OVERVIEW: UNCERTAINTIES REMAINING IN SEVERE ACCIDENT PHENOMENOLOGY

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ABSTRACT

Severe accidents have been considered since the beginning of commercial nuclear power. Initially, only the possible consequences of a severe accident were assessed. With the increased understanding derived through critical thought, fundamental experiments, integral experiments and industry experience, this has evolved into a discipline where issues are considered on design specific and accident sequence specific bases.

Fundamentally there are two reasons for pursuing an understanding of severe accidents.

1. To evaluate the response of a given reactor/containment design for a spectrum of severe accident conditions to determine whether design modifications are warranted. This could be for either an operating plant or a future design and the modifications could relate to accident prevention and/or mitigation of the consequences.

2. To understand the nature of severe accidents, how they progress and how they can be stopped. Such knowledge is generally used to determine the most effective actions for an accident situation as well as to understand what actions are sufficient.

With the above areas, it is important to periodically step back from the phenomenological details and examine where the nuclear community (the industry and the regulators) stand with respect to the two major uses of this knowledge. Only through such periodic assessments can the community focus the limited resources towards addressing the remaining issues. This is one individual's assessment of where we stand.

1.0 BACKGROUND

Nuclear safety assessments began with the WASH-740 report in 1957 (AEC, 1957). As is commonly known, this study considered the hypothetical consequences of a nuclear accident and had no concept of the reactor design, the accident sequence, or the containment building. WASH 1400 (NRC, 1975), the Reactor Safety Study, was a more in-depth investigation which addressed both BWR and PWR designs, the containments associated with the reference designs and a spectrum of accident sequences. This investigation demonstrated that severe accident understanding must consider the specific Nuclear Steam Supply System (NSSS) design, the specific containment design and a spectrum of accident sequences since these influence the system operation, the timing of major events and corrective operator actions. A fundamental conclusion arising out of the reactor safety study was that the Design Basis Accident (DBA) concept, while a powerful concept in developing designs with substantial margin, does not address all of the issues related to severe accidents. In fact, this investigation concluded that other accidents were more likely and potentially more challenging to the reactor system and containment. Needless to say, the conclusions of the reactor safety study were controversial and in some areas highly criticized and discredited (Lewis et al., 1978).

As the WASH-1400 controversy quieted, the accident happened at Three Mile Island Unit 2. Many of the accident attributes had been described in WASH 1400. For example, it was a small break LOCA and not a DBA event, the interactions between the operator training, the automatic systems and the instrumentation resulted in the confusion with respect to what was happening in the reactor coolant system (RCS). Once it was clear that an accident was occurring there was further confusion on what actions to take. Public confusion increased as the media speculated on phenomenological issues such as in-vessel steam explosions, secondary recriticalities, hydrogen burns within the reactor coolant system, etc. with no understanding of what they were describing. Hardly the environment for rational thought.

After the TMI-2 event, the NRC raised concerns with possible severe accidents at high population sites, specifically Zion, Indian Point and Limerick. As a format for rationally evaluating the risk associated with these sites, the utilities operating each site performed a plant specific probabilistic risk assessment (CECo, 1981; PASNY/ConEd, 1982; PECo, 1982). With these plant specific studies came the assessment of physical processes on a plant specific basis. This was the first time that attention was devoted to plant specific design features that could influence phenomena such as steam explosions, in-vessel debris cooling, reactor pressure vessel (RPV) failure, high pressure melt ejection (HPME), and ex-vessel debris coolability. In the absence of an analytical model that would express the progression of phenomena, these studies used containment event trees to describe the interaction of physical phenomena on a plant specific and sequence specific basis. Such considerations led the way to more rational thought, experiments and analyses on the mutual influence of these phenomena. These studies were followed by other plant specific evaluations and the industry sponsored IDCOR program directed at providing an adequate technical basis for assessing severe accident behavior.

