TMI-2 ACCIDENT SCENARIO DEVELOPMENT[a]

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ABSTRACT

A best-estimate accident scenario describing the important mechanisms that controlled the core damage progression during the TMI-2 accident has been described in previous papers and reports. Several important questions were identified in these documents for which additional analysis and/or data are necessary to develop an adequate understanding. This paper summarizes recent analytical work relating to: (a) configuration of the degraded core based on interpreting the source range monitor data, (b) the coolability of the upper core debris bed, (c) potential crust failure mechanisms and the interaction of the molten core material with the reactor vessel coolant, and (d) potential reactor vessel damage.

INTRODUCTION

The TMI-2 accident resulted in extensive damage to the reactor core and significant release of fission products from the fuel. Defueling data has confirmed that approximately 30% of the original core material (50 metric tons) achieved melting temperatures and an estimated 15 metric tons of molten core material relocated to the lower plenum region of the reactor vessel.[1,2] Because of the extensive core damage, the TMI-2 accident offers a unique opportunity to extend our knowledge of important physical mechanisms affecting core damage progression and fission product behavior for a severe accident under achievable reactor system conditions.

The TMI-2 Accident Evaluation Program[3] is being conducted for the U.S. Department of Energy as a severe accident research effort to develop a consistent understanding of the mechanisms controlling the core damage progression and resulting fission product behavior during the TMI-2 accident. This goal is being achieved through:

- Inspection and characterization of the end-state core material distribution and damage state of the core, core support assembly (CSA), and reactor vessel,

- Interpretation and qualification of the TMI-2 data recorded during the accident as it relates to the reactor system thermal hydraulic response, and

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- Analysis work to integrate these data into a consistent scenario of core damage progression and fission product behavior.

Details of the core damage progression (accident scenario) have been documented in previous papers[4,5] and reports.[6] A summary of the timing and major physical mechanisms hypothesized to have controlled the core damage progression is given in Table 1. Important questions relative to the mechanisms that controlled the core damage progression have been further investigated this year. These include the following:

1. What was the extent of core material relocation before the pump transient (and upper debris formation)?

2. What was the coolability of the upper core debris bed formed as a result of the pump transient at 174 min?

3. What was the mechanism that caused crust failure at 224 min?

4. What was the interaction of the molten core material with the reactor vessel coolant?

5. What was the potential damage to the reactor vessel?

Discussions of recent analytical work relative to each of these areas are provided in the following sections.

TIMING AND EXTENT OF CORE MATERIAL RELOCATION

The available TMI-2 data indicate that severe damage of the core had occurred by between 150-160 min and that a major relocation of core materials occurred between 224-226 min (see Table 1). The source range monitor (SRM) located outside the reactor vessel at the core mid-plane provided a signature of the changing conditions within the reactor vessel. Figure 1 compares the measured SRM response to the normal detector count rate after shutdown. Three features of the SRM response are important relative to the core configuration: (a) the rapid drop in detector count rate coincident with the B-pump transient (174 min), (b) the rapid increase between 224-226 min, and (c) the longer-term response between 400-1500 min showing a slowly increasing and then decreasing trend.

The rapid decrease in the SRM response at 174 min provides a unique benchmark to evaluate the degraded core configuration. Notice, however, that the SRM count rate did not decrease fully to the normal shutdown level. Thus, it can be hypothesized that (a) the core region was not filled with water, and/or (b) the core configuration had changed significantly. The previous interpretation of SRM data[7] assumed that the core was intact, thus giving no insight into the effect of core material relocation on the detector response.

Recent neutronic analysis[8] has been completed to evaluate the effect of core material relocation (both fuel and control rod material) on the SRM

TABLE 1. SUMMARY OF CORE DAMAGE PROGRESSION DURING THE TMI-2 ACCIDENT

Time Period	Summary of Core Damage Progression and Fission Product Behavior
0-100 minutes (Loss-of-coolant Period)	Primary coolant pumps provided cooling to the core. Coolant pump operation was terminated at 100 min.
100-174 minutes (Initial Core Heatup Period)	Core liquid level at pump shutdown was near the top of the active fuel. Core liquid level decreased due to heat transfer (decay heat) from the core. Core temperatures of 1100 K achieved by 140 min. Rapid oxidation of core started near 150 min and resulted in relocation of zircaloy cladding and UO_2 to lower regions of core. Continued core oxidation and subsequent fuel liquefaction and core slumping (melting) of fuel resulted in a large region of consolidated core material in the lower regions of the core. Gaseous fission product release from ruptured cladding occurred by approximately 140 min. Additional release occurred as a result of fuel liquefaction. Fission product release from the consolidated region was minimal because of limited diffusion from the large region.
174-176 minutes (Pump Transient)	The B-pump transient resulted in coolant injection into vessel for a short period (<1 min). Interaction of the coolant with the upper fuel rod remnants resulted in fracturing (thermal/mechanical shock) and in formation of the upper core debris. Cooling of the consolidated core material in the bottom regions of the core was negligible. Little enhanced release from the upper fuel rod remnants during the rod fracturing is estimated based on available examination data. Fission product release from the consolidated region was insignificant.
174-200 minutes (Degraded Core Heatup)	Heatup of the consolidated core material in the bottom of the core continued. Formation and growth of an interior molten region are postulated. Little fission product release from the consolidated region is thought to have occurred due to the limited diffusion through the large region of consolidated material and the solid surrounding crust.

TABLE 1. (contianued)

Time Period	Summary of Core Damage Progression and Fission Product Behavior
200-224 minutes (Degraded Core Heatup)	Continued heatup of the degraded core regions resulted in a large molten region within the consolidated core region. Heat loss from the region was minimal because of the insulating ceramic crust. Fission product behavior within the molten pool was likely dominated by the convective flow and chemistry of the participating materials (fuel, cladding, control rods, and core structure). No significant release from the consolidated region is expected based on the small diffusivity in the ceramic crust.
224-226 minutes (Major Core Relocation)	Localized failure of the core crust in the east quadrant occurred, due to thermal attack or stress induced failure. The upper core debris settled into the molten core zone. Molten core material was displaced from the consolidated core region and flowed downward into the lower plenum region and outward into the core former/baffle plate region. Most of the flow was directed downward into the lower plenum. Fission product release during the molten core material relocation was likely controlled by the interaction between the molten core material and the coolant in the lower core and plenum regions.
Post-226 minutes (Core Cool Down Period)	The relocation of the molten core material resulted in a more coolable geometry. The upper core debris and lower plenum debris were likely cooled in a matter of tens of minutes after the relocation event. The consolidated core region became thermally and mechanically stable after the relocation event, but its complete cooldown could have taken weeks because of its large size, low thermal diffusivity, and continuing decay heat generation. Fission product release was terminated shortly after the relocation event and formation of the lower plenum debris. Examination of the lower plenum debris will provide information to assess the integral release up to the 224 min relocation.

