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IN-VESSEL RETENTION - RECENT EFFORTS AND FUTURE NEEDS

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KEY WORDS

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ABSTRACT

In-vessel retention (IVR) of core melt is a key severe accident management strategy adopted by some operating nuclear power plants and proposed for some advanced light water reactors (ALWRs). If there were inadequate cooling during a reactor accident, a significant amount of core material could become molten and relocate to the lower head of the reactor vessel, as happened in the Three Mile Island Unit 2 (TMI-2) accident. If it is possible to ensure that the vessel head remains intact so that relocated core materials are retained within the vessel, the enhanced safety associated with these plants can reduce concerns about containment failure and associated risk. However, it is not clear that the external reactor vessel cooling (ERVC) proposed for existing and some advanced reactors would provide sufficient heat removal for higher-power reactors (up to 1400 MWe) without additional enhancements. This paper summarizes recent efforts to enhance IVR and identifies additional needs to demonstrate that there is sufficient margin for successful IVR in high power reactors.

1. INTRODUCTION

If there were inadequate cooling during a reactor accident, a significant amount of core material could become molten and relocate to the lower head of the reactor vessel, as happened in the Three Mile Island Unit 2 (TMI-2) accident. If it is possible to ensure that the lower vessel head remains intact so that relocated core materials are retained, the enhanced safety associated with these plants can reduce concerns about containment failure and associated risk. For example, the enhanced safety of the Westinghouse Advanced 600 MWe PWR (AP600), which relied upon ERVC with a flooded reactor cavity, resulted in the U.S. Nuclear Regulatory Commission (US NRC) approving the design without requiring certain conventional features common to existing LWRs.

However, it is not clear that the ERVC proposed for the AP600 could provide sufficient heat removal for higher-power reactors (up to 1500 MWe) without additional enhancements. Because many of the advanced LWRs being proposed by reactor designers (e.g., AP1000, APR1400, SWR1000, EPR) have power levels significantly higher than the AP600, additional efforts are underway to demonstrate the viability of IVR in their plants. Clearly, the ability to show that there is substantial margin associated with IVR has the potential to improve plant economics (owing to reduced regulatory requirements) and increase public acceptance (owing to reduced plant risk). This paper identifies key issues related to IVR and the status of research efforts to resolve these issues. Additional needs to demonstrate the margin associated with IVR in high-power reactors are discussed.

2. BACKGROUND

Advanced and existing light water reactors (LWRs) consider ERVC a key accident management strategy for IVR of relocated corium materials. If water covers the external surfaces of the lower head, significant energy (i.e., decay heat) could be removed from relocated corium materials through the vessel wall by nucleate boiling on the vessel outer surface. As long as the wall heat flux from the core melt does not exceed the critical heat flux (CHF) limit for nucleate boiling on the vessel outer surface, the reactor vessel could be cooled sufficiently that it prevents vessel failure and release of corium materials to the containment. However, previous licensing interactions have identified several key issues that must be addressed in order to assure ERVC.

2.1 ERVC

Several existing and advanced LWRs, such as the AP600 in the U.S. and the Loviisa VVER in Finland incorporated ERVC as part of their licensing basis. However, modifications were incorporated into the design of these plants to insure that they could take credit for ERVC. As discussed in Theofanous, et al., (1996), there were concerns about the ability of the AP600 insulation panels around the reactor vessel to remain in place during reactor cavity flooding. In addition, there were questions about whether the insulation design allowed sufficient water to ingress and steam to egress during cavity flooding, Hence, a revised design was developed that included a water inlet at the base of the vessel, vessel-to-insulation clearances for water and vapor flow, and structural reinforcements for the insulation. Kymäläinen (1997) describes several plant modifications that were implemented in the Loviisa plant to insure efficient natural circulation of coolant outside the reactor pressure vessel (RPV). Loviisa modifications included: increasing the minimum gap between the RPV wall and the lower neutron shield/lower head thermal insulation assembly; installing passively operated valves into the thermal insulation surrounding the RPV so that the steam-water mixture can freely escape, increasing the number of channels from the steam generator compartment into the reactor cavity; adding screening equipment in the reactor cavity to trap particles, such as debris from fibrous thermal insulation and other impurities; and installing a reactor coolant system (RCS) depressurization system to insure that the mechanical loads to the degraded vessel walls remain low and that enough water is available in the reactor cavity to submerge the RPV during relevant severe accident scenarios.

2.2 Debris Heat Loads

In the Westinghouse AP600 design certification effort, a major area of discussion was related to the assumed "bounding" debris configurations for evaluating challenges to vessel integrity. Westinghouse proposed two bounding states. The first bounding state was dominated by transient forced convection and jet impingement effects; and the second bounding state, which occurred for depressurized events, was dominated by natural convection in the final steady state. Analyses in the University of California- Santa Barbara (UCSB) report submitted by Westinghouse showed that vessel failure would not occur as a result of jet impingement. Thus, thermal loads to the vessel for the final steady state were considered bounding and analyzed in detail.

As described by Theofanous, et al., (1996), the proposed "FInal Bounding State" or "FIBS" assumes a molten, homogeneous, ceramic pool lies beneath a metallic layer (see Figure 1). The ceramic pool was assumed to contain essentially all of the core's fuel core and some oxidized cladding material. As noted above, turbulent steady-state natural convection associated with volumetric heat sources was assumed to govern heat transfer from this ceramic pool. The heat transfer on the external surface of the reactor vessel was assumed to correspond to values for a fully-flooded reactor cavity. Hence, the molten pool experienced sufficient cooling that it was surrounded by thin crusts imposing uniform temperature boundary conditions at the outer pool surface. Above this ceramic pool is a metallic layer that contains all unoxidized components within the reactor vessel that become molten during an accident (e.g., the core plate, the reflector, lower vessel internal structures, etc.). In order for the stratification assumed in this configuration to occur, gravitational forces associated with the heavier oxide material must overcome natural convection forces in relocating materials. Analyses of heat fluxes for this configuration suggest that peak heat fluxes occur near the top of the molten pool in the oxide layer and remain constant across the metallic layer. It should be noted that the presence of any fission products in the metallic layer was ignored by UCSB.