Entering into integral assessments of severe accidents for different designs was not without pain. Considerations of sequence specific behaviors led to the conclusion that a reactor vessel may fail while at substantial pressure. As a result, considerations were raised with respect to whether the debris could be transported within the containment and the subsequent consequences. This led to considerations of Direct Containment Heating (DCH) for the containment, thereby raising a new issue that must be addressed by the NRC since this was speculated as potentially challenging containment integrity shortly after vessel failure. For BWRs, a similar issue arose with respect to the possible failure of a Mark I liner as debris would be discharged from the reactor vessel (NRC, 1985). In this evaluation, NRC contractors concluded that molten core debris discharged from the reactor vessel could flow out of the pedestal region, across the drywell floor, contact the containment liner and melt through this liner into the gap between the liner and the biological shield. This was speculated to then cause a release of fission products to the environment. Furthermore, it was concluded that this thermal attack could occur even if there was substantial water on the drywell floor.

At the conclusion of the IDCOR program, the industry assessment was that the likelihood of a severe accident at U.S. nuclear plants was low, the public risk associated with the operation of such plants was low and that the specific risk associated with the given plant was dependent upon the plant design, its operation and the training of the operating staff. In 1988, the NRC issued Generic Letter 88-20 requesting all operators of nuclear power plants to perform an individual plant examination searching for vulnerabilities to severe accidents (NRC, 1988). In this generic letter the operators were requested to assess both the likelihood that an accident would occur and the severe accident issues that would specifically affect the containment performance; sometimes designated as the back-end analysis. Through this assessment, individual utilities evaluated severe accident phenomena for their specific design(s).

In some cases, EPRI and/or individual utilities performed their own experiments to improve the understanding of specific phenomena. For example, challenges to the Mark I liner integrity after RPV failure (Malinovic et al., 1989), DCH (Henry et al., 1991) and RPV external cooling (Henry et al., 1993). In addition, the NRC initiated its own update of severe accident risk using five reference plants (NRC, 1990). In NUREG-1150, the NRC and contractor laboratories concluded that the likelihood of having a severe accident at U.S. plants was low and that the risk to public health and safety associated with severe accidents was also low in comparison with other risks to which the public is exposed. Furthermore, NUREG-1150 concluded that the NSSS and the containment design would have a substantial influence on many of the phenomena.

Lastly, the nuclear industry, through EPRI initiated a study to provide the technical basis for developing severe accident management guidelines (SAMGs) (EPRI, 1992). As part of this Technical Basis Report (TBR), the status of the severe accident phenomena knowledge base, and its relationship to Accident Management (AM) considerations were summarized. Where appropriate the summaries of each phenomenon was integrated with a set of candidate high level actions that could be used to recover from a severe accident. This provided a perspective on how a specific phenomena would be influenced by one or more of these actions. An update of this assessment is given at this meeting (Chexal, 1995). Subsequent to the TBR, significant scale experiments have been performed to further investigate some of the phenomena addressed in previous studies. These are considered in the next section.

2.0 WHERE DO WE STAND AND WHY?

As briefly outlined in Section 1, various phenomena have been considered in different severe accident evaluations. In numerous cases, these phenomena have been considered as being of sufficient importance that they substantially influenced entire evaluations. Table 1 provides a list of some of the phenomena that have influenced particular studies. As noted in this table, several of these phenomena have either been resolved, or are approaching resolution. In this section we will briefly consider these and how resolution has been achieved or is being structured. To provide an assessment on physical phenomena, we first need to develop a list of pertinent phenomena for severe accident analyses.

Table 1
Phenomena Which Have Influenced the Uncertainty "Pictures"

Phenomena	Studies	
In-vessel steam explosions.*	WASH-1400	1975
Debris coolability.	Zion Study	1981
Debris dispersal (DCH).*	Zion Study NUREG/1150	1981
In-vessel natural circulation.*	EPRI/IDCOR	1985
Mark I liner.*	IDCOR/NRC NUREG-111500	1988
Hot leg creep rupture.*	EPRI/NRC	1988
External RPV cooling.*	FAI/CECo/UCSB	1989
Steam inerting.	IDCOR/EPRI/NRC	1988
Ex-vessel cooling.	EPRI/NRC	1988
In-vessel cooling.	FAI/INEL	1993

* Resolved or approaching resolution.

