APPLICATION OF UNCERTAINTY ANALYSES WITH THE MAAP4 CODE

Uncertainty Working Group of the MAAP User's Group

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ABSTRACT

Uncertainty analyses are an important element associated with using integral computer codes to evaluate the response of a reactor/containment system to off-normal situations. The more severe the off-normal transient, the more important the uncertainty analyses. How should such uncertainty analyses be formulated? How should the results of the uncertainty approach be applied? To address these questions for the MAAP4 code, an approach has been developed to uncertainty evaluation defining the importance of individual physical process (Table 1) and establishing a structure on how phenomena should be evaluated and quantified with respect to the integral assessment.

Table 1
Preliminary Assessment Characterization of Severe Physical Phenomena

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

Documentation of the technical basis for uncertainty bounds is essential to meaningful uncertainty analyses. In particular, the technical basis for determining oxidation rates, cooling rates, combustion rates, etc. must come from a composite of separate effects and integral experiments, as well as industrial experience. How this technical basis is developed and how it should be used must be documented so that the user has a clear understanding what is, or is not, included in the technical basis for the phenomena of interest.

This paper will discuss the approach to developing the technical basis for uncertainty evaluations related to the phenomenon of RCS failure which includes the influence of natural circulation within the reactor coolant system. This discussion is an example of how relevant experiments and analyses must be documented to create the uncertainty bounds for each of the physical processes of interest. How these uncertainty bounds should be used in plant analyses will be discussed.

As addressed by the plant specific PSAs/IPEs, there is a low frequency, for which severe accidents could occur and the core debris would not be cooled within the vessel, i.e. the reactor vessel would fail and core debris would be released to the containment. Under these conditions, the objectives of accident management actions remain the same, i.e. cool the core debris by submerging it in water. However, the debris cooling rate has significant uncertainties under these conditions.

1.0 INTRODUCTION

Considering the list of phenomena in Table 1, it is clear that analyses related to severe accidents involve highly complex and interrelated physical processes. As a result, the evaluations require that the models used in the calculation be qualified with experimental results. These qualifications also characterize the uncertainty associated with the particular form of the model. Specifically, the uncertainties associated with a given model should be identified by comparing with the spectrum of experimental results available in the literature (Nagashima et al., 1995).

In this paper we will discuss how such a qualification can be established for a particular phenomena, namely the challenge to the RCS pressure boundary for a PWR with inverted U-tube steam generators given a severe accident situation. Note the evaluations of the natural circulation between the overheated core in the upper plenum and the potential for material creep in the hot legs, eventually leading to a failure condition. For these two different phenomena, the first represents the thermal source term to the RPV upper plenum, the hot legs and the steam generator tubes, the second determines the likely failure state under these conditions. Hence, this represents the types of phenomena that must be quantified with the spectrum of experiments and industry experience so that the uncertainties involved in the evaluations are clearly understood, in the context of a specific model formulation, and the method for identifying the spectrum of uncertainties to be evaluated are clearly identified and documented. Both of these are essential for performing meaningful uncertainty analyses.

2.0 CORE-TO-UPPER PLENUM NATURAL CIRCULATION

The fact that such natural circulation occurs for an open lattice PWR core was clearly illustrated by the experiments reported by Stewart et al. (1986). There is also substantial evidence (Vinjamuri et al., 1985) that upper plenum structures in the TMI-2 reactor were substantially overheated. With the complexities of the TMI-2 accident scenario, it is not clear whether this overheating is due to natural circulation flows, once-through flows, or a combination thereof. However, in the evaluation of natural circulation behavior, the TMI-2 experience is of particular importance, i.e. Bayless, et al. (1995) show two different control rod lead screws in the upper plenum. The temperatures near the top of the core were 1255 K in the central region and 1033 K near the core periphery. In the upper regions the temperature in the central region is estimated to have reached 666 K, and 723 K near the core periphery. Other experiences to be considered are the limited extent of melting of the fuel assembly upper grid as well as the upper fuel assembly and fittings (Russell and McCardell, 1989 and Marley et al., 1989).

With the observations from the TMI-2 post accident examinations, it is clear there was significant heating of the upper plenum, but very little melting of the structures immediately above the fuel region during the core degradation process. While the 2B pump was restarted approximately one hour after the core was uncovered, thereby cooling the reactor core and upper plenum structures, the temperature estimates and the observations from the upper grid plate can certainly be considered as representative of the conditions immediately before the pump was started. Thus, this is an important benchmark to assure that models for natural circulation between the core and the upper plenum are consistent with these observations regarding strength of the circulatory flows. For open lattice core configurations, like those used in PWRs, the natural circulation flows in MAAP4, illustrated in Fig. 1, include those from the core to the upper plenum, the upper plenum to the steam generator inlet plenum, and for the inverted U-tube steam generators, the circulation through the steam generator tubes that is outward through some of the tubes and return through the remainder. As the core overheats, the higher temperatures in the core central region mean a lower density, hence, the higher density regions in the colder peripheral zones push inward and accelerate the lighter fluid upwards. This natural circulation flow must be taken in context with the “purge flow” characteristic of the steaming from the covered regions of the reactor core, as well as possible flashing if the system is depressurizing. As the core temperature increases, oxidation of the zircalloy cladding by steam replaces the steam with hydrogen on a mole-for-mole basis. With the hydrogen’s lower density, the potential for circulation is further increased. This circulatory pattern remains valid as long as the core configuration remains unchanged.

