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Pre-Project on Development and Validation of Melt Behavior in Severe Accidents

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Abstract

Reactor safety's concern with severe accidents, since the TMI-2 accident led to almost twenty years of intense research efforts, which have resolved a number of severe accident issues. Lately, research has been concentrated on accident management and a number of LWR plants, around the World, have adopted severe accident guidelines (SAMGs) and strategies.

In NKS, the safety advancements expected from the planned research work in the DELI-MELT Project includes (a) an assessment of the adequacy of the accident management schemes adopted currently for Nordic BWRs and PWRs, with respect to melt coolability, accident stabilization and basemat melt-through, (b) evaluation of the reasons for low explosivity of corium, (c) datebase and prediction methodology for lower head failure mode and timing, and (d) resolution of new issues (e.g. melt stratification).

This report mainly consists of three chapters, the assessment of severe accidents, the remaining, unresolved issues of severe accidents and proposed research efforts to resolve these issues. This report reviews the state of the art of the various melt/debris coolability situations and ex-vessel steam explosions during the postulated severe accident scenarios, addresses the unresolved issues concerning the core melt loadings during the severe accidents, and further suggests the experimental facilities in the Nordic countries which could be potentially useful to resolve the issues.
We believe that these issues in the order of priority are;

- in-vessel coolability of the melt pool or particulate debris,
- ex-vessel coolability of the melt pool or particulate debris,
- energetics and fragmented debris characteristics of a steam explosion endangering the integrity of the BWR containments and
- characteristics of vessel failure

Key words

Severe accidents, melt coolability, steam explosions, vessel failure, containment failure

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Final Report on

PRE-DELI-MELT

Pre-Project (PRE) on
Development & Validation (DELI) of
Melt Behavior (MELT) in
Severe Accidents

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by

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Abstract

Reactor safety's concern with severe accidents, since the TMI-2 accident led to almost twenty years of intense research efforts, which have resolved a number of severe accident issues. Lately, research has been concentrated on accident management and a number of LWR plants, around the World, have adopted severe accident guidelines (SAMGs) and strategies.

In NKS, the safety advancements expected from the planned research work in the DELI-MELT Project includes (a) an assessment of the adequacy of the accident management schemes adopted currently for Nordic BWRs and PWRs, with respect to melt coolability, accident stabilization and basemat melt-through, (b) evaluation of the reasons for low explosivity of corium, (c) datebase and prediction methodology for lower head failure mode and timing, and (d) resolution of new issues (e.g. melt stratification).

This PRE-DELI-MELT project aims to develop a detailed project plan for the DELI-MELT project for conducting research in Nordic Countries specifically on core melt loadings on the BWR and PWR reactor containments. It is foreseen that both experimental and analysis development activities can be pursued and the experimental facilities in the Nordic countries can be employed. It is also expected that the DELI MELT Project will benefit from the research activities currently, and in future, on-going in the Nordic countries, supported by EU and other organizations.

This report mainly consists of three chapters, the assessment of severe accidents, the remaining, unresolved issues of severe accidents and proposed research efforts to resolve these issues. This report reviews the state of the art of the various melt/debris coolability situations and ex-vessel steam explosions during the postulated severe accident scenarios, addresses the unresolved issues concerning the core melt loadings during the severe accidents, and further suggests the experimental facilities in the Nordic countries which could be potentially useful to resolve the issues.

In this proposed plan, research efforts will focus on severe accident management issues of most interest to the Nordic power companies and government regulatory organizations.

We believe that these issues in the order of priority are;

- in-vessel coolability of the melt pool or particulate debris,
- ex-vessel coolability of the melt pool or particulate debris,
- energetics and fragmented debris characteristics of a steam explosion endangering the integrity of the BWR containments and
- characteristics of vessel failure.

During the last decades of research in the Nordic countries on the nuclear safety, in particular, severe accidents, rich resources such as experimental facilities and database were produced and accumulated. These infrastructures in the Nordic countries could be employed to pursue further research needed for resolving the remaining key issues.

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Chapter I

INTRODUCTION

1. BACKGROUND

Unlike the development of the fast reactors, wherein the severe (class IX) accidents were considered very early in their safety assessments, the LWRs did not consider severe accidents (SAs) in their safety assessments until the development of the landmark WASH-1400 Risk study (NRC, 1975). That study, explicitly evaluated the consequences and probabilities of SAs for two LWR plants and established that their risks were indeed considerably small.

The TMI-2 severe accident occurred in 1979, a few years after the publication of WASH-1400, which led to the initiation of the SA research for the Western LWRs. Thus, the history of this research is not very long; nevertheless, phenomenal progress has been achieved in the last twenty years, primarily due to the concentration of much of the LWR safety research on this topic, as for example by the European Commission in the Framework Programmes Nos. 3 and 4.

The safety design of the current LWRs was based on the regulatory framework established in the early 1970s in which the large LOCA served as the enveloping design-base accident. The provision of a strong containment was fortunate since it did not allow the release of radioactive fission products to the environment during the TMI-2 accident (Broughton et al., 1989). The post-TMI-2 safety analyses (Wolf, J. R., et al., 1993), and other considerations, prompted the regulatory authorities, and the plants, to implement additional measures to mitigate the consequences of the severe accidents. Some of these measures are true back fits, while others are improved (e.g. symptom oriented) procedures and actions. All of these have now become the parts of Severe Accident Management (SAM) guidelines, which have been, or are being, implemented in the reactor plants. We will attempt to provide a brief listing of the mitigation measures that have been, or are being, implemented in the plants.

There is a deep connection between the results obtained with the SA research and the mitigation measures implemented in the plants. Needless to say, the SA research, pursued in the laboratories of the various western countries, always had the objective of understanding the phenomenology in order to reduce the consequences of the severe accidents. Clearly, the mitigative measures chosen by the plants, and approved by the regulatory authorities, have a solid backing from the SA research, e.g. the experiments conducted on the removal of fission product aerosols by sprays, and those on the modes of hydrogen combustion with igniters, conducted in the Nevada test facility. We shall be describing this connection between the SA research on core melt loadings and the design and the implementation of the mitigative measures in the plants. Furthermore, the unresolved issues of the core melt loadings during severe accidents will be addressed. Therefore some detailed research plans to investigate the unresolved issues will be suggested. However, we will not be providing a listing of the many excellent publications on the various phenomenological studies of SA research, since the length of the paper is very limited. Most of the relevant references are listed in earlier papers (Sehgal, 2000, 2001a, 2001b, 2003a).

2. SCOPE AND OBJECTIVES

The objective of the project is to develop a detailed project plan for conducting research in Nordic Countries on core melt loadings on the BWR and PWR reactor containments. It is foreseen that both experimental and analysis development activities will be pursued and the experimental facilities at the NPS Division of KTH, Fortum Nuclear Services and at VTT will be employed. It is also expected that the DELI MELT Project will supplement and benefit from the research activities currently, and in future, on-going at KTH, VTT and Fortum Nuclear Services, supported by EU and other organizations.

In the area of severe accident management, the specific unresolved issues for which there are still very large uncertainties will become the focus of the research. These are in the order of their priority:

- Melt (debris) Coolability and Accident Stabilization,
- Steam Explosions,
- Containment Structural Integrity,
- Lower Head Failure Mode and Timing, and
- In-vessel Melt Retention.

The safety advancements expected from the planned research work in the DELI MELT Project would be;

- An assessment of the adequacy of the accident management schemes adopted currently for Nordic BWRs and PWRs, with respect to melt coolability, accident stabilization and basemat melt-through,
- Evaluation of the reasons for low explosivity of corium,
- Datebase and prediction methodology for lower head failure mode and timing, and
- Resolution of new issues (e.g. melt stratification).

The safety advancements afforded by the research work in DELI MELT are of interest to all elements of the nuclear enterprise in the Nordic countries. Information on the results of the research and its implications will be exchanged with various institutions in the Nordic countries. The research work performed during education at NPS/KTH transforms students and post-doctors into competent researchers and plant personnel. New experts are created as well. Work at VTT and Fortum will enhance capabilities of their scientists.

Chapter II

Assessment of Severe Accidents

Severe accidents (see Figure 1) posed, to the reactor researchers, a most interesting and most difficult set of phenomena to understand, and to predict the consequences, for the various scenarios that could be contemplated. The complexity of the interactions, occurring at such high temperatures (~2500ºC), between different materials, which are changing phases and undergoing chemical reactions, is simply indescribable with the accuracy that one may desire. Thus, it is a wise approach to pursue research on SA phenomena until the remaining uncertainty in the predicted consequence, or the residual risk, can be tolerated.

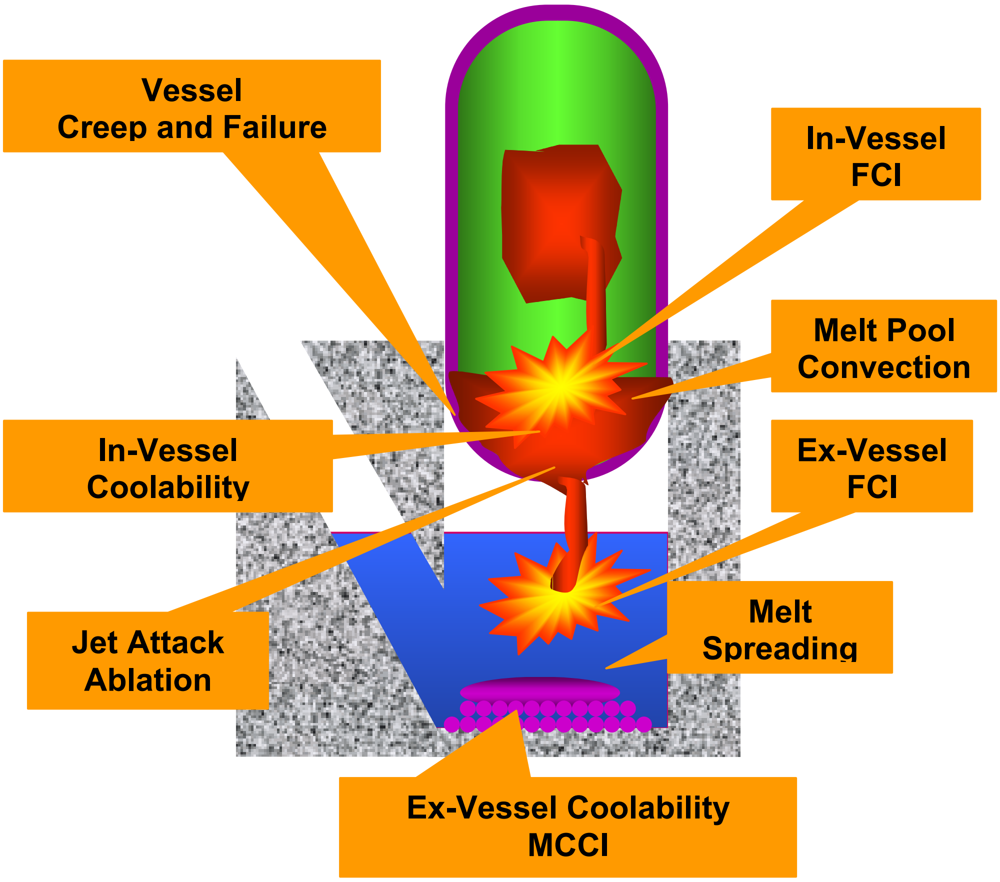

Figure 1 Severe Accident Phenomena

1. EARLY CONTAINMENT FAILURE

It soon became clear from the early work in SA research that the protection of public required the maintenance of containment integrity for at least 4-6 hours, since it was found that a large majority of the fission product aerosols, formed in the containment, deposit on the containment surfaces within 4-6 hours, and are not available for release. Thus, the focus of the SA research immediately shifted to the study of the phenomena, which could cause early failure of the containment. The phenomena identified were from the events:

- In-Vessel Steam Explosion,
- Ex-Vessel Steam Explosion,
- Hydrogen Detonation,
- Direct Containment Heating, and
- Melt Attack on Mark I-BWR Containment Wall.

The research on the phenomena from these events, pursued in the U.S. laboratories, provided differing estimates of the consequences of these events. For example, the early results obtained in Sandia National Laboratory were extrapolated to indicate quite high probability of early containment failure from each of these events. One difficulty of the SA research, as can be easily conceived, is to perform proper scaling analysis for the phenomena and then to perform experiments at proper scales of geometry and materials, so that the results obtained can be extrapolated to the prototypic size and conditions (Zuber, N. et al., 1998).

The confused and rather alarming early experimental and assessment results obtained for the above-listed events were slowly rectified by the later studies incorporating realistic phenomenology coupled with probabilistic analyses. We will describe the state of the resolution of these early containment failure issues in the following paragraphs.

1.1. In-Vessel Steam Explosion

It should be understood that this issue is concerned with a steam explosion, which has sufficient mechanical energy content to fail the upper head of the vessel, which flies away with sufficient velocity to fail the containment at the location where it hits it. The more interesting results are those on the conversion ratio of the steam explosion and the mechanisms that limit its value; and on the mechanical energy required to fail the upper head. It has been found experimentally in the FARO and the KROTOS experiments that the conversion ratio for energetic interaction of the UO_2-ZrO_2 melt mixtures with water is very low indeed. Although, a full explanation of these observations has not been obtained, some of the mechanisms which may prevent the steam explosion and/or limit the mechanical conversion have been identified. The assessment based on the recent FZK experiments on the magnitude of mechanical energy that would be able to fail the bolts of the upper head has confirmed the conclusion reached by the steam explosion expert review group (SERG) (SERG2, 1995) that the in-vessel steam explosion-induced containment failure is of extremely low probability. ***Thus, this SA issue has been resolved.***

1.2. Ex-Vessel Steam Explosion

The ex-vessel steam explosion is an early containment failure issue only for those reactor plants, which either have a deep pool of water under the vessel or/and have a relatively weak containment. The reactor plants that satisfy these criteria are (a) the Swedish and Finnish BWRs, which establish a deep pool of water in the lower drywell as a SAM measure and (b) the Westinghouse PWRs, which flood the reactor vessel cavity, according to their SAM guidelines. The mechanisms, which limit an in-vessel steam explosion, will also limit the explosion yield for an ex-vessel steam explosion. ***Nevertheless, this early containment failure issue has not been adequately resolved.***

1.3. Hydrogen Detonation

Hydrogen combustion was the first issue that received the attention of the USNRC immediately after the TMI-2 accident. The hydrogen management rule spawned much research in hydrogen combustion. The Nevada experiments established the feasibility of employing igniters to burn hydrogen as it is produced, so that accumulations, which can produce detonations are avoided. The BWRs were inerted and the ice condenser and the Mark-3 BWRs were fitted with igniters. The European plants opted not to implement igniters in their containments but to instead provide recombiners, which would reduce the hydrogen concentrations. The only sub-issues that remain for future investigations are (1) hydrogen distribution in a complex geometry containment and (2) the inadequacy of the recombiners to mitigate rapid hydrogen generation. The GASFLOW code (Royl et al., 2000) seems to be able to provide reasonable predictions. ***Perhaps the closure of the hydrogen distribution sub-issue can be accomplished in near future.***

1.4. Direct Containment Heating (DCH)

DCH was found to be potent cause of early containment failure in the early work in NUREG 1150 (NRC, 1990), the study updating the WASH-1400 study (NRC, 1975). The experimental work initiated at SURTSY facility in SNL and at the Argonne National Laboratory (ANL) showed that for the Westinghouse containments, most of the particles formed were trapped in the lower compartment and the containment pressurization was tolerable. Later a two-volume model developed in SNL provided good predictions and extrapolations. The DCH issue for the European plants also prompted experimental investigations and no adverse conclusions have been reached. The VVER plants, perhaps, have not been extensively analyzed for the postulated DCH event. It has become clear, however, that the best way to resolve this issue is to depressurize a PWR if the core steam exit temperature goes above a set temperature. Another mitigative feature is the failure of the primary piping (surge line to the pressurizer due to natural circulation) during the high-pressure scenario, which results in depressurization of the system. **We believe that this issue has been resolved by the SAM action of deliberate depressurization.**

1.5. Melt Attack on BWR Mark I Containment Wall

This phenomenon is particular to the BWR Mark I plants designed by General Electric. The wall of the steel containment for this reactor is quite close to the discharge location of the melt from the vessel. The analysis of this issue by Prof. Theofanous concluded that adding water to the drywell would reduce the containment failure probability by 2 to 3 orders of magnitude. **Thus, this issue was also resolved by another SAM measure, i.e. adding a layer of water to the drywell which limits the rate of the heat transfer from the melt to the containment wall.**

2. LATE FAILURE OF CONTAINMENT

The late failure of the containment, although less harmful to the public, still should be prevented. In this respect, there are differences in the approach to licensing between USA and Europe. The U.S. regulations admit the possibility and feasibility of evacuation of the population around a plant. Thus, they prescribe that the integrity of the containment be maintained for at least 24 hours and that the conditional (in the event of core damage and vessel failure) probability of containment failure should be ≤ 0.1. The European approach with respect to late containment failure is to prevent it so that the evacuation of the nearby population is unnecessary.