It is well known that fission products can be released from the fuel matrix as a result of overheating that would occur during an accident. As a defense against this, the containment is designed to be isolated and contain essentially all the fission products. Hence, the uncertainties associated with fission product behavior are of little concern as long as the containment remains isolated, which depends on the thermal hydraulic behavior associated with the accident progression and recovery from the accident state. Therefore, here we focus on the uncertainties associated with thermal hydraulic issues in severe accidents. Table 2 is a general categorization of thermal hydraulic phenomena. Several of these are a combination of other smaller phenomena, but this characterization serves the purpose for this evaluation.

Table 2 Severe Accident Physical Phenomena	
Phenomena	
1.	Clad oxidation.
2.	Core melt relocation.
3.	Molten pool in core.
4.	Crust formation and failure.
5.	RCS failure modes.
6.	In-vessel steam explosion.
7.	In-vessel steam generation
8.	In-vessel debris formation.
9.	RPV failure models.
10.	In-vessel cooling mechanism(s).
11.	RPV external cooling.
12.	Ex-vessel steam explosion.
13.	Direct containment heating.
14.	Mark I liner attack.
15.	Ex-vessel debris cooling.
16.	Steam inerting of the containment.
17.	Hydrogen burning in containment.

Clad Oxidation, Core Melt Relocation, Molten Pool and Crust Behavior

These have an extensive basis from experiments at various scales, the TMI-2 experience and numerous system calculations using RELAP/SCDAP and the MAAP codes. Assessments for a large variety of plants and accident sequences show that the extent of hydrogen produced generally lies within the range of 30% to 75% of the active cladding being oxidized. Hence, the direction given by the NRC following the TMI-2 accident with respect to the extent of cladding oxidation are still appropriate. Furthermore, the experience in the TMI-2 vessel investigation project (Wolf and Rempe, 1993), as well as the integral investigations of RELAP/SCDAP and the MAAP4 codes is that much of the metallic Zircaloy relocates to the lower regions of the reactor core and the material which drains into the RPV lower head is principally oxidic, i.e. UO_2 and ZrO_2. This suggests that a substantial fraction of the zirconium would result in a blockage of the lower core region with little oxidation potential and, while this could be oxidized during ex-vessel core-concrete attack, it would not be substantially oxidized in the RCS. This further supports the direction given by the NRC that individual designs must consider the oxidation of 75% of the active cladding for severe accident conditions.

RCS Failure Modes

RCS failure modes have been a relatively recent addition to the list of phenomena and have evolved from the considerations of natural circulation within the primary system. In particular, this has focused on the potential failure of hot legs in PWR designs for conditions where the reactor core is uncovered for a substantial interval. Overheating of the hot legs, and potentially the steam generator tubes for the inverted U-tube designs, has been the subject of a detailed scaled experiment (Stewart et al., 1986) and numerous analyses with the NRC (RELAP/SCDAP) and industry (MAAP4) integral system codes. All of these conclude that natural circulation would be established between an overheated core and the upper plenum region and additional natural circulation circuits would result in the hot legs and within the steam generator tubes. Furthermore, if injection to the reactor coolant system is not recovered, creep rupture of one of the hot legs (or perhaps the surge line) would occur well before the steam generator tube integrity would be challenged. Given the substantial differences in approach taken by the analyses and the existence of scaled experiments, there is sufficient information available to conclude that this material creep behavior would occur and this should be considered in developing the SAMGs. Hydrogen releases from the RCS depend on where the RCS fails and other phenomena such as High Pressure Melt Ejection (HPME), depend on whether the RCS failure occurs.

In-Vessel Steam Generation, Steam Explosions and Debris Formation

With the TMI-2 experience it is clear that debris drainage into the RPV lower plenum must be considered even when the damaged core is completely submerged in water. It was also clear that no explosive interaction was created in the accident, which has been attributed to the elevated RCS pressure. Substantial works have been reported on the influence of pressure to suppress explosive interactions (Henry and Fauske, 1979; Hohmann et al., 1979; Hohmann et al., 1982). Recent experiments performed in the FARO facility (Magallon and Hohmann, 1993) and the ALPHA test in Japan (Yamano et al., 1993) have further supported the influence of elevated pressure with the latter experiments showing that a pressure of 1.6 MPa is sufficient to suppress an explosive interaction. (This information is consistent with that presented by the previous experimental programs.)