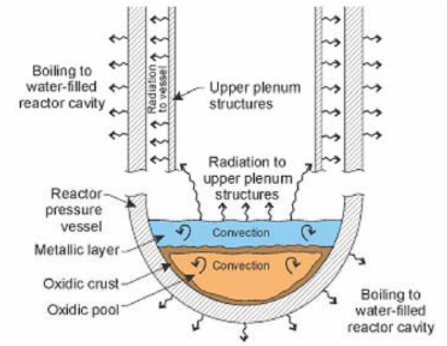


Figure 1. UCSB-assumed FIBS for the AP600. (Theofanous, et al., 1996)

In an NRC-sponsored review of the Westinghouse submittal (Rempe, 1997), it was concluded that the UCSB-assumed FIBS may not necessarily be the most bounding or plausible debris configuration. INEEL calculations suggest that peak heat fluxes, which were predicted to occur where the vessel is in contact with the metallic layer, could exceed CHF in these alternate configurations. Key differences in INEEL input assumptions included a smaller assumed metallic layer mass (as suggested by independent calculations), a reduced metallic layer emissivity, higher decay heats (associated with earlier relocation times predicted in independent calculations), and the inclusion of metallic layer heat sources due to fission product and actinide retention. In addition, several material property assumptions differed significantly from values found in the literature.

2.3 Status

As advanced LWR designers consider ERVC and the analyses that must be performed to demonstrate the viability of IVR, several factors should be considered:

- *Differences in power level.* As noted above, most advanced LWR designs proposed by U.S. vendors have higher power level ratings than the 600 MWe AP600.
- *Cavity flooding.* As noted above, it is essential that safety systems can flood the cavity before melt relocation. Hence, sufficient water must be available to rapidly flood the reactor cavity.
- *Recent experimental data confirming the potential for alternate debris configurations.* Recent RASPLAV and MASCA tests (OECD, 2004) indicate that alternate debris endstate configurations, such as some of the endstates suggested in the NRC-sponsored INEEL review of the AP600, may be possible and that fission products tend to distribute according to their chemical composition.

The above factors have led advanced reactor designers to seek methods to enhance the margin for IVR. As discussed below, several avenues are being pursued.

3. RECENT EFFORTS TO ENHANCE MARGIN

Several options are available to enhance the margin for IVR. This section describes each of these options and efforts to evaluate the magnitude of enhancement each could offer.

3.1 Narrow Gap Cooling

Experimental observations

Questions about the coolability of a continuous mass of relocated corium were raised during the TMI-2 Vessel Investigation Project (VIP). [Wolf and Rempe, 1993] Post-accident examinations indicate that nearly half of the material that relocated to the vessel lower plenum during the TMI-2 accident formed a cohesive or "continuous" layer. TMI-2 VIP results and other evidence suggest that conduction through this continuous layer of solidified corium materials was assisted by other cooling mechanisms, such as a narrow gap between relocated materials and the vessel wall.

Available information suggests that there is sufficient evidence to conclude that enhanced cooling, beyond that possible with conduction, occurs when corium materials relocate to a water-filled lower plenum of a reactor vessel during a severe accident. Prototypic material evidence provides general insights about this enhanced cooling. The TMI-2 post-accident examinations and analyses suggest that corium materials did not attack the vessel wall. In fact, nozzle examinations suggest that this material "protected" vessel and nozzle materials near the wall. Video examinations also suggest the presence of gaps and cracks within the corium materials. Analyses suggest that the vessel would have failed if enhanced cooling associated with coolant flowing through these cracks and gaps were not present. Data from experiments investigating prototypic material behavior are consistent with TMI-2 evidence. FARO tests (Magallon, et al., 1999) suggest the presence of a gap between relocated corium materials and the test plate and "furrows" and "interconnected porosity" within the solidified corium materials. FARO test data also suggest that gaps are larger in tests with water present. Similar phenomena were also observed in corium materials from the CCM-2 tests (Spencer, 1994). Unfortunately, prototypic material evidence is insufficient to estimate key parameters required to model this cooling. For example, there isn't sufficient evidence to estimate the size and density of cracks within solidified corium, the gap sizes that may form between the vessel and relocated corium, or the heat transfer from relocated corium materials to coolant flowing in these cracks or gap.

More detailed data are available from tests with simulant materials. In particular, time-dependent temperature for test vessels subjected to Al_2O_3 pours are available from Korea Atomic Energy Research Institute (KAERI) LAVA (Kim, et al., 1999) and Japan Atomic Energy Research Institute (JAERI) ALPHA (Maruyama, et al., 1999) tests. These tests also have measurements for gaps that formed between the test vessel and the solidified Al_2O_3. Although test data provide key insights about in-vessel debris coolability, they use simulant materials and no attempts were made to quantify the size of cracks and gaps present within the solidified melts as a function of time. Furthermore, material properties associated with crack and gap formation are not well known for either corium materials or aluminum oxide. Those properties that are known and observations of the debris surface in contact with the test vessel suggest that the simulant materials may behave differently than prototypic materials. Nevertheless, it is suggested that enhanced cooling models be benchmarked using simulant test data. It is also suggested that additional material properties be obtained so that differences between prototypic and simulant material behavior may be better understood.

Narrow gap heat transfer

Even if it can be shown that relocated corium experiences cracks and/or gaps during cooling, data are insufficient to predict the enhanced cooling associated with water ingress into these cracks. Likewise, data are needed to predict heat transfer in the "engineered gap" between an in-vessel core catcher and the reactor vessel wall. Of concern is the potential for CHF in a narrow gap. CHF in a narrow gap should be distinguished from CHF in a pool. At the top of the narrow gap, the small flow area restricts water ingress. Generated vapor in the gap flows upward against the penetrating water. This vapor flow is restricted at the top end of the narrow gap due to counter-current flow limitations (CCFL). A large amount of wall superheat on the surface of relocated corium and/or the RPV lower head would vaporize all of the water independent of the boiling regime. Hence, the heat flux at the surface is limited by the restricted water ingress and/or vapor egress. This heat flux limit corresponds to the latent heat of penetrating water. In other words, CHF in a narrow gap is determined by CCFL.

Kim, et al. (2004) summarizes key investigations of heat transfer in a narrow channel. Kim, et al. (2004) reports that most investigators performed experiments investigating CHF and concentrated on determining either surface orientation or gap size effects. Several experimental studies have been performed to evaluate the impact of surface orientation on CHF in pool boiling by correlating CHF data. Kim (2004) also notes that some recent investigations report the existence of a transition angle at which CHF rapidly decreases. Several correlations have been developed to predict CHF as a function of key fluid properties and geometrical parameters (e.g., fluid and vapor density, surface tension, gap size and length of channel, etc.). Recently, many researchers have started to examine heat-transfer during rewetting of hot vertical surfaces in narrow gap channels (e.g., Murase 2001). Results suggest that during the heating and cooling processes there are three key heat-transfer modes: film boiling (FB), transition boiling (TB), and nucleate boiling (NB); and two critical conditions: minimum film boiling heat flux (MFB) and CHF.