As core heatup continues, the core geometry would begin to change due to melting and/or liquefication of the core constituents. Typically the melting of stainless steel is the first change in core material geometry, potentially liquefying some surrounding structures. Furthermore, at temperatures of 2030 K, the zircalloy metal which has not been reacted with steam can melt and liquefy some of the uranium dioxide fuel, and at temperatures approaching 2700°K, there are numerous potential eutectics. These temperatures are well within those observed in the TMI-2 accident and result in substantial changes in the core configuration. Due to these changes, the

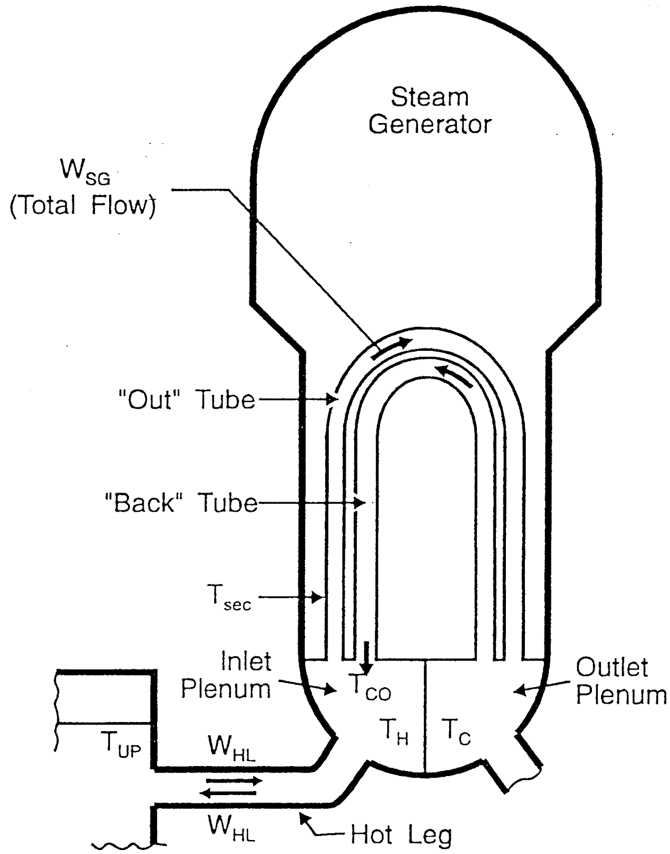

Figure 1, Natural circulation flows between the upper plenum and steam generator inlet plenum as well as the circulation flows through the steam generator tubes envisioned by the MAAP4 model.

effective frictional pressure drop increases through the core, and in cases where the molten material "bridges" between fuel pins, the local pressure drop is equal to the weight of the material. For configurations with a large cross-sectional area and substantial differences in the time at which this melting occurs, the flow is diverted around the melting region and local "bridging" occurs. This is viewed in a simplified way in MAAP by providing various types of degraded fuel pin geometry configurations, like those shown in Fig. 2; intact fuel types (Type 1), collapsed fuel pellets (Type 2), thickened fuel pins that contact as a result of relocation and freezing (Type 3), and an essentially blocked frozen region (Type 4). A fifth type is also used, characterizing a fully molten node. However, as these changes occur within given nodes, the frictional coefficient is not amplified to reflect the net result of surface deformation, bridging, etc. It is certainly true that the natural circulation flow, driven by relatively small density differences, cannot support the weight of a substantial mass of core material. Thus, if bridging occurs to any significant degree, the flow is potentially "shut-off" in the locale. This is discussed further in the section on quantification of uncertainties.

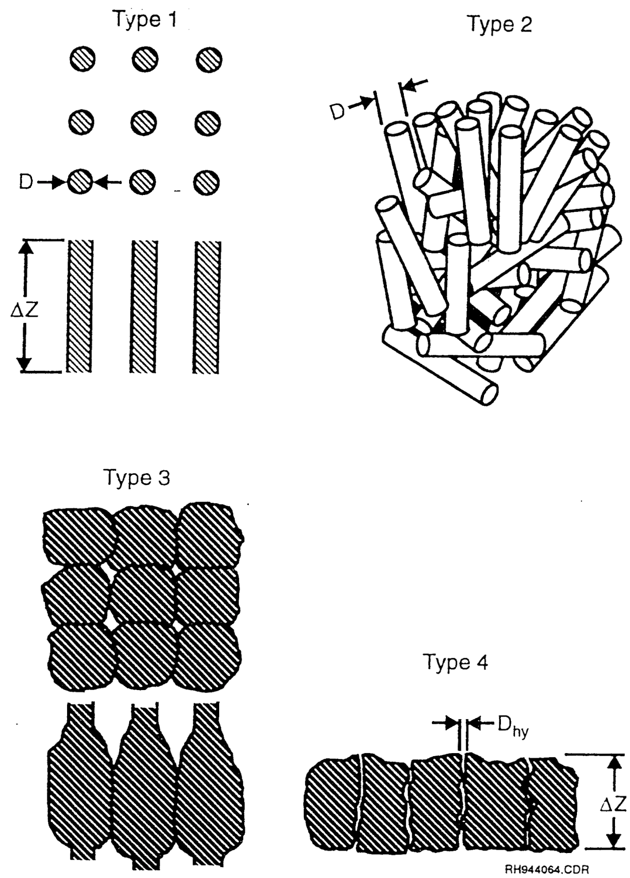

Figure 2, Types of fuel configurations considered by the MAAP4 code.

Since the core-to-upper plenum natural circulation flow provides the thermal source term for potential threats to other components of the RCS pressure boundary, this is one of the elements to be addressed in uncertainty evaluations. If the flow is strong, the transport of energy to the upper vessel regions and to the hot legs and steam generator tubes for PWRs can be large. On the other hand, if the circulation flow is limited as a result of the core degradation processes, challenge to these parts of the RCS boundary will be reduced, and potential reduced substantially. In this context, one aspect of the natural circulation process not evaluated by the integral codes at this time is the influence of high density gases and aerosols created as a part of the core degradation. Specifically, noble gases are released as a result of the core degradation and at the same time that maximum circulation flows occur. Furthermore, inert aerosols are created, from the vaporization of cadmium in those reactors with silver-indium-cadmium control material, from vaporization of materials in the steel, as well as other fission products such as cesium and iodine. These are all heavy components, created during the highest core temperatures, and have a substantial influence on suppressing the natural circulation behavior. Such uncertainty should also be accounted for in evaluating the challenge to the RCS pressure boundary.

