance under accident conditions. First, the possibility of HEPA filters plugging due to the large amount of radioactive and nonradioactive aerosols expected (no accounted for in refs. [2] and [3]). Second, the potential volatilization which could occur within the air filtration unit: CsI retained as particles in HEPA filter or demisters could turn into I\(_2\) under the influence of strong radiation fields existing in the filter (10\(^5\) Gy) increasing notably the absorbent unit load. Third, the potential volatilization of iodine previously trapped if H\(_2\) deflagrations occur. And, finally, the influence of gaseous chlorine species on the capacity and retention of iodine species; although there are no experimental data on this topic, it is possible to predict the interaction between HCl (strong acid) and impregnants of charcoal beds (alkaline compounds) and, hence, reducing radioiodine removal efficiency of the charcoal; what is more, even if no reaction took place, HCl could decrease efficiency by occupying charcoal beds surface sites.
From the accident management point of view it should be considered the possibility of switching off atmosphere filtration units during the aerosol input into the containment because of high efficiency of natural removal mechanisms is expected during these accident phases; this way HEPA clogging could be prevented and, subsequently, radiation fields within filter would be lower. Likewise, the effect of undesirable high relative humidity level expected during spray operation could be minimized if filters are turned off during sprays operation. Finally, a good mixing of containment atmosphere to prevent H₂ deflagrations should be an additional concern from the iodine revolatilization from both filters and other sinks; in this regard, spray operation should be accompanied by an atmosphere mixing measure to avoid potential stratification which would allow high H₂ concentrations in the upper regions of containment.

5. pH Control.

In section 2.1 an overview of how pH evolves during severe accidents has been given. More information about sources and release magnitudes of materials determining pH may be found in ref. [5]. Here attention is drawn to buffer solutions and to the key role of pH in severe accident scenario.

From the iodine point of view, pH control must be one of the priorities in accident management for mitigating potential consequences of severe accidents. In order to control pH, buffer materials exist in some plant designs. Analysis of buffer capacity indicate that borate and phosphate are the most suitable chemical additives in terms of pH range and chemical stability [5].

The phosphate is typically in the form of trisodium phosphate (Na₃PO₄) that is held dry in baskets that are located in baskets that are located in containment. Borate buffers are made up of sodium hydroxide and boric acid that are stored separately and combined in spray systems. Boric acid would also be used in sprays in passive pH control of trisodium phosphate. Two potential difficulties may be cited with regard to passive trisodium phosphate: its removal due to precipitation as calcium phosphate as it interacts with calcium hydroxide or calcium oxide from core concrete aerosols, and its possible reaction with CO₂ during storage.

At the end of ACE programme a critical assessment of the status of the data base on iodine behaviour in severe accidents with respect to its adequacy for predicting/controlling the distribution of iodine in containment was carried out [50]. It identified under what circumstances existing uncertainties are critical to safety. Table VI summarizes the major conclusions drawn from this work.

It is noteworthy that pH is a crucial factor, along with containment integrity, for iodine volatility in containment. In other words, if pH is controlled during severe accidents only a few technical issues not fully understood yet could become important from the safety standpoint; some of them as iodine-surface interactions or resuspension (revolatilization) have been commented along this paper. However, if pH becomes uncontrolled, uncertainties existing in almost every technical issue become relevant, being pH evolution one of the most important of these critical areas.
### TABLE VI

Assessed adequacy of the current data base for calculating iodine volatility in containment during severe accidents

<table>
<thead>
<tr>
<th>Issues</th>
<th>pH Controlled</th>
<th>pH Uncontrolled</th>
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<tr>
<td></td>
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<tr>
<td></td>
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</tr>
<tr>
<td></td>
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<td>A/B</td>
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<tr>
<td>Organic iodide reactions</td>
<td>A</td>
<td>B</td>
</tr>
<tr>
<td>Aerosol interactions</td>
<td>A</td>
<td>A/B</td>
</tr>
<tr>
<td>Structure Interactions</td>
<td>A</td>
<td>B</td>
</tr>
<tr>
<td>Aerosol behaviour</td>
<td>A</td>
<td>A</td>
</tr>
<tr>
<td>Revolatilization</td>
<td>A</td>
<td>B</td>
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<tr>
<td>Impurity Effects</td>
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<td>A</td>
</tr>
<tr>
<td>Mass transfer, Redox, T</td>
<td>A</td>
<td>A</td>
</tr>
<tr>
<td>Radiation dose &amp; pH</td>
<td>A</td>
<td>B</td>
</tr>
<tr>
<td>Partitioning data</td>
<td>A</td>
<td>A</td>
</tr>
<tr>
<td>High Temperature</td>
<td>A</td>
<td>B</td>
</tr>
</tbody>
</table>

A, existing uncertainties in the data base are not critical to safety and source term calculations
B, some of the uncertainties in the data base are critical to safety and source term calculations
C, many of the uncertainties in the data base are critical to safety and source term calculations

An assessment of chemical forms of iodine in containment was carried out for seven severe accident sequences in LWR plants [7]. One of the major findings was the direct relation between volatile iodine production and pH levels of the water pools (Fig. 11). Failure to control the pH at or above 7 was calculated to be able to increase I₂ in the atmosphere up to 80000% for BWRs and up to 33000% for PWRs, as compared with the case where pH was controlled.

### 6. Conclusions.

Along this paper a review of traditional accident management strategies to trap and retain iodine within containment during severe accident conditions in sight of current knowledge of iodine chemistry has been done. Likewise, it has been remarked some specific iodine reactions through which an improvement of iodine depletion could be achieved. Next, a set of conclusions are presented in order to summarize the major findings of this work:
Current knowledge on iodine phenomenology in terms of both "in-containment" source term and chemistry points out the need of a reassessment of engineering safety features (ESF) involved in fission product clean-up function in case of a severe accident.

Among all boundary conditions affecting iodine chemistry within containment, pH is acknowledged as a key parameter determining iodine evolution. Even though introduction of chemical additives to control pH was taken into account in the design of some ESFs, recently it has been found that some materials within containment can behave as sources of acids which may threaten the buffer capability of such additives.

Radiation fields resultant from fission product decay has shown to be a crucial factor for iodine volatility as well. It has been found that large volumes of water cause a decrease in dose rates and, subsequently, in the amount of gaseous iodine produced in pools. Therefore, addition of large volumes of water into containment seems to be an advisable means of reducing volatile iodine concentration in containment atmosphere.

Unsolubility, stability and formation feasibility in γ fields make silver-iodine compounds (AgI) be seen as a potential strategy to control not only iodine volatility but iodine revaporization as well. Such behaviour of AgI precipitate as a final sink for iodine suggests the possible convenience of introducing a soluble salt of silver in the sump before iodine input to trap it definitively.

Introduction of oxidizing agents in containment atmosphere once airborne iodine is dominated by gaseous species may lead to form condensed compounds which are more easily depleted following aerosol physics. A feasibility analysis on such reactions enhancement appears to be worthy; however, attention should be paid to possible influence on other phenomena such as H₂ deflagrations and detonations.

The importance acquired by organic species of iodine in recent studies remarks the need of a reassessment of chemical additives used in some ESFs (sprays, pools, ice condensers, etc). In this regard, reducing agents in basic solutions have been seen as a good choice; however, again attention should be given to other associated complications involved in their use. Substances such as hydrazine or thiosulfate have been suggested in literature.

Recirculation mode operation of sprays could become inconvenient if pH in pools is not maintained high. Enhancement of surface exchange area in this operation mode could lead in case of acidic pH to promote iodine partitioning between droplets and atmosphere. Hence, if no pH control can be assured it should be considered the possibility of no spray operation in recirculation mode.