The phenomenon contributing most to the probability of late containment failure is the pressurization caused by the interaction of the melt discharged from the vessel with the concrete and water in the containment. This pressurization is on top of that caused by the release of the steam and hydrogen to the containment during the in-vessel accident progression. This additional pressurization can only be prevented if the melt is cooled down below the concrete ablation temperature. Thus, the phenomena concerned with late containment failure are the molten corium concrete interaction (MCCI), the melt coolability, and containment venting.

2.1. Molten Corium Concrete Interaction (MCCI) and Basemat Melt-Through

This issue is of particular importance to the French and the German PWRs, which do not allow any water entry to the vessel cavity. The melt discharged from the vessel will continue to attack the concrete basemat and pressurize the containment, which for the French plants can be vented through a sand-bed filter. The greater concern is with the basemat melt-through since with the crust formation on top of the melt pool, there is not much heat loss and most of the heat generated is delivered to the basemat causing further ablation. It is imperative to predict the time taken to melt through the basemat for devising emergency measures around the plant. Currently, there is no data on two dimensional MCCI to validate the predictive methods. Clearly, large radial heat transfer would lead to longer time for basemat melt-through and vice versa. The total amount of the concrete ablated and the gas produced can be predicted quite readily from the heat balance considerations. Experiments are, currently, under way at ANL on 2-D MCCI in the OECD sponsored MCCI Project.

2.2. Melt/Debris Coolability

Melt coolability is perhaps the most vexing unresolved issue, since it is not clear how to cool and quench a melt pool interacting with a concrete basemat in the current plants. Clearly, the easiest SAM action to stabilize and terminate the accident is to flood the PWR vessel cavity, or a BWR drywell, with water, to quench and retain the melt in the containment. The MACE program at ANL has performed experiments at different scales, with prototypic melt pools, flooded with water. Unfortunately complete coolability was not achieved, primarily due to the attachment of the insulating crust to the wall and thereby the detachment of the melt pool from the crust. Three modes of heat transfer from the melt pool to the water were identified. They are (a) initial melt-water contact (b) water ingression into melt and (c) melt eruptions into water for which separate-effect experiments are being performed at ANL in the MCCI Project. Currently, it is not clear that melt coolability by a water overlayer can be certified. Perhaps, at plant scale, with spans of several meters, the top crust will be unstable and there would be periodic contact between melt and water to eventually quench the melt. It is clear, however, that some basemat ablation will occur during this process. Another benefit of the water overlayer is the scrubbing of most of the fission products produced during the MCCI.

Since melt coolability with a water overlayer may be hard to achieve, alternative and innovative means have been explored to quench the melt. Experiments have been performed at the COMET facility (Tromm, et al. 1995) in FZK in which water is introduced at the melt bottom, with a slight overpressure, either through nozzles or through porous concrete substrate. It has been found that melt quenching is achieved quite readily and no steam explosions occurred even with an Al_2O_3 melt pool. The COMET concept can only be accomplished in current plants with containment modifications. Another innovative concept is that of employing downcomers to channel the water from top of the melt to the bottom of the melt. Experiments on this concept are being performed at KTH and some initial success has been achieved in scaled experiments.

Coolability of particle debris beds has been investigated as an adjunct to that of melt-pools. Particle debris beds are generated in both in-vessel and ex-vessel accident progression when melt jets come in contact with water and break up. Establishing a water pool in the Swedish BWR drywells and flooding of the vessel cavity in Westinghouse PWRs result in formation of particulate debris beds. Clearly, particle debris beds are easier to cool, except when the bed porosity and particle size are very low, and a particle debris bed could convert into a melt pool. Low porosities can result in a debris bed if small size particles produced, for example in a steam explosion, are mixed with the larger size particles produced during the melt jet breakup process. The POMECO experiments (Konovalikhin, M. J, 2001) performed at KTH have shown very substantial increases in the dry out heat flux and the quenching rate, when downcomers are employed in low porosity debris beds.

2.3. Fission Product Release and Transport

The ‘source term’, i.e., the magnitude, the chemical and the physical form of the fission product (FP) source distribution in the containment atmosphere is, perhaps, as important for the consequences of a severe accident as the containment failure since its release to the environment is the risk to the public health and safety of nuclear power. Research in the ‘source term’ has been extensive and long lasting; even today the PHEBUS program (Schwarz, 1999) is generating confirmatory data and has found that there are still some discrepancies in the prediction of the iodine compounds and the organic iodine that may be in the containment atmosphere. Some of these discrepancies and uncertainties may be attributed to the paints, while the others may be in the prediction of the iodine compounds e.g., formation of AgI in the PWRs. Another uncertainty in the in-vessel part of the SA is the deposition of FPs in the primary system, which can act as a ‘source term’ later in the scenario due to revolatilization induced by the heat up of the primary system.

The ex-vessel release of FPs due to the concrete ablation gases sparging through the melt pool has not been found to be of large consequence since the release rates of refractory FPs have been found to be quite low. The presence of a water over-layer greatly diminishes the ex-vessel source term.

2.4. Containment Venting

Containment venting is needed for small volume and relatively weak designs. The BWR containments in the Nordic countries employ a filtered vent, which opens at a set pressure and the filter decontaminates the released gases by passing them through a venturi and water pool. A hard vent is being constructed in some U.S. BWRs from the condensation pool. A sand bed filtered vent is placed in the French PWRs. The vent design is based on the very large-scale tests performed in the LACE Project. Large decontamination factors (DFs) can be achieved. Vents have not been adopted by the U.S. PWRs.

3. MITIGATION MEASURES IN CURRENT PLANTS

As mentioned earlier the SA research results have led to the mitigation measures of backfits and of accident management and procedures which have enhanced the safety of the plants. A representative list is provided below:

- Hydrogen control with ignitors and catalytic recombiners
- Improved safety valves on PWRs
- Water addition to the Mark-1 drywell to prevent liner failure
- Vessel depressurization for DCH protection
- Use of BWR suppression (condensation) pools for fission product decontamination
- Hard vents for BWRs from the suppression pool
- Flooding of PWR vessel cavity for Westinghouse PWRs
- flooding of drywell for Swedish BWRs

- Additional water delivery sources for accident termination
- Reinforcement of containment penetrations
- Pressurized thermal shock prevention procedure
- Filtered venting
- Long term management of iodine in the containment
- Long term cooling of the containment to maintain heat sink

The research results have also provided the rationale for the deliberate decisions of not requiring any backfits or mitigation measures. Examples of these are:

- No inerting of Mark-3 BWRs
- No backfits for protection against alpha mode failure

Chapter III

UNRESOLVED ISSUES CONCERING CORE MELT LOADINGS IN SEVERE ACCIDENTS

Reactor safety's concern with severe accidents, since the TMI-2 accident led to almost twenty years of intense research efforts, which have resolved a number of severe accident issues. Lately, research has been concentrated on accident management and a number of LWR plants, around the World, have adopted severe accident guidelines (SAMGs) and strategies. There are still several severe accidents issues, which remain unresolved as mentioned in the previous chapter of this report. A suggested prioritization (Sehgal, 2000, 2001a, 2001b, 2003) is as follows:

(1) Ex-vessel Melt/Debris Coolability,

(2) Ex-vessel Steam Explosion Loads,

(3) Basemat Melt-Through,

(4) Lower Head Failure Mode and its Timing, and

(5) Core Quenching

Perhaps, the issue which most affects the mitigation strategy in the current plants is that of ex-vessel melt/debris coolability, since the stabilization and termination of the accident depends on it. The current SAM measures either avoid flooding the PWR vessel cavity and the BWR drywell or depend on such action for coolability, except that the flooding action opens the issue of the vulnerability of containment to steam explosion loads. Currently, neither mitigative measure is clearly preferred.

It is clear that long term stabilization and coolability of the melt after a severe accident is one of the goals of the near-future new LWR plants. Two lines of mitigation measures have been focused in the new designs: in-vessel coolability of debris/melt and retention in the AP-600, AP-1000 and the SBWR designs and the ex-vessel coolability and retention in the EPR and the VVER-1000. Both of these concepts have progressed to commercialization and design certification stage and tremendous research activities, conducted over several years, have supported the development of these mitigative designs. The in-vessel melt retention concept has been installed in the LOVIISA VVER-440 containment through backfits. The VVER-1000 employs a core catcher in which the melt released from the vessel reacts with oxide material to reduce its temperature. The core catcher vessel is cooled from outside, similar to the vessel cooling in the in-vessel melt retention concept.

An innovative mitigation measure to cool the ex-vessel melt pool in the vessel cavity has been investigated at FZK (COMET Program) and KTH (DECOBI Project) (Paladino, 2002). This involves the idea of adding water at the bottom of the melt pool to take advantage of the much greater cooling efficiency of co-current two phase flow than that of the counter-current two phase flow as occurs in top flooding. This can be incorporated in new plant designs and, perhaps, is more cost-effective.

Another innovative mitigation measure which could even be employed for the current plants is to provide a set of downcomers in the cavity under the vessel which could bring water from the top of the melt pool/debris bed to the bottom. A natural circulation flow field is established for the co-current cooling flow. Substantial increase in coolability has been observed in the preliminary experiments conducted at KTH.

These remaining issues are complex and difficult to resolve. In fact, that is the reason they still remain unresolved. The issue of melt/debris coolability is of particular significance since a severe accident can not be characterized as stabilized and terminated until the core melt/debris has been cooled and quenched and kept in the latter state for a long time. The public has to be assured that a severe accident, if it ever occurs, will be managed and terminated quickly, with no release of fission products out of the containment. Achieving and maintaining coolability of the melt/debris is paramount, since fission product release and non-condensable gas generation stops as the melt/debris temperature drops below $\cong$1000°C, and containment integrity is not seriously challenged any more.

The basemat melt-through issue is of particular concern for dry cavity PWRs and the BWRs, which do not add water to their dry wells although the wet cavity plants are also unable to assure its avoidance. The scenario assumes that no coolability of the melt/debris can be, or has been, achieved and the attack of the melt on basemat continues till its melt-through, resulting in possible ground and water contamination (the China syndrome).

Ex-vessel steam explosion loads, for some BWRs and PWRs, arise due to the accident management action of establishing a pool of subcooled water under the reactor vessel, whose interaction with a melt jet, released from the PWR or BWR vessel, could result in an energetic steam explosion threatening the integrity of the containment. The objective of this accident management action, which forces the interaction of the melt jet with water, is to fragment the melt and to form a coolable particulate debris bed. Thus, the ex-vessel steam explosion, basemat melt-through and the ex-vessel melt/debris coolability issues are inter-related, i.e. easier coolability may involve the risk of steam explosion, avoiding the risk of steam explosion may involve the risk of basemat melt-through.

The melt/debris coolability has been recognized as the 'Achilles-heel' of the current LWR designs. The solutions proposed are incorporated in (a) the in-vessel melt retention (IVMR) accident management scheme (Theofanous et al., 1995) of the Westinghouse AP-600 and AP-1000 designs, the FANP's BWR-1000 design and the KEPCO's 1400 MWe Advanced PWR design, (b) the melt spreading and cooling compartment of the EPR design (Fischer, 1999) and (c) the core catcher of the VVER-1000 design under construction in Tian Wan, China. Each of these innovative designs needed new knowledge. Much knowledge has been gained, however there are still some gaps and considerable uncertainties. Nevertheless, it is clear that concerted efforts have been made to find design solutions for assuring melt/debris coolability and thereby the stabilization and termination of a postulated severe accident in LWRs.

The postulated severe accident scenarios require melt/debris coolability and quenching not only during the ex-vessel phase, but also during other phases of the progression of the accident. The very early phase of the accident may require quenching of the core with intact rods at high temperature and, perhaps, with some local melting. Water addition may result in shattering of the fuel rods and formation of a particulate debris bed (as occurred in the TMI-2 accident). Later, a melt pool may be formed underneath the particulate debris, if the water supply is not restored. Further in the scenario, the movement of the melt from the core region to the lower head can create particulate debris beds and later a re-circulating melt pool with a metal layer. The in-vessel melt/debris configuration for coolability depends on the recovery time of the accident. A variable is the probability of the occurrence of a steam explosion (both in-vessel and ex-vessel), which can create very small size particles, which can change the porosity of a particulate debris bed. Melt and debris composition can also vary during the progression of the accident; in particular, the zirconium, boron and carbon content of the melt/debris can create configurational changes due to chemical reactions, which may produce heat, hydrogen or different density regions (Asmolov et al., 2003).

The following sections of this report will review the state of the art of the various melt/debris coolability situations and ex-vessel steam explosions during the postulated severe accident scenarios, address the unresolved issues on the core melt loadings during the severe accidents, and thereby suggest the experimental facilities available in the Nordic countries which are potentially useful to resolve the issues.

1. CORE QUENCHING

The first and the best opportunity to stabilize a severe accident is to catch it before it progresses further. This refers to introduction of water (if it is available) into the core as soon as there are indications of an escalation in the temperature in the hot leg or in the core itself. Core quenching is not a straightforward management action since the steam formed may aggravate the accident by increasing the zircaloy oxidation, which adds oxidation heat to the core and increases the core temperatures. The QUENCH (Miassoedov et al., 2002) research project has obtained valuable data, over many years at FZK, on the transient hydrogen generation rate during the process of quenching a hot fuel rod subassembly heated electrically. Control rod assemblies containing B_4C control rods were also tested. The major conclusions were;

(a) the oxidation rate is a function of the thickness and stability of the oxidic film deposited on the fuel rods during normal operation,

(b) greater oxidation takes place when cracks are formed in the oxidic layer, and

(c) the control rods melt early and the boron reacts with steam to generate additional hydrogen.

The main objective of the studies performed on the CORQUENCH facility has been to predict the additional hydrogen that will be produced very rapidly during the water addition phase. Clearly, if the water is added at a slow rate, i.e. with high or medium pressure ECCS, large volume of hydrogen may be produced. The key action is to reduce the clad temperature quickly by adding a large volume of water at a rapid rate. This may lead to structural damage in the core but the accident would be stabilized and less hydrogen will be produced. No experimental facilities related to this issue are available in the Nordic countries. This issue can be considered separately from the core-melt loading issues.