3.0 RUPTURE OF THE RCS PRESSURE BOUNDARY

The challenge to the RCS pressure boundary is one of heating the various components to temperatures where they are not capable of sustaining the stress induces by internal pressure. The MAAP4 method of evaluating the structural capabilities in systems with substantial temperature gradients through the structural wall is outlined in the MAAP User's Manual (EPRI, 1994). It is clear from this evaluation that the temperature gradient through the RCS components must generally be accounted for when assessing the structural capabilities, i.e. unless the component has a small temperature gradient, using an average temperature can be misleading. Maile et al. (1990) performed a test on a full size hot leg (Figure 2-8) with the material described as 20 MnMoNi 55. Separate effects tests were performed to characterize the rupture time as a function of the imposed stress. A comparison of these results with those determined for the TMI-2 lower head vessel steel (Wolf et al., 1994) shows the materials are very similar in terms of material creep behavior.

In the test, a 0.7 m diameter hot leg with a 47 mm wall was pressurized to 16.3 MPa and heated externally in stages. As the average temperature of the hot leg approached 700°C, plastic deformation was observed and the heating was allowed to stabilize into a “holding phase”. There was a significant temperature difference across the pipe wall with the outer surface being hotter since the system was heated externally. During this time, the radial deformation increased exponentially and approximately 1100 seconds after the hold phase was initiated, the pipe ruptured.

4.0 QUANTIFICATION OF UNCERTAINTY TIES TO BE CONSIDERED

4.1 Core-To-Upper Plenum Natural Circulation

Given the MAAP representation for this natural circulation flow, consistent with the experiments performed by Stewart et al. (1986), the principle uncertainty associated with the circulation is the effective frictional pressure drop for both the radial (FFRICX) and axial (FFRICR) flows. Typically, MAAP analyses are performed with a nominal value of 0.25 for the radial friction factor and 0.1 for the axial friction factor. These are representative of the radial flows through pin bundle configurations, and the axial flows through a fuel pin array supported by grid spacers. As the core degrades, these friction factors will increase as a result of an increased effective surface roughness due to oxidation, the rupture of fuel pins and control rod materials, but principally due to the influence of molten material when the temperatures reach values where one or more of the core constituents become molten. At this juncture, the effective frictional coefficient would experience large changes, i.e. orders of magnitude. Because of the complexity of the system, perhaps the most informative means of investigating changes in the effective friction factor is examining integral calculations for the TMI-2 accident. This is one of the continual benchmarks for the MAAP code during its evolution to MAAP4 (Kenton et al., 1986; Sharon et al., 1989; and Paik et al., 1995). In particular, the MAAP4 code models recovery of the damaged core material, including the transport of molten material to the lower plenum and cooling of the core debris within the RPV lower head. Since interest here is the natural circulation flow, we will focus on the accident interval before 174 minutes (NSAC, 1980).

Figure 3 shows a comparison of the MAAP4 calculation of the TMI-2 accident sequence from the accident initiation until five hours into the accident sequence. With the core being uncovered approximately two hours and recovered approximately three hours into the accident sequence, our attention is focused on this one hour interval. The integral evaluations in Fig. 3 are performed for nominal values of the frictional coefficients as well as assuming a value of ten for both coefficients. Additional calculations for much higher values of the frictional coefficient demonstrated this value was sufficient to essentially shut off the natural circulation flow. Examining the comparisons shown in Fig. 3, it is apparent there is no significant influence on the primary system pressure, the mass of hydrogen created or the pressurizer level as a result of these changes. However, the flow between the core and upper plenum (WGUPCR) is reduced to a value about one third of that calculated using the nominal parameters. This has a significant influence on both the gas temperature in the upper plenum and the temperatures of the upper internals and the dome plate. As illustrated, the peak temperatures of the upper internals is decreased approximately 400°F (220°C).

Comparing the temperatures for the upper internals with those discussed from the TMI-2 lead screw evaluations shows these temperatures are significantly greater than the average temperatures observed. Furthermore, the gas temperatures calculated, using the nominal values for frictional coefficients, would have melted all of the structures immediately above the reactor core, which is not consistent with the post-accident examinations discussed earlier. The results for the high frictional value case are much closer to the observed behavior.

Another aspect of the calculated behavior for TMI that should be included is the Babcock & Wilcox design of holes in the core baffle plates providing water into the baffle region to cool the steel structures and act as a neutron shield and reflector. Typically, in the MAAP4 calculations, the flow is forced downward through the baffle region, inward through these holes and upward through the core. As a result, the strength of the circulatory flow can be limited by the area available for flow through the core former plates. In the evaluation for natural circulation flow uncertainties, the TMI-2 accident scenario can be examined assuming that the circulation is of the character typically evaluated for a Zion-like reactor system. In this case, the natural circulation flow is evaluated as downward through the outer core regions, radially inward and upward through the central core region. This flow path is activated by using the MAAP model parameter FNCBP = 0. A comparison of the influence for this model parameter and the nominal frictional coefficient is illustrated in Fig. 4. As illustrated, there is virtually no difference in the calculated system pressure, the hydrogen created, or the pressurizer level before the pumps start at 174 minutes. However, the circulation flows are even greater, as are the gas temperatures and upper plenum structural temperatures when the circulation is evaluated as it is for the Zion-like design. The character of this circulation is an uncertainty in the evaluation and is a likely combination of the two different paths considered. However, it can be concluded that if the circulation is downward through the outer core regions and upward through the central core, the TMI-2 experience clearly indicates that the effective frictional coefficients are much greater than the nominal values currently used in the MAAP4 evaluation that come from intact geometry studies such as that examined by Stewart et al. (1986).

