NUREG/CR-1518
SAND80-0701
R-7

ASSESSMENT OF CORE PENETRATION OF A PWR REACTOR VESSEL AND PARTICULATE DEBRIS COOLABILITY IN TMLB', S2D, AND ABG ACCIDENTS

Ronald J. Lipinski

Contributions By:

P. B. Bleiweis**

J. Tills***

Manuscript submitted: March 1980
Date Published: July 1980

Sandia National Laboratories
Albuquerque, New Mexico 87185
operated by
Sandia Corporation
for the
U. S. Department of Energy

Prepared for
Division of Reactor Safety Research
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, DC 20555
Under Memorandum of Understanding DOE 40-550-75
NRC FIN No. A1054

**Science Applications, Incorporated, Albuquerque, NM
***Civil Systems, Incorporated, Albuquerque, NM

8008270222

ABSTRACT

A brief analysis of events surrounding a PWR reactor vessel failure following a core meltdown was performed. The purpose of the analysis was to assess the impact of such events on a containment building filtered vent. Specific accidents considered included a loss of AC power and auxiliary feedwater (TMLB'), a small-break LOCA with ECCS failure (S2D) and a large-break LOCA with failure of the containment heat removal system (ABG). The MARCH computer code[1] analysis of these accidents (with respect to the Indian Point 3 and Zion reactors) was used as a basis for comparison. The major findings are as follows:

1. The location and size of a vessel rupture in the TMLB' accident could significantly affect the pressure history in containment and the subsequent loading on the filtered vent.

2. High internal reactor pressure (from rapid debris slumping into the lower head water) could cause steam generator tube failure and thus failure of secondary containment. A similar failure could be caused by thermal shock if cold feedwater enters a dry steam generator after AC power restoration.

3. A significant containment building pressure rise could occur from molten debris dropping into the reactor cavity if there is adequate water in the cavity for complete quenching. The TMLB' and ABG accidents nominally do not have sufficient water in the cavity for complete quenching; however, other accident sequences may.

4. The coolability of total-core in-vessel or ex-vessel particle beds by natural circulation (assuming an adequate coolant supply) can neither be assured nor excluded at this time. However, current data and models suggest that both in-vessel and ex-vessel beds may be coolable in some cases.

Suggested research to resolve uncertainties in the above items is discussed.

2

CONTENTS

1.0 INTRODUCTION

In a general PWR meltdown with failure of the primary vessel or piping, the release of radioactive materials is prevented by the containment building. The prevention of uncontrolled leaks in the building in such accidents is important. One possible cause of leakage is overpressurization of the building from hydrogen generation (via zircalloy and steel oxidation by steam) and from steam generation (via flashing of primary water or boiling of emergency coolant). A vent which filters radioactive materials could relieve the pressure buildup and prevent an uncontrolled leak. However, since a vent would have a limited release capability, a rapid generation of steam (e.g., from a sudden mixture of hot debris with water) could build up pressures even with a vent.

The MARCH computer code[1] calculates the progress of various core-meltdown accidents and the subsequent pressurization of containment. An important stage in that calculation is reactor vessel failure after core melt. At that time there is the possibility of rapid steam release from within the reactor vessel and rapid steam generation from contact of molten core materials with accumulator water and reactor cavity sump water. It is the purpose of this study to consider the events around the time of primary vessel failure with particular attention to steam generation, the appropriateness of the MARCH modeling assumptions, and the possible impact on filtered vent designs. The accidents which will be considered are a loss of offsite and onsite AC electrical power with failure of restoration for one to three hours along with a failure of all auxiliary feedwater (TMLB'), a small-break loss of coolant with failure of the active emergency coolant injection system (but not the passive accumulators) (S2D), and a large-break loss of coolant with failure of electrical power to the engineered safety systems and failure of the containment heat removal system (ABG). In addition, the consequences of delayed power restoration at various times in the TMLB' accident will be investigated since it is reasonably likely and could affect the load on containment. These accidents will be considered primarily with respect to the Indian Point 3 reactor with additional reference to the Indian Point 2 and the Zion reactors.

2.0 PRIMARY CONTAINMENT FAILURE

Meltdown of a PWR core after scram normally results from uncovering of the core. In a loss of AC power and feedwater (TMLB') accident, there is an eventual loss of heat sink when the steam generators boil dry on the secondary side. Subsequent heating of the primary water increases the primary pressure until the pressurizer relief valve opens. The water then boils and escapes the primary system at a rate governed by the decay heat generation rate. Thus the time until the start of core uncovering is long and the rate of decrease in water level is slow. In a small-break LOCA (S2D), the response is similar if the break is small enough. However, if the break is large enough, the rate of water loss will be greater since the system is depressurizing and flashing can occur. However, in this case, the rate of water loss is limited by the size of the break. In a large-break LOCA (ABG), the rate is much faster. In all three cases, when the core melts, there is still some water in the lower plenum which has not boiled off. The molten core must penetrate this water, as well as the lower plenum structures, before it can start heating the primary vessel. The rate at which the melt vaporizes the water can greatly affect the pressure inside the primary vessel and inside containment just prior to vessel failure. In the MARCH code models A and B, the molten fuel is assumed to remain in the core region until a large fraction of the core is melted. The molten debris is then assumed to fall rapidly into the lower plenum and vaporize the water. Once the lower plenum water is vaporized, assault on the primary vessel walls begins. In the MARCH code model C, the fuel is assumed to stream into the lower plenum as it melts. Thus the lower plenum water is vaporized earlier and at a more uniform rate. Furthermore, failure of the reactor vessel may occur earlier (with less of the core molten). Due to the limited scope of this study, the consequences of model C will not be considered.