2. COOLABILITY OF THE IN-VESSEL PARTICULATE DEBRIS BEDS

The addition of water to the very hot core can create a particulate debris bed, whose coolability is the next challenge for the stabilization and termination of the accident. This event occurred during the TMI-2 accident; thus quenching of particle debris beds, either by flooding from top or from bottom has been studied and reported in literature (Ginsberg et. al 1982, Cho and Bova 1983, Hall and Hall 1981, Tung and Dhir 1983, 1986). It was found that the quenching process of the bed with top flooding has to fight the counter current flooding limitation (CCFL) in which the steam formed limits the downward ingression of water in the bed. There is no such CCFL limitation for the bottom flooding. The rate of heat transfer is limited only by the available driving pressure or the coolant flow rate. The heat removal process was found to be very efficient in this case. Heat removal rates as high as 8-10 times those obtained during top flooding have been observed.

The situation of an in-core particulate debris bed is not easily determined since much of the core could be blocked from the bottom and the resumption of water supply would result only in top flooding of the blocked portion of the debris bed and the quenching rate could be very low. The particulate beds may remelt and become melt pools as it did occur during the TMI-2 accident.

Particulate debris beds are also formed when the melt from the core drops into the lower head full of water. The jet break up process has been studied in the FARO Program (Magallon et. al., 1999) with prototypic melts and in the MIRA Program (Haraldsson, 2000) with simulate melts. Particle size distributions have been measured in both Programs. The MIRA data was fitted to various distributions and a sequential fragmentation distribution (Brown et al., 1983) has been recommended. The particulate beds formed in the FARO tests were found to be non-uniform. In particular, cake type regions were found. In general the steam formed in the lower head would lift smaller size particles which would fall down later and form a layer on top, or mix with larger size particles to reduce the bed porosity. The explosive interaction of the corium jet has been found to be very low in the FARO and KROTOS experiments (Magallon et. al. 1999, Huhteniemi et. al., 1999). Nevertheless even a small steam explosion can generate many very fine particles which can either form stratified layers or mix in with the larger particles to form low porosity and low mean particle size debris beds.

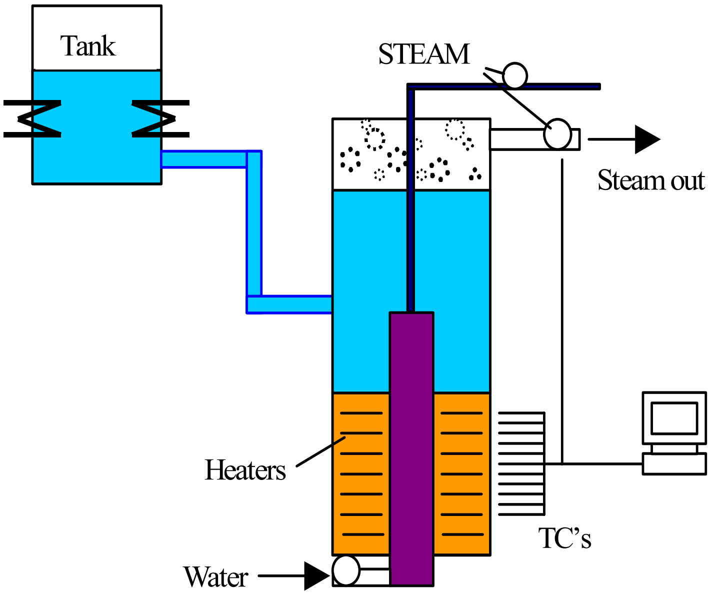

Figure 2 The POMECO Facility at KTH

Recently, coolability of low porosity and small mean particle size beds have been studied at KTH, Sweden in the POMECO (porous media coolability) facility (Konovalikhin et al., 2000; Konovalikhin and Sehgal, 2001). This facility shown in Figure 2, incorporates a parallelepiped test section ≈ 350×350×400 mm in which sand beds are constructed with different porosities and mean particle sizes. A bed with porosity of 0.26 was made by mixing sands of different mean particle sizes. The beds are heated with thin electrical resistance heaters to impart prototypic uniform heat generation rates of ≈ 1 MW/m^3. Both bed dry out and bed quenching experiments were performed for uniform and stratified beds. It was found that the dry out heat flux and quenching rates are very low for low porosity and low mean particle size beds and clearly, these beds cannot be cooled for the prototypic decay heat generation rates. Beds with mean particle size of ≥ 2 mm and porosities of ≈ 40% are coolable. The dry out heat flux and quenching rate in stratified beds with layers of different porosities and mean particle size are those for the layer with lower porosity and/or lower mean particle size.

Thus, it is clear that the in-vessel particulate debris of relatively high particle size and porosity are coolable as was the TMI-2 particulate debris bed in the lower head. Coolability is generally not assured since there may be regions of low porosity and of small mean particle size, because of possible energetic interactions between the melt jets and water.

The boiling water reactor (BWR) is a special case for the coolability of a debris bed in the lower head, since it may incorporate ≈ 100 control rod guide tubes (CRGTs) in which there is a continuous flow of subcooled water. These CRGTs, if they survive, which they could for a certain time window, provide a very large surface heat removal path, which supplements the regular debris bed cooling. Demonstration of this was provided (Konovalikhin et. al., 2003) in experiments performed in a modified POMECO facility in which a unit cell of prototypic size of a CRGT, contained in a particle bed providing prototypic heat generation, was constructed. The CRGT was provided with prototypic flow rate of water at different subcoolings. Preliminary results showed that the heat flux to the CRGT could be in the range of 50 to 100 kW/m^2, which would substantially help in cooling the low porosity and low particle size beds.

However, further detailed investigation on the in-vessel coolability through the CRGTs is needed to conclude the applicability of the CRGTs as the in-vessel coolability. The POMECO facility at KTH and the STYX facility at VTT can be a candidate for this investigation.

3. COOLABILITY OF IN-VESSEL MELT POOLS

The TMI-2 accident developed a large in core melt pool of primarily oxidic mixture (UO_2-ZrO_2), which was flooded by water, formed a thick crust all around and was not coolable. Similarly, if a large melt pool develops in the lower head of a PWR, in the event of a debris bed melting due to the low value of the dry out heat flux, or due to the complete evaporation of water before restoration of water supply, its coolability by water flooding is doubtful. There have been suggestions that the coolability of the TMI-2 debris in the lower head was achieved by gap cooling, since a part of the vessel reached very high temperatures (hot spot) and then was quenched. It has been postulated that water flowing through the gap between the crust at the bottom of a melt zone and the vessel wall was able to cool the vessel wall. It has been suggested (Henry et. al., 1996) that a small (1-3 mm) gap between the crust and the vessel wall would admit water to cool the vessel wall and ensure the integrity of the vessel.

We performed two experiments in the FOREVER facility (see Figure 3) to test this hypothesis and to obtain data on the extent of coolability that can be expected for a large melt pool in the lower head. The experiments EC-FOREVER-5 and –6 employed a binary oxide mixture (30% CaO + 70% B_2O_3) to simulate the corium (80% UO_2 – 20% ZrO_2) in a 1/10th scale vessel. The melt was maintained at temperature of $\approx$ 1500 K by heating it with a special heater, inside the vessel, providing $\approx$ 30 kW of resistance heating. The vessel pressure was maintained at $\approx$ 25 bars to represent a depressurized severe accident. The melt pool filled the lower head to $\approx$ 2 cm below the vessel equator level. These experiments were performed to favor the formation of a gap by holding the vessel for about 2 hours at 25 bar pressure and the prevailing temperature with 30 kW of heat addition (maximum wall temperature of $\approx$ 900°C about 20° below the vessel equator). The creep of the vessel wall was measured continuously with linear differential transducers (LDT) and when the maximum vessel displacement of $\approx$ 5% (1 cm at $\approx$ 45° below equator and $\approx$ 5 mm at 2 cm below equator) was reached, subcooled water was added to the top of the melt pool and a water overlayer was maintained with subsequent additions of water. The vessel and the melt temperatures were measured as a function of time as well as a video of the vessel was recorded.

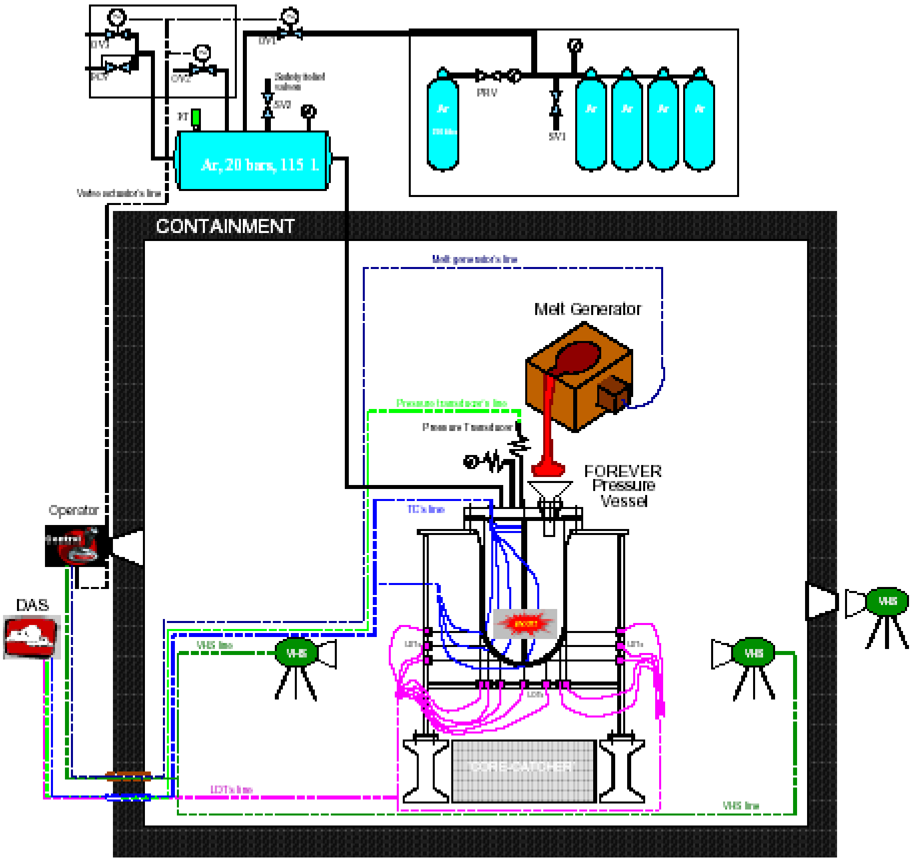

Figure 3 Diagram of the FOREVER Facility.

The results of the experiments for gap cooling were negative. The vessel wall was not cooled and it did not cool even near the top face of the melt pool. The melt pool itself cooled only partially. The thermocouples near the bottom of the pool did not show reduction to the level of water saturation temperature i.e., the water ingression did not proceed to the bottom of the melt pool. The vessel was cut open after the test and it was found that the top 6-7 cm of the melt was fragmented, i.e., was quenched by water, while the bottom 11-12 cm of the melt cooled down from the molten state (see Figure 4). The maximum upward heat flux was ≈ 1.8 MW/m^2 when the initial bulk cooling took place. Later the heat flux decreased to ≈ 0.3 MW/m^2 in about 300 sec. after formation of the crust. It degraded to lower values later. These results obtained are very similar to those obtained in the MACE Project ex-vessel coolability experiments (Farmer et. al. 2000) conducted at different scales over many years with water flooding of prototypic corium pools interacting with concrete.

Figure 4 Post-test View of Solidified Melt for EC-FOREVER-5 Experiment.

We believe that coolability of a large melt pool in the vessel with in-vessel water flooding will not be possible. The in-vessel melt coolability and retention strategy involves external cooling of the vessel with a water pool maintained in the containment. We shall discuss this in a latter section in which future designs are described.

The BWR CRGTs could survive in a melt pool if there is sufficient flow in these tubes to avoid reaching the critical heat flux condition. The crust formed on the exterior surface of these tubes provides a protective barrier. Experiments performed in the POMECO facility (Konovalikhin et al., 2003) show that heat fluxes of 350 Kw/m^2 could be available for heat removal from the melt pool to the CRGT water. This heat removal promotes the ingression of water into the melt pool. Although, this is a very recent observation, the very large surface area provided by the large number of CRGTs could help in cooling and retaining the melt inside the BWR lower head in the event that the CRGTs have survived. Therefore, the POMECO, COMECO and FOREVER facilities at KTH are candidates to investigate the in-vessel melt pool coolability.

4. COOLABILITY OF EX-VESSEL PARTICULATE DEBRIS BEDS

The march of the severe accident continues to vessel failure if a large convecting melt pool is formed inside the lower head. The release of the melt from the vessel to the containment in the form of a jet either encounters water in the vessel cavity, or not, depending upon the accident management actions followed by the individual plants. The German and the French reactor containments do not allow any water in their vessel cavities, while the Swedish BWRs and the Westinghouse PWRs will fill, respectively, their lower drywells and their vessel cavities, with water. The presence of water breaks up the melt jet from the vessel into a particulate debris bed. In fact the Swedish BWRs depend upon the coolability of the particulate debris beds, since the basemat sits on top of useable rooms underneath, as well as important cables. Similar considerations have prompted the Westinghouse Owners Group to adopt the guideline of adding water into the vessel cavity to form a particle bed and to cool the lower head from outside to retard vessel failure, and, hopefully, retain the melt in the lower head.

The coolability process in the debris beds formed in the containment is very similar to that in the beds formed inside the lower head. The only difference is that inert gases sparge through the bed due to interaction with concrete. The effect of these inert gas flows on the quenching of the bed has been measured by Tung et. al. 1986 and it also has recently been measured in the POMECO facility for two beds with different porosities (Jasiulevicius and Sehgal 2003). It was found in the POMECO experiments that gas flows, corresponding to the expected average concrete ablation rate, reduce the quenching rate considerably for the very low porosity bed ($\varepsilon \approx 0.26$) but they do not affect the quenching rate in beds having porosity of ≈ 0.4. The effect of non-condensible gas sparging on the dry out heat flux does not appear to be large.

Figure 5 The STYX-1 Test Facility at VTT (From http://www.vtt.fi/pro/tutkimus/finnus/moses.htm)

In Finland, under the MOSES (Modelling and Simulant Experiments of Severe Accidents Phenomena) program (Kyrki-Rajamäki, 2002), the STYX facility as shown in Figure 5 at VTT was constructed to investigate the ex-vessel debris-bed coolability for the Olkiluoto BWR power plant with a uniform debris-bed. The facility employed the Olkiluoto specific debris-bed particle characteristics to examine the dryout heat flux of the debris bed. The facility successfully performed the tests at 0.1~0.7 MPa pressure for uniformly mixed debris-bed with various range of debris sized and shapes. The results showed that the dryout heat fluxes increased with pressures, for instances, 232 kW/m^2 at near atmospheric pressure and 451 kW/m^2 at 0.6 MPa overpressure. Recently, the STYX facility conducted the debris-bed coolability for stratified beds under the SANCY (Severe accidents and Nuclear Containment Integrity) project in the SAFIR (Safety of Nuclear Power Plants – Finnish National Research Programme) program (SAFIR, 2003) for the Finnish nuclear power plants.

The coolability of the ex-vessel debris beds in the BWR dry well and in the Westinghouse PWR vessel cavity is determined primarily by the dry out heat flux, since the bed will be water-logged. The bed will, most probably, be radially and axially stratified. It could also have very low porosity and a small mean particle size if a steam explosion occurs, which will produce very small size particles. Similar conclusions apply as for the in-vessel debris beds, except that the ex-vessel beds will be larger in diameter due to the greater space available and they may as well be in the shape of mounds. We believe that the ex-vessel debris beds will be more three dimensional, and stratified, than the in-vessel debris bed. We believe there will be transverse paths available for water to penetrate the bed and cool the regions where dry out may occur. Three - dimensional flows through the debris beds have been calculated by the WABE code (Cognet et al., 1999) but there is no validation. Most of the debris bed experiments performed so far have provided one - dimensional addition of water, either from top or bottom. A few small-scale experiments at UCLA were performed with addition of water from the side.