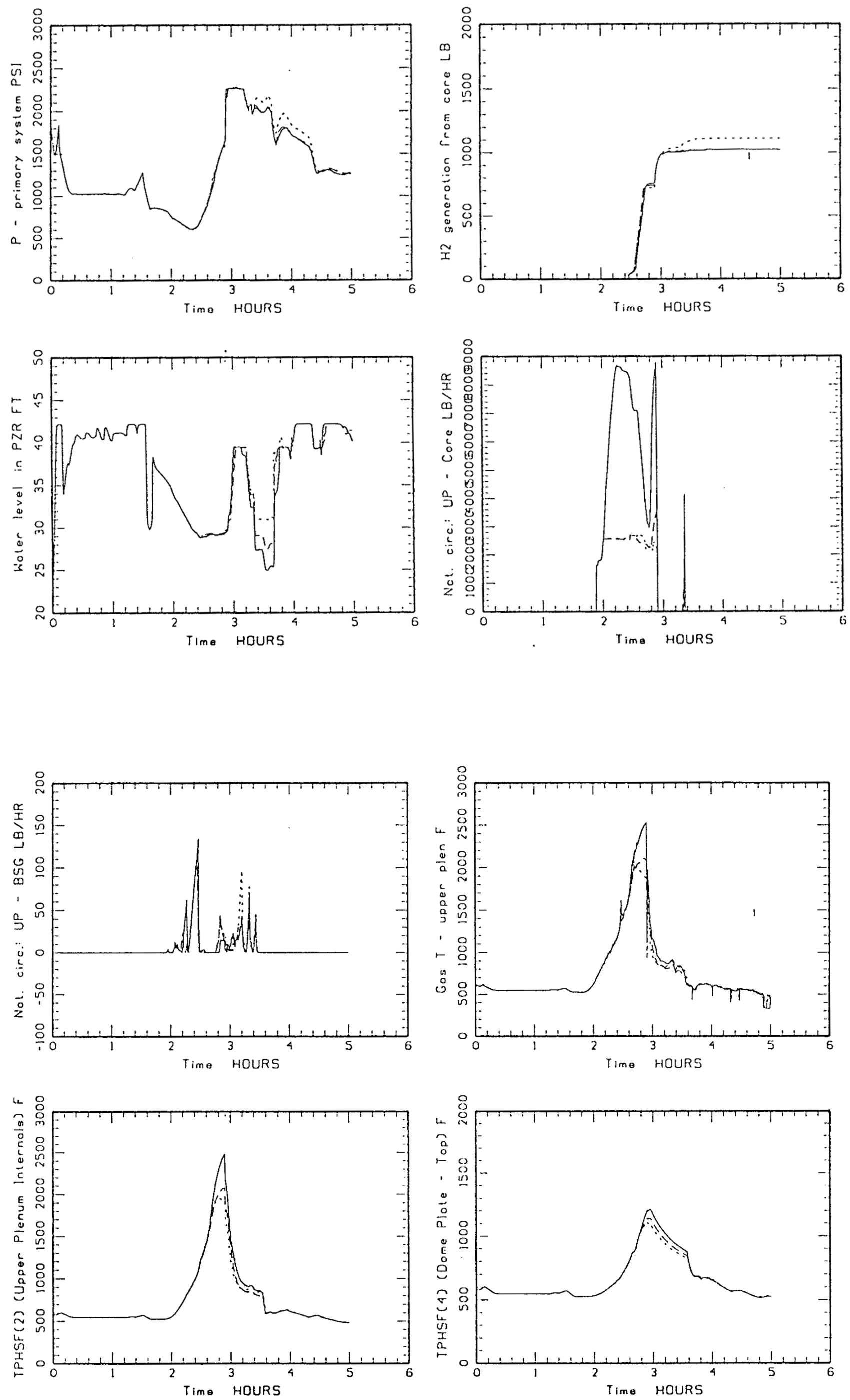

Figure 3, Examination of the influence of the frictional coefficient for natural circulation (FFRICX, FFRICR) on the MAAP4 calculation of the TMI-2 accident. The variations are: FFRICX = 0.45 with FFRICR = 0.1 (——), FFRICX = 10.0 with FFRICR = 10.0, and FFRICX = 1.E4 with FFRICR = 1.E4.

