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Thermophysical, Hydrodynamic and Mechanical Aspects of Molten Core Relocation to Lower Plenum

Kune Y. Suh, Chang Wook Huh
Seoul National University
Department of Nuclear Engineering
56-1 Shinrim-dong, Kwanak-gu, Seoul, 151-742, Korea
Phone: +82-2-880-8324, Fax: +82-2-889-2688, Email: kysuh@plaza.snu.ac.kr

Abstract

This paper presents the current state of knowledge on molten material relocation into the lower plenum. Consequences of movement of material to the lower head are considered with regardt to the potential for reactor pressure vessel failure from both thermal hydraulic and mechanical standpoints. The models are applied to evaluating various in-vessel retention strategies for the Korean Standard Power Plant (KSNPP) reactor. The results are summarized in terms of thermal response of the reactor vessel from the very relevant severe accident management perspective.

I. INTRODUCTION

As emphasized in the TMI-2 accident, the transfer mode of molten material from the degraded core to the lower head of the vessel is an important factor in the evaluation of the in-vessel progression of a severe accident in nuclear reactors. The understanding of this complex phenomena is pivotal to the in-vessel debris cooling and potential vessel failure analyses [1-5].

Early phase (metallic phase melting) and late phase (ceramic phase melting) core damage progression affect the initial conditions for material relocation. Metallic blockage formation in the lower core region, melt pool growth and crust failure behavior are important factors in determining the quantities, rate and mode of melt/material transfer to the lower head. Melt relocation process into the lower plenum may also depend on the differences in core damage conditions relevant to the type of nuclear reactor under examination. Generally, the TMI-like "wet-core" scenario, typical for PWRs, contrasts with a BWR-typical "dry-core" scenario. As demonstrated by the TMI-2 accident, the "wet-core" conditions enhance large molten pool formation with coherent molten mass relocation after crust failure; while gradual melt relocation is more likely under "dry-core" conditions.

II. PHYSICO-CHEMICAL PHENOMENA RELEVANT TO IN-VESSEL RETENTION

Fuel degradation begins after the coolant inventory in the reactor vessel falls below the top of the reactor core. Flowing steam, which is generated by boiling taking place in the still covered fuel region of the core, is initially sufficient to convectively cool the exposed fuel rods, but as the water level continues to drop, the flow of steam diminishes to the point that the heat removed by convective cooling is less than the decay heat generation rate in the exposed fuel. At that point, the fuel rods begin to heat up and fuel degradation processes follow. The major core degradation processes that follow are listed in approximate chronological order: (1) steam oxidation of zircaloy cladding with hydrogen generation, (2) cladding failure by ballooning or by local perforation in a grid spacer location by eutectic interaction, leading to the gap release of fission products to the reactor coolant system, (3) failure of control rods (PWR) or blades (BWR) leading to relocation of molten control materials to lower cooler regions of the core, (4) melting of the metallic zircaloy cladding and in the case of BWR designs, of the channel boxes, and relocation of these materials to the lower core region, (5) release of noble and volatile fission product species (Xe, I, Cs, etc.) to the reactor coolant system (RCS), (6) degradation and melting of the ceramic fuel materials remaining in the upper hotter regions of the core, (7) release of lower volatile fission products (Sr, Ba, Ce, La, U, etc.) to the RCS, (8) formation of a high temperature ceramic molten pool, and (9) the migration of the molten fuel pool region to a core boundary where release of the material to the lower vessel region takes place.

Each of the core degradation processes described above is affected in degree by variations in accident conditions (e.g. pressurized or depressurized conditions), and by the reactor type, PWR or BWR. The core melt progression processes determine the timing of core material/melt transfer to the lower vessel region as well as the total mass transferred, the rate of transfer, and the composition (metallic/ceramic) and temperature of the relocating materials. These conditions ultimately determine the conditions for subsequent accident progression behaviors, including (1) melt/coolant interactions in the water-filled lower plenum, (2) thermal loading characteristics of the lower vessel head, (3) likelihood and manner of pressure vessel failure, (4) magnitude of containment pressurization in the event of a pressurized melt ejection, and (5) the degree of ex-vessel core/concrete interactions.

Failure of the debris crust in the core would allow the molten material to relocate into the lower region of the reactor vessel. Potential crust failure mechanisms [6-10] may be:

1) thermal melt-through of the crust by natural convective heat transfer driven by decay heating,
2) thermal/mechanical stress failure due to temperature and pressure differences across the crust layer, and
3) eutectic interaction between the crust material and the interior material.