Therefore, the large-scale experiment in which water will be introduced from the sides of the bed is needed. The facility should employ a prototypic bed non-uniform radially and axially; top flooded water may reach the more dense sections of the beds from the side or the bottom after going down the more porous parts of the bed. We expect that 3-D coolability of a particulate debris bed will be much more efficient than the one-dimensional top flooding. This type of the facility has been proposed at KTH, so-called, the POMECO-GRAND.

5. COOLABILITY OF EX-VESSEL MELT POOLS

Coolability of a melt pool interacting with a concrete basemat by a water overlayer was under intense investigation in the MACE Project (Sehgal et al., 1992, Farmer et al., 2000), sponsored by an international consortium and managed by EPRI. The experimental work was performed at ANL. Three experiments were performed successfully in which melt pools of $30\times30\times15$ cm depth, $50\times50\times25$ cm depth and $120\times120\times20$ cm depth were generated on top of limestone common sand (LCS) concrete basemats and water added on top. The melt material contained Uranium oxide, Zirconium oxide, Zirconium and some concrete products. The decay heat generation in the melt was simulated through electrical heating. It was found that for these three tests, the effect of the sidewalls dominated the phenomena, since an insulating crust was formed, which attached itself to the sidewalls. The crust prevented intimate melt-water contact and the heat transfer rate slowly decreased from approximately 2 to 0.1 MW m^{-2}, which is less than the decay heat input to the melt.

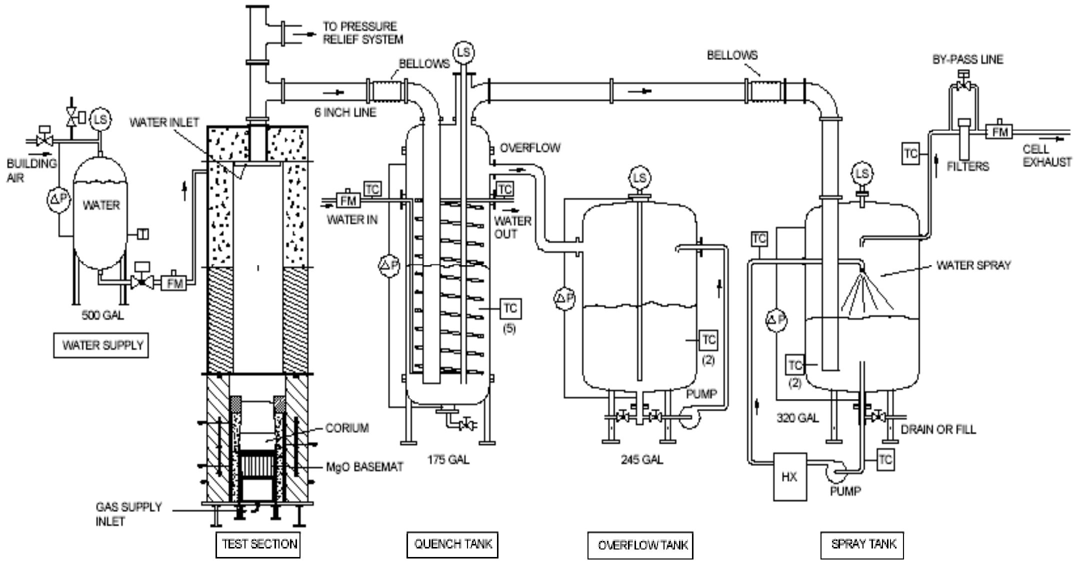

Figure 6 Basic Elements of the MET Apparatus.

Three modes of heat removal from the melt pool have been identified. These are the (1) initial melt-water contact; (2) the conduction through the crust; and (3) melt eruptions into water, when the heat generated in the melt is greater than that removed by conduction or water ingression through the crust. In the large test (120×120×20 cm), it appears that significant water ingression occurred since after the test the crust (or cooled melt) was 10 cm thick, i.e. about half the melt was cooled. Continued concrete ablation led to the separation of the melt pool from the suspended crust, and the conduction heat transfer decreased substantially. A 50×50×25 cm integral melt coolability test with siliceous concrete was performed whose results were approximately the same as for the earlier tests. The siliceous concrete has much less gas content than the LCS concrete. Its volume reduction due to ablation is also less than that of LCS.

The integral test program was modified to investigate the three modes of heat transfer through separate effect tests with the intent of developing validated models which could be employed for the evaluation of prototypic consequences. A test simulating the melt eruption was performed in which gas injection rate at the bottom of the melt pool was varied and melt eruptions into overlaying water were generated. Data on the entrainment coefficient were obtained.

A new project named MCCI has started last year under the sponsorship of OECD. The objective is to continue the separate effect tests (see Figure 6) to obtain sufficient information to model the heat transfer processes occurring. Tests have been performed to study the water ingression mechanism through which the melt pool is cooled slowly and the crust thickness increases as it was found to occur in the 120×120×20 cm test. The test results are being reported in a paper presented at the forthcoming ICAPP meeting. The tests appear to find that the water ingression mechanism is melt material dependent and, in particular, it was found that the addition of concrete products to the oxidic melt pool decreases the water ingression rate markedly. The other mechanism of heat removal from the melt pool: melt eruption into water depends on the gas generation rate from concrete ablation. This mechanism will not be as active in the ablation of the siliceous concrete found in Europe since its gas content is quite low. The ablation of the limestone-common sand concrete may be able to support melt eruptions due to the larger gas generation, however, it is not clear what fraction of the melt pool could be cooled with this mode of heat transfer.

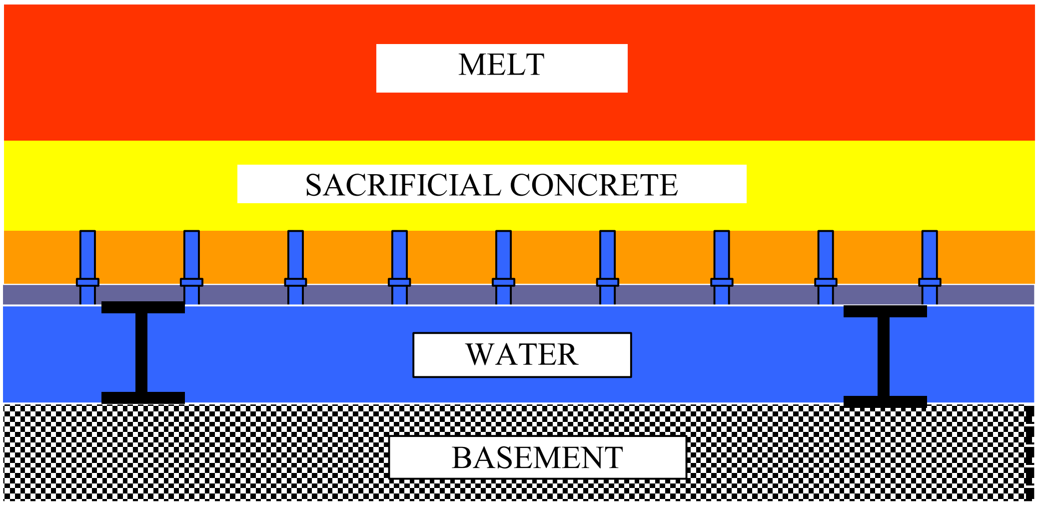

Figure 7 COMET: Bottom Water Injection through Plastic Tubes embedded in Sacrificial Concrete.

Currently, it is not evident that coolability of a corium melt pool by a water overlayer can be certified. Perhaps, at plant scale, with spans of several meters, the top crust will be unstable, and there would be periodic contact between the melt and water to eventually cool and quench the melt. It is clear that some basemat ablation will occur during the coolability process. One benefit of the water overlayer should be mentioned: the water will scrub most of the fission products that are produced during the molten corium concrete interaction (MCCI).

Since melt coolability with a water overlayer may be hard to achieve, alternative and innovative means have been explored to cool and quench the melt. Experiments have been performed (Alsmeyer et al., 1998) at the COMET facility (see Figure 7) in FZK in which water is introduced at the bottom of the melt pool with a slight overpressure, either through nozzles or through a porous concrete substrate. It has been found that melt cooling and quenching is quite readily accomplished and that no steam explosion occurred even with the Al_2O_3 melt. It appears that addition of sacrificial concrete in the Al_2O_3 melt reduces its explosivity considerably. Another reason may be that the water is injected in the melt pool at very low rate; and it evaporates readily and does not provide the conditions for forming a pre-mixture, which can lead to a steam explosion. The COMET design is currently being optimized through a series of experiments at different scales. This concept has merit since it uses the same principle as in the coolability of a particulate debris bed with water injection at the bottom. The co-current water and steam flow are much more efficient in cooling and quenching a melt pool than the counter-current flow that occurs when the melt pool is flooded at the top. It should be noted here that the water injected at the bottom creates porosity in the melt which provides paths for the water and steam to cool the melt. Presently the physical mechanism that creates the porosity cannot be described accurately. Experiments performed at KTH in the

DECOBI Program (Paladino et al, 2002) have delineated the influence of the melt viscosity and melt structure (e.g. ceramic). It is found that porosity is difficult to create for melts with greater viscosity. Thus it is advisable to inject water into the melt bottom boundary before a large quantity of silecious concrete mixes with the corium melt.

The COMET concept can only be accomplished in the current plants with extensive modifications in their containments. Another concept which is under study at KTH is that of downcomers built into the containment which channel the water from top of the melt pool to the bottom of the melt pool, thereby utilizing the already proven high cooling efficiency of the bottom water injection. The downcomers, we consider, are cylindrical tubes which are taller than the expected melt height, open at top and bottom, constructed in the vessel cavities of PWRs and the lower drywells of BWRs. The downcomers could be protected by a ‘hat' at the top to prevent entry of particles and melt into it. The particulate bed or the melt pool would surround a set of these downcomers and as the top of these is flooded, the downcomers will be filled with water, which is lead to the bottom of the debris bed. Clearly, there is no driving head to establish a two-phase natural circulation loop between the top and bottom of the debris bed or melt pool. A loop however, is expected to be established in which water goes down in the downcomer and the steam rises (after the evaporation of water entering at the bottom through the debris bed or the melt pool). Thus, the quenching of the top flooded debris bed or melt pool, which has to fight the CCFL is enhanced by the much more efficient co-current cooling process brought on by the availability of water at the bottom. We believe that such an innovative cooling system can be installed in existing PWRs and BWRs, quite easily, without jeopardizing the regular functioning of the plant, or the periodic shutdowns or inspections.

Experiments performed in the POMECO facility (Konovalikhin and Sehgal, 2001) have demonstrated the benefits of the downcomers through a several fold enhancement of the dry out heat flux and the quenching rate. Currently similar experiments are being performed in the COMECO (<u>CO</u>rium <u>ME</u>lt <u>CO</u>olability) facility with a melt pool, around a downcomer, flooded from top. Preliminary results indicate substantial benefit of the downcomer. However further experimentation is needed and will be performed.

6. EX-VESSEL STEAM EXPLOSIONS

The steam explosion loads on the containment were first considered in the WASH-1400 and, because of the assumptions made about the nature of this event at that time, the failure of containment (due to in-vessel steam explosion generated missiles) contributed a substantial fraction of the probability for early containment failure. The work on steam explosions (Theofanous, 1987) since that time, led to more realistic estimates of the probability of containment failure due to in-vessel steam explosions. A steam explosion review group (SERG) established in 1995 (SERG2, 1995), deliberated on the phenomenology of the steam explosion and provided expert estimates on the probability of the containment failure as a result of an in-vessel steam explosion.

Much experimental and analysis-development work is in progress, presently, on in- and ex-vessel steam explosions. Experiments have been performed with less than gram quantities to several kilogram quantities of heated particles and molten materials. Elaborate three-field analysis codes: MC3D, IVA, ESPROSE.m and PM-ALPHA have been developed. Some of the insights gained are: (1) steam explosion probability is much reduced due to the extensive water-depletion that occurs around the fragmented particles of a jet in the premixture; (2) super-critical steam explosions, however, cannot be excluded.

Ex-vessel steam explosion loads on PWR and BWR containments are also an issue, since (a) in some PWRs, water discharged from the reactor primary system accumulates in the reactor cavity under the vessel; and (b) in some BWRs, a deep water pool is established under the vessel, prior to vessel failure: an accident management strategy employed in the Swedish BWRs. The ex-vessel water is generally highly subcooled and the extensive voiding, that develops in the premixture in a saturated pool, may not occur in the subcooled pool. Additionally, it has been found that the median particle size, obtained during the break-up process, may be much smaller for the subcooled water than for the saturated water. Contrary to these effects, which may argue, on heuristic grounds, for a larger probability of a steam explosion, there are the effects of cooling and solidification which argue for a reduction in the probability of a steam explosion. The corium melt may be a complex mixture of metals and oxides, however, predominantly it is a mixture of UO_2–ZrO_2–Zr, whose phase diagram, in general, shows a liquidus curve and a solidus curve, which are apart from each other by at most 100 to 200 K. For the UO_2–ZrO_2 mixture the difference between the liquidus and the solidus curve is only 50–75 K. As the corium mixture solidifies its properties change drastically. In particular, the viscosity, which is infinite in the limit of solidus, changes radically. The process of break up of a corium melt jet during its interaction with water results in many corium melt droplets of complex shape undergoing solidification from the exterior surface to the interior of the droplets. The changes occurring in the physical properties of the droplets affect the potential for the participation of the droplets in the steam explosion process. For example, it has been found that a thin high viscosity layer on the surface of a spherical droplet will greatly impede its subsequent fragmentation by a pressure wave, or shear forces. The most remarkable experimental observations are derived from the experimental program employing prototypic corium melt (UO_2–ZrO_2) in the FARO (Magallon, 1999a) and (UO_2–ZrO_2) and Al_2O_3 in the KROTOS facilities (Huhtiniemi, 1999) at Ispra, Italy.

It is clear that if a relatively large steam explosion occurs near the bottom of the 7 to 11 meters pool; the Swedish BWR containment (in particular the pedestal) can fail. It is also clear that the existing experiments, so far, indicate that the conversion ratio (or energetic yield) in a triggered UO_2–ZrO_2 explosion is significantly less than that in a triggered Al_2O_3 or stainless steel melt explosion. There are some ‘limiting mechanisms’ which reduce the yield for non-eutectic oxidic mixtures. In this context we still have to establish if the UO_2–ZrO_2–Zr mixture will behave differently from the UO_2 – ZrO_2 mixture.

These significant observations point to the important role that the melt physical properties may be playing in the steam explosion process. Some research on this aspect was pursued in Europe under the auspices of the European Commission. Some physical mechanisms have been identified. However, no comprehensive answer has yet been given for the question on the role of the melt physical properties on energetics of steam explosions (‘limiting mechanism’).