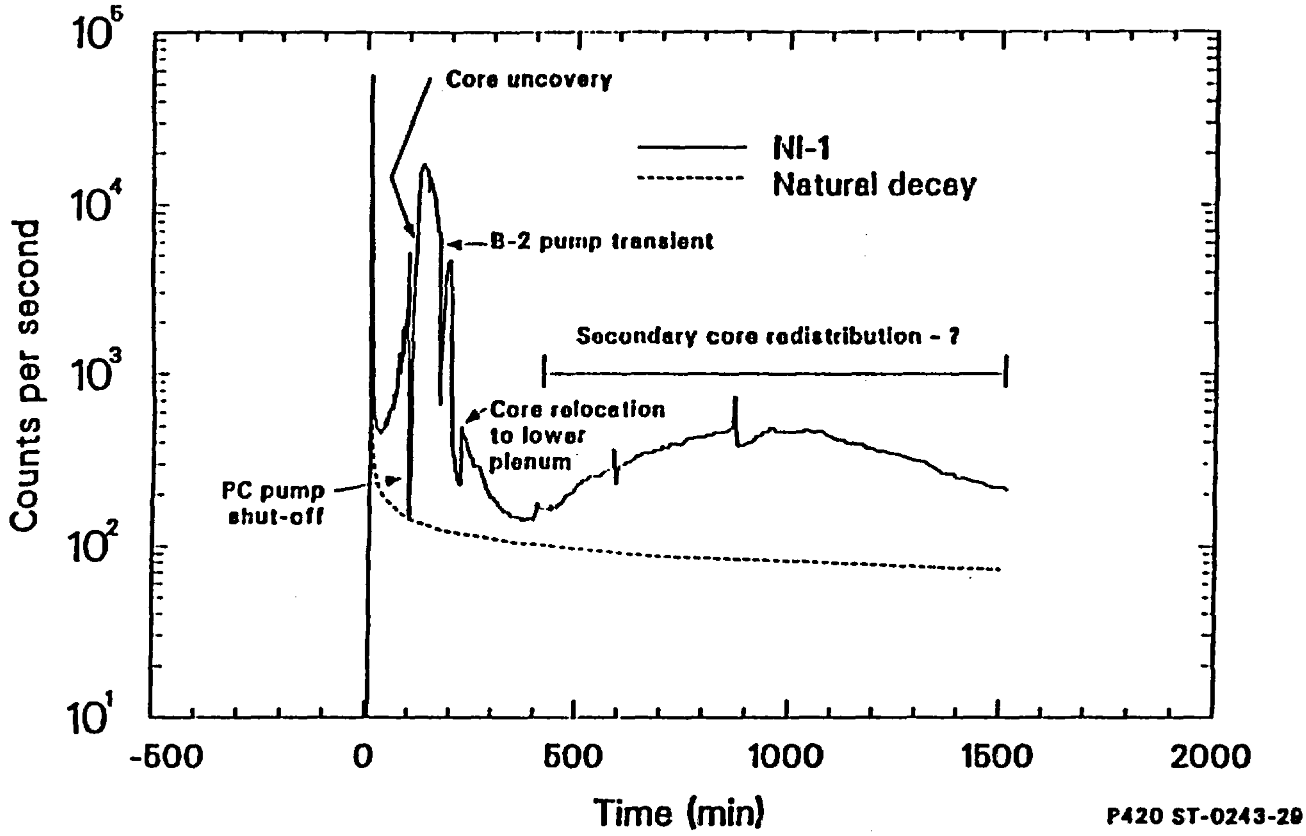

Figure 1. Source range monitor response during the TMI-2 accident.

response. Sensitivity calculations have confirmed that the limited decrease in SRM response at 174 min can be explained by core material relocation and is generally consistent with the core relocation as shown in Fig. 2.

The effect of core material in the lower head region of the reactor vessel was also investigated via two-dimensional neutronic calculations. The end-state degraded core configuration simulated in the neutronic model is shown in Fig. 3. Calculations indicate that relocation of between 10 metric tons of UO_2 and 80% of the core control rod materials is necessary to result in the observed SRM count rate. These calculations generally agree with the known mass of the lower plenum debris[9] and results confirm that a major core relocation occurred between 224-226 min as proposed in the accident scenario.

Additional SRM analysis is now underway to investigate the effect of core material in the core barrel assembly as described in Ref. 1. Also, the sensitivity of the SRM response due to differing configurations of the degraded core material in the lower plenum is being investigated. Possible explanations of the yet unexplained, long term response of the SRM between 400-1500 min are also being evaluated.

UPPER CORE DEBRIS COOLABILITY

The upper core debris study[10] was conducted to evaluate the coolability of the upper debris bed. The debris bed characteristics are summarized below:

- The debris bed axial height varied from 0.75 m to 1.25 m.
- The debris bed mass is estimated to contain from 20-25% of the core materials.
- The debris bed was heterogeneous, containing both Zr and UO_2.
- Approximately 90% of the particles ranged between 1 and 5 mm.

Based on the above data and sample examinations of the debris material[11], the upper core particle bed can be approximated by a right-circular cylinder which has a height of 0.9 m and a diameter of 2.8 m. The bed was modeled with an average particle diameter of 0.9 mm[a] and a porosity of 0.54. Between 3 and 5 hr after reactor scram, the power density in the debris bed is estimated to be about 0.75 MW/m^3. Using these parameters and assuming that all the heat generated in the bed was transferred upwards, the heat flux from the particle bed is compared to the dryout heat flux of the particle bed in Fig. 4. The Lipinski deep bed model[12] is used to calculate the dryout heat flux. Also shown in Fig. 4 is the heat flux from the total debris in the core region if all the heat generated in the debris was transferred upwards through the particle bed. For the particle bed only, the heat flux

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a. A 0.9 mm particle diameter results in the same effective debris surface area as estimated using the actual particle size distribution.