The following arguments go along the line of the first two boundary conditions, the third mechanism still carrying considerable uncertainties because of difficulties in precisely determining the material compositions in the molten and solidified debris.

When a molten pool has been created in the core, the boundary condition to cause a localized collapse of a molten pool crust is the crust thickness which would support the melt above. The functional expression to describe the strength may be of the Larson-Miller parameter which relates the applied stress to the time to rupture given the crust average temperature. Numerous investigations for UO_2 fuel have shown that the material is susceptible to creep. The stress can be related to the depth of molten material accumulated and the available area for resisting this applied load. The bending stress may be used to evaluate failure of the side crust, while the shear stress may be adopted for failure of the bottom crust. The other variable to be evaluated is the crust average temperature. Typically this value would be determined by the temperatures of the overlying melt and the material below. Hence, this can be approximated by conduction behavior through the crust to the material below.

The tensile strength and rupture strength of UO_2 at room temperature are given as 7.5 MPa and 220 MPa, respectively, by Ma [6]. This gives an indication of the strength of the material such that a functional relationship for stress versus the Larson-Miller type parameter for the crust can be established. In addition, the TMI-2 experience revealed that a stable crust could be strong enough to hold a debris pool of about 20,000 kg for at least one hour and likely much longer. A functional relationship may be characterized so that the crust parameter can be computed for an imposed stress. Then the time at temperature until rupture would occur can be evaluated. This provides for an assessment of downward melt relocation versus a sideward progression as determined by the molten pool circulation and effective strength of the debris crusts. This functional representation for the creep rupture failure mechanism can provide for a time-at-temperature behavior and demonstrate a weakening condition as the debris temperature increases. The rupture time for a crust may be determined by summing the fractional progression of the time to rupture during the accident.

An approximate failure mechanism of this nature can also be used for the side crust at the core outer boundary. A similar evaluation can be performed for bending of the crust where the stress acting on the individual crust is the height of molten material above the bottom of the core node in consideration. Also this evaluation should be performed for each node with crusts in the outer radial region. If a failure condition is calculated, this should be considered to be a local failure which allows molten debris to flow into the bypass region and then down to the lower plenum ablating and perforating the core shroud (as in BWRs) or the core baffle (as in PWRs).

Once the crust is breached, the molten material may flow from the core region in three possible modes [6]:

1) narrow discontinuous streams distributed over the duration of the core meltdown,
2) a narrow continuos pour over a period of fractions of minutes to several minutes, or
3) a relatively massive, coherent pour occupying a few seconds or less.

The third mode may be broken up by the core peripheral and support structures and bottom-head-entry control rod guide tubes (as in BWRs) and the instrumentation thimble guides (as in PWRs). The timing of the discharge may be related to the mode of relocation and to the degree of core melting. Thus, if a large amount of core is molten at the onset of discharge, a massive relocation can take place (mode 3 for mostly PWRs). On the other hand, if only a small fraction is molten, a smaller, gradual relocation may result (modes 1 and 2 for mostly BWRs). If the relocation takes place alongside the core periphery depleted in water, the molten jet may candle down the core boundary structure wall partly freezing on the surface and partly ablating the structure. The solidified debris may partially hold up the progression of the melt through the annular gap.

Core debris relocation into the lower head may involve a coherent mixing of a large fraction of the core inventory with water in the lower plenum. However, a gradual, rate-limited relocation of the debris may also result from a local breach of a crust formed in the core region. Moreover, the presence of structures containing flow paths of varying sizes may intercept and redirect the melt streams or rivulets. These structures' geometric configuration and thermal/mechanical conditions will determine the size and the number of molten jets draining down to the lower plenum. The flow paths themselves will vary in size and configuration due to possible freezing and remelting of the heat generating debris on the surface of the structures.

Because of the structures between the core and the lower plenum, the most likely mode of debris entry into the water would be of small diameter pour streams. These streams or jets would undergo breakup when flowing through water. The rate at which the jets disintegrate influences the steam generation rate and hydrogen production, debris bed coolability, reactor vessel lower head thermal and pressure load, creep and potential failure. If the water depth is shallow where the debris jet enters the lower plenum, some fraction of the molten debris within the pour stream may not be entrained. Rather, it will accumulate as a non-particulate continuous layer. On the other hand, that portion of the jet that is particulated will become frozen particles of fairly large characteristic size in the range of 1 - 5 mm or larger [6].

The debris jet particulation can be represented as the erosion of a cylindrical jet using an entrainment correlation. The jet velocity can be computed as gravity driven drainage, and then the maximum length of the intact jet can be evaluated. The molten debris entrainment can be evaluated from the decrease in the jet area and the jet flow rate with the particle diameter assumed to be of the capillary size.