Other experimental and analytical studies have focused on the potential for fine scale particulation and mixing of large quantities of high temperature melt and water (Angelini et al., 1993 and Fletcher and Denham, 1993). Both investigations concluded that it is extremely difficult to mix large quantities of high temperature melt with water and also that substantial steam is formed during the premixing which depletes the water in the interaction zone. This is a more refined assessment of the mechanism proposed by Henry and Fauske (1981) which suggests that vapor formation during the premixing limits the molten material involved.

In 1985, the NRC formed the Steam Explosion Review Group (SERG) (NRC, 1985) and chartered this group to assess the likelihood of α-mode failure. The consensus was that explosive interactions sufficient to rupture the primary system and therefore the containment were very unlikely. During the 1993 CSNI-FCI Special Meeting in Santa Barbara, similar questions were asked, and again, the consensus was that the work performed since the 1985 meeting supported the conclusions made by the SERG. In many instances, additional work had further refined key arguments related to the inability to establish the necessary initial conditions. In June of 1995, the NRC is sponsoring a workshop to update the understanding with respect to steam explosions. With the additional experiments provided in the FARO (Magallon and Hohmann, 1993) and ALPHA facilities, as well as the additional analyses performed by Angelini et al. (1993) as well as Fletcher and Denham (1993) it would appear that there is a developing consensus on the α-mode failure issue.

Debris particulation and in-vessel steam generation are also part of the understanding related to steam explosions. In particular, significant work has been performed with respect to the breakup of molten jets as they pour through water (Berger et al., 1993). Moreover, the FARO experiments provide a substantial scale, real material demonstration of the steaming rate during this process. From the information accumulated to date, including the TMI-2 Vessel Inspection Project (VIP) (Wolf and Rempe, 1993), some particulation may occur, but the assessment of debris within the lower plenum must also consider that there is a substantial material layer which does not particulate. Particulate debris causes a net steam generation to the RCS with some potential for additional pressurization as was observed in the TMI-2 accident. These integral system details are part of the lower plenum modeling in the MAAP4 code (EPRI, 1994). With the extensive information available on debris particulation and the net steaming rate from molten material draining into the lower plenum, the major issue is how much material is not particulated since this results in a potential threat to the RPV wall integrity.

RPV Failure Modes

RPV failure modes were initially considered in the Zion Probabilistic Safety Study (CECo, 1981). In this study, molten core debris draining into the lower plenum was postulated to challenge the limited depth welds that anchor the in-core penetrations in the RPV. Similar considerations were used for the BWR evaluation of the Limerick plant (PECo, 1982). As the core removal progressed in the TMI-2 vessel and more was learned of the accident scenario, including the drainage of molten core material into the lower plenum, it became apparent that these welds were more robust than was previously credited. Since this is a generic issue, the details of how the vessel would fail are not particularly important in the IPEs which searched for plant specific vulnerabilities. However, for AM evaluations, the time available to recover from an accident and keep the core debris in the RPV, become of key importance. In particular, retaining debris within the RPV lower head eliminates the uncertainties associated with ex-vessel debris behavior and therefore minimizes the uncertainties to be considered by the AM team. EPRI sponsored full scale experiments on the challenge to in-core penetrations by molten material, with particular emphasis on the possibility that molten debris could flow through the central passage used by the traveling in-core probe (TIP) (Hammersley and Henry, 1994). These experiments revealed that the in-core penetrations experience molten debris traveling through the TIP passage but that this material quickly froze and plugged this flow path so completely that there was no further depressurization of the simulated RCS. Moreover, there was no challenge to the supporting weld for the penetration and analyses indicated that these welds would have to be essentially melted before the penetration could be ejected. Hence, these experiments provided the fundamental insights to create mechanistic models for the MAAP code to represent this lower plenum behavior. These models show that the penetration behavior is far removed from a failure condition when molten debris drains into the lower head, even if the RCS pressure is at the nominal operating values.