Zhang, et al. (2003) reports efforts to develop complete narrow gap heat transfer curves using data from vertical, annular channels with gaps ranging from 0.5 to 7.0 mm. Figure 2 compares results from the Zhang tests with selected correlations. In the film boiling region, Zhang tests suggest that the CCFL causes heat transfer coefficients to decrease with smaller gap sizes. For gap sizes of 1.0 mm, the boiling curve corresponds to a correlation calculated by a heat transfer correlation for vapor laminar flow (assuming a Nusselt number of 4); and for the gap size 4.0 mm, the boiling curve corresponds to values calculated with the Bromley correlation (1950) for pool boiling. For a gap size of 2.0 mm, the Zhang data are consistent with values predicted with the vapor laminar flow and the Bromley correlations. In the transition boiling region, Zhang data qualitatively correspond to values calculated with the Murase TB correlation (although this correlation overpredicts the heat transfer). In the nucleate boiling region, the Murase NB correlations agree well with the Zhang data in the low superheat and high superheat regions. For a gap size of 7.0mm, the Zhang data are consistent with values predicted by the Kutateladze (1952) NB correlation, which that the heat-transfer characteristic is close to that of pool boiling. For CHF, the Chunlin Xia (Xia, et al.,1996) correlation was used.

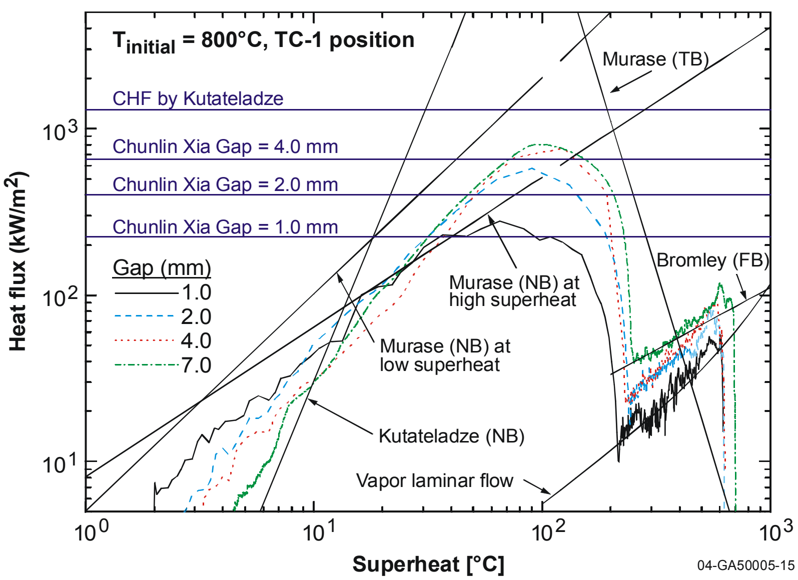


Figure 2. Comparison of Zhang data with narrow gap correlations. (Zhang, et al., 2003)

As part of a Korean and U.S. sponsored International Nuclear Energy Research Initiative (INERI), a comprehensive investigation of narrow gap heat transfer with CCFL is underway at Seoul National University (SNU) using one- and two-dimensional facilities. (Kim, et al., 2004) In the GAMMA 1D configuration, CHF and boiling data are obtained for gaps of 1, 2, 5, and 10 mm as a function of heated surface orientation at atmospheric pressure. In addition, the flow pattern during heat transfer through the narrow gap as a function of surface angle are observed using high-speed photography. For the GAMMA 1D configuration, a high resistance film heater is soldered to the underside of a copper block [26 x (26 to 46) x 5 mm^3]. The heating assembly is placed in a stainless steel housing and submerged in the test pool, as shown in Figure 3. Pyrex-glass is imbedded into the edge of the housing, maintaining gap sizes of 1 to 10 mm. The housing is rotated to observe the impact of inclination angle on heat transfer. The GAMMA 2D configuration (see Figure 4) allows SNU to investigate boiling and CHF with CCFL as a function of gap size and pressure using a two-dimensional, slice geometry test section (with radius of 250 mm). In the GAMMA 2D slice facility, CHF and boiling data are obtained for gaps of 1, 2, 5, and 10 mm as a function of heated surface orientation for pressures up to 1 MPa. As shown in Figure 4, GAMMA 2D is a closed loop with a stainless steel pressure vessel that is manufactured to provide gap sizes of 1 to 10 mm between the copper shell and the pressure vessel.

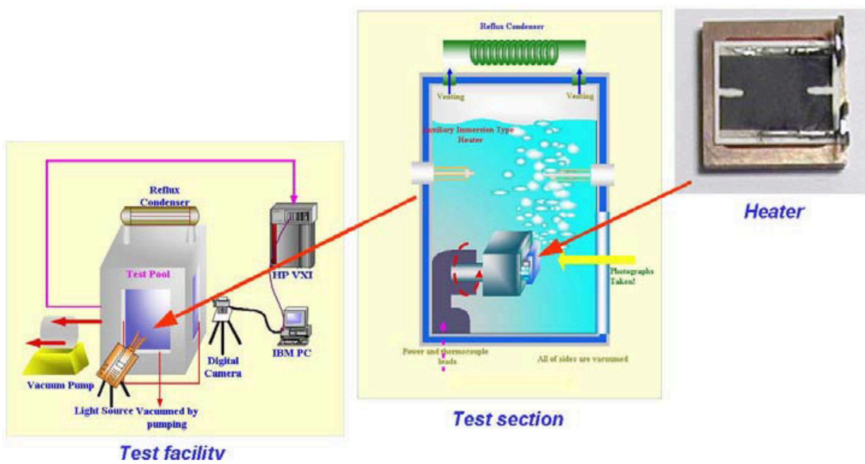


Figure 3. GAMMA 1D facility for narrow gap cooling tests. (Kim, et al., 2004)