As the high temperature core material enters the lower plenum water, an interaction zone can be formed in which water circulates into the bottom and steam exhausts from the top. With the steam void developed in the zone, the density difference between the water surrounding the zone and the voided region would drive the circulation. This density difference must also support the weight of core material in the zone. As debris enters the interactive zone, the water flow would slow the downward penetration, reverse the particle flow and push it out through the sides of this region. This water velocity can be estimated by the levitation of an individual core debris particle by the water. This can be assumed to be the velocity of water circulating through this interaction zone on account of density differences. In addition to treating the cooling of the settling particles, one must deal with the cooling of the debris bed accumulated in the lower plenum. The settling distance can be determined from the point of jet entrainment through the water to the upper surface of debris bed.

Regarding steam generation due to debris particulation and circulation within the water, a simplification can be made such that the rate of energy removal from the debris in the process of quenching can be evaluated based on the sensible and latent heat in the material and the material entrainment rate. This can be divided into that which causes net steam generation and that which is transferred to the circulating water in the interaction zone.

The time over which the energy is extracted from the debris can be determined by the fall height of the entrained droplets and their terminal velocity in water. With this, the rate of energy transfer to the water as the particles settle to the debris bed can be calculated along with the energy addition rate to the debris bed. As the molten debris jet is broken up and entrained, the debris particles can undergo surface oxidation.

The initial diameter of the debris jet presents uncertainty since it dictates the entrainment, and hence, the resultant steaming and to a limited extent the hydrogen generation in the lower plenum water. This jet diameter is likely to be determined by the size of holes in the structure below the reactor core, i.e. core plate, diffuser plate, etc. The debris particle size has a minor influence on the fraction of Zr oxidation. Typically, the particles are expected to be several millimeters or centimeters [6]. All sizes greater than 1 mm have cooling rates well in excess of decay power and the oxidation is also not greatly changed.

Local flooding or dryout may be assumed to be the controlling process for cooling the particulated debris bed. Since the particle settling time is considerably smaller than the cooling period, one can assume an initially, fully established debris bed of particles of representative sizes and the initial effective temperature. With this assumption, the quenching of the debris bed may be estimated. Both of these models predict effective cooling for particle sizes of a few millimeters. Hence, there is no significant influence with respect to the debris bed quenching model. The combined radiative and conductive heat transfer out from or into the debris bed may be represented as a semi-infinite optically thick slab.

Once the molten debris lands on the lower head, as a result of stresses imposed by the internal pressure and the dead weight of the material, the reactor vessel lower head may experience creep at elevated temperatures. As the material creeps, the brittle oxidic core material is not subject to the same pressure stresses, hence it would not experience the same creep and would tend to separate from the reactor vessel surface. Consequently, if such a creep mechanism were established, the lack of adherence of the debris to the reactor vessel wall and the differential structural response would result in a stretching of the reactor vessel wall with respect to the core material. This would then create paths for water to ingress between the debris and the reactor vessel wall. This relative growth of the reactor vessel wall compared to the core debris, while quite small (gap dimension of the order of 100 μm), would be extremely important in terms of the thermal response of the reactor vessel wall.

III. REACTOR APPLICATION AND SUMMARY

The foregoing models were applied to the KSNPP reactor high pressure (station blackout) and low pressure (large-break LOCA) sequences to quantify the consequences of the core material relocation onto the lower head. A total of eight (8) cases (see Table 1) were chosen each, and the reactor vessel temperature results are presented in Figure 1 for the station blackout case in which the lower head creep was more conspicuous due mostly to the high primary system pressure. The parameters varied include various vessel outer and inner wall heat transfer coefficients, recovery time, allowance of thermal and mechanical elongation of the lower head wall. As can obviously seen, the sequences with recovery after some time since the first movement of the molten core material into the lower plenum all have attained fairly low temperatures throughout the wall thickness. This theorizes the potential benefit of flooding the reactor vessel either from inside or outside or both. One thing to note here, though, is that the commercial power reactors are equipped with the thermal shield during normal operation. If one would decide to go with the reactor cavity flooding, the themal shield must soehowbe taken down so that the coolant water may effectively find its way to the vessel lower head. Also the lower penetrations must be given special considerations that they shall not suffer quench crack at the time of external flooding of the vessel. On the other hand, the internal injection of emergency coolant can meet with high primary system pressure so that depressurization may become a prerequisite. Also, the difficulty of the coolant to soak through the narrow gap formed between the debris and the lower head wall should be thoroughly examined to realistically quantify the risk of jeopardizing the vessel integrity during a severe accident.

