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ACCOMPLISHMENTS AND CHALLENGES OF THE SEVERE ACCIDENT RESEARCH

B.R. SEHGAL
Nuclear Power Safety,
Royal Institute of Technology,
Stockholm, Sweden

Abstract

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

1. INTRODUCTION AND BACKGROUND

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

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

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

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

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

2. In-Vessel Accident Progression

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

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

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

a) the magnitude and rate of hydrogen generation,

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

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

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

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

f) the fission product chemical species.

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

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

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

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

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

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

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

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

3. Early Failure of Containment

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

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

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

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

4. Late Failure of Containment

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

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

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

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

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

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

5. Direct Containment Heating

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

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

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

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

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

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

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

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

6. MELT ATTACK ON BWR MARK 1 CONTAINMENT LINER

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

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

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

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

7. HYDROGEN COMBUSTION

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

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

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

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

8. IN-VESSEL AND EX-VESSEL STEAM EXPLOSION

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

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

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

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

9. MELT DEBRIS COOLABILITY

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

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

10. NEW REACTOR DESIGNS

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

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

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

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

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

11. CONCLUSIONS

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

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