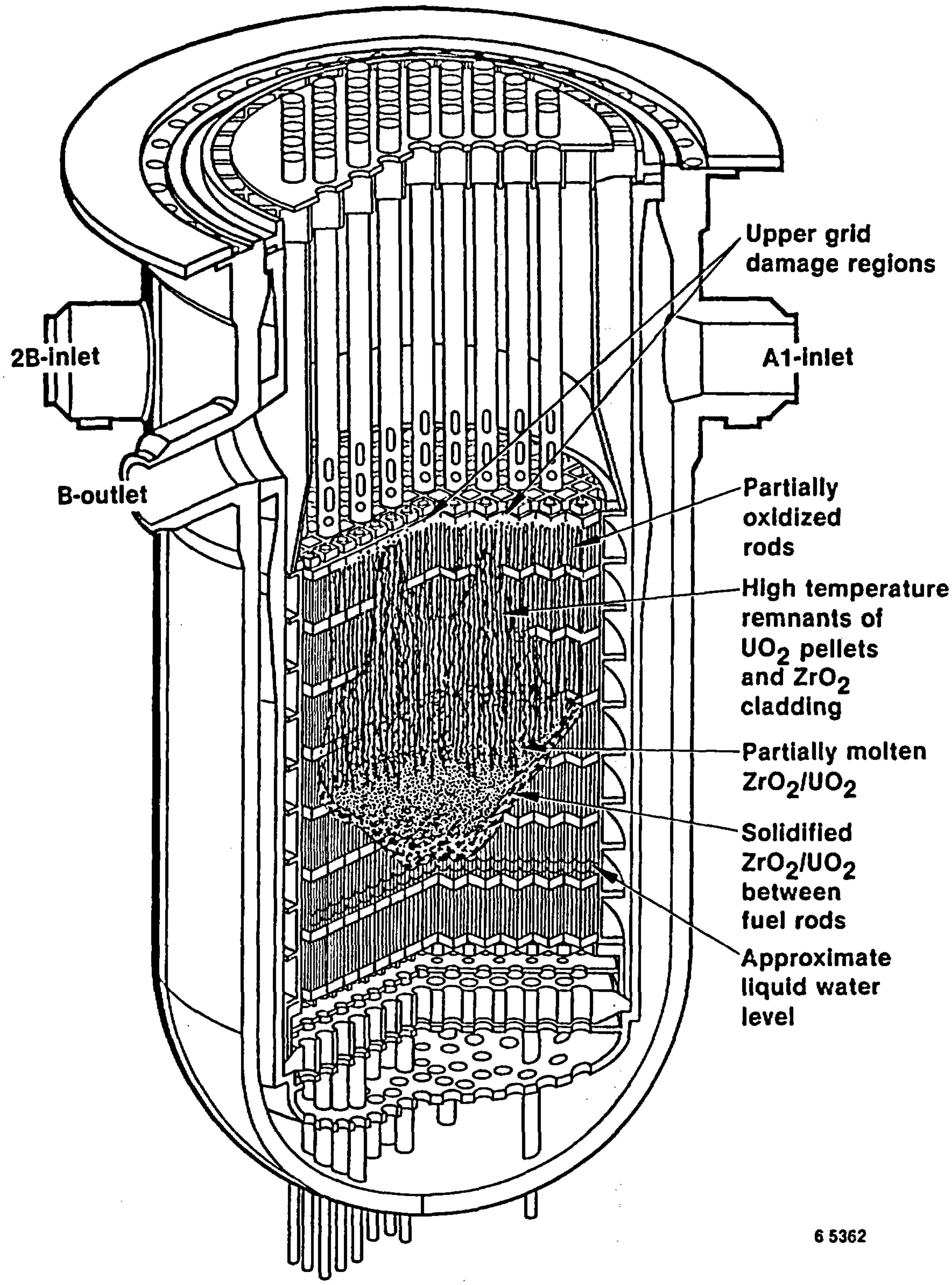


Figure 2. Hypothesized core configuration just prior to 174 m (pump transient).