Other experiments were performed on the 5 cm (2 in.) water filled drain lines that are in some BWR vessels. Two experiments were performed and there was no indication of significant strain in the drain line even though the entire line was filled with molten oxidic material. In fact, these experiments showed a considerable potential for creating a significant contact resistance between the debris and the drain line which influenced the energy transfer from the molten material to the wall. Other experiments were carried out with molten material draining into a dry lower plenum without penetrations and also with water in the lower plenum. These tests also demonstrated the formation of an interfacial contact resistance if the molten debris drains through water. This was the foundation for this model in the MAAP4 code and also provides insights into the possibility of in-vessel cooling that is discussed below.

As a result of the experiments on RPV failure modes, it is clear that the lower head penetrations are not nearly as susceptible to failure as was once considered. In fact, there appears to be virtually no potential for failing a penetration immediately after core debris would drain into the lower head. If the core debris accumulates in the lower head, dries out, heats the reactor vessel wall sufficient to cause extensive strain of the wall, then there is a potential that the penetrations could be the failure site. This mostly depends on the accident sequence and is certainly dependent on whether water has been added to the reactor coolant system or whether external cooling of the RPV is developed. There is a potential for a much higher heat flux toward the equator of the hemisphere, however, the accident sequence generally dictates that molten debris arrives in the bottom of the reactor vessel long before the melt accumulates sufficiently to approach filling the RPV lower head. As a result, there are uncertainties in where vessel failure would occur if the RPV head is dry, but it is clear that there would be no immediate failure as a result of debris entering the lower plenum.

In-Vessel Cooling

As part of the TMI-2 core removal, there was some excellent detective work performed on the lower head, to assess the thermal response. Consideration of this thermal-mechanical response has resulted in a proposed mechanism (Henry and Dube, 1994) for in-vessel cooling which characterizes the cooling of the RPV wall with water that is in the RCS. In this mechanism, limited wall strain, of the order of a few hundred microns, is sufficient to enable water to ingress between the RPV wall and the debris to cool the wall and prevent further strain, and therefore RPV failure. Hence, there is an important feedback between the core overheating the RPV wall, straining of the wall and water ingression into the small gap between the debris and the wall. It is important to note that the extent of strain is very small compared to that which would potentially threaten the RPV integrity. It is also important to note that the RPV lower head will not fail without substantial strain. This model has been added to the MAAP lower plenum models including material creep in the presence of a strong radial temperature gradient in the carbon steel wall. Benchmarking with the TMI-2 accident gives a consistent picture with molten debris draining into the lower head, substantial overheating of the wall and cooling of the RPV wall when limited strain occurs. While there are other possible explanations, it is important to note that substantial strain must occur before failure and integral code calculations must consider such strain, and its implications, before assessing the potential for RPV failure.

This is a relatively new mechanism but it also has substantial importance to the accident management assessments because there is an obvious benefit to keeping core debris within the reactor vessel. This can be done by either cooling the debris within the RPV, external cooling of the RPV, or both. Certainly adding water to the RCS is the appropriate action when injection is available. For many reactor containments, external cooling is accomplished in a relatively easy manner. On the other hand, there are also numerous designs in which it is difficult to flood the containment and it may also be difficult to cool the RPV lower head because of a vessel support skirt. In these cases, flooding the containment under conditions in which the ECCS injection has been restored may not be an appropriate action. Containment flooding is difficult and in many cases eliminates some of the important features, such as pressure suppression for the Mark I and Mark II designs, as well as potentially flooding key instruments. Therefore while in-vessel cooling can not be considered as resolved because it is so new, it is important to perform the necessary work to understand this cooling potential, because of the importance to accident management decisions.