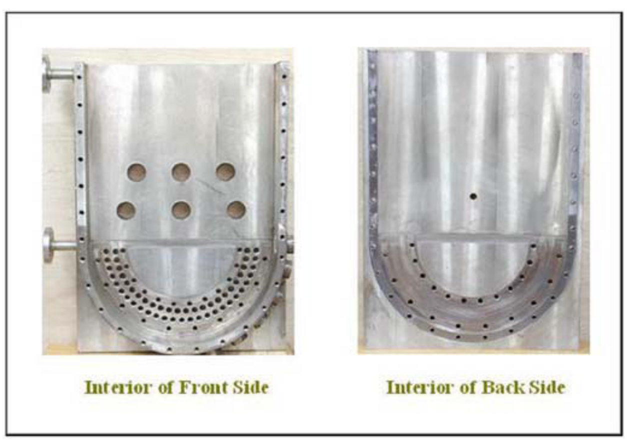


Figure 4. GAMMA 2D facility for narrow gap cooling tests. (Kim, et al., 2004)

Results indicate that CHF in gap boiling is affected by the gap size as well as by the induced flow within the channel. Consistent with other investigations, results show that CHF increases as the gap size increases at the vertical angle (90º). In the downward-facing (180º) position, however, the vapor movement was enhanced by the gap structure, and CHF increases as gap size decreases. As shown in Figure 5, GAMMA 1D data indicate that there is a “transition angle” where CHF changes rapidly for each gap size and that the transition angle increases as the gap size increases. The transition angles for the 2, 5 and 10 mm gaps were found to be 165º, 170º and 175º, respectively. However, no transition angle was detected for the 1 mm gap and pool boiling in an unconfined space.

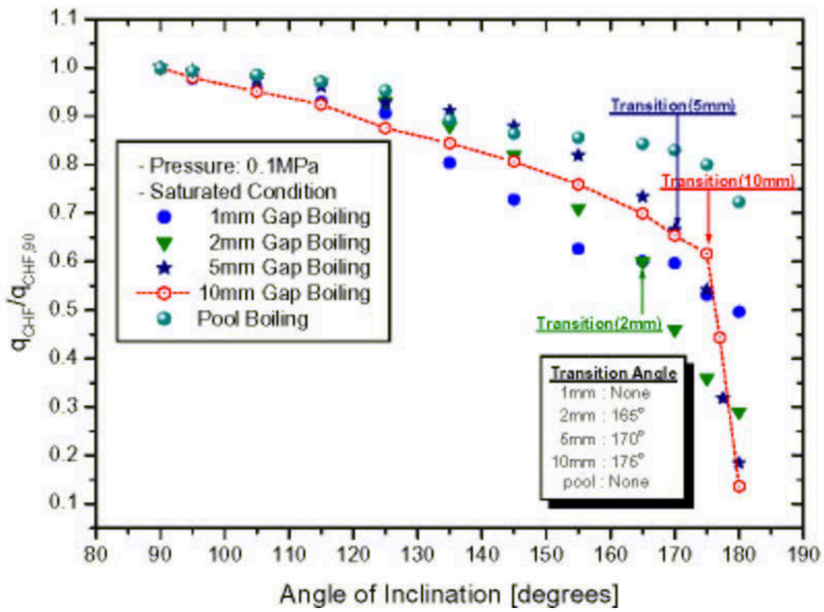


Figure 5. Effect of surface inclination and gap size on CHF.

In summary, the ability to predict heat transfer from a narrow gap has several applications in assessing the potential for IVR. It is needed to predict heat transfer in cracks that develop within the relocated core materials, between relocated core materials and the vessel or the "engineered gap" that may occur if an in-vessel core catcher is placed within the reactor vessel. Significant progress has been made toward developing a boiling curve for predicting heat transfer in narrow gaps with CCFL. Results indicate that surface angle, gap size, pressure, and dimensional effects must be considered.

3.2 ERVC

As part of a Korean/U.S. INERI, Cheung, et al., (2003) is investigating two design modifications that have the potential to enhance the margin for IVR via ERVC: an enhanced insulation /vessel configuration, and coatings on reactor vessel external surfaces.

Enhanced insulation /vessel configuration

The thermal insulation design may significantly impact local CHF limits, primarily because of the annular channel configuration that forms between the reactor vessel and the insulation structure. The impact is due to two main factors. One is the gap size or more specifically, the minimum gap of the annular channel. The other is the flow configuration of the annular channel. If the thermal insulation design is chosen so that an annular channel is formed that streamlines the steam venting process, then a higher coolant mass flow may be achieved for a given wall heat flux. Such a design leads to a higher local CHF limit owing to a higher coolant flow rate.

In general, a higher value of CHF can be obtained by using a design with a smaller gap size. This is true provided that the gap size remains large enough to allow effective steam venting. However, if the gap size falls below a certain critical value, choking of the steam flow could occur in the annular channel between the reactor vessel and the insulation structure. In that case, premature dryout could occur on the vessel outer surface with CHF limits significantly decreasing. For a given reactor vessel/insulation design, therefore, an optimum gap size exists that maximizes CHF values.

In most ALWRs, the annular channel that forms between the reactor vessel and the insulation structure does not have a uniform gap size. The APR1400 vessel/insulation design, for example, has a highly non-uniform gap size that varies significantly from the flow inlet at the bottom plate to the equator of the reactor vessel. In that case, the CHF data obtained for uniform gap sizes such as those reported by Theofanous et al. (1996, 1997), Rouge et al. (1997, 1999), Chang and Jeong (2002) and Jeong et al. (2003) cannot be directly applied.

For annular channels having variable gap sizes, the most important geometrical factor is the minimum gap size. Physically, this minimum gap represents a bottleneck of the channel that could potentially choke the steam flow at very high heat flux levels. On one hand, the local CHF is expected to increase as the gap size decreases. On the other hand, choking of the steam flow and premature dryout might occur when the minimum gap size falls below a certain value. Cheung and Liu (1998, 2001) observed a considerable decrease in the CHF for both the AP600 and APR1400 as the minimum gap size became very small. They found the decrease in the CHF to be a direct consequence of the relatively large pressure drop through the minimum gap region and the difficulty in venting the steam through the bottleneck. As the local CHF limit was approached, the vapor slug tended to occupy the entire cross-sectional flow area at the minimum gap, thus preventing the supply of fresh liquid water to the heating surface. This resulted in a premature dryout of the surface, leading to a considerably smaller value of the local CHF limit.

Cheung and Liu (1998, 2001) observed that for both the AP600 and APR1400 insulation geometries, the nucleate boiling heat fluxes and the local CHF limits for the case with insulation were consistently higher than the corresponding values for the case without insulation. These differences in the nucleate boiling heat transfer and the CHF were attributed to the buoyancy-driven two-phase flow effect. For the case without thermal insulation, an external boundary layer flow was induced by buoyancy on the vessel outer surface. The mass flow rate so induced was relatively small. On the other hand, with an insulation structure, a much higher mass flow was induced in the annular channel between the test vessel and the insulation structure. Hence, a stronger flow effect was present in the case with insulation relative to the corresponding case without thermal insulation, thus resulting in higher nucleate boiling and CHF.