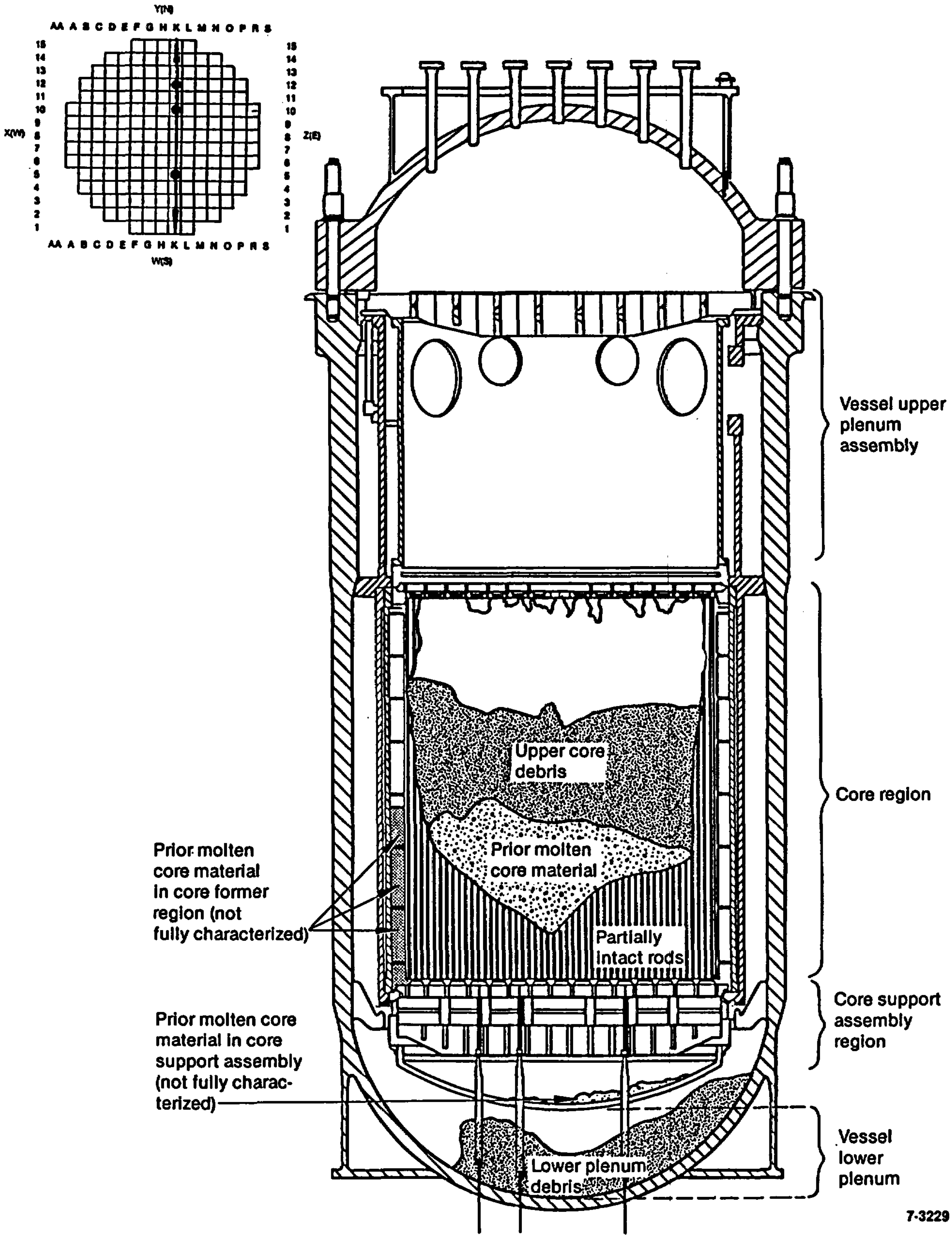


Figure 3. End-state core configuration.

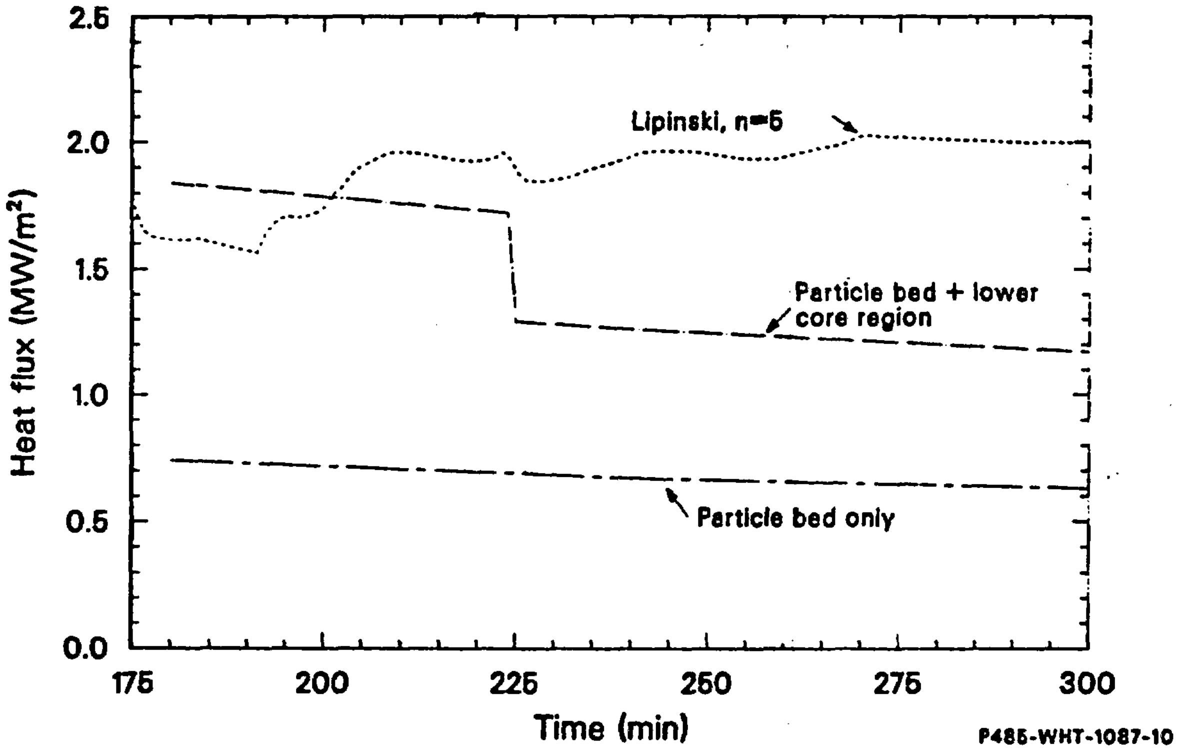


Figure 4. Upper core debris bed - comparison of predicted dryout heat flux to estimated actual bed heat flux for the upper core debris bed.

was much lower than the dryout heat flux and the particle bed was coolable in the presence of water. The debris bed heat flux (assuming most of the heat from the consolidated region was transfered upward into the debris bed) was comparable to the dryout heat flux before the relocation (174-224 min). Thus, the debris bed may not have been cooled appreciably during this time period. After the major core relocation at 224 min, however, the heat flux in the debris dropped due to the removal of fuel and the particle bed became coolable even if all the heat in the lower consolidated region was transferred upward through the particle bed.

Once the particle bed became coolable (i.e. the debris bed heat flux was less than the dryout heat flux), quenching took place in the presence of water. A simple energy balance formulation for the quench time was used to estimate the debris bed quench time. The energy balance model is shown in Fig. 5. It was assumed that heat transfer to the water at the quench front was the same as the dryout heat flux. Thus, the difference between the total dryout heat flux and the internal heat generation rate plus the heat transferred into the debris bed from below, lowered the temperature of the debris bed and led to quenching. Two estimates of the quenching time of the particle bed were carried out. The shortest time was associated with no heat transfer into the bed from below, i.e., only heat generation within the debris bed was considered. The longest time assumed 80% of the heat from the consolidated region was transferred into the upper debris bed. Assuming the emergency core cooling water flooded the core by 207 min and provided the source of cooling and the initial temperature of the debris bed was 2000 K, the earliest predicted quench time was about 18 min, putting the bed quenching time at around 225 min. The latest quench time was predicted to be about 38 min and would have resulted in a final quench time around around 245 min.

POSSIBLE CRUST FAILURE MECHANISMS

Identification of possible crust failure mechanisms is important because the mode of crust failure determines the extent and timing of the molten core relocation and the thermal challenge to the core support structures and reactor vessel. Three possible failure mechanisms were identified in preliminary work to assess crust failure mechanisms.[13,14] The first is melting of the crust. Calculations show that failure of the lower crust is not likely, because in the presence of water the crust thickness is on the order of several inches. However, upper crust melting is possible because of crust thinning due to the predominant upward convective heat transfer from the molten pool.

The second mechanism is structural failure of the crust due to thinning of the upper crust as the molten pool grew. The pressurizer relief valve was opened at approximately 220 min, approximately 4 min before the major core relocation event, causing the reactor system pressure to decrease by about 300 psi. The pressure reduction outside the molten core interior thus increased the pressure differential across the crust and may have led to failure of the crust.