In spite of high decontamination factors found experimentally in subcooled pools, it should be kept in mind that iodine chemistry can make iodine re-evolve from pool to atmosphere in long term unless pH is kept alkaline. This
indicates the convenience of coupling scrubbing and chemistry to assess pool retention efficiency. Therefore, attempts to turn pools into a final sink for iodine would require to adopt additional measures such as enhance formation of iodine precipitates or ensuring high pH all over the accident.

Both controlled and uncontrolled containment depressurization can yield a substantial resuspension and revolatilization of iodine. Saturation pools are much less effective removing fission products than subcooled ones. The fact of becoming airborne means that surface exchange is increased and, subsequently, iodine partition is promoted. Accident management strategies conducted to depressurize containment should consider this impact on iodine.

Performance of air filtration and adsorption units within containment is submitted to large uncertainties: particle overloading of HEPA filters, CsI revolatilization from particulate filters by radiolysis or pirolysis and so on. Despite these uncertainties, it seems that a length increase of adsorption units would minimize them partially.

Formation of chlorine compounds, which are highly reactive, are supposed to compete with iodine species in several processes. This could alter substantially the current view of iodine scenario in containment in many aspects: organic iodine formation, pools pH, filter efficiency trapping iodine, etc. Therefore, in order to carry out a more accurate ESF reassessment a peer analysis of the potential impact of chlorine on iodine phenomenology is thought to be worth.

REFERENCES.


Figure 1. Chemical scenario of iodine within containment under accident circumstances.

Figure 2. Radiolytic conversion of I to I₂.
Figure 3. Concentration of $I_2$ during run A-4 of CSE programme.

Figure 4. Influence of pH in iodine removal from air by water spray.
Figure 5. Excited states of CH$_3$I

Figure 6. Removal of iodine compounds from steam/air mixtures passing through pools.
Figure 7. Effect of boric acid ice loading on iodine removal.

Figure 8. Effect of sodium tetraborate ice loading on iodine removal.
Figure 9. Effect of temperature and relative humidity on TEDA charcoal bed retention.
Figure 10. Effect of weathering on charcoal bed retention.

Figure 11. Atmospheric elemental iodine release as a function of pH.
Session 3:

Survveillance and protection of containment function.

Chairman: B. De Boeck
Selection of scrubber as a filtered venting device.

B. Hamnér, Kärnkraftteknik AB, Sweden

To be presented at the OECD specialist meeting on selected containment severe accident management strategies, Stockholm 13th-15th June 1994.

Abstracts

Since the TMI accident occurred all Swedish nuclear power plants have been equipped with passive operated containment venting devices, to retain radioactivity in case the containment must be vented in an accident.

In 1980-85 a stone condenser was raised at one of the plants, aimed as the main mitigations tool in case of an accident.

In 1986-88 high performance scrubbers were raised at the remainder power plants. They may be operated automatically but are regarded as back up tools beside other mitigations methods and strategic maintenance.

The development from a passive perspective into a more accident case directed outlook was carried out by extensive investigations on severe accident phenomenology.

The article gives a brief description on considerations being made, leading to adoption of scrubber technology, and a description on how the view on severe accidents was developed in Sweden during the 1980 decade.

The TMI accident

Following the TMI accident in 1979 the Swedish government required that filtered venting should be constructed for the Barseback power plant in south of Sweden.

The Barseback plant, including two BWR's each of 1800 MWth, is located in a relatively densely populated area, with a distance of 20 km to the Danish capital Copenhagen.

For the other Swedish power plants it was decided that research and investigations should be carried out on severe accidents and that recommendations should be submitted to the government a few years later, regarding suitable mitigation actions.

Therefore three other major projects, named RAMA\(^1\), OKG- RAMA and MITRA\(^2\), were started simultaneously with the condensor FILTRA project in Barseback.

The RAMA project was a joint project between the Swedish authorities and power companies, defining the knowledge base and validating codes used for analysing severe accidents. The MITRA and OKG- RAMA were projects in which the utilities made plant specific analysis.

Based on these studies, the view on how to conduct analysis and mitigate accidents was radically developed, from a pronounced passive into a more active attitude regarding mitigation efforts.

\(^{1}\) RAMA = Reactor Accident Mitigation Analysis.

\(^{2}\) MITRA = Mitigation of Reactor Accidents.
Considered accident cases in 1980.

Barsebäck was the first plant in the world being equipped with a containment pressure relief system. Consequently one had to define events leading to both containment over pressure and release of radioactivity.

To achieve these consequences a design base accident was defined comprising large LOCA, total blackout (loss of AC-power) and impaired Pressure Suppression (PS/BWR) function. It was also assumed that the operator personnel remained passive for 24 hours and that by this time the AC power was back on line, restoring residual heat removal.

This combination of events, each with low probability, will undoubtedly lead to releases of radioactivity as well as possibly containment rupture due to over pressure, provided mitigation actions is not being made.

First generations filter.

Based on these assumptions a huge passive operated condensor was raised, sized in order to take care of steam and gases containing radioactivity.

Gross volume amounts to 10'000 m³ respectively the net gas volume is in the order of 4'000 m³, filled with gravel stones to cool and condense emerging steam with. Due to its size, the condensor has became a common and general symbol for accident mitigations measures.

A summary of efficiency regarding particle retention is given in figure 1. Obviously efficiency rate increases at condensing conditions while it diminishes at non condensing operation condition.

Improved knowledge on severe accidents.

BWR accidents starting with pipe rupture and reduced PS- function leads to very fast pressure build up within containment and consequently pressure release will have to be operated within seconds. However the time duration for this case is very short and therefore the containment can be closed again a few minutes later.

Within the RAMA- project it was clarified that in such a short time core damages can not occur even if further complications would occur simultaneously, such as total loss of AC- power.

Therefore pressure release regarding cases with reduced PS- function will not have to be carried out as filtered venting, however it is suitable to close the containment wall when pressure have been released.

Considering on the other hand, PWR and BWR accidents including reduced decay power cooling capacity, pressure increases much slower. Therefore increased pressure is expected not to threaten containment integrity within several hours or days after the primary event.

Meanwhile released radioactive aerosols will be deposit in the containment.

To conclude, it has been stated that different accident cases are separated in time regarding plant integrity consequences. Therefore mitigation measures can be directed and adjusted in different ways.
Radioactivity transportation and land contamination.

From the RAMA- projects, the knowledge were much improved concerning radioactivity release from melting fuel and the transportation and retention in primary systems and containment. For instance, it was clarified that most activity is released in the form of small particles (aerosols) and that release rates, for different matters, strongly depends on the core temperature.

The most important substances leading to land contamination are iodine and caesium. Both these substances are volatile and are mostly in the chemical form CsI and CsOH. The ground dose rate is dominated the first month by the relatively shortlived iodine isotopes and thereafter the caesium isotopes.

This was confirmed by the Tjernobyl accident, which contaminated parts of Sweden with caesium for a range of years. Actually, this activity is still measured.

Noble gases play a special role since they can not be retained or else affected. But consequently they don't contribute to land contamination and can therefore only cause damages during the acute situation.

Calculations shows that high but not lethal doses can occur close to a plant if noble gases would be emerged shortly after shut down. Within a distance of some kilometres from the plant evaluated but low doses, due to dilution in the atmosphere, may occur.

It should be pointed out that early radioactivity release to the atmosphere, within hours after shut down, is not regarded as a realistic case but rather a "failure of strategy ".

To conclude, in 1985 much more was known than some years earlier, regarding radioactivity behaviour and possible environmental consequences, following for example a station black out accident. This is important, when making investment decisions as well as strategy plans on mitigation actions.

Conclusions and requirements made in 1986

The stone-bed filtered venting was implemented 1985 in Barseback power plant. In the beginning of 1986 the government issued the requirements for the remaining power plants at Oskarshamn, Ringhals and Forsmark, totally 7 BWR's and 3 PWR's.