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Table 1. KSNPP Reactor Application Cases for In-Vessel Retention Study

Case	*External Cooling (Recovery Time)*	*Internal Cooling (Recovery Time)*	*Creep*
1	No	No	No
2	No	No	Yes
3	Yes(15 min)	No	No
4	Yes(90 min)	No	No
5	Yes(210 min)	No	No
6	No	Yes(0 min)	No
7	No	Yes(15min)	No
8	No	Yes(45min)	No

* Recovery Time : Elapsed Time from Core Material Relocation into Lower Plenum

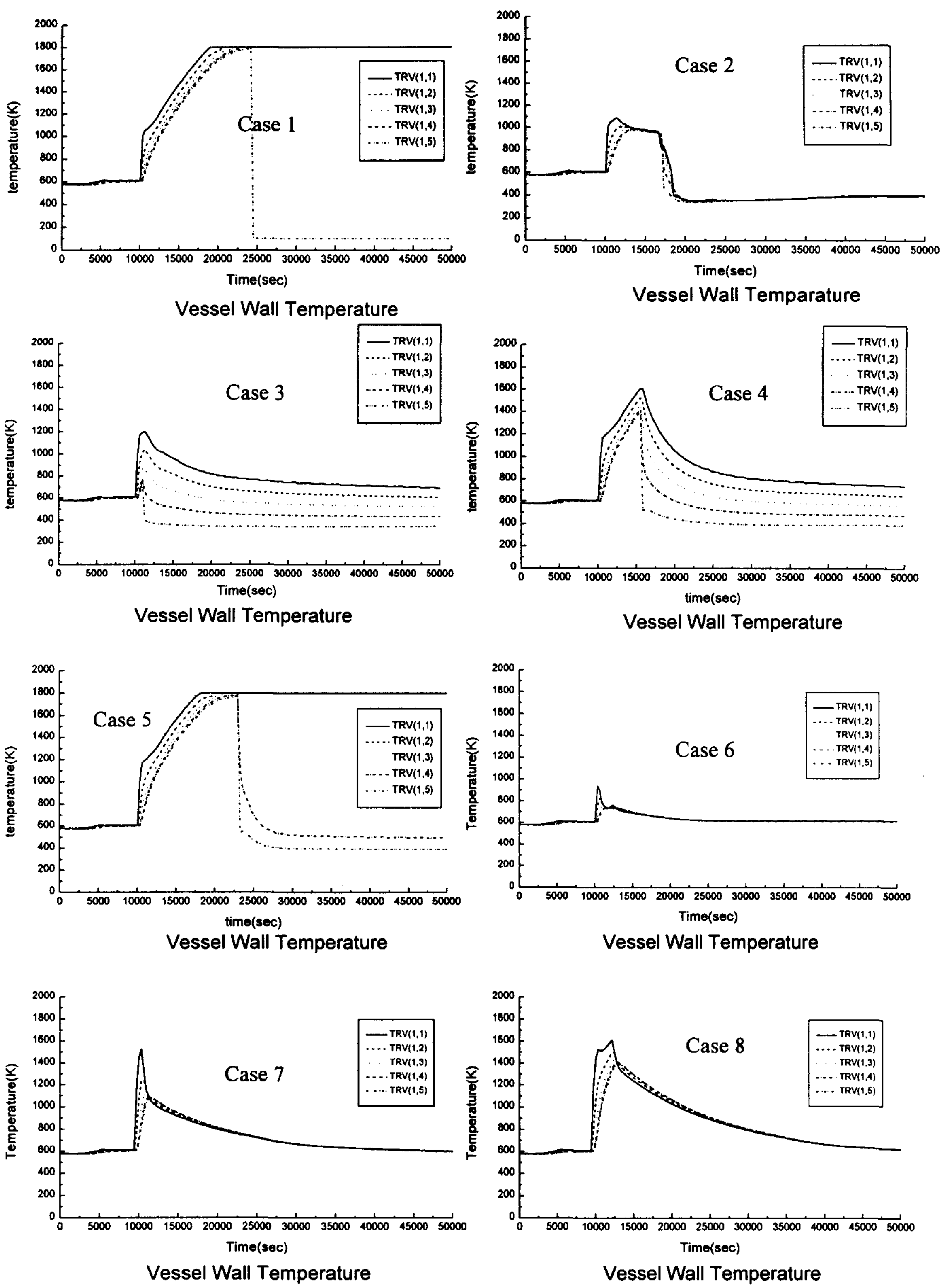

Figure 1. Reactor Vessel Wall Temperature Response