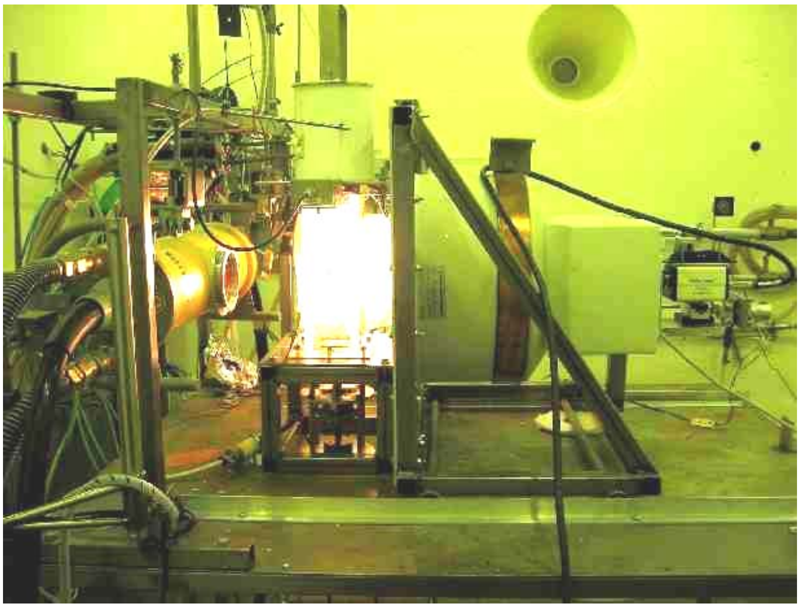

Figure 8 The MISTEE Facility at KTH

At KTH in Sweden, a small laboratory scale experimental facility (Park, 2004) was constructed to investigate the detailed explosion phenomena with different materials. The facility called MISTEE (Micro Interactions in Steam Explosion Experiments), as shown in Figure 8, is uniquely equipped with high-speed X-ray radiography system to visualize the fast transient MFCI (Molten Fuel-Coolant Interactions). This capability of the system facilitates to observe the dynamic fine fragmentation process with consistent sequential visualization, which was not possible in the previous studies. We applied this real-time continuous high-speed X-ray radiography to visualize the fine fragmentation process during a single molten drop interaction with water. The detailed qualitative observation by X-ray images, pressure, temperature measurements will provide an insight on steam explosions with different materials. The facility is also capable of performing the small jet steam explosions.

7. SEVERE ACCIDENT STABILIZATION AND TERMINATION

It is clear that long term stabilization and coolability of the corium melt is one of the goals of the near-future new LWR plants. Two lines of mitigation measures have been focused in the new designs: (a) in-vessel coolability and retention and (b) ex-vessel coolability and retention.

7.1. The In-Vessel Melt Retention (IVMR) Strategy

The in-vessel coolability and retention is based on the idea of flooding the PWR vessel cavity or the BWR drywell with water to either submerge the vessel completely or at least submerge the lower head. The PWR or BWR lower head containing the melt pool is cooled from outside, which keeps the outer surface of the vessel wall cool enough to prevent vessel failure. This concept is employed in the LOVIISA VVER-440 in Finland, where it has been approved by the regulatory authority STUK. The concept is also employed in the PWR designs: AP-600, AP-1000, Korea Advanced PWR-1400 and in the FANP's SBWR design.

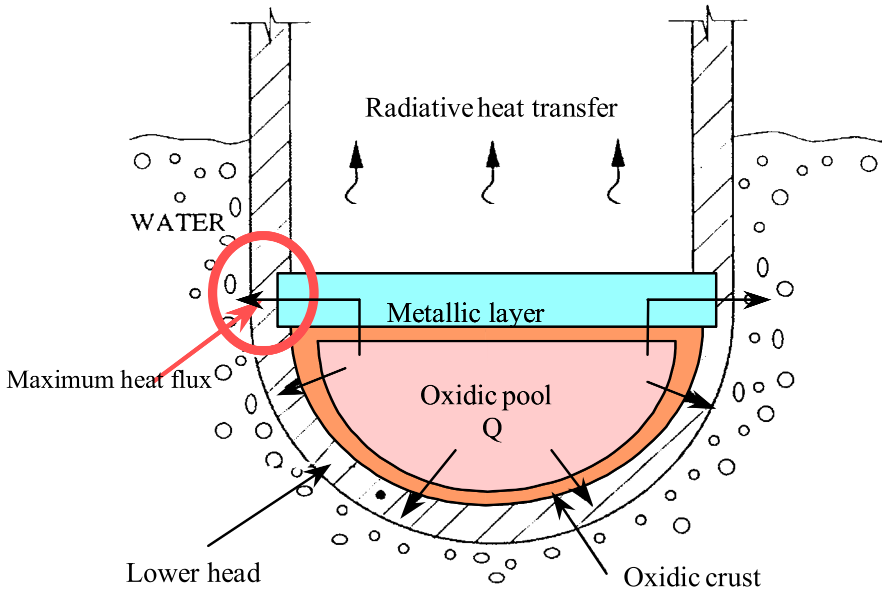

Figure 9 In-vessel Melt Retention.

The AP-600 design was analysed in Theofanous et al. 1995 with a bounding accident assumption of the lower head full of convecting melt pool. They found that the heat flux varied with angle, peaking near the equator. Fortunately the heat removal by the water outside also varied with angle reaching highest value also near the equator. It was found that for a uniform corium pool for the 600 MWe AP-600 reactor, there was sufficient margin between the critical heat flux (CHF) on the water side and the incident heat flux from the corium pool. This margin of safety, however, may be reduced to zero in case there is a metal layer present on top of the oxidic corium pool. The metal layer results from the steel present in the PWR and the BWR lower head which is melted by the corium pool and since it is lighter it rises to the top of the corium pool (see Figure 9). At IVO, Finland, the COPO experiments (Kymäläinen, 1997) were carried out to investigate the molten pool heat transfer in a vessel lower head for the LOVIISA plant. The facility as shown in Figure 10 has a large-scale two-dimensional geometry with an elliptical lower head different from the hemispherical lower heads of the Western LWRs. From their results with water simulant they concluded that the thermal loads on the LOVIISA RPV wall from molten corium pool remain at sufficiently low values and thus melt-through of the RPV are unreasonable in all scenarios in which the reactor cavity is flooded. The COPO II facility (see Figure 11) was also constructed to investigate heat transfer behavior of a molten corium pool using simulant fluids. The facility is a two-dimensional slice of the lower head of a reactor vessel in the linear scale 1:2 (larger scale than the SIMECO facility at KTH 1:8). The heat generating corium was simulated with water-Zinc sulphate solution with direct joule heating and are cooled by liquid nitrogen. The main objective of the COPO-II experiments was to perform similar kind of stratified pool experiments performed at KTH with the SIMECO facility (see Figure 12). Another objective was to compare the data obtained to those obtained in the earlier COPO uniform pool and the stratified pool experiments, in which an intermediate plate and an upper layer of distilled water was employed. This large-scale experiments confirm that the upward heat flux was significantly reduced compared to that measured in the earlier uniform pool COPO experiments.

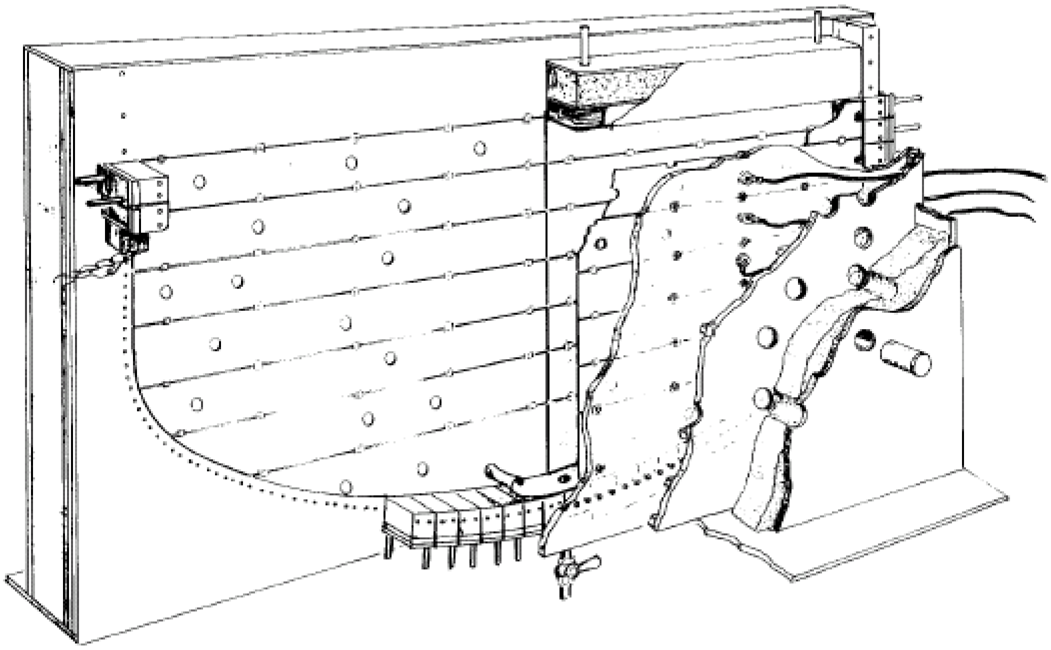

Figure 10 The COPO-I Facility at IVO in Finland (From Kymäläinen, et al., 1997)

The metal layer receives heat from the corium pool and performs Raleigh-Benard convection which transfers heat transversely to the vessel wall, which is then subject to a highly elevated heat flux. This heat flux focusing is most intense for a thin metal layer since the transverse area for heat transfer is smaller. It was found that for metal layers of ≈ 20 cm depth the focused heat flux could overwhelm the critical heat flux near the equator. For the AP-600, it was found that the metal layer would be thick and there was sufficient margin available between the focused heat flux and the CHF outside.

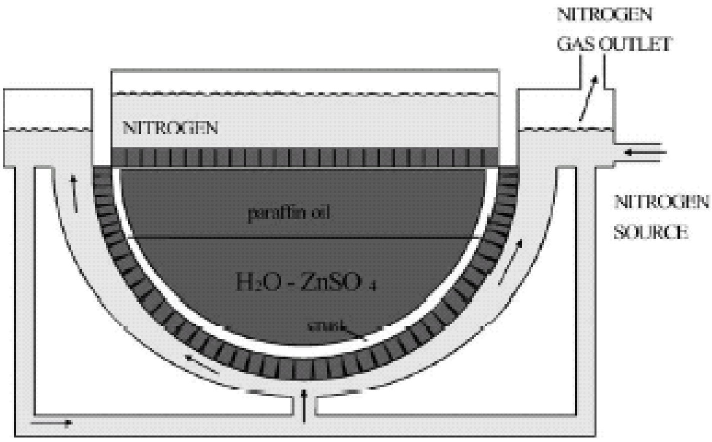

Figure 11 The COPO II-AP Facility at IVO in Finland (From Sehgal et al., 2003b)

The power of the AP-1000 is 60% larger than that of AP-600 and that of Korea Advanced PWR by 230%. For the 1400 MWe reactor, the focused heat flux would be greater than the CHF on the water side. The strategy of the Korean plant is to simultaneously flood the metal layer with water inside the vessel, which could remove sufficient heat from the upper face of the metal layer to reduce the focused heat flux to values less than the CHF. A dedicated water system has been installed in the plant for water injection to reach the lower head at the appropriate time.

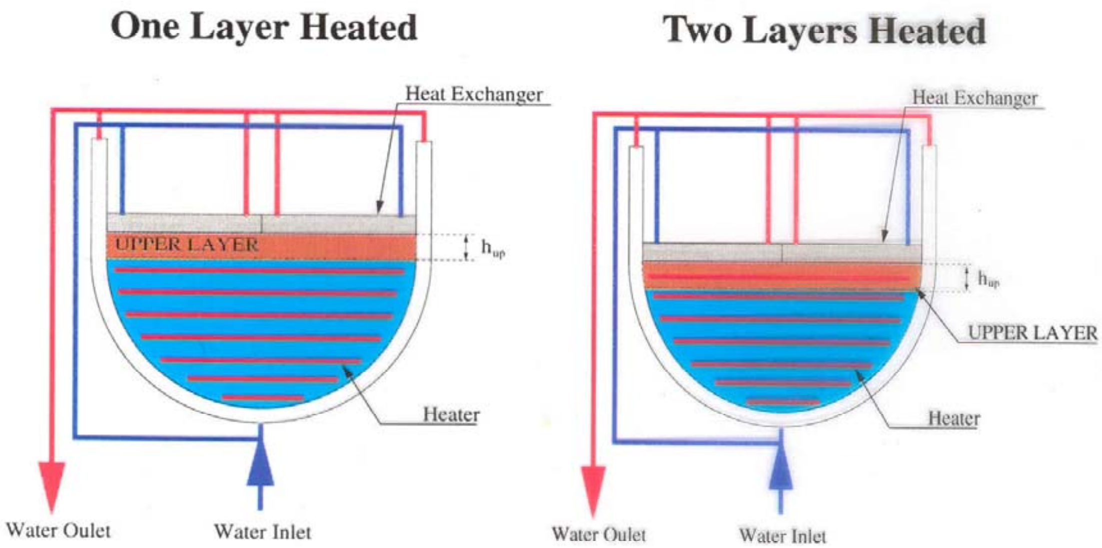

Figure 12 The SIMECO Test Facility at KTH.

Further complications have been introduced recently by the findings in the RASPLAV and the MASCA Projects (Asmolov et al., 2003) of chemical reactions between the melt constituents which may create different layer configurations in the melt pool. For example it was found in the RASPLAV Project that presence of even small amounts of carbon in the system promotes the stratification of the melt pool by separating the oxides from the metals in the melt, thereby forming a light melt layer, rich in metals, residing on top of the oxide-rich melt pool. Another finding from the MASCA Project is that of the combination of the steel components with Uranium to form a metal compound which is heavier than the oxidic pool, which sinks to the bottom of the oxide-rich melt pool. It is not clear whether all the steel will combine with the Uranium metal. The initiator of this steel-Uranium combination is the unoxidized Zr present in the melt. The worst situation would be in which some of the steel is taken by Uranium metal to the bottom of the pool, while some remains at the top to form a thin metal layer which can provide strong focusing of the heat on the vessel wall. The melt pool composition and configuration situation is quite confused presently. There is also no data on pool convection in a three-layer pool and the heat flux distribution its convection imposes on the vessel wall. At KTH, we have obtained data on a two-layer pool simulating a two oxide-layer pool, i.e., a two-layer pool of metal and oxide layers, in the SIMECO facility (see Figure 12). The SIMECO facility can be easily modified to the three-layer pool to investigate the heat flux distribution in this configuration. These experiments will provide data on the angular distribution of the heat flux as a function of layer heights, layer heat inputs and layer composition and density. Such data are needed to clarify the arguments about the safety margins in the in-vessel melt retention (IVMR) accident management strategy.

7.2. The Ex-Vessel Melt Retention (EVMR) Strategy

This strategy has been adopted by the EPR design and by the new Russian VVER-1000 design for China and India. The EPR design (see Figure 13) spreads, cools and retains the discharged corium, mixed with sacrificial concrete, on a flat surface made with Zirconia bricks which are cooled from bottom with a heat exchanger and the spread melt pool is flooded from top. The idea behind this design is that with spreading the depth of the melt pool will be reduced to values which can be cooled by a water overlayer. Sacrificial concrete is mixed with corium discharged from the RPV in a concrete vessel to reduce its temperature, and more so, its solidus temperature. Thus the mixture remains a liquid over a much larger temperature range, and, in fact, will spread more easily and over a larger floor area.