These requirements included fundamental efforts which, in accordance with the power companies recommendations to the government, should be made primarily. These recommendations included efforts such as,

- BWR's should be provided with possibility to release pressure shortly after shut down, in order to extend margins at fast pressure build up accidents (pipe rupture combined with reduced PS-function).

- Water supply underneath the reactor should be submitted in all dry containment's in order to protect the concrete floor if the core should melt through the reactor vessel bottom (total loss of AC-power).

- Methods and strategies should be developed regarding restoring residual heat removal and suitable accident management.

These efforts were judged to be the most important in order to keep containment integrity for several hours after shut down.
Furthermore the government required that pressure relief systems should be adopted and designed so that operation could be made manually or automatically. They should give guarantees for limited release of radioactivity within specific limits given by the government (0.1 % from the core inventory in a 1800 MW reactor regarding Cesium isotopes 134 and 137).

Filter technology

Generally

Considering the increased knowledge about radioactive aerosols behaviour and transportation, it was in 1986 possible to reflect on many filter alternatives, including conventional technology.

Initially cyclone technology was proposed, which rather soon guided into venturi technology and other so called wet filter methods. Reasons are illustrated in picture 2, showing that conventional technology very well matches the specifications made for the project.

It should be noticed that for industrial purposes, construction and operation costs are often predicting and restricting the efficiency level. Therefore one could be rather convinced that better capacity would be obtained when developing filters without these restrictions.

Wet filter design considered.

Due to reasons like those given above, a range of designs were proposed on wet technology generally and on venturi technology especially. A few typical examples are showed in picture 3:

Figure 3a shows a scrubber containing venturies submerged in water. The venturi is operated by the containment pressure and water is supplied from the surrounding pool. This is important since proper operation at total AC- power failure is required. Figure 3b and 3c shows two other interesting examples, where the filter devices have been integrated with the containment.

Pool- bubbling and spraying towers are examples of other wet filter methods considered. It was then observed that pool bubbling actually occurs when steam blows down into the cooling water pool in BWR containment's and that the containment itself works as a spray tower when the containment's spray system is operating.

As a consequence the containment spray systems have been modified and can now be operated from mobile diesel driven pumps or from the fire extinguishing system.

Dry methods considered

In 1986 so called wire mesh- filters were being tested and developed in Germany by the nuclear research center in Karlsruhe (KfK). The concept is that lots of thin metal wires are closely crowded in boxes, likewise conventional ventilation's fibre filters. Very good efficiency have been obtained.

This type of filter fell off early due to that Germany, by the time, stated that pressure release efforts would not be necessary within 24 hours after shut down. However Swedish analyses predicted that automatic release could occur within 5 to 8 hours.

The French sand filter concept also became dispatched early since it, based on the same 24 hours philosophy as in Germany, was at the time outlined for efficiency limited to 90 %. This was insufficient to meet with the Swedish requirements.
The Barsebäck stone-condenser concept was also accurately considered. It was not judged to be competitive regarding costs and efficiency, although a much smaller version was studied compared to the Barsebäck condenser. But with a volume of only 2000 m$^3$ and being connected to plants with more than 3000 MW$_{th}$ power, the condensing capacity was much reduced compared to the condenser raised in Barsebäck.

Conclusions

Following studies on alternative filter types and designs, the power companies agreed on some common decisions, guiding the ongoing projects:

- Some kind of wet filter scrubbers should be adopted due to foreseen advantageous regarding efficiency, volume and costs. They should be located outside but close to the containment's, considering that erection would have to be carried out during plant operation.

- The same kind of technology should be applied in order to facilitate component standardisation as well as the licensing procedure, so that project time could be saved.

- Reactor manufacturers should be chosen as head suppliers since basic knowledge on containment behaviour and interaction with connected equipment is of concern for proper operation.

Based on this common platform tenders on water scrubbers were given by two reactor manufacturers which was compared and evaluated.

**Scrubber design and function.**

The scrubber which was later on chosen is named Mult Venturi Scrubber System (MVSS); picture 4. It was developed in Sweden, based on Swedish patents.

It contains over 800 venturi tubes submerged in a circular shaped container. The tubes are arranged and connected in a special way in order to permit operation within wide ranges of flow rates at maintained high efficiency. This is important since it allows manual operation at possibly low flow rates as well as automatic operation at high gas or steam release rates from the containment.

A venturi tube can be regarded as a special designed pipe. Inside the tube there is a pinch and nearby there are a number of holes drilled in the tube wall, all special designed.

At operation gas is forced by the containment pressure to pass through the tubes and small amounts of water is sucked into the tubes from the surrounding pool. Due to the pinch the gas is accelerated to very high speed, typical 100 m/s. As it collide with the water, the water is "atomized" into a cloud of small droplets with typical size of 50 to 150 μm.

Due to the combination of high relative speed and fine sized water droplets, the particles in the gas are easily collected on the droplets, influenced by high values of inertia forces.

The water droplets containing radioactive particles are then easily captured in the tubes and in the water pool covering the tubes. In the water pool further retention mechanisms are going on, such as diffusion and chemical reactions.

Placed above the pool on top of the scrubber, there is another filter device which prevents water droplets which may be released from the pool surface, to reach the environment. It contains the same kind of gravel bed as in the condenser in Barsebäck. However in this case it's operating on comparatively very large water droplets.
The venturi function has been known since over 100 years and the technique has been utilised within metallurgical and other process industries for many decades.

**Verifications program**

In order to verify performance, extensive calculations and laboratory tests were carried out in 1987/-88. Different conditions was simulated regarding gas and water flow rates, - compounds, - temperatures and other important operations conditions.

The calculations, carried out with different mathematical models, showed that decontamination factors exceeding several hundreds up to several thousands should be expected.

Calculations also indicated that lowest performance would be expected for high density medias which corresponds to release of plain nitrogen gas from containment. This is conservative since other, lighter, gases will have to be present during pressure relief operation.

Only particle capture due to inertia forces within the venturi tubes was considered in the calculations. Consequently, even higher values of decontamination factors was expected to show up in the laboratory tests.

Many of the laboratory tests were performed in a test rig containing one single full scaled venturi tube. This test rig was designed so that real operating conditions could be properly simulated for any one of the real scrubber tubes. Thus, the full sized scrubber could be verified at full scaled conditions.

The laboratory tests confirmed as expected that very high performance should occur at a real operation, with DF ranging from at least 500 for BWR- cases and 1500 for PWR- cases up to tens of thousands.

It should be pointed out that the aim of the verification was not to predict real performance at real operation, but to prove that decontamination above requirements in all cases would be achieved, that is 100 for BWR- cases and 500 for PWR- applications.

Therefore only required decontamination, and not the very high values emerging from the performance tests, were considered when the consequences of the radioactive release was calculated.

**Acknowledgement**

I would like to direct many thanks to the Swedish state power board and to OKG AB who sponsored the article and made analysis documentation's available.
Shutoff valve
Gravel filter:
Volume 10,000 m³
D = 20 m
H = 40 m
Stone size 25-35 mm

Schematic drawing – filtered venting of reactor containment.

Total penetration = mass of penetrating particles
mass of injected particles

x) Particles = Test aerosol with median mass-equivalent
diameter of 2.0 micrometers.

Influence of condensing steam on penetration of particles through a gravel column
with a length of 7.5 m at different gas velocities.

315
**DUST COLLECTOR CHARACTERISTICS**

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**Flygaska från kolpulverad penna**

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**Kvalitetsgrad**

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**Partikelstorlek µ = 1/1000 mm**

**Diagram:**

- Ultramikroskopsiska partiklar
- Mikroskopsiska partiklar
- Partiklar synliga för blotta ögat

**Filter:**

- Cycloner > 8 mm
- Sändcykler < 150 mm
- Slägghylsor
- Desintegratorer
- Yesterlovsfilter
- Elektr filtration

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A. Scrubber with venturis and manifolded plate, creating small bubbles. Retention by inertia and diffusion.