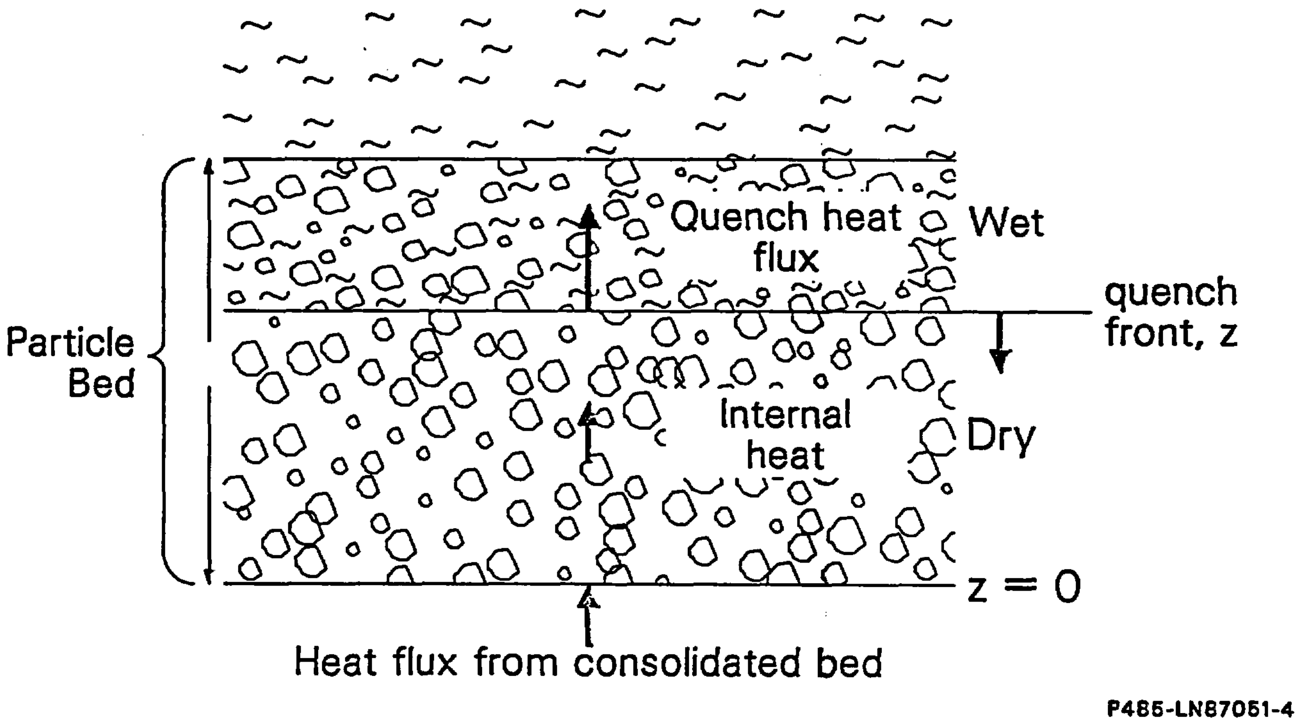


Figure 5. Upper core debris bed quench model.

The third possible mechanism is the potential interaction of the degraded core materials with the core barrel assembly at the core periphery. The degraded core region was skewed to the east side of the vessel and, as the degraded core heated up, it may have caused melting of the core barrel structures which chemically attacked the crust, resulting in crust failure.

Details of a hypothesized crust failure scenario[15] have been proposed based on the thermal properties of the degraded core material as shown in Fig. 6. Because of the ceramic properties of the consolidated material, the temperature profile across the consolidated region would be expected to be relatively constant, as shown in Fig. 6. Significant heat transfer into or out of the consolidated region would occur only at the thermal boundary layer adjacent to the molten region, δ_1, and at the upper coolant interface, δ_2, as shown in Fig. 6. The estimated thickness of these thermal boundary layers is only a few millimeters. Thus, little heat is transferred from the consolidated region, and the internal heat generation results primarily in melting the interior region and formation of a molten pool as shown in Fig. 7(a). As the molten core material regions grows, eventual interaction of the two thermal boundaries shown in Fig. 6 will occur.

Calculations indicate that for a molten pool of 1.25 m radius in equilibrium with the surrounding coolant, the equilibrium crust thickness at the outer surface would be approximately 8 mm. Previous estimates of the crust thickness necessary to support the upper debris bed mass indicate that a crust at least 2.5 cm thick is required. Experiments to measure heat flux variations in convective pools indicate that nonuniform heat fluxes would likely occur in the molten pool, resulting in thinner crusts at the top and at the periphery. These trends, together with a slight skewing of the degraded core region towards the east side of the vessel, are hypothesized to have led to localized failure of the crust near the core periphery as shown conceptually in Fig. 7(b). As the upper crust failed, the upper debris bed would fall into the molten pool, displacing the molten core materials from the core region as shown in Fig. 7(c). Estimates of the time it took to displace the molten core material were made by balancing the drag and gravity forces on the debris particles as they settled into the molten pool. The time required to displace the molten core material in the form of a liquid was calculated to be about 12 s. This is somewhat shorter than the maximum relocating time of about 1 min as inferred from the source range monitor data. The time difference can be explained by considering solidification of the molten ceramic in the interstices of the particles, which is calculated to have prolonged the settling time by an estimated 1 min.

MOLTEN FUEL COOLANT INTERACTION

An evaluation of the interaction of the molten core material with the water in the reactor vessel is also documented in Ref. 15. Breakup of the molten core material stream was analyzed in terms of the growth and detachment of unstable capillary waves or surface ripples on the outer surface of the molten stream or jet. The rate of stream breakup, via the surface wave instability theory, has a square-root dependence on the fluid density surrounding the jet. If the water along the path of jet movement was saturated, the fluid responsible for the breakup of the jet would have been