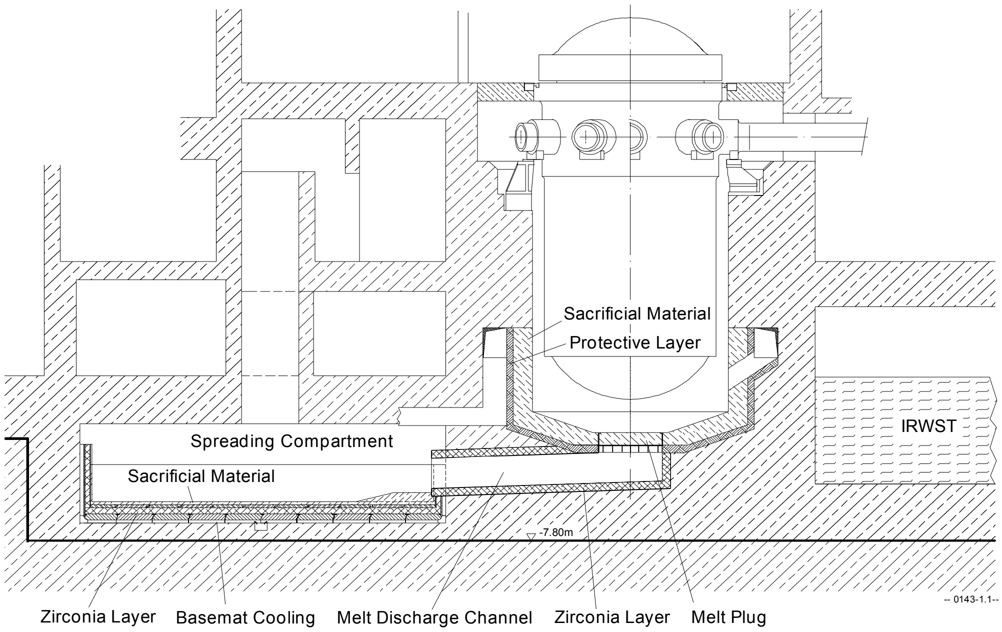

Figure 13 EPR Core Catcher.

Much research was performed on the efficiency of spreading of the melt at various European laboratories including at KTH. We developed a very innovative scaling theory for spreading (Dinh et al., 1998) which has been able to predict most of the spreading data obtained with simulant and prototypic melt materials. The EPR melt spreading analysis was also performed with this model and it was found that even with conservative assumptions, uniform spreading of the discharged melt and concrete mixture can be obtained in the EPR design. The depth of the melt unfortunately is greater ($\approx$ 40 cm) than that can assure melt coolability with water flooding alone. The cooling coils built in the base of the spreading chamber will be needed to cool the melt. It appears, however that it could take several days before the center part of the spread melt pool will solidify.

The Russian VVER-1000 design employs a core catcher in the traditional sense. This core catcher shown in Figure 14 is a separate vessel installed under the RPV with an intake designed to cover almost the whole surface of the bottom head so that the melt discharged from the RPV is deposited in the core catcher even if the RPV failure occurs at an angular position close to the equator (which it will). The core catcher is like a lower head but of much larger volume and it is cooled from outside by a water pool as in the IVMR concept. The core catcher is full of bricks made of oxidic material containing Fe_2O_3 and other oxides. The purpose is the same as in EPR: to reduce the temperature of the discharged corium and to keep it liquid over a larger range of temperature. The core catcher walls are steel but they are lined with oxide bricks. The chemistry of the materials with the corium has been subject of several experiments and the chosen oxide composition is such that the Uranium and the metals in the corium combine to form a dense metal layer which sinks to the bottom of the melt pool. There, supposedly is no metal layer on the top of the oxidic pool. The melt pool is flooded with water without fear of a steam explosion since the metal is at the bottom under the oxidic material pool.

The melt pool in the Russian core catcher design also may remain molten for long time and will perform natural convection. There are some misgivings about that and also about the possibility of chemical attack on the core catcher lining to fail the core catcher vessel.

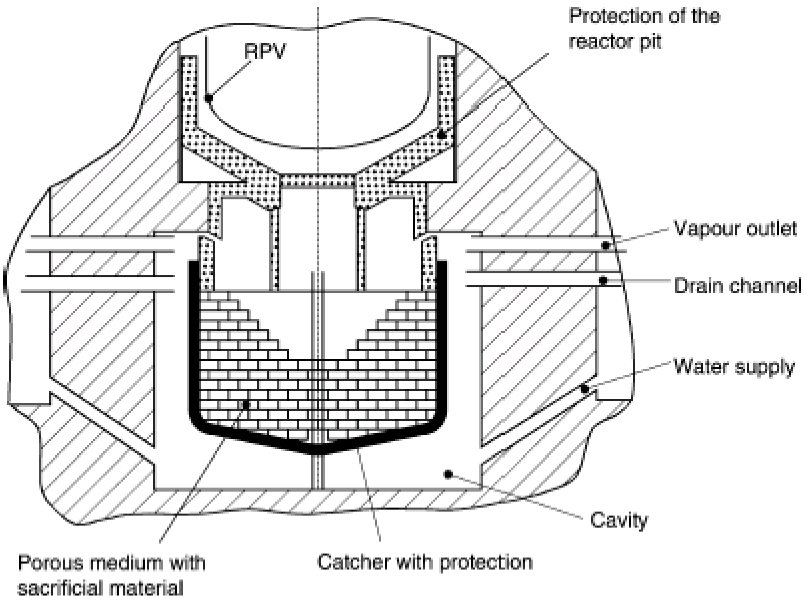

Figure 14 Tian Wan Core Catcher (From Seiler, J. M., 2003).

Chapter IV

PROPOSED PROJECT PLANS ON CORE MELT LOADINGS IN SEVERE ACCIDENTS

The prioritized remaining issues identified in the previous chapters suggests some future research efforts on the remaining, unresolved issues of most interest to the Nordic power companies and government regulatory organizations concerning core-melt loading in severe accidents. We believe these future research efforts in the order of priority should be on the areas of;

- in-vessel coolability of the melt pool or particulate debris,
- ex-vessel coolability of the melt pool or particulate debris,
- energetics and fragmented debris characteristics of a steam explosion endangering the integrity of the BWR vessel and containments, and,
- characteristics of vessel failure.

In the following sections of this chapter will describe the suggested plans to resolve the issues.

1. PROJECT PLAN FOR IN- AND EX-VESSSEL COOLABILITY OF MELT AND/OR PARTICULATE DEBRIS

1.1. In-vessel Coolability

We suggest that the unresolved issues on in-vessel coolability can be investigated at two different facilities shown in Appendix, i.e., the COMECO (corium melt coolability) facility and the POMECO (porous media coolability) facility. In particular, the three-dimensional coolability can be investigated at the modified and larger version of the POMECO facility, called POMECO–GRAND facility. The work will also be directed towards representation of the BWR lower plenum with its forest of control rod guide tubes (CRGTS). The scaling is based on a unit cell containing a CRGT with its associated melt, or particulate debris, volume around it receiving the prototypic amount of heat through internal or external heaters.

The previous scoping experiments employed two flow rates in the COMECO facility but detailed measurements of the axial variations of heat transfer to the CRGT were not obtained. The COMECO experiments are very complicated, time consuming and expensive since they are with high temperature melt, with special high temperature heaters around the test section supplying heat to the melt. The previous experiments provided reasonable results but it is needed to confirm those with thermocouples inserted in the CRGT at various levels to obtain detailed measurements on heat transfer as a function of time. The experiments with different flow rates, subcooling and for different scenarios of the depth of water pool on top of melt to determine the effect of the water entry through bypass holes will evaluate the applicability of the CRGTs as a measure for the in-vessel retention in Nordic BWRs. The scenario, we believe, of interest is that of the re-melting of the particulate debris bed which is formed in the lower head of BWR when the melt is discharged to the lower head.

The POMECO–GRAND facility can determine the coolability of the three-dimensional particulate debris beds. Previous experiments were performed with the POMECO facility in which a full scale CRGT was installed and data on heat removal capability of the CRGT was determined for different flow rates, subcooling and scenarios of water height.

No radially-stratified debris bed experiments were performed in the POMECO facility, although axially-stratified debris beds were constructed and their characteristics measured. The purpose of the POMECO–GRAND facility is to (a) add more than 1 CRGT in the debris bed to determine the interaction, (b) add radial stratification and (c) determine 3-D effects in coolability. The POMECO–GRAND facility will also be used for the ex-vessel coolability investigations. This facility may have a test section 60 cm x 60 cm in cross section and 75 cm height to build deep debris beds of different porosity and stratification (both radial and axial). Water entry into the bed could also be made from the sides of the bed in order to have a 3-D distribution of cooling. Since the volume of this facility is about 5 times that of the POMECO, it will require approximately 250 kW of power supplied by a large complex of thin heaters which will be built in the test section. A large array of thermocouples will also be provided.

The SIMECO facility will be used for investigating the heat transfer behavior in three-layer configuration of the corium pool and comparing with the rich database on single and two-layer melt convection obtained from the previous SIMECO and COPO experiments.

The specific items to be investigated in the POMECO and POMECO-GRAND are the effects of (a) radial porosity, (b) non-uniform heat generation, (c) homogeneous bed and uniform heat flux, and (d) pressure on dryout and quenching of debris bed. The specific items to be investigated in the COMECO are the melt coolability and water ingression with different melt properties during the top water flooding and water injection at the bottom of the melt pool using the CRGTs. Lastly, three-layer SIMECO experiments of various combinations of simulant layers with different heat generation rates and melt pool heights can be investigated. The experimental parameters are water injection (a) on the top of the melt pool, (b) through multiple CRGTs at the bottom of the melt pool in the POMECO, POMECO-GRAND and COMECO programs, (c) the internal heat generation rates, (d) melt pool density and (e) melt heights in the SIMECO program.

The proposed research can be performed with at least three man-years for 3 years.

1.2. Ex-vessel Coolability

For the ex-vessel coolability issues, the COMECO and the POMECO–GRAND facilities can again to be employed, except that they will employ air injection from the bottom. The emphasis in this research is on the enhancement of coolability due to the incorporation of downcomers for both the particulate debris beds and the melt pools.

The COMECO facility can employ one downcomer unit cell, while the POMECO–GRAND facility can employ several downcomers, in order to observe any interaction effects.

The POMECO–GRAND will measure the 3-D distribution effects and the coolability of both radially and axially stratified particulate debris beds. This task will bear 50% of the construction costs for the POMECO–GRAND and for the modified designs of the COMECO facility. The COMECO facility will also employ the TiO_2–MnO_2 melt, which has a different material structure than that of the CaO–B_2O_3 melt. We believe that there may be differences in the water ingression efficiency in the melt pool due to material structure differences. The TiO_2–MnO_2 melt has a ceramic structure when it cools down, while the CaO–B_2O_3 melt is of glass structure as it cools. There are also differences in the viscosity of the two melts; the CaO–B_2O_3 melt has much higher viscosity.

In the POMECO (not POMECO–GRAND), different type of sands or particles which have greater density than of the sand used in the previous POMECO experiments can be employed, since there may be effect of particle density on debris bed coolability.

The specific items to be investigated in the POMECO and POMECO-GRAND are the effects of (a) radial porosity, (b) non-uniform heat generation, (c) homogeneous bed and uniform heat flux, and (d) pressure on dryout and quenching of debris bed. The experimental parameters are (a) water injection (only through center downcomer,only through side pipes, center downcomer and side pipes, and all downcomers and side pipes), and (b) noble gas injection in addition to air injection. The proposed research can be performed in at least two man-years for 3 years.

2. PROJECT PLAN FOR ENERGETICS AND FRAGMENTED DEBRIS CHARACTERISTICS AND LIMITING MECHANISMS OF EX-VESSEL STEAM EXPLOSION

2.1. Experiments in Steam Explosion

It is clear that if a relatively large steam explosion occurs near the bottom of the 7 to 11 meters pool; the Nordic BWR containment (in particular the pedestal) can fail. It is also clear that the existing experiments, so far, indicate that the conversion ratio (or energetic yield) in a triggered UO_2–ZrO_2 explosion is significantly less than that in a triggered Al_2O_3 or stainless steel melt explosion. There are some limiting mechanisms which reduce the yield for non-eutectic oxidic mixtures. In this context we still have to establish if the UO_2–ZrO_2–Zr steel-mixture will behave differently from the UO_2 – ZrO_2 mixture.

Since it is infeasible to perform large-scale steam explosion experiments with UO_2–ZrO_2 or UO_2–ZrO_2–Zr and it is very difficult to establish a scaling relationship, we believe that a more fundamental investigation will bear fruit in terms of identifying the limiting mechanisms. We accomplished this for the jet break-up phenomenon and now we have constructed the micro interaction steam explosion experiments (MISTEE) facility shown in Appendix, wherein, currently, we will be observing the differences between the character of the explosion phase of a steam explosion, for a single droplet, of different material (metal, single oxide, binary oxide mixture, binary oxide and metal mixture, etc.) melts. We obtain conversion ratio by employing a very fast pressure transducer. As experimental parameters, we will vary the subcooling of water, trigger strength and the melt droplet superheat. The objective is to study and model the limiting mechanisms for steam explosions.

In addition, relating to the coolability issues mentioned in the previous section, the information of melt debris size distribution due to steam explosion will be of importance to determine the configuration (for instance, debris porosity) of melt debris inside or outside the vessel relocated during severe accidents. In this study, therefore, the characteristics of fragmented debris in terms of thermal and mechanical conditions imposed in steam explosion process and material properties can be carefully examined. In so doing, a molten jet can also be employed with the X-ray visualization to investigate the jet break-up during the pre-mixing phase and the fine fragmentation during the explosion phase.

The specific items to be investigated are (a) metallic single drop steam explosions, (b) oxidic single drop explosions (single oxide and binary oxide melts), (c) metallic melt jet and (d) oxidic melt jet (single oxide and binary oxide melts). The experimental parameters are (a) thermal conditions (melt and coolant temperatures, etc.), (b) dynamic conditions (triggering strength, drop or jet velocity etc.,), and (c) thermo-physical properties (eutectic and non-eutectic composition, density, heat capacity, viscosity of melt, etc.). The quantitative experimental outputs are (a) X-ray radiography and photography images, (b) transient melt fragmentation and break-up distribution, (c) dynamic pressure and local temperature histories, and (d) fragmented and break-up particle distribution.

The proposed work requires at least two man-years per year for three years.

2.2. Analysis of Steam Explosion

Steam explosion phenomenon consists of various distinctive sub-phenomena classified with four phases as the phenomenon progresses, such as pre-mixing, triggering and explosion, propagation and expansion. The complexity of the phenomena naturally leads to the difficulties in analytical modeling.

Recent experimental results from the MISTEE facility showed the detailed fine fragmentation and the vapor dynamics of a single molten liquid droplet during the explosion pictured by the high-speed X-ray radiography and high-speed photography, respectively. These results provide the re-look of the triggering mechanism of steam explosion phenomena. In addition, the quantitative measurement of transient fine fragmentation during the explosion after a series of image processing of the X-ray images with fragment mass calibration provides the new experimental data to verify or develop the fine fragmentation model which is considered as one of the key unresolved models to evaluate the energetics of steam explosions. The appropriate fine fragmentation model can also be used for predicting the final debris size distribution after the steam explosions.

Experimental data from the MISTEE experiments with a molten jet can be used for validating and developing the jet breakup models. The MISTEE experimental data are uniquely different from the a number of previous experimental data because the transient breakup of a molten jet can be quantified and visualized by the X-ray radiography, which was impossible in the other previous experiments with the photography since the melt jet can not be visualized by the photography due to the existence of vapor film around the melt jet during the mixing.

The existing mechanistic computer models, such as TEXAS, PM-ALPHA, ESPROSE.m, IVA, MC3D and COMETA codes, can be a platform to evaluate these new types of experimental data. At KTH, the COMETA code is recently under evaluation and development to enhance its capability to predict the consequences of steam explosions.

The specific items to be investigated are (a) evaluation and re-modelling of triggering and explosion models based on the MISTEE results, (b) evaluation and re-modelling of jet break-up models based on the MISTEE results, (c) verification of the existing large scale experimental data from FARO (JRC, Ispra), KROTOS (JRC, Ispra) and TRIO (KAERI, Korea) and (d) prediction of energetic steam explosions in the prototypic conditions.