B. Nozzles operated by pressure in containment. Off-gasses through main chimney.

C. Pools and venturis connected in series. Off-gasses through main chimney.
INTRODUCTION
In the event of an accident arising in an EDF (Electricité de France) pressurized water reactor, the IPSN Emergency Technical Centre (CTC) would be entrusted to provide the safety authorities with technical assistance. As part of this responsibility it would analyse and forecast the barriers status (fuel clad, reactor coolant system, containment building) and the related safety functions (subcriticality, water inventory, primary pressure and temperature control, confinement). Its assignments would also require it to evaluate the kinetics and magnitude of a possible fission products release, in progress or to come, so as to advise necessary counter measures in order to ensure the surrounding population's protection, should the occasion arise.

A realistic assessment of the release requires a good knowledge of the containment quality. It is therefore important to detect potential isolation faults (isolation failure or by-pass of the third barrier) as soon as possible. Once these leaks are identified, the plant operator will set required corrective arrangements rapidly. The CTC would follow up his actions closely. If none of them happen to be effective, containment leakage would be taken into account when estimating releases.

Work of the emergency team lies indeed in two main points:
- early localization of isolation failure or containment by-pass,
- following up of the plant operator actions meant to remedy the problem.

In order to meet these two requirements IPSN has developed an expert system named ALIBABA. Part one presents its advantages, whereas part two describes the expert system. Part three deals with the software environment and part four offers an example of the help provided by ALIBABA.

ADVANTAGE OF AN EXPERT SYSTEM
In order to detect containment leaks, CTC emergency teams have the following information at their disposal:
- report of the containment building isolation valves position,
- global activity measurements in the ventilation ducts and at the stack,
- local activity measurements in the auxiliary buildings (measurements above sumps or near pipes).

In a simple case such as a local activity alarm triggering, the task may be achieved manually with the help of piping and instrumentation diagrams. Let us consider a threshold overshoot on an activity measurement set above the sump of the containment spray system pump room during recirculation stage. The cause of the alarm triggering would probably be a leakage on one of the system lines. The isolation of the faulty train
and the use of the other train could then be recommended, provided the leakage position and the systems' availability are satisfactory.

On the other hand using a manual method may become very hard when more complicated situations occur. Following are three illustrations of this statement.

A first instance could concern a single unleaktight penetration with a leakage towards the process drain sump of the auxiliary service building. Numerous penetrations would then be suspected and make the assessment uneasy.

The second situation deals with the failure of several isolating valves connected to different penetrations. Let us also imagine the triggering of ventilation duct alarms because of high activity. The questions to be answered are not as simple as those previously raised. Are the isolating valve failures connected with the high activities detected by duct sensors? If so, which are the possible penetrations inducing the contaminated fluid release? What actions should be taken? Several information sources must be confronted in order to answer these questions, and the task may prove slow and tedious, therefore hard to be carried out in case of emergency.

The third example is a similar situation with loss of electrical power supplies. Before asking the same questions, one should think about the effects of power source losses on measurements (sensor supplies), and on the interpretation of these measurements (fan supplies).

In any of these three cases answers are not immediate. Information about the supply of systems, the location of circuits and measurements, and the connections between systems and complementary isolation means are needed. In order to provide better performance, expert system ALIBABA was developed by IPSN with the collaboration of the Direction for Nuclear Reactors (Atomic Energy Commission, CEA).

**DESCRIPTION OF EXPERT SYSTEM ALIBABA**

**Knowledge base**

Data of expert system ALIBABA concern penetrations (potential leakage sources) and systems (instrumentation and various equipment).

**Penetrations**

Penetrations of the reactor building can be divided into two categories, electrical penetrations and mechanical penetrations:

Electrical penetrations have cables running through, and are continuously pressurized with reliability. Moreover they are checked at regular intervals. Their likelihood to leak is low. They are therefore not taken into account by the expert system.

Mechanical penetrations have pipes running through, and have usually an isolating valve placed on each side. Because these components may prove defaulting, they are considered by ALIBABA in its knowledge base. However some of the penetrations were not selected because of their too low probability of outwards leakage:

- penetrations inside which the fluid advances towards the containment building with a pressure high enough and maintained with sufficient reliability (buffer tanks, supply redundancy, etc.) to ensure a very low probability of a reversing of the fluid direction;
- penetrations directly connected to steam generators (feedwater, steam lines and blowdown lines). Such lines associated to a fluid coming out of the reactor building possess activity measurements (loss of coolant accident detection) that may identify them easily, should release occurs;
- withdrawn, standby or plug obturated penetrations;
- penetrations used by the accumulator test lines, because of the high number of isolating components on the sampling tap line.
All in all ALIBABA models 76 penetrations, from which 48 are bound by a containment isolation signal. Each penetration of the knowledge base is described on the following pattern:
- relevant elementary system,
- connected isolating valves and types of the valves,
- features of the fluid traversing through (nature, source, temperature),
- isolation class,
- diameter.

Important components (pumps, filters, etc.) are described downstream from the isolating valve, each of them being located in a room of the plant. By default, the research takes into account leakage paths with only three closed isolating valves placed in series. Beyond this point, a leakage is very unlikely to occur. All systems can therefore be displayed in form of dynamic schemes, the component status depending on the plant unit configuration. The following sketch sums up the description:

**Systems**

Fifteen elementary systems are described in ALIBABA. Three types of modelled elements are inserted in the knowledge base:
- **Electrical sources**

Electrical panels providing ventilation systems and activity measurements with power supply are modelled in the expert system. However no relationship is established between the different switchboards.
- **Ventilation**:
  Modelled systems consist of the ventilation of the BAN (nuclear auxiliary building), of the BW (peripheral building where containment penetrations are established) and of the BK (fuel building). Each following feature is described for all systems:
  - number of parallel lines,
  - fans with their electrical supply and capacity,
  - nominal volume flow rate.

- **Instrumentation**:
  The first instrumentation type known by ALIBABA concerns activity measurements. They are divided in two categories: measurements linked to a ventilation system and measurements connected to a special place.
  The first category gathers activity measurements at the stack (normal and accidental ranges) and measurements in ventilation ducts that are scanned on a cyclic basis. Information associated to this type of activity measurement consists of:
  - the different rooms controlled by the measurement,
  - the ventilation system onto which the measurement is carried out,
  - the power supply of the sensor,
  - the measuring range,
  - the accuracy.
  The second category consists of measurements placed at sensitive points, for instance above sumps or along pipes. Information associated to this type of activity measurement is the following:
  - the component near the measurement point,
  - the power supply of the sensor,
  - the measuring range,
  - the accuracy.

  The second instrumentation type known by ALIBABA is the flow rate measurement at the plant stack. Relevant information in the knowledge base is the following:
  - the power supplies,
  - the measuring range,
  - the accuracy.

  The expert system models as well:
  - even and uneven plant units,
  - the two safety systems, safety injection and containment spray systems, either in waiting state, direct injection or recirculation,
  - the reinjection of contaminated primary wastes.

**Method of assessment**
The method of assessment consists of the four following stages. It starts up with the checking of the availability of equipment and sensors, in order to prove the information the expert system has at its disposal. The next three tasks make up the strictly speaking leakage search ("upstream" approach, "downstream" approach and resulting qualitative balance). Two other tasks provide additional information in order to detect potential release through non instrumented channels and to restore isolation.