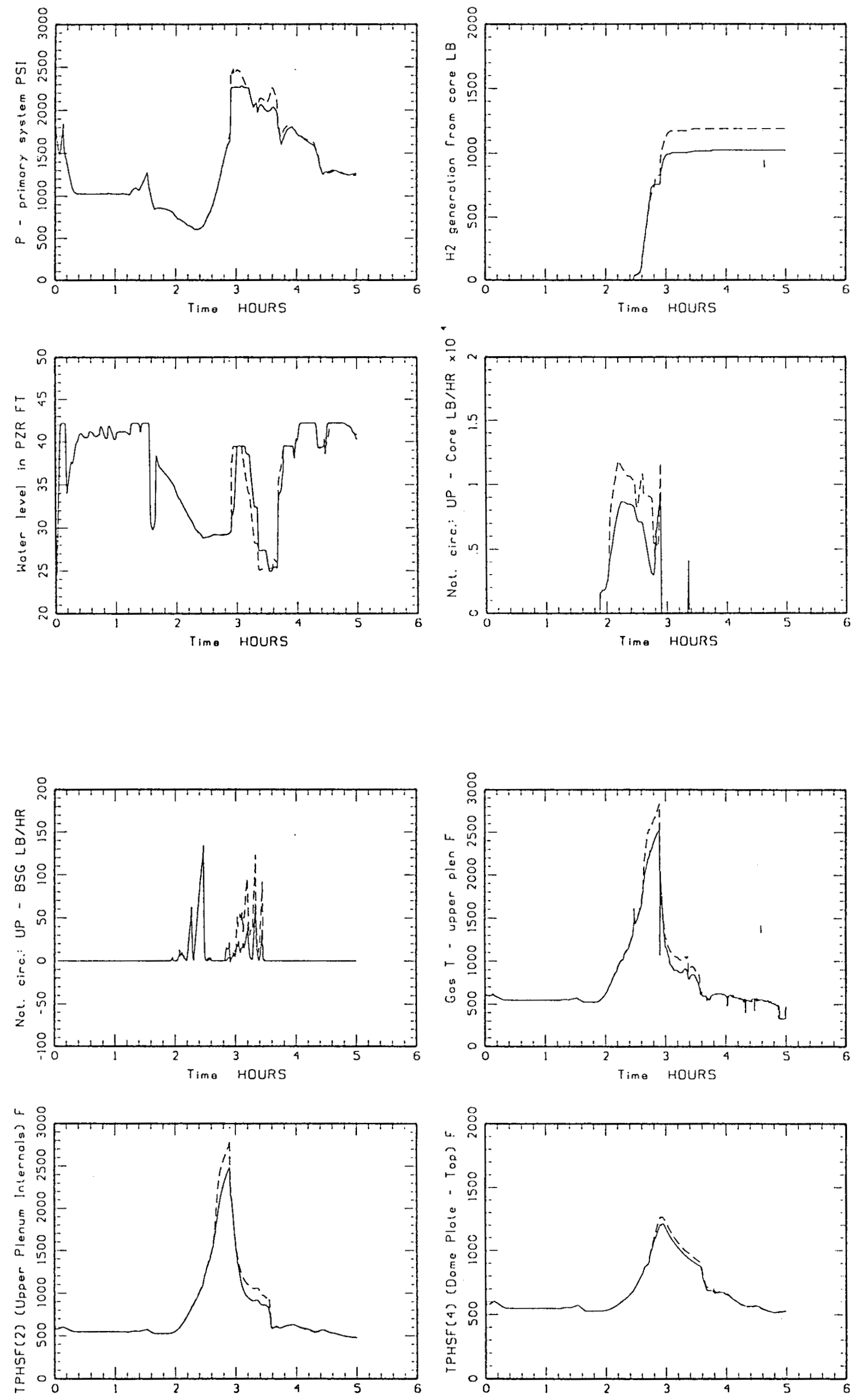

Figure 4, Influence of the different circulation path for the MAAP4 calculation of the TMI-2 accident. (FCNBP = 0 (X X X) forces the downward circulation to be through the baffle region and inward through the holes in the core former plates while FCNBP = 1 (——) enables downward circulation through the outer fuel assemblies and upward circulation through the middle.)

The TMI-2 accident experienced a release of most of the noble gases in the core and a substantial fraction of the iodine and cesium. Hence, the release of these heavy gases and vapors would tend to counteract the upward circulatory flows induced by heating of the steam and the production of hydrogen. Therefore, the MAAP4 comparison with the TMI-2 accident scenario, particularly with respect to the energy releases to the upper plenum region, is a most meaningful test in terms of the uncertainties to be assessed in the natural circulation flows.

One other element to be assessed (as part of the core melt progression that could influence the flow of energy into the upper plenum) is the parameter controlling the collapse of the core material when it stands at elevated temperature for a significant interval (LMCOL). Figure 5 shows the influence of variations in these parameters from the value used in the TMI-2 benchmarking for the MAAP 4.0 code (LMCOL = 53) to lower values of 50 and 45. Because of the logarithmic behavior of this parameter, this reduction in LMCOL results in a substantial "weakening" of the fuel material compared to that considered in the MAAP 4.0 benchmark. Since there is no data available for the incipient collapse of the fuel pins, there is no "right value" for this parameter, but substantial variations in the parameter need to be investigated. As illustrated in the comparison, the change in this parameter has no significant influence on the primary system pressure, but does somewhat influence the total hydrogen created; all parameters result in values generally consistent with the post-accident observations. This figure also shows some decrease in the gas and upper plenum internal temperatures as this parameter is decreased, but this is less influential than the effective friction for the circulatory flows.

Given the above discussions, evaluations for challenges to the RCS pressure boundary should be assessed using the nominal values for the frictional coefficients as well as values which substantially limit the circulation flows when temperatures elevate. To achieve this latter case in a simplified manner, the natural circulation flows should be examined with frictional coefficients at least as high as ten. Examining comparative results between the nominal values and those equal to ten should provide the necessary insights into the influences of such circulatory flows on the potential challenges to the RCS pressure boundary.

4.2 Quantification of the Rupture Potential of the RCS Pressure Boundary

The uncertainties related to the possible rupture of the hot leg are the thermal source term (natural circulation from the core to the upper plenum), the material properties and the strain-to-rupture for the material in question. Note that there are potentially different materials in the reactor vessel nozzle, possibly carbon steel, and the hot leg, which is generally stainless steel for Zion-like designs. Therefore, the evaluations need to consider the possibility of the carbon steel nozzle as the initial failure site, and that the shape of the nozzle (thicker wall) could prevent its failure such that the stainless steel hot leg is the failure location. These effects are typically larger than the variation in properties between materials of a similar type.

The extent of creep necessary to result in rupture was illustrated in a full size hot leg experiment reported by Maile et al. (1990). Hence, once the significant plastic deformation is initiated, there are only a few minutes until failure. Also, since the data is characterized as the time to rupture with a given stress, the extent of strain does not enter the MAAP4 hot leg