The proposed work requires at least one man-year per year for three years.

Summary

SUMMARY

Reactor safety's concern with severe accidents, since the TMI-2 accident led to almost twenty years of intense research efforts, which have resolved a number of severe accident issues. Lately, research has been concentrated on accident management and a number of LWR plants, around the World, have adopted severe accident guidelines (SAMGs) and strategies.

This report mainly consists of three chapters, the assessment of severe accidents, the remaining unresolved issues of severe accidents and proposed research efforts to resolve these issues. This report reviews the state of the art of the various melt/debris coolability situations and ex-vessel steam explosions during the postulated severe accident scenarios. It addresses the unresolved issues concerning the core melt loadings during the severe accidents, and finally suggested the experimental facilities in the Nordic countries which could be useful to resolve the issues.

In this report, the issues in severe accidents was categorized into two parts; (a) early containment failure and (b) late containment failure. For the early containment failure, the phenomena identified were from the events, i.e., (a) in-vessel explosion, (b) ex-vessel explosion, (c) hydrogen detonation, (d) direct containment heating (DCH) and (e) melt attack on MARK-I BWR containment wall. The current status of the resolution of these issues were identified in this report describing that issues on en-vessel and hydrogen detonation were still unresolved or partially resolved, respectively. The phenomena concerned with the late containment failure, i.e., (a) molten core-concrete interaction (MCCI), melt coolability, and containment venting were not yet completely resolved.

The review on the phenomena in this report, which threaten the integrity of the containments during severe accidents provided a list of remaining, unresolved issues concerning core melt loading in severe accidents in the order of priority; i.e.,

(a) ex-vessel melt/debris coolability,

(b) ex-vessel steam explosion loads,

(c) basemat melt-through,

(d) lower head failure mode and its timing, and

(e) core quenching

Based on the consideration of these prioritized remaining issues and research infrastructures in Nordic countries, some future research efforts on the remaining, unresolved issues of most interest to the Nordic power companies and government

regulatory organizations concerning core-melt loading in severe accidents are suggested. These future research efforts in the order of priority are on;

- in-vessel coolability of the melt pool or particulate debris,
- ex-vessel coolability of the melt pool or particulate debris,
- energetics and fragmented debris characteristics of a steam explosion endangering the integrity of the BWR vessel and containments, and,
- characteristics of vessel failure.

During the last decade of research in the Nordic countries on the nuclear safety, in particular, on severe accidents, rich resources, such as experimental facilities and database were produced and accumulated. These infrastructures in the Nordic countries could be employed to pursue further research needed for resolving the remaining key issues.

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Appendix

APPENDIX

A.1 LISTS OF EXISTING RECOURSES IN NORDIC COUNTRIES

KTH, Sweden

Experimental Facility

- COMECO Facility
- DECOBI Facility
- FOREVER Facility
- MISTEE Facility
- POMECO Facility
- SIMECO Facility

Computer Codes

- ANSYS code
- CORQUENCH (from Argonne National Laboratory, USA)
- COMETA code (from JRC, Ispra)
- MVITA code

VTT, Finland

Experimental Facility

- SYTX Facility

Computer Codes

- BEDEXP code
- PASULA code

IVO, Finland

Experimental Facility

- COPO-I Facility
- COPO-II Facility

A.2 DESCRIPTION OF EXPERIMENTAL FACILITIES IN NORDIC COUNTRIES

COMECO FACILITY AT KTH IN SWEDEN

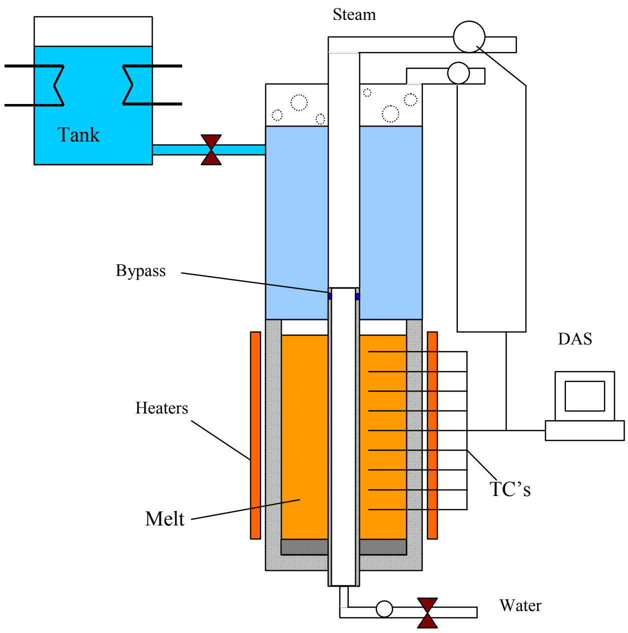

Figure 15 The COMECO Test Facility at KTH

The COMECO facility consists of a test section (200 × 200 mm cross section), with the maximum of 300 mm of melt height. The test section walls are made from 25 mm thick carbon steel and 24 thermocouples are placed within the test section. The test section is connected to the upper tank (which is 1000 mm high). Water is supplied to the upper tank via the water line from the heated water storage. A water level gauge is installed in the upper tank to monitor the water level variation during the experiments. The Control Rod Guide Tube (CRGT) model (with the outside diameter d_o=50 mm and the inside diameter of d_i=45 mm) is placed in the centre of the test section. The CRGT is connected to the water line at the bottom. Water at different subcooling temperatures could be supplied through the water line. Two bypass openings (of the diameter d = 9 mm) are made in the upper part of the CRGT model. Two flowmeters are installed on the steam outlet lines from the CRGT and the upper tank. A flowmeter is installed also on the water supply line, to measure the water flowrate through the CRGT. In the COMECO facility, the melt layer is heated directly by KANTHAL heaters, located on the sidewalls of the test section. Four heaters are installed on the four sides of the test section. The maximum power of 16 kW could be delivered to the melt pool. Thus, the COMECO experiments could be conducted at the maximum power density of 1.33 MW/m^3. Water to the upper tank is supplied from a water supply tank. The melt temperatures are measured at various locations within the melt pool. The temperature readings are obtained from 24 thermocouples, uniformly distributed within the melt pool. The thermocouples are placed at 8 axial elevations and at 3 radial locations in the melt pool. The steam flow rates, generated within the CRGT and the upper tank, are measured by the two Vortex type flowmeters made by Omega company. The measurement range of the flowmeters was up to 200 litre/sec. The heat removal rate is evaluated from the steam flow rate from the CRGT and the upper tank. The CRGT water flow rate is also measured using a liquid/gas flow meter, produced by the Omega company.

POMECO FACILITY AT KTH IN SWEDEN

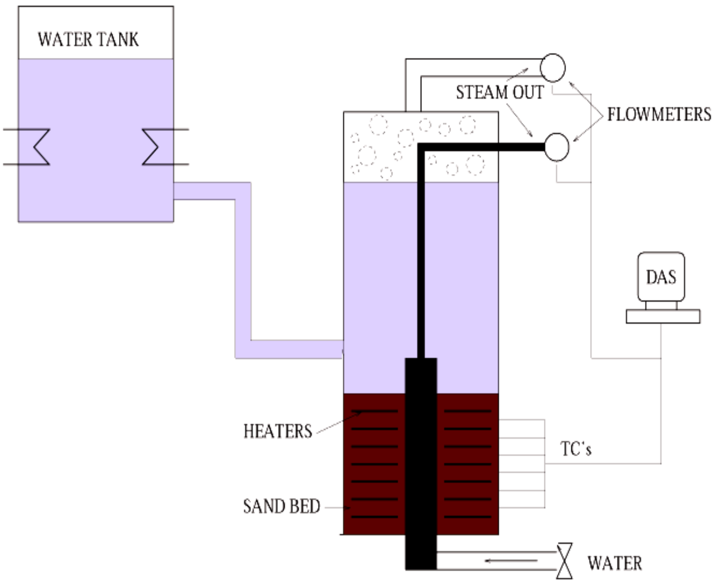

Figure 16 The POMECO Test Facility at KTH

The schematic of the modified POMECO (POrous MEdia COolability) facility, designed and constructed at the Nuclear Power Safety Division of the Royal Institute of Technology. The test section is a stainless steel vessel whose details of the test section are presented on the Figure. The cross-sectional area of the test section is 350×350 mm square. The height of the lower part is 500 mm and the height of the upper part is 900 mm. The maximum height of 370 mm can be reached for the sand bed. The POMECO facility contains an annular pipe of same dimensions as the actual CRGT in the BWR lower head inserted in the debris bed. The pipe has four holes at the top. These holes are of the same flow area as the bypass openings in the prototypic CRGT. The holes are designed to be open or closed. The CRGT annular pipe is led out of the POMECO facility so that the steam generated in the annular pipe can be measured separately from that generated in the water pool. The CRGT tube is connected to a water line at the bottom. Water flow rate is measured using flowmeters. A water level gauge is installed in the upper part of the test section to monitor the water level variation during the experiment. Sand particles of various parameters can be chosen to build up the porous particle beds of different mean particle sizes and porosities. In the POMECO facility, the sand bed is heated internally by a number of the thin electrical resistance heaters that are capable to provide up to 44 kW power (power density of up to 1 MW/m^3). Water is supplied from a water supply tank, as shown in Figure 1. Two heaters are installed in the water tank to keep the specified supply water temperature. Thirty-three thermocouples are distributed at different axial and radial locations in the particle bed. In addition, 9 thermocouples are embedded in the CRGT wall at three different wall depths and at 3 different elevations in order to obtain temperature distribution within the wall. The steam flow rates are measured by a Vortex Flow Meters made by Omega Company, which are installed on the steam lines (as shown in the Figure). The measurement range of these meters is up to 200 liter/sec.

POMECO GRAND FACILITY AT KTH IN SWEDEN

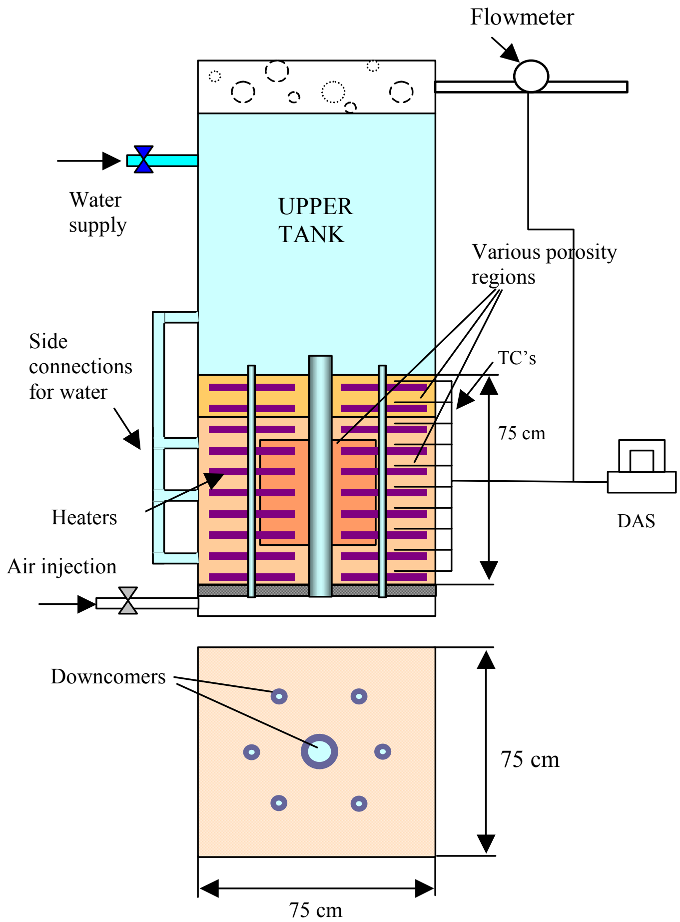

Figure 17 The Proposed POMECO-GRAND Test Facility at KTH

The test section is a stainless steel vessel with a cross section of 750 mm x 750 mm and 750 mm height. There is an upper tank connected to the test section of the same cross section but of 900 mm height. The upper tank will contain water, while the test section will contain a debris bed, which will be constructed with sands of different mean particle sizes.

The test section will contain several pipes representing (i) BWR control rod guide tubes (CRGTs) and (ii) downcomers of different cross sections. The CRGT pipes will be annular and each will have 4 holes at the top having the same flow area as for a prototypic CRGT. The holes will be designed to be either open or closed. The CRGT pipe will be designed to be connected to a pipe, which could be lead out at the top of the facility in order to measure the steam generated inside the CRGT.

The CRGT pipes will have water entry at the bottom with a valve and a flow meter. The downcomer pipes will also be designed to be either open or closed at the top. The main part of the test section will contain a series of thin heaters, which will disperse heat in the sand debris bed. The heaters will be spread evenly in the test section to achieve uniform temperatures in the debris bed. The temperature limit for the bed will be consistent with the temperature limitation on the heaters. It is expected that the bed temperatures will be close to 500˚C, with the heaters employed earlier for the POMECO facility.

The water entry into the bed will be from the top face, but connections from the upper tank water will also be made to the bed on the sides. At least 3 connections will be provided to each side of the test section. Each of these connections will be provided with a valve and a flow meter to control the magnitude of the side flows into the test section. This flexibility will allow measurement of the three dimensional effects in debris coolability. The bottom of the test section will contain a porous plate whose function is to allow water entry from bottom but to be a barrier for the sand particles. The porous plate is constructed with a series of small mesh screens through which water can flow. A chamber is built under the sand-bed test section in which the water transported by the downcomers from the upper pool is collected. There is also provisions for injecting air at the bottom of the bed, representing the gas generated by the molten corium concrete interaction (MCCI).

A series of thermocouples (TC's) will be added in the test section at different elevations and at different 'radial' locations to be able to map the temperature in the sand bed. The steam flow rates will be measured by a Vortex flow meter made by the Omega company. The meters will have the appropriate measurement range.

MISTEE FACILITY AT KTH IN SWEDEN

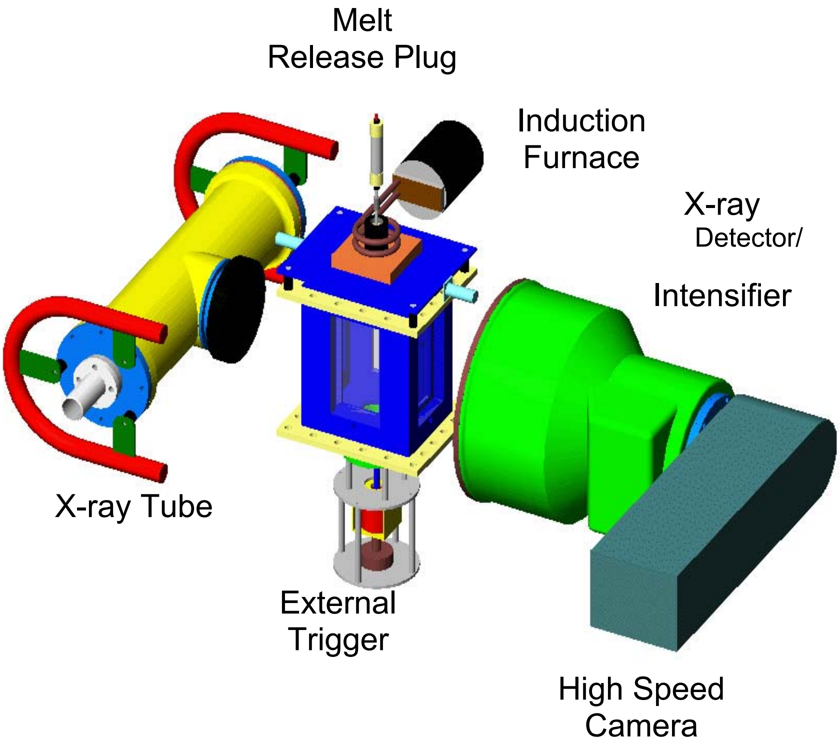

Figure 18 The MISTEE Facility at KTH

Steam explosion is observed when high temperature liquid comes into contact with cold and volatile liquid. In this phenomenon, rapid heat transfer between the high temperature liquid (e.g., molten materials) and cold liquid (e.g., water) produces explosive vapor generation, resulting in strong shock waves, which provide a hydrodynamic loading to a surrounding system. Our research activities on steam explosions are focus on: (1) investigating the triggerability and explosivity in a well-controlled facility of a high temperature melt droplet with an external trigger, (2) identifing the influence of melt thermo-physical properties on triggerability and explosivity of the melts, (3) acquiring quantitative data on the volume fractions of melt, coolant and vapor in the interaction zone during the fine fragmentation process in the explosions, and eventually (4) developing scaling methodology for the explosion phase of a steam explosion.