**Availability checking**
A component may become unavailable during the accident, either for mechanical reasons or as a consequence of an electrical supply loss. A component can therefore be forced unavailable with the system interface (in case of a mechanical failure for instance), or electrical supplies can as well be forced unavailable. In this late case, all components dependant on the one supply source become unavailable as well. When in addition a
ventilation system undergoes consequences because of the component unavailability, the following rules are adopted:
- if ventilation is partially available, activity measurement in the duct is valid,
- if ventilation is unavailable, activity measurement is assumed out of order too.

Yet the expert system user may choose to search for leakage paths taking into account all measurements, even those found unavailable.

"Upstream" approach
This approach consists in checking the state of containment penetration isolating valves. These data were input by using containment isolation reports based on travel stops of the isolating valves that stand on each side of the penetration. ALIBABA assumes any penetration, for which at least one isolating valve has been found unclosed, as possible leakage path.

"Downstream" approach
This approach is based on activity measurement use. Depending on the situation, it ensures the link:
- between an activity measurement and the room it comes from through the ventilation system, in case of a sensor assigned to duct scanning or in case of a measurement at the stack,
- between a measurement and a component, in case of a local measurement (above a sump or near a pipe).

It then proceeds towards penetrations, according to circuits described in the knowledge base.

Qualitative balance
Qualitative balance is the synthesis of the two former tasks. Penetrations selected by previous processes are exclusively binned into three categories (a penetration cannot belong to several categories). The first category (red colour on the computer screen), first to be referred to, holds all penetrations selected by the "upstream" approach (at least one isolating valve unclosed). The second category (orange colour on the computer screen) consists of penetrations found by the "downstream" approach, due to the detection of local measurement activity. Finally, the third category (yellow colour on the computer screen) draws up the list of all penetrations selected because significant activity was detected in ventilation ducts or at the stack (scanning measurements or measurements at the stack).

Measurements inside each category are binned according to a priority order with the following criteria:
- a coefficient is assigned to each penetration and is incremented any time the penetration is selected by search paths,
- the value of the increment factor depends on the reason why the penetration was selected.

It is worth:
+10 when selection occurs through the "upstream" approach,
+5 when selection occurs as a consequence of activity detection by a local measurement,
+1 when selection occurs as a consequence of activity detection by a measurement in a ventilation duct or a the stack.

A last classification takes into account the fluid nature (gaseous or liquid), and its source (potential contamination degree).
Results are presented in tabular form where each column (red, orange or yellow) contains the selected penetrations binned according to priority order. When selecting the penetration number on the computer screen, the diagram of the associated systems appears in the graphic window (see example in appendix). This technique enables the
user to see the penetration in its environment, and if necessary to advise or to follow up the corrective actions being undertaken by plant operators.

**Quantitative balance**

Quantitative balance may come as a complement of qualitative balance in order to detect potential leaks through non instrumented paths. It consists of a volume balance and of an activity balance. The two following equations summarize the principles:

\[
\sum Q_i + Q_{nc} = Q_{ch}
\]

\[
\sum Q_i A_i = Q_{ch} A_{ch}
\]

with:
- \(Q_i\) representing flow rate in ventilation \(i\),
- \(Q_{nc}\) representing ventilation flow rate in transfer channels without activity measurements,
- \(Q_{ch}\) representing volume flow rate at the stack,
- \(A_i\) representing activity in duct \(i\),
- \(A_{ch}\) representing activity measured at the stack.

The following scheme sums up the various elements the balance takes into account:

After analysing these two balances, the expert system reaches the following conclusions:
- if the activity rate at the stack is greater than the one in the ventilation ducts, presence of activity is assumed in non instrumented areas,
- if the activity rate at the stack is lower than the one in the ventilation ducts, direct leakage to the atmosphere is assumed.

Three important points must still be mentioned in order to understand fully the assessment made by ALIBABA:
- ventilation systems must be completely available in order to avoid any uncertainties about flow rate values in ventilation ducts and at the stack;
- the expert system controls the consistency of data from the user. When a value happens to be below (above) the measurement lowest (highest) threshold, the system considers the threshold value instead of the registered value;
- the software takes into account uncertainties concerning measurements of volume and activity flow rates.

**Restoration of isolation**
After identifying the isolation fault the plant operator must suggest and carry out containment restoration. He is thus led either to repair the defective isolating component, or to find another one that can be operated. It may be a manual or automatic valve, usually opened, which can be closed. The expert system user can follow up this research since a certain amount of complementary isolating components are described in the knowledge base, although they do not appear on schemes. Selecting a circuit section on the figure displays the list on the screen.

**SOFTWARE ENVIRONMENT**
Expert system ALIBABA was developed and is now operating with the assistance of software environment SPIRAL, which enables artificial intelligence techniques representation. This CEA developed environment was written in C language for portability purposes. Its formalism provides Object Representation (classes, inheritance, etc.). ALIBABA is implemented on work stations.

**APPLICATION EXAMPLE WITH EXPERT SYSTEM ALIBABA**
Let us consider the case of a normal operating plant unit whose safety injection and containment spray systems run in direct injection state. And let us suppose that significant activity is detected concerning the process drain sump of the nuclear auxiliary building. In this situation activity is found through the following activity measurements:
- sensor in the room where the sump is located,
- cyclic scanning measurement of the room containing the sump,
- scanning measurement above the sump,
- measurement at the stack.

The scheme in appendix presents the information arrangement on screen, with a diagram of one selected penetration.

Faulty assumed penetrations are gathered by ALIBABA according to the binning previously mentioned:
- no penetrations appear in the red area (left part of the "traversées retenues" window) since no discordant isolating valves were selected,
- two penetrations are suggested in the orange area (middle part of the "traversées retenues" window) due to the local activity detection, and according to a maximal number of isolating valves of three in series,
- the yellow area (right part of the "traversées retenues" window) sums up all other penetrations selected because of significant activity measurements in ventilation ducts and at the stack.

In this example, expert system advantage lies on the possibility to watch on a single screen the whole information needed for a better assessment by the user. After selecting one of the suggested penetrations, the operator can follow the leakage path, determine
the fluid nature, identify components likely to be discordant and ask for confirmation from plant operators, and lastly search complementary valves that would restore isolation.

CONCLUSION
Expert system ALIBABA present version enables early detection of containment by-pass situations for PWR 900 MWe. It was shown to experts from the IPSN Emergency Response Centre (CTC) during training periods. This very first use at CTC brought back various comments which the next version will take into account. It was again successfully tested as a safety assessment tool, when analysing an incident communicated by the operating team on a gaseous waste circuit.
ALIBABA is a precious tool for emergency crisis teams. It enables them to quickly identify containment by-passes, and provides them with information about systems in form of diagrams, as well as in form of texts for specific components. It certainly eases the following up of corrective actions undertaken by plant operators in order to restore the isolation of the containment building.
Further developments are planned after a first experiment feedback. The next version is scheduled for 1994 and should provide the following improvements:
- implementation of the latest version of software environment SPIRAL (version 3),
- improvement of the man-computer interface concerning data input and display of expert system information,
- improvement of result display in order to take into account the nature of the fluid which progresses through the circuit,
- consideration of links between different electrical panels.
Development of a version for PWR 1300 MWe will be considered depending on emergency drills feedback.
CONTAINMENT LEAK TIGHTNESS AFTER SEVERE ACCIDENT

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ENEL SpA - R & D Division, Nuclear Department
Rome, Italy

Next generation Nuclear Power Plants are characterized by safety features more advanced than current generation and will be even safer with some improvements on present design. After a severe accident, plant operators must be sure that containment isolation occurs and consequently leak tightness of containment comply with design criteria to limit dose to the population within design value for the whole duration of accident. Enhancement on plant monitoring allows the plant operators to be early informed, after the start of accident, of the isolation devices leak tightness. Taking into account the radioactive isotopes released and their physical properties in terms of diffusion and plate-out inside the containment, the paper discusses some ideas about methods to inform the plant operators of proper containment isolation system performance. Such information is essential to perform possible simple recovery actions.