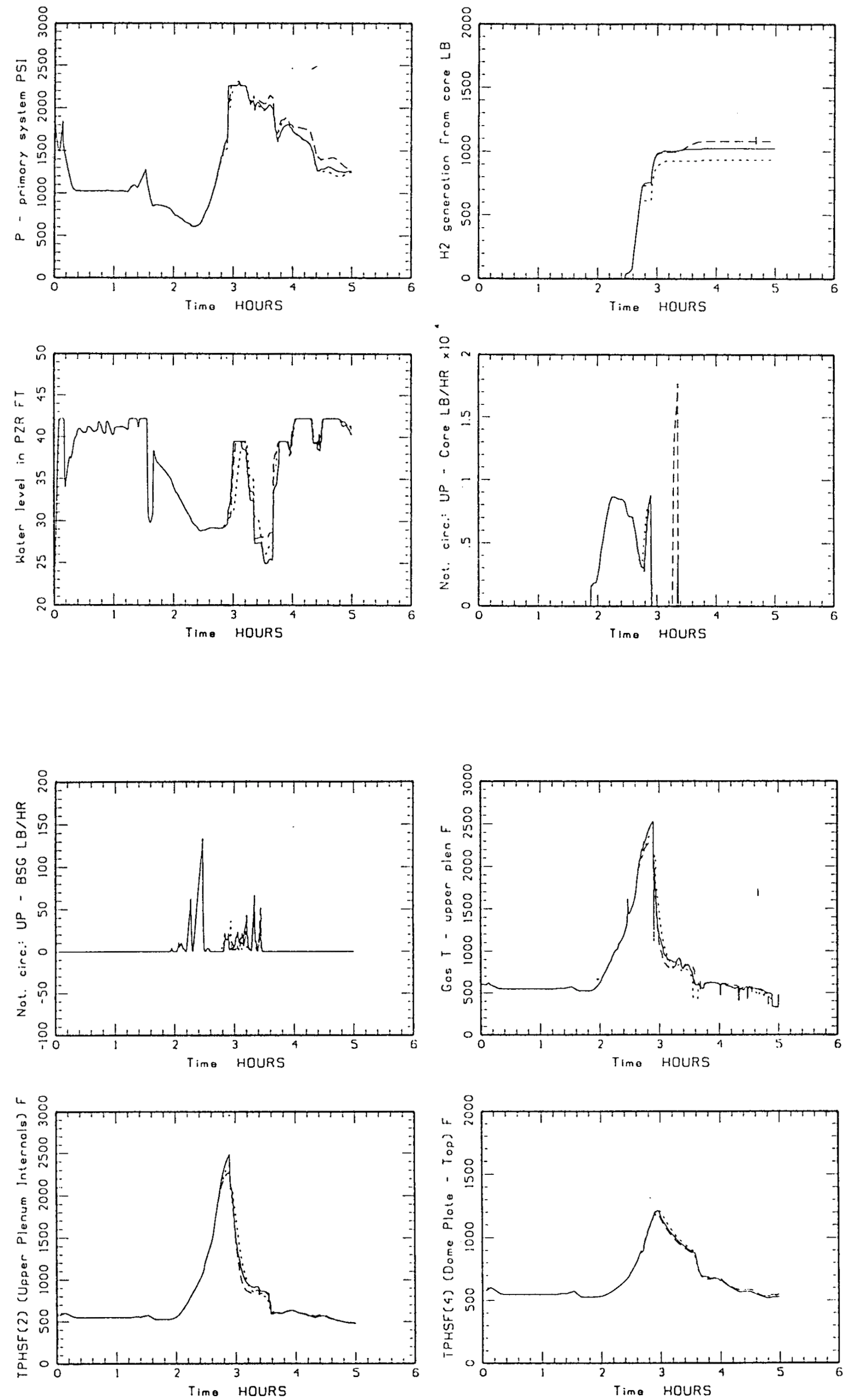

Figure 5, The influence of the core collapse parameter LMCOL on the MAAP4 calculation for the TMI-2 accident. The variations in LMCOL are: 53 (——), 50 (X X X) and 45 (-------).

calculation. However, the structural properties of carbon steel vs. stainless steel do enter this calculation. Hence, the influence of challenges to the hot leg should be evaluated using (a) the structural properties of carbon steel and (b) the structural properties of stainless steel for material creep.

Lastly, the strength of the steam generator tubes should be considered as an uncertainty to be evaluated. At one end of the spectrum, the strength of full thickness tubes should be evaluated along with the hot leg to determine if the thermal transient results in challenges to the tube integrity before hot leg failure. In addition, steam generator tube degradation can occur with aging. The information on the degradation of tube strength for Inconel 600 material has been documented (Vagins et al., 1979), observing that a substantial degradation of the tube wall was required before the steam generator tubes could not withstand the internal pressure. Since this degradation is specific to given operating histories, age, etc., we will consider a simplified view in which the tube wall is uniformly thinned by 50%. This is a large degradation, one easily determined from the standard steam generator tube surveillance activities typical of industry practices. If this large degradation does not result in a significant challenge to the steam generator tubes in comparison to the hot legs, then the current degradation management activities are sufficient to limit the uncertainties to values less than those causing a challenge to the tube integrity. Thus, the calculations are performed considering full tube thickness and one half the nominal thickness.

5.0 INTEGRAL ANALYSES TO CHARACTERIZE THE INFLUENCE OF UNCERTAINTIES

Analyses were performed for a Zion-like Westinghouse 4-loop plant for conditions in which the natural circulation behavior would be the strongest, as would the potential threat to RCS pressure boundary, namely station blackout-like conditions in which the RCS pressure is at the PORV relief or safety valve setpoint pressures. Since both the energy transport by natural circulation and the stress on the RCS structures is maximum at the higher pressures, this is the accident sequence that will be used to investigate the influences of uncertainties. In particular, the coefficients for turbulent flow through the core region are incrementally increased from the nominal values to 0.5, 1 and 10. The results for these incremental increases are illustrated in Fig. 6. As expected, this has no influence on the primary system pressure and only a limited influence on the extent of hydrogen created. However, the circulation flow from the core to the upper plenum is substantially reduced as are the temperatures of the upper plenum internals. All of these calculations eventually result in creep rupture of the hot leg and a rapid depressurization of the reactor coolant system. Consequently, the overall perspective of the accident sequence was not changed even though substantially less energy was connected to the upper plenum. In this regard, the hot leg was found to be substantially weaker than the steam generator tubes since the cooling by the high density gases on the secondary side resulted in a temperature increase substantially less than that observed by the hot leg. Consequently, as seen by these calculations, the types of effective increases for the frictional coefficients in the convective flows, which provide the best agreement with the TMI-2 observations, still result in creep rupture of the hot leg but do not challenge the steam generator tube integrity. This is further supported by the information shown in Fig. 7 in which the steam generator tube thickness is one half the nominal value. As observed from the integral calculations, the behavior is identical to that calculated