Recently, Cheung et al. (2003) numerically performed a full-scale flow simulation of an enhanced vessel/insulation design having the same geometry as the APR1400 system. Results suggest that the coolant mass flow was significantly higher with an enlarged flow area near the shear keys and a smooth transition section in the insulation structure (see Figure 6). Using the Subscale Boundary Layer Boiling (SBLB) facility at the Pennsylvania State University (PSU), tests were conducted to validate numerical predictions and perform a visualization study of the downward facing boiling that occurs on the external surface of a vessel surrounded by an enhanced insulation structure. The SBLB facility consists of a sealed pressurized water tank with a condenser unit, a test-module sliding mechanism, a heated hemispherical test vessel surrounded by a scaled insulation structure, a data acquisition system, a high-speed photographic system, and a power control system. The water tank is designed and built for studying CHF phenomenon for downward-facing boiling on a simulated reactor vessel (see Figure 7). The water tank has several viewing windows for flow observation and can be operated under constant pressure conditions up to 2.5 bars. It has an air chamber at the top above the water level that can be used to preheat the test vessel. There are three immersion heaters located near the bottom of the tank that can be used to preheat the water to a desired temperature before a run. The heaters are spaced evenly in order to provide uniform heating. In addition, there is a reflux condenser assembly located on top of the tank. The condensate from the condenser is returned to the water tank from the bottom to maintain a constant water level within the tank.

Figure 8 (from Yang, 2004) compares the local CHF values for three different cases: a vessel without any insulation, a vessel with the originally-proposed APR1400 insulation, and a vessel with the enhanced insulation proposed in Figure 6. As shown in Figure 8, SBLB data suggest that improved steam venting through the narrow gap location occurs, which in turn leads to significantly enhanced CHF values.

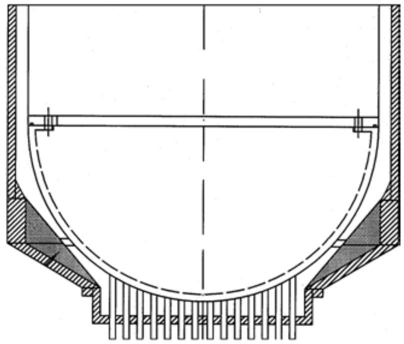


Figure 6. Proposed enhanced APR1400 insulation configuration. (Cheung, et al., 2003)

Figure 7. SBLB facility for ERVC testing (test section with insulation). (Cheung, et al., 2003)

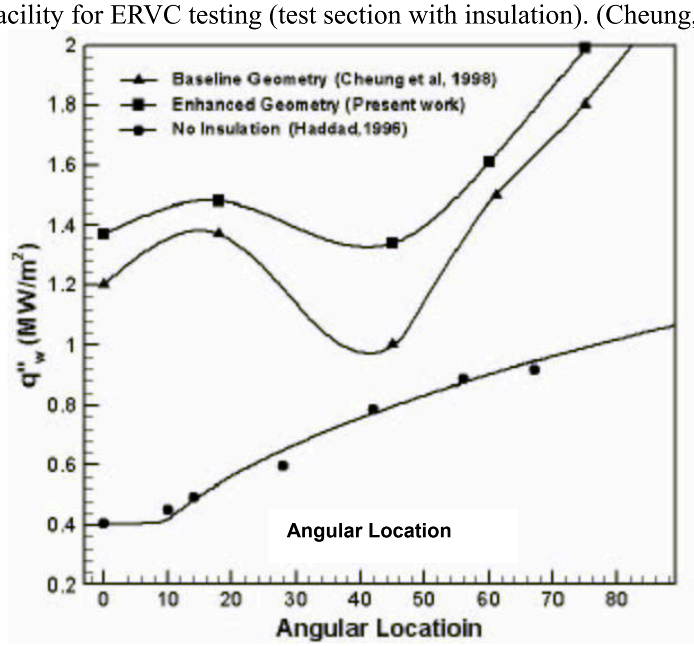


Figure 8. Comparison of CHF as a function of angle. (Yang, et al., 2004)

Vessel coatings

Under IVR conditions, the outer surface of the reactor vessel could be altered due to water chemistry, oxidization, and/or aging. Thus, it is necessary to address the potential effect of surface condition on the local CHF limits. From the numerous studies of conventional pool boiling and flow boiling reported in the literature, it is widely recognized that the surface condition could have a significant effect on the CHF. This turns out to be also true for downward facing boiling during ERVC under IVR conditions. The surface condition could substantially alter the vapor dynamics on the heating surface, thus affecting the local CHF limit at which local dryout would occur.

Theofanous et al. (2003) observed a strong effect of surface condition in their experiments at the UPLU-2400 facility for Configuration V. Test results indicated that the sand particles used to roughen the copper surfaces modified the surface molecules properties, at least temporarily, due to a deposition of aluminum molecules (the sand particles contained aluminum oxide). Because of the surface modification, different CHF limits were obtained. Moreover, they found that the molecular deposition of aluminum could be dissolved by de-ionized water used in some of their tests, leading to a significant reduction in the CHF values in subsequent tests. This degradation effect was not observed in those tests using tap water rather than de-ionized water.

Recently, Dizon et al. (2003) performed an extensive study of the effect of vessel coating on the local CHF limits in the SBLB facility. They developed a spray coating technique to form thin micro-porous metallic coatings on hemispherical test vessels. Their data clearly revealed that the coated vessels consistently increased the local CHF values from around 40% to more than 110% compared to those obtained under identical boiling conditions on uncoated, plain vessels (see Figure 9). Unlike the trend observed for plain vessels, the local boiling curve for coated vessels did not shift monotonically upward and to the right as the angular position was increased from the bottom center toward the equator of the test vessel. The local CHF limit at the bottom center was actually higher than the values for adjacent downstream locations up to θ = 28º. The local CHF exhibited a minimum at the θ = 14º location rather than at the bottom center.