1. INTRODUCTION

The correct management of a severe accident in the NPPs involves some aspects not yet completely known or very complex to control. Therefore it needs supplementary systems in order to improve plant control and to decide immediate recovery actions. In this context we are working to define a monitoring system suitable to reveal a possible failure in containment leak tightness by radionuclide detection. During this preliminary phase a set of parameters and working hypotheses has been identified to select a release radionuclide spectrum for a correct and effective input to appropriate instrumentation of the system. A detailed analysis of the containment penetrations and the related isolation system components to interface with the system has been carried out and eventually a conceptual idea of the system has been conceived with particular reference to the ALWR Plants. A further detailed program, including possible experimental tests, will be performed to verify the feasibility of the system.

2. ACCIDENT SOURCE TERMS

Source term evaluation under severe accident has been considered of great interest after TMI accident and a continuous research efforts was made to better understanding the fission products transport and release. Recently the NRC issued a draft report (NUREG 1465) [1] in which the results of previous researches have been reviewed and a definition of accident source terms for application in the future LWRs is proposed. Given our interest mainly in the ALWR Passive Plants, we assume, for our purpose, such data right for the in-containment release considerations.

As stated in the said report, the severe accident phases may be standardized, for the same family of reactors, independently from the plant design as indicated in the following list:
1. Coolant Activity Release
2. Gap Activity Release
3. Early In-Vessel Release
4. Ex-Vessel Release
5. Late In-Vessel Release.

The fission product composition was determined by computer code elaboration, 54 radionuclides were considered and divided into 9 major groups (STCP: Source Term Code Package) taking into account their similar chemical behaviour, as shown in table 1.

<table>
<thead>
<tr>
<th>Group</th>
<th>Element</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Xe, Kr</td>
</tr>
<tr>
<td>2</td>
<td>I, Br</td>
</tr>
<tr>
<td>3</td>
<td>Cs, Rb</td>
</tr>
<tr>
<td>4</td>
<td>Te, Sb, Se</td>
</tr>
<tr>
<td>5</td>
<td>Sr</td>
</tr>
<tr>
<td>6</td>
<td>Ru, Rh, Pa, Mo, Tc</td>
</tr>
<tr>
<td>7</td>
<td>La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y</td>
</tr>
<tr>
<td>8</td>
<td>Ce, Pu, Np</td>
</tr>
<tr>
<td>9</td>
<td>Ba</td>
</tr>
</tbody>
</table>

The activity release into containment has been evaluated by reactor type (PWRs and BWRs) as fractions of the core inventory and subdivided into the phases of the accident, considering also the studies and the results reported in NUREG 1150 [2]. The release rate during the phase period is assumed to be constant and the "Coolant Activity Release" has been omitted because it is not significant for our purpose. The "Early In-Vessel" release phase, and then before the reactor vessel failure, is strongly affected by RCS pressure that produce aerosol retention during the core degradation in case of high value. On the contrary, low pressure causes a rapid aerosol release without a significant retention. Probabilistic analysis of accident
sequences has been carried out for various BWRs and PWRs and the results indicates that a significant fraction of the cases examined occurred at low pressure. For these reasons the source term configuration has been chosen in condition of low pressure in the RCS during the core degradation; They are listed in table 2, including the duration of each phase.

In our case we refer to a typical PWR Passive Plant and in table 3 are shown the core inventory and the design basis coolant activity.

### Tab. 2 PWR Release into Containment (*)

<table>
<thead>
<tr>
<th>Phase</th>
<th>Gap</th>
<th>Early In-Vessel</th>
<th>Ex-Vessel</th>
<th>Late In-Vessel</th>
</tr>
</thead>
<tbody>
<tr>
<td>Noble Gases</td>
<td>0.05</td>
<td>0.95</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Iodine</td>
<td>0.05</td>
<td>0.35</td>
<td>0.29</td>
<td>0.07</td>
</tr>
<tr>
<td>Cesium</td>
<td>0.05</td>
<td>0.25</td>
<td>0.39</td>
<td>0.06</td>
</tr>
<tr>
<td>Tellurium</td>
<td>0</td>
<td>0.15</td>
<td>0.29</td>
<td>0.025</td>
</tr>
<tr>
<td>Strontium</td>
<td>0</td>
<td>0.03</td>
<td>0.12</td>
<td>0</td>
</tr>
<tr>
<td>Barium</td>
<td>0</td>
<td>0.04</td>
<td>0.1</td>
<td>0</td>
</tr>
<tr>
<td>Ruthenium</td>
<td>0</td>
<td>0.008</td>
<td>0.004</td>
<td>0</td>
</tr>
<tr>
<td>Cerium</td>
<td>0</td>
<td>0.01</td>
<td>0.02</td>
<td>0</td>
</tr>
<tr>
<td>Lanthanum</td>
<td>0</td>
<td>0.002</td>
<td>0.015</td>
<td>0</td>
</tr>
</tbody>
</table>

(*) Fraction of the core inventory

### Tab. 3 Core Inventory and Coolant Activity

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Core Inventory</th>
<th>Coolant</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>(Ci)</td>
<td>(μCi/g)</td>
</tr>
<tr>
<td>Kr85m</td>
<td>1.4E+07</td>
<td>5.8E-01</td>
</tr>
<tr>
<td>Kr85</td>
<td>5.4E+05</td>
<td>2.0E+00</td>
</tr>
<tr>
<td>Kr87</td>
<td>2.7E+07</td>
<td>3.2E-01</td>
</tr>
<tr>
<td>Kr88</td>
<td>3.8E+07</td>
<td>1.0E+00</td>
</tr>
<tr>
<td>Xe131m</td>
<td>3.9E+05</td>
<td>5.5E-01</td>
</tr>
<tr>
<td>Xe133m</td>
<td>1.6E+07</td>
<td>5.0E+00</td>
</tr>
<tr>
<td>Xe133</td>
<td>1.1E+08</td>
<td>8.0E+01</td>
</tr>
<tr>
<td>Xe135</td>
<td>2.2E+07</td>
<td>8.4E-02</td>
</tr>
<tr>
<td>Xe135m</td>
<td>3.6E+07</td>
<td>2.3E+00</td>
</tr>
<tr>
<td>Xe138</td>
<td>9.1E+07</td>
<td>1.5E+01</td>
</tr>
<tr>
<td>H2O</td>
<td>8.4E+05</td>
<td>2.7E-03</td>
</tr>
<tr>
<td>H2O2</td>
<td>8.0E+07</td>
<td>3.0E-01</td>
</tr>
<tr>
<td>H2O3</td>
<td>8.0E+07</td>
<td>5.2E-01</td>
</tr>
<tr>
<td>H2O4</td>
<td>1.1E+08</td>
<td>5.2E-01</td>
</tr>
<tr>
<td>H2O5</td>
<td>1.2E+08</td>
<td>1.2E-01</td>
</tr>
<tr>
<td>H2O6</td>
<td>1.0E+08</td>
<td>3.5E-01</td>
</tr>
<tr>
<td>Cs131m</td>
<td>9.0E-06</td>
<td>2.6E-01</td>
</tr>
<tr>
<td>Cs133m</td>
<td>2.8E-06</td>
<td>6.2E-01</td>
</tr>
<tr>
<td>Cs137</td>
<td>6.1E-06</td>
<td>1.9E-01</td>
</tr>
<tr>
<td>Te129m</td>
<td>4.7E-06</td>
<td>1.9E-03</td>
</tr>
<tr>
<td>Te131m</td>
<td>8.4E-06</td>
<td>3.1E-03</td>
</tr>
<tr>
<td>Te132m</td>
<td>7.9E-07</td>
<td>3.2E-02</td>
</tr>
<tr>
<td>Sr90</td>
<td>5.3E-07</td>
<td>4.8E-04</td>
</tr>
<tr>
<td>Sr90m</td>
<td>4.4E-06</td>
<td>2.0E-05</td>
</tr>
<tr>
<td>Ru103</td>
<td>8.9E-07</td>
<td>5.9E-05</td>
</tr>
<tr>
<td>Ru106</td>
<td>2.9E+07</td>
<td>0.0E+00</td>
</tr>
</tbody>
</table>