RPV External Cooling

External cooling was discussed extensively in the TBR, both in terms of the physical processes involved and the design specific features that could limit this cooling. At the time, there was only limited experimental data, but this data showed a substantial margin between the surface heat removal required to maintain RPV integrity and that which would cause a boiling crises. Since then additional experimental works have been reported to further support that external cooling is a viable mechanism for preventing RPV failure as long as water has access to the RPV wall. In particular, the two dimensional critical heat flux experiments by Theofanous et al. (1994) showed the heat removal capabilities to be far in excess of the energy transfer that would be expected under accident conditions. In addition large scale three-dimensional experiments were performed at Sandia National Laboratory on a vessel with an elliptical lower head. With this particular head design, the larger radius of curvature provided a conservative representation of the capabilities for boiling heat transfer on the downward facing surface (Chu et al., 1994a and Chu et al., 1994b). Both of these experiments further supported the capabilities for RPV external cooling. Hammersley et al. (1993) have performed experiments for vessel with support skirts and have shown that this can preclude water from contacting the RPV lower head. With the various experiments that have been performed, this issue can be considered as resolved for accident management behavior. Of course the plant specific features, such as a vessel support skirt, must be considered before external cooling is credited.

Direct Containment Heating and Ex-Vessel Steam Explosions

These are considered together since much of the experimental work on ex-vessel steam explosions comes from the DCH experiments. In particular, a number of experiments have been performed with water in the simulated reactor cavity. These have measured significant dynamic pressurizations of the reactor cavity but these interactions have not led to any other consequences other than expelling the debris from the reactor cavity and producing steam that could oxidize metal in the molten debris.

As part of the Zion IPE study, Commonwealth Edison sponsored experiments to investigate the DCH potential in a 1/20th linear scale representation of the Zion containment (Henry et al., 1991). At about the same time the NRC initiated a scaling methodology committee, chaired by Dr. Novak Zuber, to address scaling issues for severe accidents using direct containment heating to both develop the methodology and demonstrate its usefulness (Zuber, 1993). Linear scaling was judged as the most appropriate means of addressing DCH and the NRC initiated programs at two different scales at Argonne National Laboratory (1/40th) and Sandia (1/10th) for Zion-like systems. These tests provided an excellent application of the proposed methodology and experimentally provided direct insight into scale related issues. These experiments (Binder et al., 1994 and Allen et al., 1994) showed remarkable similarity in the containment dynamic behavior for counterpart experiments. Hence, these demonstrated confidence in the scaling methodology as well as that the containment pressurization exhibited was much less than that which would challenge the containment integrity. Specifically, the net pressure increase for a substantial inventory of melt ejected from the cavity was approximately 1 bar for an inerted containment and 2.5 bars if hydrogen combustion occurred. Further it was noted that pre-existing hydrogen in the containment atmosphere was not burned on a timescale that was meaningful with respect to the containment pressurization. Hence, the hydrogen that was burned during the pressurization was that created by oxidizing high temperature metals during the transport through the reactor cavity and the steam generator region. With the development of the scaling methodology and the successful completion of the Zion-like experiments at two different scales and the resulting pressure increases that are much less than those that would challenge the containment integrity, this issue has been closed and documented in Pilch et al. (1994).

A similar set of experiments were performed at 1/10th and 1/6th scale mockups of the Surry containment in facilities at the Sandia National Laboratory (Blanchat et al., 1994). Containment compartmentalization had a substantial influence on the pressurization that could occur within the containment as a result of high pressure melt ejection. Here again the pressure increase in containment was approximately 2.5 bars for conditions in which hydrogen combustion could occur but the hydrogen burned was essentially created during the event. Hydrogen that was previously in the containment atmosphere did not appear to be consumed on a timescale that significantly influenced containment pressurization. With the successful completion of the test program at the two different scales and the examination of the conditions in the RCS that could result in depressurization due to hot leg creep rupture, the DCH issue for Surry-like containments has also been closed. This has recently been documented for the NRC by Sandia National Laboratory personnel (Pilch et al., 1995).