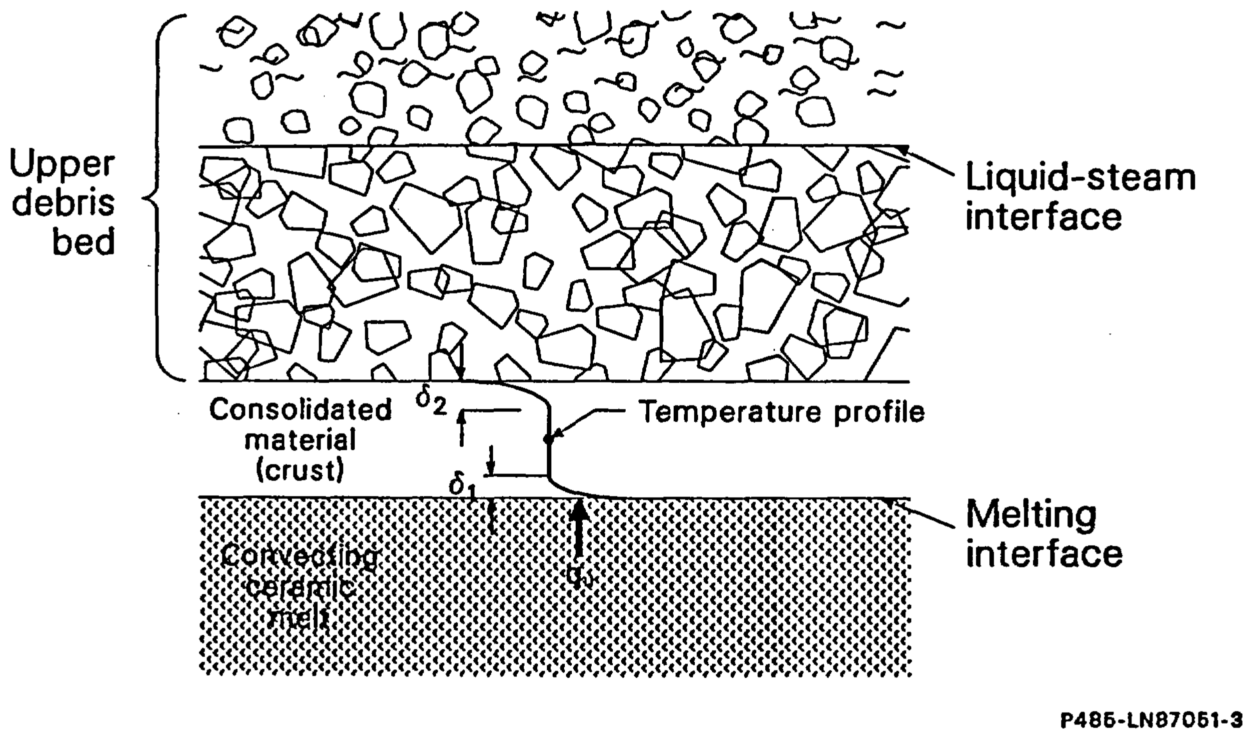


Figure 6. Degraded core thermal model.

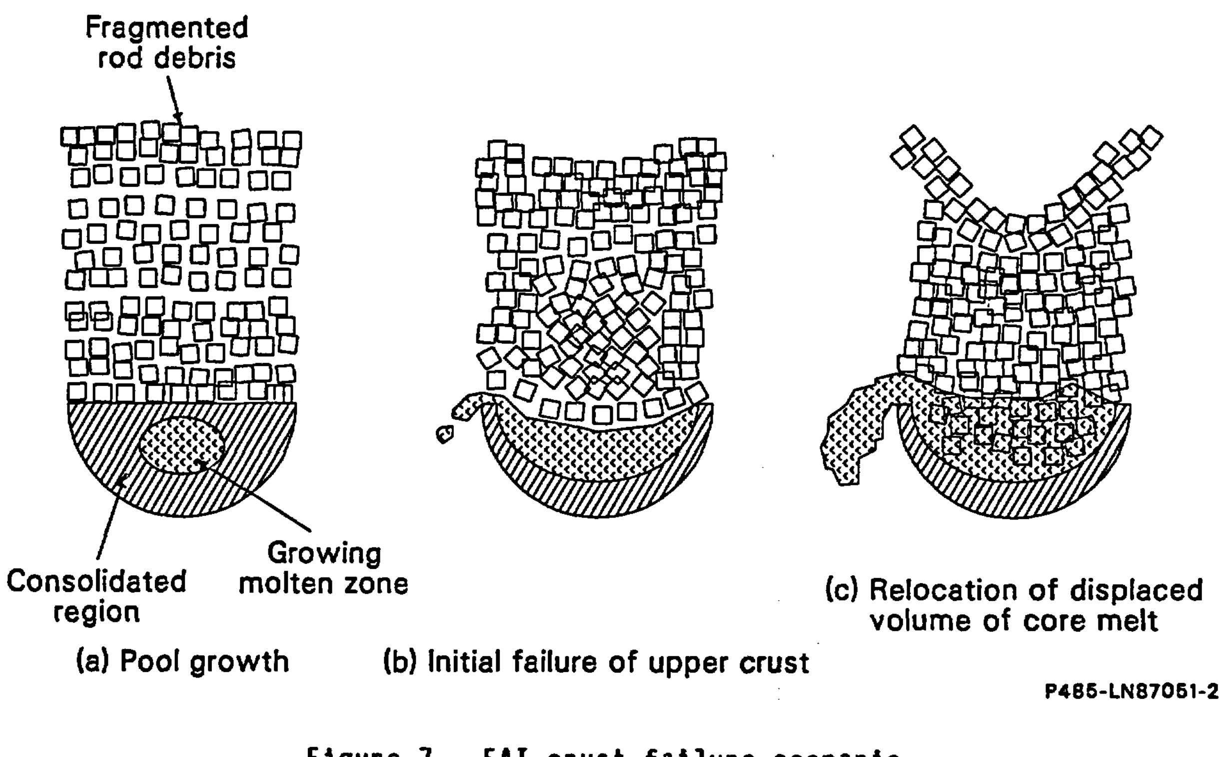


Figure 7. FAI crust failure scenario.

primarily steam, generated from film-boiling at the surface of the stream. For a stream velocity of 3.7 m/s, and a diameter of 0.08 m (based on an assumed relocation flow pathway of one fuel assembly and a relocation time of 1 min), it would require a distance of about 7 m for complete stream breakup in saturated water. The distance from the mid-core elevation to the bottom of the lower head is about 4 m. Therefore, complete breakup of the jet would not have been possible. In this case, the molten stream may have eroded the vessel head at the point of impingement.

If the water surrounding the jet was subcooled by about 80 K, the steam layer at the jet interface would have been thin, thus allowing interaction of the surrounding water with the jet surface resulting in jet breakup. Due to the square-root dependence of the breakup rate on the fluid density, breakup of the jet is estimated to occur over a traveling distance of about 2 m, which is about half the distance from the core mid-plane to the lower head.