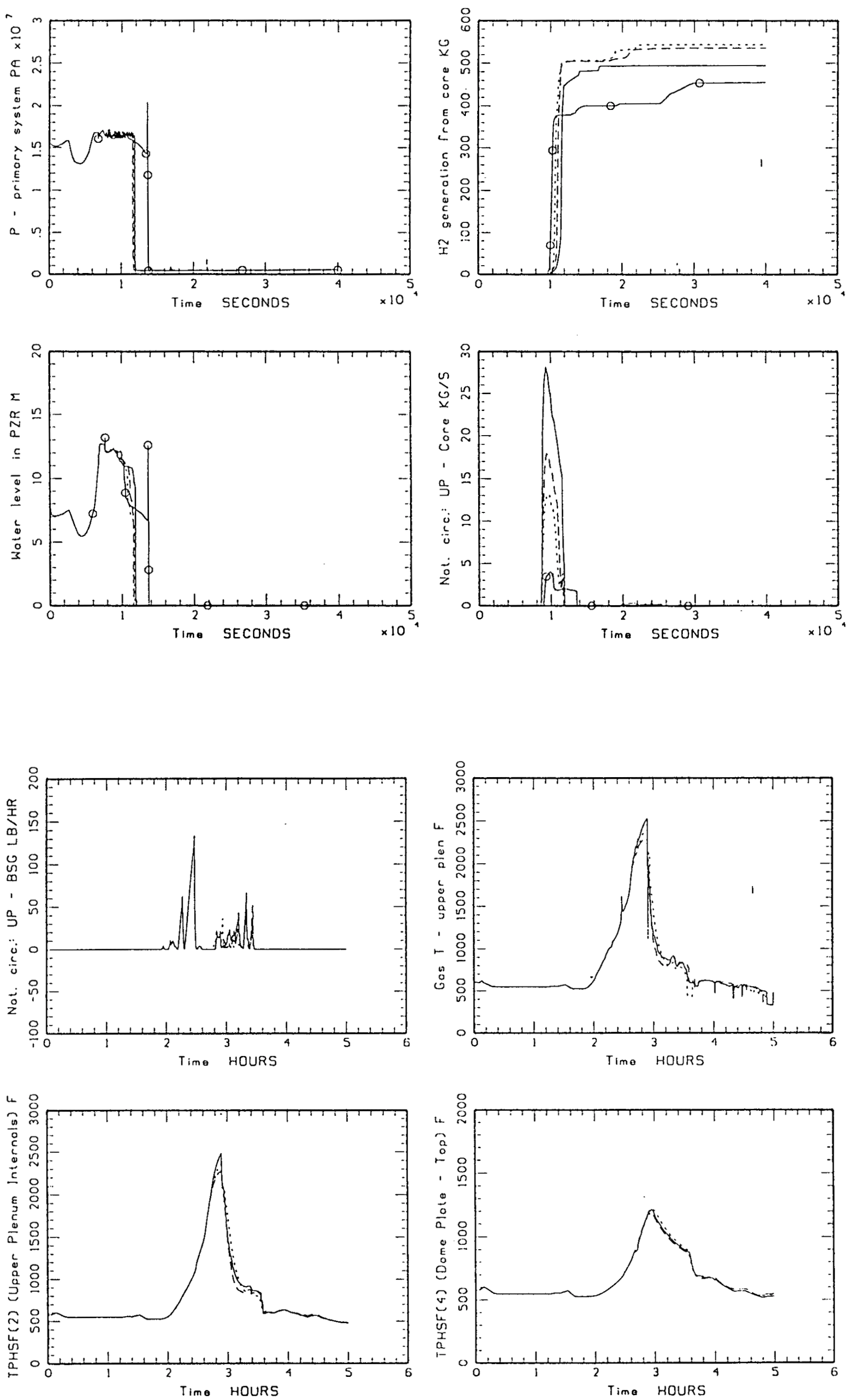

Figure 6, Sequence designator RCS/W/Z//FFRIC//TMLB HL Creep, Seal LOCA with variations in the frictional coefficients for radial (FFRICX) and axial (FFRICR) flow of FFRICX = 0.25 with FFRICR = 0.1 (———), FFRICX = 0.5 with FFRICR = 0.5 (X X X), FFRICX = 1.0 with FFRICR = 1.0 (-------) and FFRICX = 10.0 with FFRICR = 10.0 (—θ—).