The MISTEE systemconsists of an interaction chamber, an induction melt furnace, an external trigger system, an operational control system, and data acquisition and visualization systems. The test section is a rectangular 304 stainless steel water tank (180x130x250mm) with 4 view windows (70x150mm), where at its bottom, a 1kW immersion heater is installed. A piezoelectric pressure transducer (sensitivity 75.0-750mV/MPa, rise time < 1.0μs), connected to the four-channel ICP signal conditioner, is flush-mounted on the center of a test section wall. K-type thermocouples are used to measure temperatures of the molten droplet at the furnace and water temperature inside the test section. The induction melt furnace with the voltage and current up to 260V and 40A, respectively, consists of a graphite cylinder (40mm O.D. x 50mm) and an alumina crucible (20mm I.D. x 30mm) with a 4.1mm hole at the center of the bottom. A Boron-nitride plug (10mm O.D. x 20mm) as a melt release plug is used to block the crucible bottom hole during the melting and it is lifted by a pneumatic piston to release the melt drop. The external trigger, located at the bottom of the water tank, is a piston that generates a sharp pressure pulse similar to a shock wave. The hammer is driven by a rapid discharge of a capacitor bank consisted of three capacitors ($400V_{dc}$ and 4700 mF each). The visualization system, photography and radiography, consists of a continuous X-ray source tube (0-320 keV and 0-22mA), an X-ray converter, an image intensifier and a high-speed video camera (4 seconds of recording time at 8000fps). The X-ray converter and image intensifier are powered by a high voltage power supply has three different magnification modes. The image size of the high-speed camera at 8000 fps is 80x70 pixels. The MISTEE facility is located inside a 0.6m thick reinforced concrete containment (3.8x3.8x3.9m) for the X-ray radiation shielding during the tests. The operation of the test is controlled remotely outside this containment.

At present, a series of tests using metallic melt (less than 1g of Tin at temperatures lower than 1200˚C in water ranging from 20 to near 100˚C) has been performed. High temperature experiments using oxidic melts, temperatures up to 2000˚C, are in progress.

STYX FACILITY AT VTT IN FINLAND

(referred from http://www.vtt.fi/pro/tutkimus/finnus/moses.htm)

Figure 19 The STYX-1 Test Facility at VTT

(From http://www.vtt.fi/pro/tutkimus/finnus/moses.htm)

Test facility has been constructed to measure the dryout heat flux in a heated, stratified debris bed. The facility is utilised to determine the dryout heat flux in conditions estimated for the Olkiluoto containment in severe accident.

Facility parameters are:

- pressure range 1 - 6 bar
- debris bed diameter 300 mm, height 600 mm
- Al_2O_3 particles simulate UO_2, generic size distribution
- a possibility for stratified beds, a fine particle layer (up to 100 mm) on top
- cooling from top
- resistance heating in 6 layers, max heat flux on top 1 MW/m^2
- ~ 60 thermocouples
- pressure, water level, coolant in, coolant out
- Dryout heat flux at various pressures has been determined in three sets of experiments during 2002

SIMECO FACILITY AT KTH IN SWEDEN

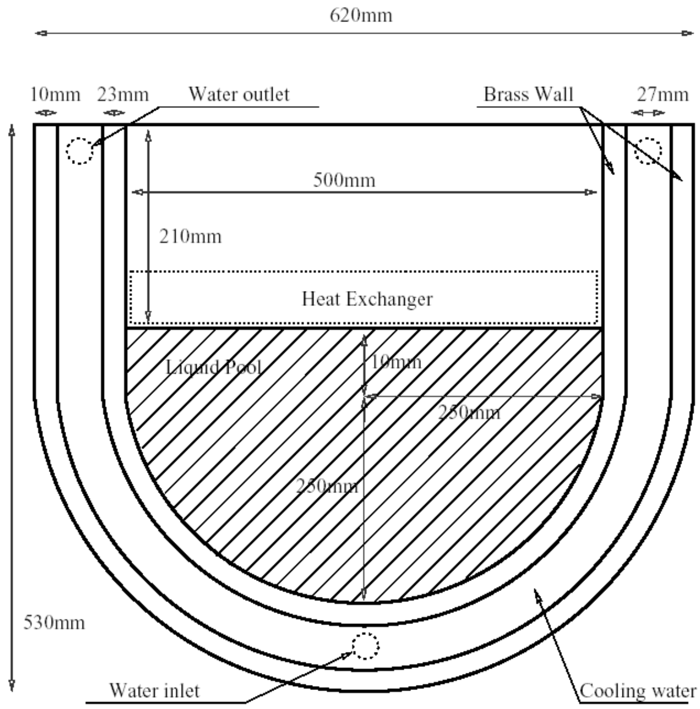

Figure 20 The SIMECO Test Facility at KTH

The SIMECO experimental facility consists of a slice type vessel, which includes a semi-circular section and a vertical section, representing the lower head of the reactor vessel. The size of the facility is scaled to be 1/8 of prototype PWR type reactors. The vessel's sidewall is represented by a thick brass plate, which is externally cooled by a regulated water loop. On the top of the vessel a heat exchanger with regulated water loops is employed to measure the upward heat transfer. The sideways and upward heat fluxes are measured by employing array of thermocouples at several different angular positions. Cable type heater with 3 mm in diameter and 4 m in length provides internal heating in the pool. Practically isothermal boundary conditions are provided at vessel boundaries with help of isothermal bath. A plate type heat exchanger mounted in the isothermal bath circuit to increase cooling capacity of isothermal bath. The cooling circuit has two parallel paths, one for sidewall heat exchange and other for top heat exchange. Top heat exchanger flow is established by isothermal bath inbuilt recirculation pump. Second external recirculation pump was mounted in order to establish necessary flow rate for sidewall heat exchange. A digital flowmeter measures sidewall flow and an analog flowmeter measures top heat exchanger flow.

The diameter and height of the test section are respectively 62.0 cm and 53.0 cm. The width of a slice is 9.0 cm. The front and back faces of the facility are insulated in order to decrease heat losses. Thickness of the vessel wall is 2.3 cm.

Total 64 K-type thermocouples are mounted to obtain data on sidewall heat flux, heat flux on top of pool, inlet and outlet water temperatures, as well as pool temperatures inside the vessel, and the upper heat exchanger. Location of thermocouples is shown in Figure 7.3.

After completion of experimental set-up, we performed one two-layer experiment and three 3-layer experiments with different conditions. Short notes on each are given below.

Two-layer experiments
Paraffin oil (880 kg/m^3) and water (996.1 kg/m^3) were used as simulating liquids for this experiment. Power applied was equal to 1050 W. Flow rate through sidewall was ~ 7.5 l/min, and through upper heat exchanger ~ 4.4 l/min. Thickness of upper layer (paraffin oil) was 21 cm, and thickness of the lower layer (water) was 8 cm. In this experiment only upper layer was heated. Total duration of the experiment was equal to 3400 seconds.

Three-layer experiments
The experiments were done with three immiscible layers, v.i.z. chlorobenzene (996.1 kg/m^3), water (996.1 kg/m^3), and paraffin oil (880 kg/m^3). During all three experimental sessions chlorobenzene served as a lower layer with 8 cm depth, the water layer with depth of 21 cm was stratified on it, and above the water layer, a paraffin oil layer with depth 4 cm was added. The heat generation was set on only inside the water layer, so chlobenzene and paraffin oil were unheated. Flow rate through sidewall was ~ 7.5 l/min, and through upper heat exchanger ~ 4.4 l/min.
Raleigh number was the only factor, which changed for these three cases by changing the heating power.

- Experiment at Ra = 2.3621×10^{13}
- Experiment at Ra = 1.9133×10^{13}
- Experiment at Ra = 1.5117×10^{13}

FOREVER FACILITY AT KTH IN SWEDEN

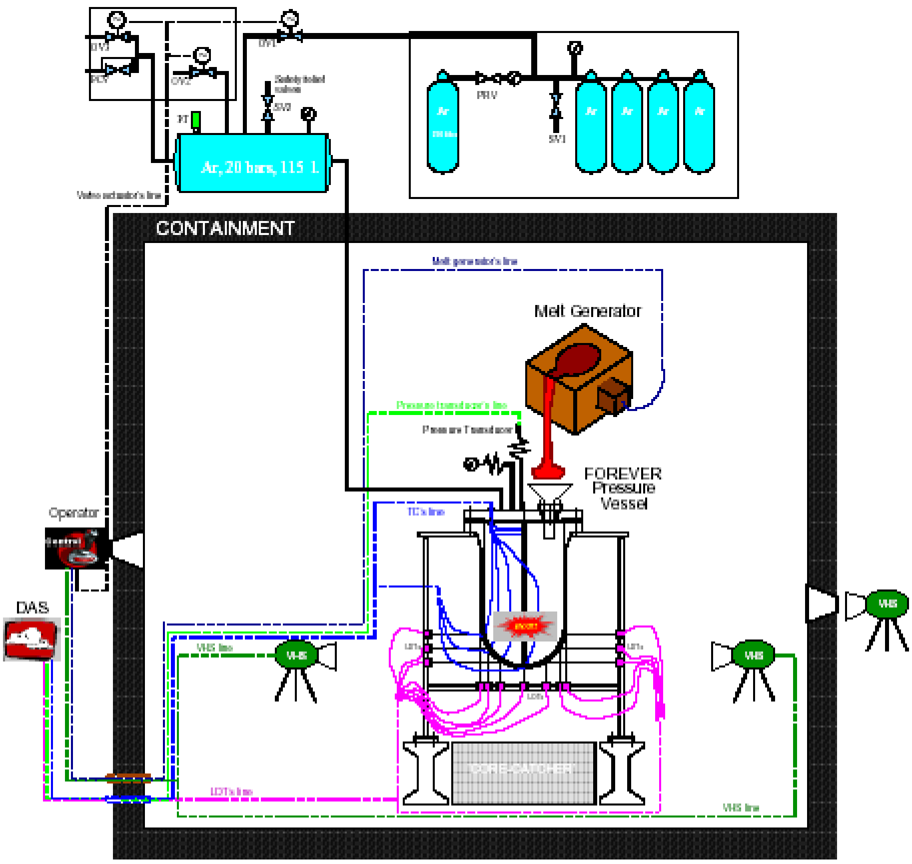

Figure 21 The FOREVER Test Facility at KTH

The EC-FOREVER-4 experiment is the fifth in the new series of experiments on In-Vessel-Melt-Retention (IVMR) strategy, supported by the ARVI (Assessment of Reactor Vessel Integrity) project in the EC fifth Framework Program. It is also the seventh of the FOREVER (Failure Of REactor VEssel Retention) tests performed at NPS/KTH in Sweden. It is an integral test to investigate the creep failure of a 1:10 scale reactor pressure vessel under the combined thermal and pressure loadings. This test simulates the late stages of the in-vessel melt progression in reactor severe accident scenarios.
The main objective of the EC-FOREVER test is to obtain multi-axial creep deformation and vessel failure data for the scaled reactor vessel geometry under prototypical thermal and pressure loading conditions. The distinguishing feature of this test in comparison to the LHF and OLHF tests (performed at SNL, USA) is the high temperature conditions in the vessel (950-1100^{0}C) as compared to ~ 800ºC in all the LHF tests; and medium pressure, prototypic, loadings (2.5 MPa) as compared to 5 ~ 10 MPa in all the LHF tests.

The facility employs a 1/10th scaled lower head (hemispherical in shape and made of SA533B, American reactor steel) of 400 mm outer diameter and 15 mm wall thickness. A cylindrical shell of 15Mo3-German steel, of 400 mm height and thickness of 15 mm, was welded to hemispherical lower head to make a complete vessel. A high temperature oxide melt (T_{liq} ≈1027 ºC) made of 30 wt%CaO-70 wt%B_2O_3 was melted in an inductively heated (50 kW max) SiC crucible. A custom made $MOSi_2$ (45 kW max) electrical heater was employed to heat up the melt in the vessel and maintain a melt pool temperature of ~1300 ^oC. The corresponding external wall temperature (based on pre-test calculations ANSYS) should be in the range 950~1000 ºC. The vessel was pressurised to about 2.5 MPa using an argon reservoir in the EC-FOREVER-4 test.

The measurements were made with 34 thermocouples, 22 LPTs (Linear Position Transducer) and one pressure transducer. The thermocouples were located at 7 angular positions on either side of the vessel wall, both inside and outside, and also, at 6 different locations along the centreline in the melt pool. There were 5 LPT on similar angular positions on either side of outer vessel wall and one LPT at very bottom of the outer vessel wall, to measure the displacement due to initial thermal expansion and creep. Thin walled (~1 mm) steel tubes, 4 mm diameter, were used to protect the internal thermocouples from corrosive action of the melt. A KANTHAL tube of diameter 10 mm was employed to protect the centreline thermocouples.

The test conditions can be summarised as follows:

- Melt Volume ≈ 12 litres.
- Power input in the melt ≈ 38 kW.
- Maximum temperature in the melt pool ≈ 1300 ºC.
- Maximum external wall temperature ≈ 950 ºC.
- Internal pressure load ≈ 2.5 MPa

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Affiliation(s)	Royal Institute of Technology, Department of Energy Technology, Sweden
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Abstract

Reactor safety's concern with severe accidents, since the TMI-2 accident led to almost twenty years of intense research efforts, which have resolved a number of severe accident issues. Lately, research has been concentrated on accident management and a number of LWR plants, around the World, have adopted severe accident guidelines (SAMGs) and strategies.

In NKS, the safety advancements expected from the planned research work in the DELI-MELT Project includes (a) an assessment of the adequacy of the accident management schemes adopted currently for Nordic BWRs and PWRs, with respect to melt coolability, accident stabilization and basemat melt-through, (b) evaluation of the reasons for low explosivity of corium, (c) datebase and prediction methodology for lower head failure mode and timing, and (d) resolution of new issues (e.g. melt stratification).

This report mainly consists of three chapters, the assessment of severe accidents, the remaining, unresolved issues of severe accidents and proposed research efforts to resolve these issues. This report reviews the state of the art of the various melt/debris coolability situations and ex-vessel steam explosions during the postulated severe accident scenarios, addresses the unresolved issues concerning the core melt loadings during the severe accidents, and further suggests the experimental facilities in the Nordic countries which could be potentially useful to resolve the issues. We believe that these issues in the order of priority are;

- in-vessel coolability of the melt pool or particulate debris,
- ex-vessel coolability of the melt pool or particulate debris,
- energetics and fragmented debris characteristics of a steam explosion endangering the integrity of the BWR containments and
- characteristics of vessel failure.

Key words: Severe accidents, melt coolability, steam explosions, vessel failure, containment failure