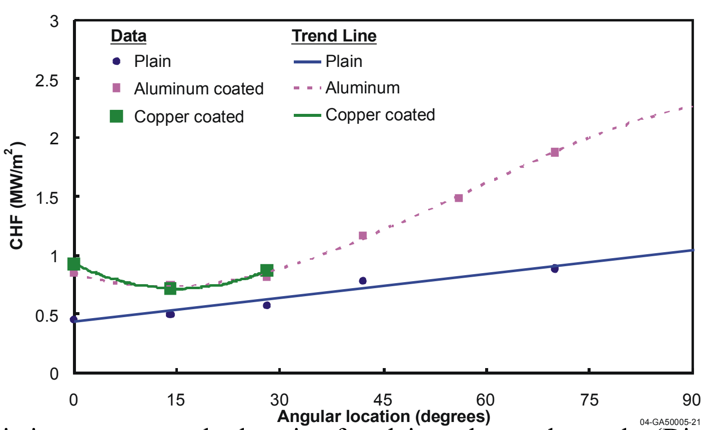


Figure 9. CHF variation versus angular location for plain and coated vessels. (Dizon, et al., 2003)

The non-monotonic behavior of the local CHF variation was found to be largely due to the capillary effect of the micro-porous coatings, where there was continuous liquid supply from all radial directions toward the bottom center. Optical and scanning electron microscope (SEM) examinations revealed that the vessel coatings had the form of a porous matrix composed of interconnected channels and different pores on the surface. Improvement in nucleate boiling heat transfer and CHF could be attributed to the structure of the porous layer itself and the capillary action it induced. The matrix of cavities and voids within the coating effectively trapped vapor, which served as active nucleation sites. These sites in turn were fed with liquid flowing through the interconnected channels.

The pores on the surface of the porous coating served as flow inlets for liquid supply to the heating surface, leading to appreciable enhancements in the local CHF limits.

Dizon et al. (2003) further observed that the micro-porous aluminum coating was very durable. Even after many cycles of boiling, the vessel coating remained rather intact, with no apparent changes in color or structure. Moreover, the heat transfer performance of the coating was found to be highly desirable with an appreciable CHF enhancement but very little effect of aging. Although similar heat transfer performance was observed for micro-porous copper coating, the latter was found to be much less durable and tended to degrade after several cycles of boiling. Dizon et al. (2003) concluded that the most suitable vessel coating material for ERVC is micro-porous aluminum coating.

3.3 In-vessel Core Catcher

The concept of a core catcher (in-vessel and ex-vessel) to retain materials that relocate during a severe accident has been proposed by several researchers. Table 1 summarizes desirable attributes for such a device. Clearly, an in-vessel core catcher must fit within the reactor vessel and not adversely affect coolant flow. In addition, it should be sized to retain materials expected to relocate during an accident without any recriticality concerns. Typically, core catcher designs rely on passive, rather than active measures. Although many are designed to initially retain the corium, it is recognized that the corium may ultimately penetrate through most core catchers. Nevertheless, these designs provide an additional barrier that delays when relocated materials attack the containment (or vessel, if the core catcher is within the vessel), providing additional time for the heat generation in the relocated materials to decay.

Table 1. Core catcher design goals

Design Goals
• The core catcher shall prevent recriticality of relocated material.
• The core catcher shall fit within the reactor vessel.
• The core catcher shall reduce the decay heat power density of relocating corium.
• The core catcher shall contain relocated material (size, structurally, thermal shock resistance).
• The core catcher shall be as inexpensive as possible.
• The core catcher shall facilitate long-term coolability using passive means.
• Interactions between core catcher materials and relocated debris shall not result in exothermic reactions or generation of combustible gases.
• The core catcher shall not present a seismic hazard.
• The core catcher shall be stable for the lifetime of the reactor.
• The core catcher shall be easily installed and maintained.
• The core catcher shall not adversely affect reactor performance or coolant circulation.

As part of a Korean/U.S. INERI (Rempe, et al., 2004), a preliminary design was developed that builds upon an in-vessel core catcher concept proposed by Hwang and Suh (2001). This new core catcher design consists of several interlocking sections (see Figure 10). The use of multiple sections reduces cost and simplifies manufacture and installation. The sections are machined such that they fit together when inserted into the lower head. For reactor designs with penetrations, the core catcher is manufactured with holes to accommodate lower head penetrations. Each section of the core catcher (see Figure 10) consists of two material layers with an option to add a third layer (if deemed necessary): a base material, which has the capability to support and contain core materials that may relocate during a severe accident; an oxide coating material on top of the base material, which resists interactions with high-temperature core materials; and an optional coating on the bottom side of the base material to prevent any potential oxidation of the base material during the reactor's lifetime.

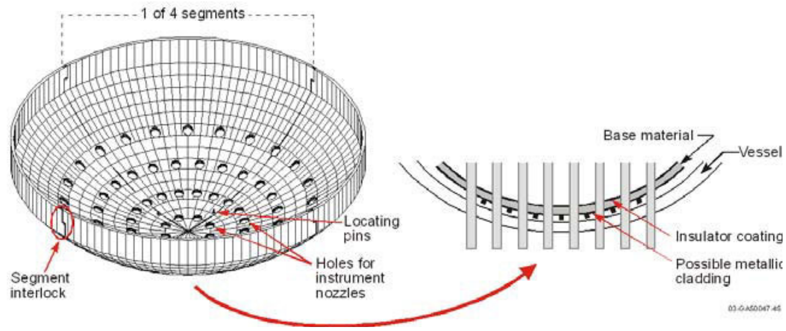
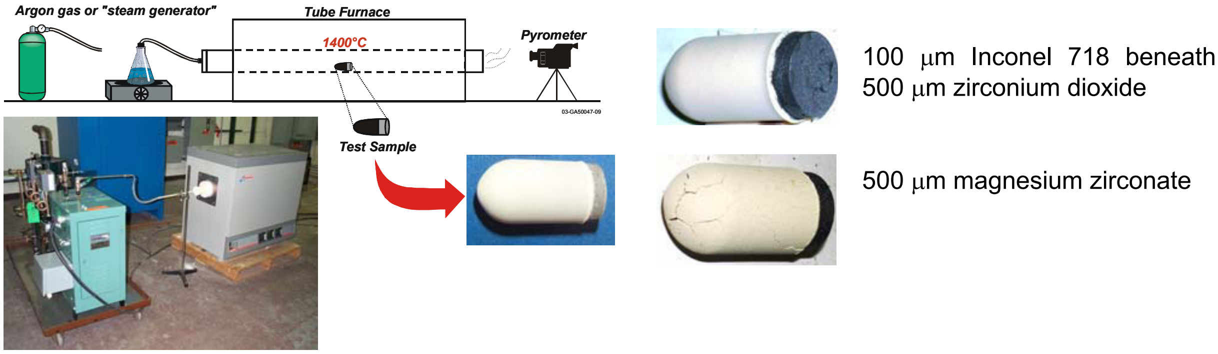