3. SOURCE TERM APPLICATION IN MONITORING SYSTEM

To use in effective manner the accident source terms it is necessary to evaluate the activity released to the containment, identify the limitations and fix some hypotheses of scenario evolution. As first step the specific activity has been calculated referring to the free volume of a typical containment (about 5.4·10^4 m^3). The activity versus the isotope is shown in the fig. 1 for each phase of the postulated accident and in the table 4 are listed the relevant values. We can observe the peaks of Noble Gases and Iodine in the phase "Early In-Vessel" (from 30 minutes to 1.8 hours
after the start of the accident sequences). Some experts on the matter support the opinion that in radiological assessments of DBA for evolutionary and passive reactors it's sufficient to consider only the two phases relevant "Gap" and "Early In-Vessel" releases. The inclusion of the "Ex-Vessel" and "Late In-Vessel" has been considered unduly conservative since the estimate frequencies of such scenario are low enough to appear not credible.

Therefore for our purpose we deemed sufficient to adopt the same schema as it results more conservative for the activity detection and for possible recovery actions (see tab. 4 and fig. 2).

Following the analyses performed it's necessary to point out some aspects that highlights limitations on the use of considered source terms and set up some working criteria:

- The values on the NUREG 1465 represents conservative data for ALWR reactors although more realistic than the past evaluations.
- In the evolution of the accident sequences the perfect homogeneization of the radionuclides in the containment atmosphere has been supposed.
- The Noble Gases are released completely to the containment atmosphere.
- On the basis of specific considerations, the iodine released to the containment should include:
  95% of that is very soluble in the pool water or plates out on wet surfaces in ionic form.
  a.
  b. Iodine 4.75% in elemental form.
  0.25% in organic form. The last two forms should contribute completely to the diffusion into the containment.

In conclusion we can affirm that, in the worst conditions for the detection, the percentage of Iodine in the containment should be the 5.95% of the total. 

_Aerosol deposition_

The main pathway of fission product transfer is the aerosol form that includes both solid particulate and fine droplets. The deposition mechanisms are quite complex (gravitational settling, diffusiophoresis, thermophoresis, diffusion, etc.) and depend on the physical properties of the medium. The gravitational settling, as natural deposition, require 4 to 10 hours, whereas containment spray system reduces the concentration by an order of magnitude in about 15-20 minutes. In the plants examined in NUREG 1150 the aerosol removal rate has been calculated and varies from 0.18 to 0.20 h⁻¹. In passive plants, complying with EPRI/URD requirements [4], the containment spray system is not included therefore in our case we consider only natural deposition in a period of 5 hours.

---

### Tab. 4: Typical Source Terms of Passive PWR

<table>
<thead>
<tr>
<th>Phase</th>
<th>Gap 0.5 h</th>
<th>Early In-Vessel 1.3 h</th>
<th>Ex-Vessel 2 h</th>
<th>Late In-Vessel 10 h</th>
<th>Gap + Early In-Vessel</th>
</tr>
</thead>
<tbody>
<tr>
<td>Isotope</td>
<td>Inventory Activity Fraction (µCi/cm³)</td>
<td>Activity Inventory Activity Fraction (µCi/cm³)</td>
<td>Activity Inventory Activity Fraction (µCi/cm³)</td>
<td>Activity Inventory Activity Fraction (µCi/cm³)</td>
<td>Activity Inventory Activity Fraction (µCi/cm³)</td>
</tr>
<tr>
<td>Kr85m</td>
<td>0.05 1.3E+01</td>
<td>0.95 2.5E+02</td>
<td>0.00 0.0E+00</td>
<td>0.00 0.0E+00</td>
<td>2.6E+02</td>
</tr>
<tr>
<td>Kr85</td>
<td>0.05 5.0E+01</td>
<td>0.95 9.5E+02</td>
<td>0.00 0.0E+00</td>
<td>0.00 0.0E+00</td>
<td>5.0E+02</td>
</tr>
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4. IDENTIFICATION OF "TRACING" RADIONUCLIDES

To detect a possible containment leak tightness failure in accident conditions, it was necessary to find discriminating elements. It was reasonable to identify such elements as a sub-set of the total radionuclide spectrum; it assumes a role like radioactive tracers in the industrial uses. In this context we tried to minimize the errors due, in particular, to the uncertainty of the involved parameters. The selection of the fission products for the purpose is based on the assumptions mentioned in the previous paragraph and, moreover, the following physico-chemical properties, significant for detection, has been considered.

- **Volutility of radionuclides**: Low volatile radionuclides (e.g., Ba and Sr) give a scarce contribution in the containment atmosphere and consequently have a low probability to diffuse in the piping where a failure of leak tightness occurred.
- **Gamma Energy**: Generally the photons with energy lower than 0.4 MeV have been excluded.
- **Significant half-life**
- **High ratio accident/coolant activity**
- **Significant decay probability**, etc.

Considering the uncertainty associated to the accident conditions, it is advisable, for radiation detection, to refer to a sub-set of the activity values included in a band delimited by an upper and lower curve limit. As lower limit we consider the coolant activity, whereas the upper limit is calculated by the initial postulated activity reduced by the quantity due to deposition, decay, plate-out, etc. The graph of said sub-set and those relevant the upper limit spectrum are shown in fig. 3 and in fig. 4 respectively; the numerical values are listed in tab. 5. In these results we can observe that the major contribution is represented by Kr88 that has good characteristics to be detected (high energy gammas, high activity ratio, etc.).

In conclusion we can affirm that the described configuration represents an initial orientation on a possible accident spectrum to be detected. More detailed modelling and calculations will allow a more wide and precise information. In any case if the activity values fall, at least, into the band of fig. 3 and the outline of the real spectrum is similar to those of fig. 4, there is an high probability of a containment leak tightness failure.

5. CHARACTERISATION OF THE CONTAINMENT PENETRATIONS

In this paragraph the containment penetrations of a typical pressurized ALWR and the related isolation system components are analysed. All the containment penetrations have been subdivided into 5 categories according to their characteristics. The first-line isolation valves (internal and external) and the possible backup valves inside and outside the containment walls are considered, including the power operator they use. The backup valves can become very important in case of the first-line valve failure because they allow the remote isolation of a faulted line. If a backup valve exists on the external side, it could be actuated also manually allowing the recovery of a failed open line, depending on the valve location, the
radiation intensity in its vicinity and its own accessibility. Manual actuation of these valves using robotised devices could also be provided. Moreover possible recovery actions for the specific penetration has been envisaged keeping in mind some considerations. The first-line valve isolation failure can occur for two reasons: mechanical failure to close or loss of closing signal. If the first failure type occurs it is very difficult that the valve can be closed manually even if it's accessible by the local operator (i.e. a leak due to imperfect closure or a loss of valve tightness is considered almost impossible to recover). If the failure is due to a signal loss, the operator can try to close the valve remotely from the control room and in case of impossibility the valve could be closed by local operator.