The IDCOR program (IDCOR, 1985) identified several types of reactor cavity configurations. Because of these variations, some additional experiments are underway at Sandia National Laboratory to address these differences. In particular, the extensive work done on issues related to hot leg creep rupture have demonstrated that an uncovered reactor core would lead to hot leg creep rupture and depressurization of the reactor coolant system before any high temperature molten material would drain into the lower plenum. Consequently, the only conditions that would be considered as realistic for an HPME event would be those in which the reactor core would be covered by water, as was the case during the TMI-2 accident. Hence, for these conditions high temperature core debris would be pushed out of the reactor coolant system by saturated water. Note that this set of conditions conflicts with the in-vessel cooling mechanism that suggests that vessel failure would not occur if water were present within the reactor system well before vessel strain were to occur. Hence, this set of conditions may not exist, but are being considered a part of closure for the DCH issue.

Mark I Liner Attack

This issue was raised by the NRC Containment Loads Working Group (NRC, 1985) and was addressed by significant scale experiments (Malinovic et al., 1989). Issue resolution was addressed by the NRC through a structured analytical and experimental program (Theofanous et al., 1990). In this approach, a general characterization of the conditions which could possibly challenge the Mark I liner were formulated and evaluated in a probabilistic structure using Peach Bottom as a reference plant. In the final analysis, the assessment concludes that if there is no water in the drywell at the time that core debris is released from the reactor vessel, there is a high probability that the containment liner could be attacked. Conversely, if there is water in the drywell at this time, attack of the Mark I liner is not physically credible. This approach to resolution of the Mark I liner issue was submitted to a peer review committee. Numerous comments were provided to the authors and additional, separate analyses were performed to evaluate and address the various comments. Eventually technical closure was developed on all comments and the Mark I containment liner issues is resolved.

Ex-Vessel Cooling

Ex-vessel debris cooling has been the subject of substantial discussion since the Zion Probabilistic Safety Study. Experiments have been performed, such as those discussed above for the Mark I liner attack, to obtain a perspective on the rate of cooling resulting from water ingression. There are substantial scale issues associated with ex-vessel cooling, including the properties of the core debris, how these physical properties could be altered as molten concrete is added to the debris as thermal attack progresses. Significant scale experiments have been performed and are discussed in the literature (Epstein, 1992) which demonstrate the possibility that there is a significant heat removal from core material to the ex-vessel mode. However, these studies have not been able to clearly identify a long term cooling mode. Furthermore, performing such experimental investigations are difficult at best, and must be performed at a significant scale, which is judged to be at least 1 meter by 1 meter, i.e. substantial quantities of uranium dioxide.

Because of the experimental difficulties and the substantial scale related issues associated with water ingression, the details of ex-vessel coolability may never be clearly understood. On the other hand, with respect to accident management actions, a detailed understanding may not be required. In all cases, the discharge of core debris from the RCS into containment requires that the debris be submerged in water for several reasons. Firstly, water can cool the debris by ingression. Secondly, submerging the core debris would scrub fission products that could be released as a result of core-concrete attack even if the debris were not coolable. Lastly, submerging the debris eliminates any substantial heat load from the core material to other containment structures that are potentially sensitive to elevated temperatures. For all these reasons, it is clear that water should be added to the containment if it is suspected that core debris has been lost from the RPV, regardless of the cooling capacity of the water itself. Hence, while a more detailed perspective of ex-vessel cooling is desired, there is enough known from current information to support the necessary decision making for accident management.

Hydrogen Burning and Steam Inerting in the Containment Atmosphere

These two are combined for obvious reasons. Substantial investigations have been performed, as well a large scale experiments (Thompson et al., 1988a and Thompson et al., 1988b), and with these, the importance of hydrogen combustion in the atmosphere is understood as are the influences of accident consequences such as steam inerting. It is important that AM decision making appreciates the possibility of de-inerting the containment atmosphere as a result of a candidate high level action, i.e. containment sprays. In particular, containment sprays both de-inert the atmosphere and increase the turbulence which can increase the burning rate (Thompson et al., 1988a). Also, the rate at which the atmosphere is de-inerted can be important since the retention of a significant steam partial pressure tends to limit the burning rate. Thus, while this is important for accident management evaluations, the existing knowledge base appears to be sufficient.