Experiments in which molten core material was dropped into water pools,[16] also show that subcooled water results in particulate debris formation and limited steam generation compared to experiments with saturated water, in which much less molten stream breakup occurred and much higher steam generation was measured.

POTENTIAL VESSEL HEAD DAMAGE

Because a localized crust failure has been hypothesized and 15-20 metric tons of previously molten core material rests on the lower vessel head, two studies were undertaken to evaluate the potential damage to the lower reactor vessel head. The first study is an evaluation of potential localized damage as a result of a highly localized relocation stream.[17] Knowing the amount of lower plenum debris to be about 15 metric tons, and assuming a localized flow area for the relocation stream, a simple gravity flow calculation provides some insight into the flow pathways and timing as illustrated in Fig. 8. The relocation flow times are estimated to be 15 s and 75 s for an assumed flow area associated with the nominal flow area of 4 and 1 fuel assemblies, respectively.

Two cases were considered in evaluating the effect of the molten stream impingement on vessel. These cases are shown in Fig. 9. The first is a relative weak jet, in which the stream turbulence at the vessel wall interface is not maintained. For this case the molten material would freeze at the vessel interface and heat conduction into the vessel wall would be limited by thermal conduction through the frozen layer of core material. The second case assumes a more turbulent stream of core material, in which the stream turbulence at the vessel wall interface is maintained. For this case, the heat conduction from the molten stream is greatly enhanced since molten core material is assumed to be adjacent to the vessel during the relocation time.

For the conduction controlled or weak jet case, damage to the lower head is not predicted. However, for the strong jet case, where turbulence at the vessel interface is maintained, limited damage to the vessel wall may have

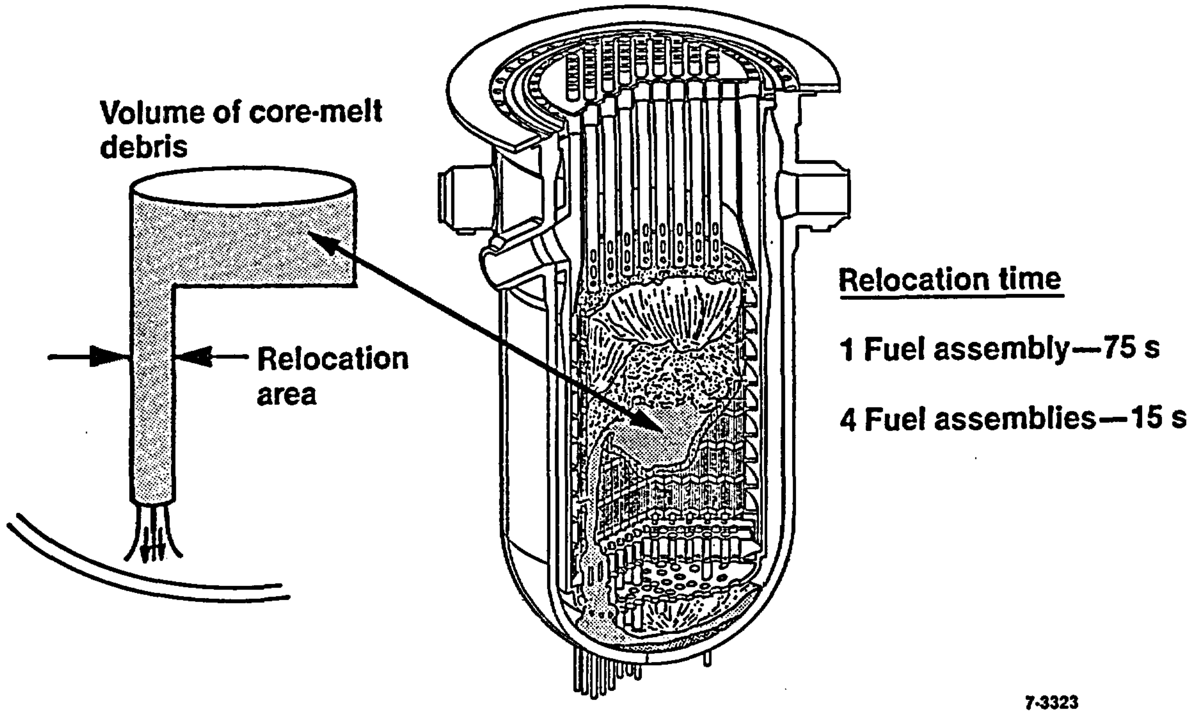


Figure 8. Model for estimating major core relocation timing.

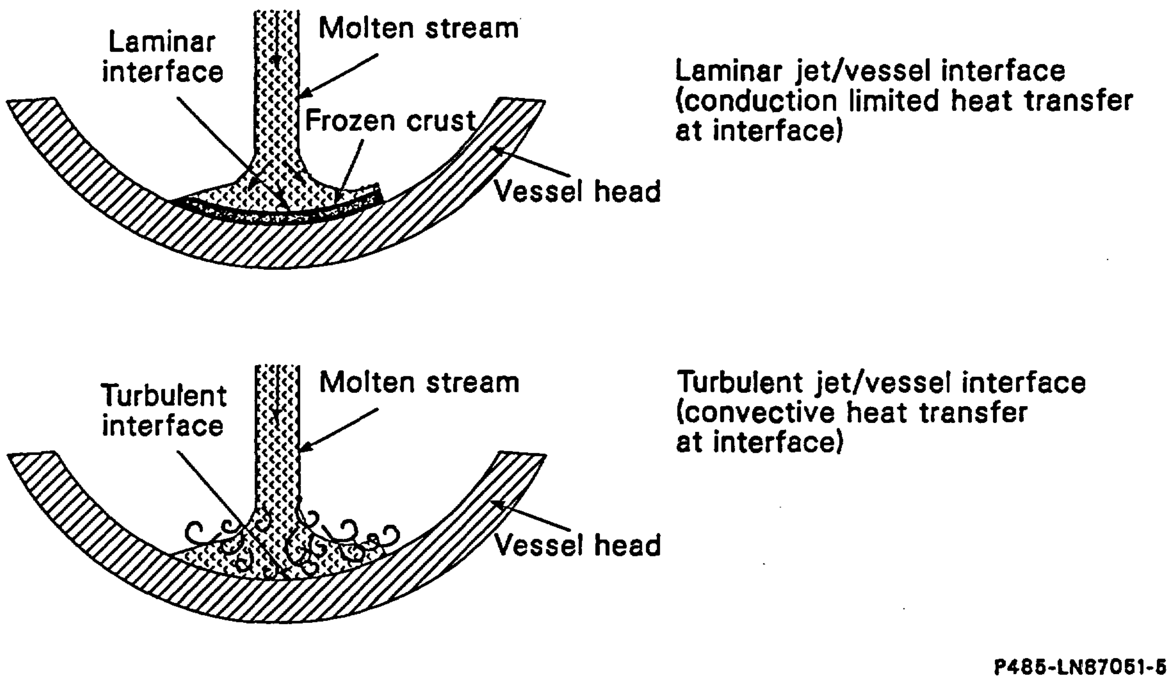


Figure 9. Model assumptions for localized core stream/vessel heat transfer.