Fig. 3: Monitoring Radionuclide Sub-set

Fig 4: Monitoring System Gamma Spectrum
If another valve exists in series to the outside containment isolation valve, it will be possible to backup the isolation using this valve. This back up can be remote if the second valve is remotely operable or local if the valve can be only manually operated. Therefore three situation of recovery may occur: No backup means that it is not possible to recover the loss of isolation through the use of a backup valve. The remote or local actuation of the first line valve to recover the failure is still possible depending on the kind of valve failure and its accessibility. Remote Backup means that valves in series to the first-line ones exist and are remotely operable. Local Backup means that valves in series to the first line ones exist outside the containment but they are operable only manually by the local operator if accessible. In the following the more common types of penetrations are listed.

5.1 Penetrations with at least a normally closed valve (or a check valve) inside and a normally closed isolation valve outside the containment

They have at least two isolation valves (one inside and one outside the containment) that are normally closed during operation. These penetrations include systems that are in stand by conditions or are not continuously running during normal operations, such as the Residual Heat Removal System or the Air Sample System. Many of these penetrations have also in series valves inside, outside or both sides of the containment.
that allow the backup of the isolation functions. These penetrations are better manageable because they are normally closed (several of them are, however, open after a reactor scram and during accidents) and, if open before the request of isolation, they have a higher probability to be correctly closed.

5.2 Penetrations with normally open valves inside and outside the containment

These penetrations have two isolation valves normally open during operation related to systems that are running in normal reactor operation, such as the Component Cooling Water System or the Chemical & Volume Control System. The valves belonging to this category require to change status upon isolation signal, and therefore they are susceptible to stack open especially for the lines belonging to system in operation for long time periods.

5.3 Penetrations with a check valve inside and a normally open valve outside the containment

These penetrations have a check valve inside the containment that stop the fluid passing through it in the in-out direction and a normally open isolation valve outside. The difference from the penetrations of point 5.2 is only related to the nature of the check valves that do not require active intervention for their closure. The reasoning about the higher possibility of stack open failure made in the previous paragraph can be repeated here, because these line too belong to system in service during normal reactor operations.

5.4 Penetrations with closed system inside containment

This category includes the penetrations of those systems that can be considered "Closed Systems" as described in the GDC 57. These systems are generally connected to the Steam Generators or related to the containment pressure instrumentation. These instrumentation, in particular, consists of containment pressure sensors to be considered as sealed system with bellows seals inside the containment, liquid filled capillaries between the seals, and the sensing element outside containment. These lines are close systems both inside and outside containment and they are designed to withstand the containment pressure and temperature conditions following a loss of coolant accident and the related dynamic effects.

5.5 Other Penetrations

Other types of penetrations are used for servicing of the internal containment systems during maintenance operation. They include, in addition to maintenance access, equipment access and personal access hatches, the containment leak rate test and the fuel transfer channel.

All these penetrations are normally sealed and, while they are little susceptible to leaks, they are also impossible to recover.

6. CONTAINMENT ISOLATION M-MIS

Adequate means to allow the operator to manage the function of containment isolation shall be provided by the M-MIS. These means include the information related to the containment isolation function status and to the single penetration conditions in terms of valve positions and radioactivity levels inside the piping. Moreover they indicate the controls to perform remote action if required and indication of local manual action when necessary and possible.

The initiation of the isolation valves closure is actuated by several signals which occurs after most of the accidents considered in the safety analysis. Generally speaking, the various containment isolation valves are closed on parameters indicative of the need to isolate. For our purpose the signals which actuate the containment isolation system do not assume the same importance in respect to the main function. In fact, the discussion of the present document is related to events that occurs later in the accident evolution. In particular, the initial scenario adopted in the following discussion is the occurring of a severe accident, with a large amount of core radioactivity released inside the containment due to a core melt and primary circuit failure, in presence of a containment isolated.

At this moment the operators are following the evolution of the accident through their normal control room workstations. One of the several displays that are accessible to them to perform the monitoring functions is related to the containment isolation system. This screen, that is dynamic, in the sense that it provides real time updated information and is capable to accept commands by the operator, is represented, as an example, in fig. 5.

The real operator screen will be obviously in colours, allowing a better understanding of all the elements contained in it. In this figure the operator can monitor the containment isolation system and the principal information provided are:

- Penetration number
- Isolation valve status
  - Power-operated valves (i.e., air, motor, electro-hydraulic or solenoid, have position indication in the main control room).
  - Manual isolation valves that have primary or backup isolation function should be provided of limit switch with indication in control room.
- Primary Plant parameter indication
  - This information is not relevant during severe accident, but it is reported in this screen because it is one of the screen that the operator has available during normal operation to monitor the isolation valve status during tests or other activities.
  - These indications include the four channel status of each parameter that is necessary to actuate the containment isolation functions.
- System interested by the various penetrations.
The operator, while looking the screen has the possibility to zoom to a specific system connected to a given penetration by clicking the screen cursor inside the related white arrows depicted on the sides of the containment.

If during the evolution of the severe accident a given penetration leaks, high radioactivity level will result in both the penetration piping and in the room that accommodates the faulty penetration related system. This occurrence, on the basis of source term spectrum described in par. 4, will generate tightness failure alarm that will result in highlighting and flashing the arrow related to the affected system in addition to audible alarm. After the operator has acknowledged the alarm in the usual way, or by clicking the flashing arrows, a new screen depicting the system affected by the containment isolation leak will appear.

In fig. 6 an example of this screen is given, with reference to the Normal Residual Heat Removal System. This screen, that is also re-callable by the operator during normal operation, is subdivided in three parts. The upper one contains the system P&ID, enough detailed for the specific purpose. On the side of the P&ID there are the alarms connected with the system, that in the specific case are:

- **High NRHR Room Radiation**
- **High NRHR Containment Inlet Radiation**
- **High NRHR Containment Outlet Radiation**
- **Isolation Valves Not Closed**

One or more of these alarms will be obviously lighted and the component (valve) that has caused the failure will change colour and flash. At the same time, on the left of the P&ID a flashing box will appear indicating the valves that should be operated to try to stop the leak. In the example the motor operated valve RNSV022, that is the valve affected by fault, will be indicated as the target of the operator action. For this purpose, in the bottom right part of the screen the control for that valve is appeared together with the related position. Using the “Close” button the operator shall try to close completely the valve. If the action succeeds the leak should stop. Otherwise the valve is stack in that position and the only possibility to stop the leak is to manually close the valve V005A and B. Therefore the operator will address the local personnel to perform this action compatibly with the environmental conditions.

In the bottom left side of the screen is reported the logic diagram of the alarm system related to the affected penetration system. This diagram can be useful to determine the right cause of the failure.

### 7. OUTLINE ON RADIATION DETECTION

To detect the radionuclides, as radioactive "tracers", to reveal the tightness failure through the penetrations analyzed above, we have to consider detectors with an energy resolution sufficient to discriminate the photons of interest in the spectrum. Said detectors shall operate in harsh conditions and then shall possess particular characteristics. The modern industry on the field has recently realized detection systems for the purpose that can be cooled by remote cryogenic devices in place of liquid nitrogen tanks below the measuring head. That make the operation and the maintenance less difficult in restricted accessibility areas. The spectrum evolution analysis, moreover, could give more indications or confirmation of the accident sequences into the containment.

### 8. CONCLUSIONS

Considering the aspects examined in the previous points, we deemed advisable to carry on with this research activity. In fact the proposed containment monitoring system by radioactivity detection may give further information very useful to the operators to manage also the very low probability accident scenarios with malfunction in isolation system. A further detailed analysis shall allow the possible choice of the penetrations to monitor and the adoption of supplementary procedures for the operators in case of high radiation beyond the isolation system. It's, in any case, obvious that the qualification of the monitoring system requires an appropriate program of experimental tests which will be necessary also for the calibration of the instrumentation.

### 9. REFERENCES


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Appendix 2:

Brief description of OECD, NEA and CSNI
The Convention establishing the Organisation for Economic Co-operation and Development (OECD) was signed on 14th December 1960.

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

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

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

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

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

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

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

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

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