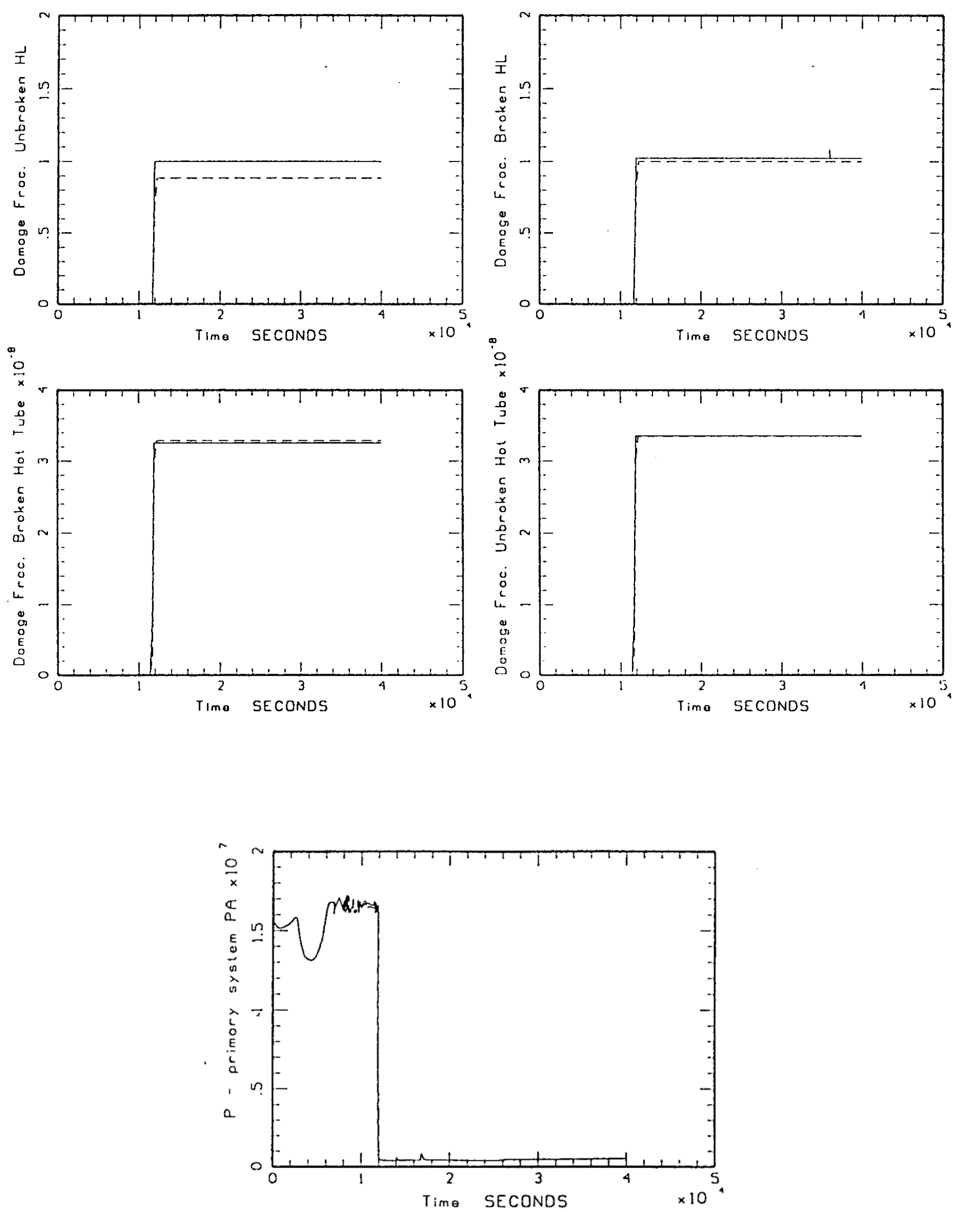

Figure 7, Sequence designator RCS/W/Z//XTSG//TMLB HL Creep, Seal LOCA with variations in the thickness of the steam generator tubes (XTSG) as a surrogate for tube degradation of 0.00333 m (——) and 0.00167 m (X X X).

when a full tube thickness was used. Thus, the challenge to the RCS pressure boundary under severe accident conditions is not substantially increased when substantial tube degradation is experienced. This is not to imply that tube degradation and tube failure do not occur, but that the tubes are essentially as strong under accident conditions as they were under normal operation, and the threat to consequential tube rupture due to the accident condition is not significant.

6.0 CONCLUSIONS

The challenges to the RCS pressure boundary are principally due to natural circulation flows from an overheated reactor core. Typically, this is a PWR issue as a result of the larger flows that could be convected to the RCS pressure boundary due to the substantial flows through the hot legs and the inverted U-tube steam generators. These substantial convective flows potentially challenge the hot leg integrity and perhaps the integrity of the steam generator tubes. Integral analyses of the TMI-2 accident show that the natural circulation flows from the core to the upper plenum apparently are limited over those calculated assuming frictional coefficients typical of an intact configuration for an open lattice core. This limitation to the circulation process is evidently due to the increased frictional coefficients associated with core degradation, including melting of the core constituents and their downward relocation. Considering the uncertainties associated with the frictional coefficients, the challenges to the RCS pressure boundary appear to be thermal challenges to the hot leg integrity and not the steam generator tubes (see Table 2 for a summary of the results). This is true even when the steam generator tube thickness is reduced to 50% of the nominal value. To represent the degradation of tubes with age, experimental information on full size hot legs undergoing a substantial thermal transient to approximately 700°C were compared with the MAAP analysis; the MAAP model was found to be in good agreement with the experimental results, hence, a challenge to hot leg integrity of the RCS pressure remaining at a value close to the PORV setpoint seems a likely event.

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Table 2
Summary of Integral Analyses of RCS Failure Modes
Matrix of Analyses for RCS Failure Modes (RCS)
Code: MAAP 4.0.2+

Physical Behavior Investigated	Model Parameter Varied & Magnitude (Other Parameters of Interest)	Accident Sequence Used	Phenomenon Ruled Out? Why?	Identifier for Analyses	Summary of Results
Friction factors controlling for the core-to-upper plenum natural circulation flow.	FFRICX & FFRICR (nominal values FFRICX = 0.1 and FFRICR = 0.25) • FFRICX = 0.1 FFRICR = 0.45 • FFRICX, FFRICR = 10 • FFRICX, FFRICR = 1.E4	TMI-2 accident evaluation with particular focus on the calculated upper plenum thermal response.	None but both circulation flow paths were exercised, i.e. downward circulation in the baffle region and upward through the core and also downward through the outer core assemblies and upward through the central assemblies. Of these, the latter had the greatest circulation and the highest upper plenum temperatures.	RCS/TMI-2// FFRIC// TMI-2	Circulation flows calculated using large value of the friction factor provide results which are much closer to the TMI-2 behavior than those in which nominal values are utilized for smooth tube type to flow.
	FFRICX & FFRICR (nominal values FFRICX = 0.1 and FFRICR = 0.25) • FFRICX = 0.1 FFRICR = 0.25 • FFRICX, FFRICR = 0.5 • FFRICX, FFRICR = 10	PWR/station blackout (TMLB without recovery actions).	None.	RCS/W/Z// FFRIC// TMLB HL Creep	Increased frictional coefficients for the natural circulation flows through the core did not substantially change the accident sequence. Specifically, hot leg creep rupture still occurred and there was no significant challenge to the steam generator tube integrity.
Core-to-upper plenum natural circulation flow path	FNCBP (nominal value FNCBP = 0.0) • FNCBP = 0 • FNCBP = 1	TMI-2 accident evaluation with particular focus on the calculated upper plenum thermal response.	None.	RCS/TMI-2// FNCBP// TMI-2	If the circulation is downward through the outer core region and upward through the central core, the TMI-2 experiment clearly indicates that the effective frictional coefficients are much greater than the nominal values currently used in the MAAP4 evaluation.
Influence of core collapse on the flow of energy to the upper plenum.	LMCOL (nominal value LMCOL = 53.0) • LMCOL = 53 • LMCOL = 50 • LMCOL = 45	TMI-2 accident evaluation with particular focus on the calculated upper plenum thermal response.	None.	RCS/TMI-2// LMCOL// TMI-2	Decreasing the core collapse parameter somewhat decreases the gas and upper plenum internal temperatures, but this influence is less than that of the effective friction for the circulatory flows.
Steam generator tube degradation.	XTSG (nominal values XTSG = 0.00333 m) • XTSG = 0.00333 m • XTSG = 0.00167 m	PWR/station blackout (TMLB without recovery actions).	None.	RCS/W/Z// XTSG// TMLB HL Creep	The challenges to the RCS pressure boundary under severe accident conditions is not substantially increased when substantial tube degradation is experienced.

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