Figure 10. APR1400 core catcher conceptual design. (Rempe, et al., 2004)

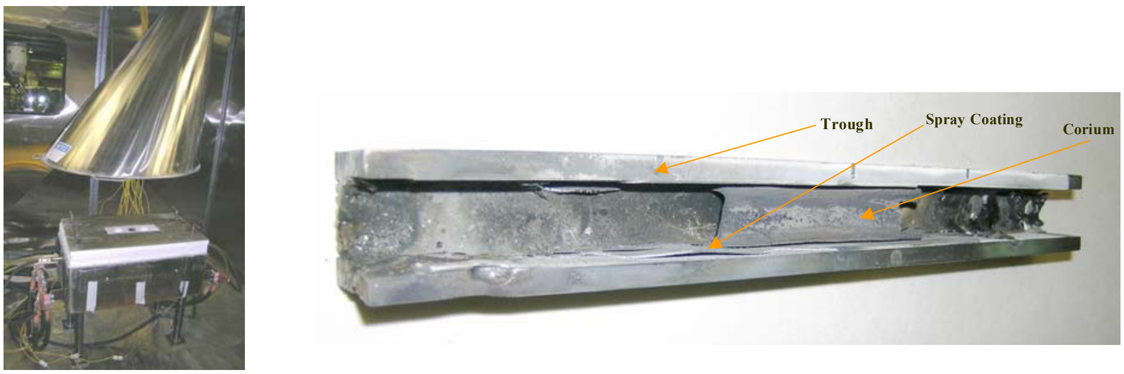
Various types of application methods, such as chemical vapor deposition, thermal plasma spraying, and painting, were reviewed; and preliminary evaluation suggests that the insulator coating should be applied via a plasma spray process. This process, which is relatively inexpensive, can provide a chemically stable, rugged, dense, and bonded coating of materials for any desired thickness.

Key properties of possible core catcher base and coating materials were reviewed; and a set of candidate materials was identified based on cost, material properties (melting temperature, ultimate strength, thermal conductivity, resistance to thermal shock, coefficients of linear expansion, etc.), and the potential for chemical interactions. Scoping thermal and structural analyses were also completed to obtain additional insights about the thickness and type of material that should be selected for each layer of the core catcher. Results suggest that coatings containing magnesium dioxide or zirconium dioxide will perform better because of their material properties. Results also indicate that the thermal performance is not significantly impacted by the type of steel (SS304 or SA533B1) selected for the base material, the thickness of the base or coating material, or the porosity of the coating material. Scoping structural analyses results suggest that the core catcher's base material should be at least 2 cm thick to support the loads associated with relocated materials during a severe accident (smaller thicknesses may be possible, depending upon heat removal capabilities associated with narrow gap cooling). Scoping analyses were also completed to determine the impact of this core catcher on reactor vessel coolant flow. Results indicate that only 2% of the RCS flow from the downcomer may be diverted beneath the core catcher if it is placed approximately 0.5 cm above the reactor vessel inner surface. Hence, the impact on RCS flow is considered negligible.

Efforts were completed to identify possible high temperature materials interactions between proposed insulator coating materials and base materials for the core catcher. In addition, sensitivity studies were performed to optimize thermal spray parameters for coating materials. As shown in Figure 11, samples were tested in steam and argon at 1400ºC. Endstates suggest that samples coated with zirconium dioxide sample over a 100 μm Inconel 718 bond coating performed best, experiencing no materials interactions and minimal cracking (Figure 11). To further assess the performance of these materials, tests are being conducted with prototypic materials expected to relocate during a severe accident at the Idaho National Engineering and Environmental Laboratory (INEEL) High Temperature Test Laboratory (HTTL). In these tests, the performance of the proposed core catcher materials is assessed when it is exposed to high temperature (> 2800 K) corium materials (UO_2, ZrO_2, U-Zr-O eutectic, etc.) in inert and oxidizing environments.



(a) High temperature testing of samples with thermal spray coatings



(b) Prototypic materials test setup and simulated core catcher test section

Figure 11. HTTL testing provides insights about core catcher materials. (Rempe, et al., 2004)

As part of a Korean/U.S. INERI (Kang, et al., 2004), the Korea Atomic Energy Research Institute (KAERI) is performing LAVA-GAP experiments (see Table 2). As shown in Figure 12, the LAVA-GAP facility consists of a thermite melt generator, a melt separator, a test section of the lower head vessel, and a gas supply system for internally pressurizing the system. The experiments have been conducted using 60 kg of Al_2O_3 thermite melt as a corium simulant with a 1/8 linear scale mock-up of the reactor vessel lower plenum. The lower head vessel was made of carbon steel with an inner diameter of 500 mm and a thickness of 25 mm, respectively. After the thermite melt was generated in the melt crucible, it was first delivered to the melt separator and then gravitationally poured into the lower head vessel through the melt delivery nozzle with an inner diameter of 8 mm. As shown in Figure 13, LAVA-GAP tests with simulant material indicate that an in-vessel core catcher will significantly reduce thermal loads and attack from relocating materials.

Table 2. LAVA-GAP Test Parameters

Test	Melt[1]	Pressure, MPa	Initial Water Subcooling, °C	Coating	Gap Size, mm
LAVA-GAP-2	Al_2O_3, 60 kg	0.1	1.5	None	10
LAVA-GAP-3	Al_2O_3, 60 kg	0.1	1.5	92 % ZrO_2, 8% Y_2O_3	10
LAVA-GAP-4	Al_2O_3, 60 kg	0.1	1.5	92 % ZrO_2, 8% Y_2O_3 95% Ni – 5% Al bond coating	5
LAVA-GAP-5	Al_2O_3, 60 kg	0.1	1.5	92 % ZrO_2, 8% Y_2O_3	5
LAVA-GAP-6	Al_2O_3, 60 kg	0.1	1.5	92 % ZrO_2, 8% Y_2O_3 Inconel-718 bond coating	10