In summary, most of the elements related to the understanding of severe accidents have a sufficient knowledge base. Table 3 summarizes the above discussions on whether these major phenomena are sufficiently understood to be considered as resolved, whether they are approaching resolution, or whether more work is needed to achieve the necessary understanding to support AM decision making.

3.0 WHAT IS LEFT TO DO?

As indicated in Table 3, the understanding for most phenomena is sufficient to support accident management decision making, and more specifically the development of SAMGs. Only in limited cases are there areas where this understanding could be further refined and these principally relate to stopping the accident progression by cooling the core debris. The focus of these activities can be determined by considering the key events for an accident sequence and why they are so important (see Table 4). As demonstrated, the last two focus on stopping the accident sequence and preventing the release of fission products from the containment. By keeping the debris within the RPV, the uncertainties associated with ex-vessel cooling and other containment issues are essentially eliminated.

Cooling within the RPV can be done by either water injected to the RCS, external RPV cooling, or both. While each has its own uncertainties, those associated with external cooling are substantially less because of the larger experimental database. However, some reactor designs do not permit effective cooling of the RPV lower head and for others, achieving this state is very difficult. Therefore, to provide a necessary basis for accident management decision making, two areas would be particularly helpful to achieve closure. The first is to provide the technical depth

Table 3 Resolution Status with Respect to Accident Management Evaluations				
	Phenomena	**Resolved**	**Virtually Resolved**	**More Work Is Needed**
1.	Clad oxidation.	√		
2.	Core melt relocation.	√		
3.	Molten pool in core.	√		
4.	Crust formation and failure.	√		
5.	RCS failure modes.	√		
6.	In-vessel steam explosion.	√		
7.	In-vessel steam generation.	√		
8.	In-vessel debris formation.	√		
9.	RPV failure modes.		√	
10.	In-vessel cooling mechanism(s).	√		√
11.	RPV external cooling.		√	
12.	Ex-vessel steam explosion.	√		
13.	Direct containment heating.		√	
14.	Mark I liner attack.	√		
15.	Ex-vessel debris cooling.		?*	?*
16.	Steam inerting of the containment.		√	
17.	Hydrogen burning in containment.		√	

* Enough is likely known for AM evaluations even though substantial uncertainties remain.

Table 4 Key Events In An Accident and Why They Are So Important	
Event	**Increase in Accident Severity (Consequence)**
Core is Uncovered	Core integrity is challenged
Major Core Damage	Fission products released to the containment.
RPV Failure	Core debris is discharged to the containment.
Containment Failure	Fission products released to the environment.

to the in-vessel cooling phenomena; specifically to create the necessary level of confidence that is required to clearly define when sufficient actions have been taken for cooling the core material. Secondly, the issues associated with external RPV cooling have only addressed those related to heat removal from the RPV lower head. Additional experiments showing the removal from the remainder of the RCS, including the RPV cylinder and the hot legs, would be beneficial. In particular, this could help address the issue of: if only external cooling was available, what water level would be sufficient to stop the accident progression?

As indicated by the above discussion, those issues necessary to support accident management decision making are rapidly approaching closure. Once technical closure has been achieved on the list of phenomena given in Table 2, the only remaining element is to establish effective training for the operating and technical staffs. This training should not be presented in a burdensome manner, but should be done in a streamlined fashion, which identifies the state of knowledge of all major phenomena and how these are addressed. It is important that this training be reinforced on a regular basis to (1) demonstrate that this is a living process which continues to take advantage of an increasing knowledge base from scientific studies, integral analyses and industry experience and (2) ensure that the influence of these physical processes is continually reviewed with the operating and technical staffs. In this regard, the structure of physical phenomena as discussed in this paper and as applied to the four key events would be an effective approach to cover both the basics of accident management as well as to touch on the more esoteric elements of individual phenomenon. With this background and training, the operating and technical staffs have the wherewithal to address the two reasons for understanding severe accidents:

1. to determine whether design modifications are warranted, and

2. to understand the nature of severe accidents, how they progress and how they can be stopped.

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