occurred. Under these assumptions, limited surface ablation of the vessel liner is calculated. However, the melt front penetration of the vessel wall is estimated to be less than 1 cm. The calculations also indicate a direct jet impingement of 15-20 min is necessary to cause melting of half of the vessel wall thickness. The TMI-2 data clearly do not support relocation times greater than about 1 min.

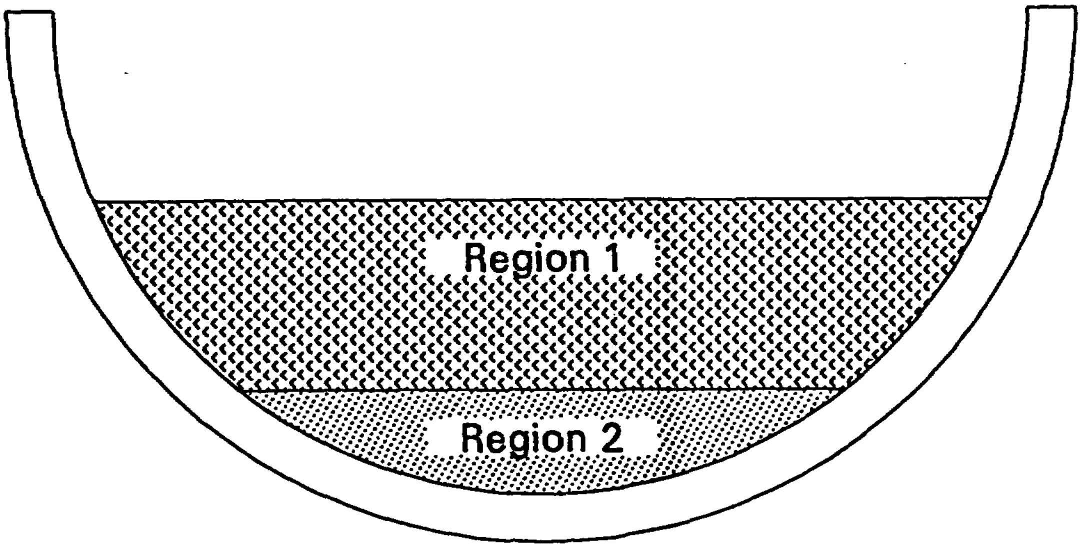
The second vessel study evaluated the global thermal response of the vessel assuming 15 metric tons of core debris on the vessel head.[18] A two-dimensional (radial, axial) heat conduction model of the TMI-2 lower plenum debris and reactor vessel was utilized to address the vessel margin-to-failure question. Because the debris configuration during the molten core relocation period is somewhat uncertain, two assumed debris configurations were analyzed. The first configureation assumed a porous debris bed resting on the vessel head. The second assumed a porous debris bed supported by a layer of previously molten but consolidated core material adjacent to the vessel head. For each of these cases, two assumptions on debris cooling were made, i.e., (a) no cooling of the debris material, and (b) heat transfer from the debris and consolidated material leading to quenching in a 20 min period. The general lower plenum debris and vessel head model is shown in Fig. 10.

The analyses show that the vessel thermal response is sensitive to both the debris configuration and cooling of the degraded core materials. For the consolidated material configuration, if the upper debris is not cooled, vessel melting is predicted to occur after several hours. However, at temperatures in the range of 1000-1100 K, creep rupture of the vessel becomes an important issue since the reactor system pressures were high (7-10 MPa). Thus, it is expected that vessel failure due to creep rupture would likely occur before vessel melting temperatures are achieved. If cooling of the porous debris on top of the consolidated material is assumed, melting of the vessel is not predicted. However, the vessel temperatures are also predicted to exceed 1100 K for this case. Thus, for the lower plenum configuration with consolidated material adjacent to the vessel, even with debris cooling, vessel creep rupture is an important issue.

For the case in which the lower plenum material is porous debris, vessel melting is not predicted; however, again vessel wall temperatures of 1100 K are predicted, indicating creep rupture of the vessel to be important. However, if cooling of the debris is assumed, vessel wall temperatures are estimated to be less than 800 K. For this case, mechanical challenge to the vessel would be insignificant.

SUMMARY

TMI-2 defueling data to characterize the core damage state and location of the degraded core materials, examination of the degraded core material from the TMI-2 core and lower plenum regions, interpretation of the TMI-2 on-line data recorded during the accident, and supporting analyses are providing a remarkably consistent interpretation of the core damage progression that occurred during the TMI-2 accident. This work has provided a baseline



Case 1: Region 1 - porous debris
Region 2 - consolidated material

Case 2: Region 1 - porous debris
Region 2 - porous debris

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Figure 10. Heat conduction model to evaluate vessel heatup.

accident scenario that defines the basic mechanisms that controlled the core damage progression and has provided a baseline from which to interpret the measured fission product distribution. The fission product behavior analysis work based on the core damage progression scenario developed to date is summarized in Ref. 19.

The analytical studies summarized in this paper have significantly improved our understanding of the mechanisms affecting the degraded core heatup, crust failure mechanism, interaction of molten core material with the reactor vessel coolant, and the potential thermal challenge to the reactor vessel. Further analytical work to be completed in the next year will add insight into the earlier phases of the core damage progression, particularly regarding the impact of core flow blockage on the core heat transfer and on hydrogen production. In addition, work is ongoing to establish a better understanding of the mechanisms leading to the damage of the upper grid structures. Work will also be necessary to interpret the most recent observation of very localized melt ablation of the lower fuel assembly grid plate in one of the centrally located fuel assemblies.

Completion of the core and lower vessel region defueling, examination of degraded core materials from these regions, and the necessary supporting analytical work to interpret the data, will complete our understanding of the accident and provide important data to assess more generic technical issues relative to core degradation and fission product behavior during severe accidents in light water reactors.

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