1. Some iron also relocated into the core catcher due to incomplete melt separation and dissolution.

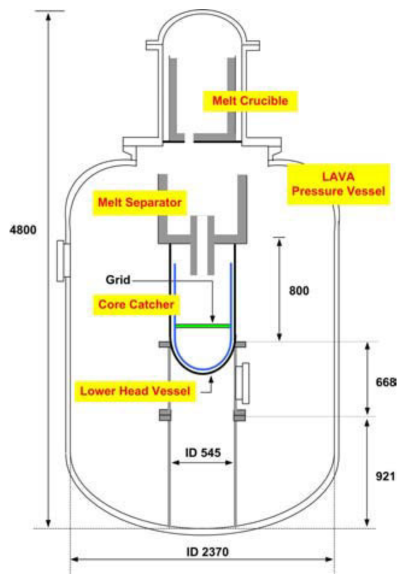


Figure 12. Schematic diagram of the LAVA-GAP experiments (Kang, et al., 2004)

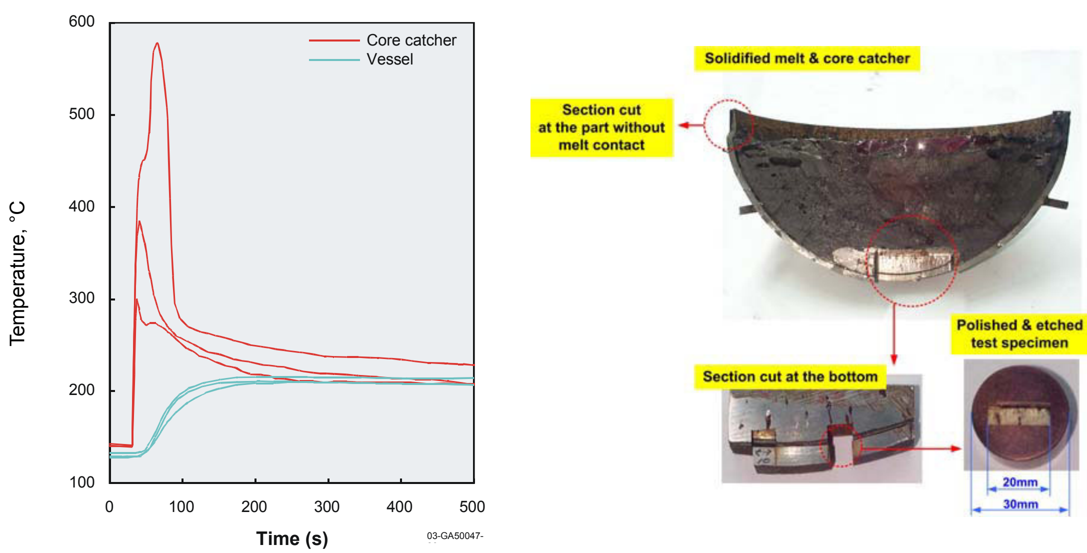


Figure 13. LAVA-GAP data show and post-test exams show that vessel thermal loads are significantly reduced with an in-vessel core catcher. (Kang, et al., 2004)

4. SUMMARY AND ADDITIONAL NEEDS

IVR is an accident mitigation strategy that is relied upon by advanced and existing reactor designers. Furthermore, IVR strategies may be applicable to several generation IV (GEN IV) reactor designs, such as the supercritical water reactor. In evaluating the viability of IVR, several factors should be considered:

- *Differences in power level.* Most advanced LWR designs proposed by U.S. vendors have higher power level ratings than the 600 MWe AP600.
- *Cavity flooding.* It is essential that safety systems flood the cavity before melt relocation. Hence, sufficient water must be available to rapidly flood the reactor cavity.

- *Recent experimental data confirming the potential for alternate debris configurations.* Recent tests indicate that several debris endstate configurations may be possible and that fission products could distribute according to their chemical composition.

Clearly, significant progress has been made to improve our understanding of mechanisms that impact IVR. Nevertheless, additional evaluations could solidify the case for IVR in advanced reactors proposed with higher power levels and for applicable GEN IV reactors for which enhanced safety must be demonstrated. This review indicates that investigations into the following areas have the potential to significantly enhance the margin for IVR.

Narrow Gap Heat Transfer – The ability to predict heat transfer from a narrow gap has several applications in assessing the potential for IVR. It is needed to predict heat transfer in cracks that develop within the relocated core materials, between relocated core materials and the vessel or the "engineered gap" that may occur if an in-vessel core catcher is placed within the reactor vessel. Significant progress has been made toward developing a boiling curve for predicting heat transfer in narrow gaps with CCFL. Results indicate that surface angle, gap size, pressure, and dimensional effects must be considered. Once narrow gap models are developed, they should be applied to predict heat transfer in relocated corium materials and in the gap between an in-vessel core catcher and the vessel. If simulant test data are used, additional material properties must be obtained so that differences between prototypic and simulant material behavior may be better understood.

External Reactor Vessel Cooling – Several investigations have been completed that explore methods to enhance the margin associated with ERVC. Results show that the efforts to enhance the insulation placed around the vessel structure can significantly improve the steam venting process (and subsequently increase local CHF values). In addition, tests show that coatings can be applied to the vessel external surface and enhance its coolability). However, additional investigations are needed in the following areas:

- Additional tests to assess the combined effect of enhanced insulation and vessel coatings.
- Correlations for predicting CHF with an enhanced vessel/ insulation configuration, vessel coatings, and a coated vessel with an enhanced vessel/insulation configuration.

These investigations will be completed as part of an on-going Korean / U.S. INERI.

In-Vessel Core Catcher - Although new experimental and analytical investigation results suggest that an in-vessel core catcher is viable and can significantly reduce the heat loads to the vessel from relocated core materials, more detailed evaluations are needed to fully assess its merit.

- More detailed evaluations of the impact of an in-vessel core catcher on coolant flow in the RCS. As discussed above, scoping calculations indicate that the presence of the in-vessel core catcher will not significantly impact coolant flow. However, detailed calculations and experimental evaluations should be completed to confirm scoping analysis results.
- More detailed experimental evaluations to confirm that proposed coatings will not degrade over the lifetime of the reactor.
- Experimental evaluations to assess in-vessel core catcher performance when subjected to sustained heat loads. Such evaluations could be conducted by placing heat sources in the simulant materials that relocate in the LAVA-GAP facility.
- As noted above, narrow gap heat transfer models should be applied to predict heat transfer observed in recent LAVA-GAP tests. Then, additional material properties should be obtained so that differences between prototypic and simulant material behavior may be better understood.

Although significant progress has been made by scoping Korean/U.S. INERI in-vessel core catcher investigations, more detailed evaluations are needed prior to implementing this safety feature into a commercial reactor vessel.

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