2.1 TMLB' Accident

After core melt and slumping into the lower plenum, MARCH calculates the time of vessel failure based on a combination of stresses (caused by internal pressure and debris weight) and reduced strength (caused by vessel heating). MARCH assumes a large break upon failure which results in an immediate pressure rise in containment caused by the release of high pressure steam from within the primary vessel and the dumping and boiling of the accumulator water. Furthermore, the opening is assumed to be sufficiently low in elevation that all of the molten debris immediately falls into the reactor cavity sump. These highly conservative assumptions may be inappropriate for a TMLB' accident.

Barring a steam explosion, vessel failure in a TMLB' accident will most likely be dominated by stresses caused by the high internal static pressure since the debris weight is negligible by comparison. The location of the initial failure will be dictated by wall thickness, stress intensification factors, and wall temperatures. The lower hemisphere-cylinder junction is a particularly weak point since there is a reduction in wall thickness there which causes stress intensification. In addition, the junction may be nearly the hottest spot since the upper portions of the lower plenum would dry out first and also would receive the largest heat fluxes from a convecting pool. For these reasons, the location of an initial pressure rupture would seem to be at the lower hemisphere-cylinder junction. Thus the initial failure may be near the top of the lower-plenum debris. This would prevent the bulk of the molten debris from falling into the reactor cavity unless the crack propagated downward. Since the steel may be cooler in the lower levels, downward propagation may be hampered. One must still consider the possibility that any melt flowing through the crack may erode it downward. (Complete circumferential ductile crack propagation is probably unlikely in this case due to circumferential non-uniformities in heating and the depressurization as the crack grows.)

One consequence of not having the molten debris fall immediately into the reactor cavity sump upon rupture of the primary vessel is that boiling of cavity water (assuming there is water in the cavity) does not immediately follow the release of reactor vessel steam into containment. This may give a filtered vent time to reduce containment pressure before facing sump water vaporization. In addition, the vaporization of accumulator water may proceed in two stages if some of it boiled in the vessel and some jetted out the break as a liquid into the cavity.

In addition to the location of the rupture possibly being higher than that assumed by MARCH, the size may also be significantly smaller. A longitudinal crack (from pressure-induced hoop stresses) would find thicker and cooler walls upward and reduced stresses (and possibly cooler walls) downward. This, as well as the decreasing pressure from escaping vapor, would tend to limit the crack growth. If the break is small enough, steam release from the pressure vessel may take many minutes. For example, a 1-meter long crack deformed into a 0.1-meter wide ellipsoidal hole would take about 4 minutes to discharge the primary system steam and the vaporized accumulator water (assuming one-phase sonic vapor flow). Thus, a small failure could limit steam discharge and the subsequent rate of containment pressurization. Unfortunately, the size and location of the rupture may be difficult to predict.

If the rupture is not small enough to significantly reduce the flow of steam out of the primary vessel, there is still the question of how swiftly the accumulator water would enter the primary vessel, how much would be vaporized in the pipes and on the downcomer walls, and how swiftly it would be vaporized by the residual hot debris in the lower plenum. A one-dimensional model was developed to address the first two issues. (See Appendix C.) It was determined that the accumulators would discharge in about 35 seconds if the reactor vessel pressure decreased rapidly to the containment pressure of about five atmospheres (i.e., if there were a very large hole). It was further determined that only about 6% of the accumulator water would be vaporized before reaching the bottom of the downcomer. Thus, the MARCH assumption that all of the accumulator liquid is delivered to the debris bed is probably reasonable for large vessel ruptures. The remaining issue (accumulator water vaporization rate by the lower plenum debris) is more difficult to calculate due to the uncertain geometry. The rates of steam generation and two-phase water release through the rupture must be left to a future study.

After a small-hole rupture and depressurization, most of the fuel could still be in the vessel. The fuel would then start to heat the vessel walls until a low pressure vessel failure occurred with likely dumping of nearly all the remaining molten debris into the reactor cavity. Thorough mixing and transfer of heat to the cavity water within a period of minutes may occur. Typically, there is little water during a TMLB' accident, so the quenching is water-starved and the pressure generation is limited. However, whatever liquid accumulator water was assumed to jet out of the small hole must now be included in the vaporizing cavity water.

Thus, a TMLB' accident would probably yield two pressure spikes separated by the time it takes to melt through the
4 primary vessel. The first spike would involve all the lower plenum water vaporizing (and partly causing the vessel failure) plus some of the accumulator water. It could be reduced and broadened in time by the limited flow through a small hole. The second spike would involve the water in the reactor cavity, which could come from condensation on the containment walls as well as from liquid accumulator water jetting out of a small hole rupture in the primary vessel.

There is another aspect of primary containment failure which merits consideration. The MARCH code indicates[2] a sudden increase in primary pressure (from 17.6 MPa (2548 psia) to 27.2 MPa (3939 psia) in the Indian Point 3 analysis) when the core slumps into the lower head water. If such a pressure increase does occur, it could fail the boundary between the primary and secondary sides of the steam generators (see Appendix A). Since the secondary side vents directly outside the containment building in a TMLB' accident, this failure would lead to a direct leak of radioactive gases from the molten fuel to the outside atmosphere.

The criterion for steam generator tube rupture depends on tube temperature and the pressure across the tubes. Since the steam generators are calculated to dry out about 223 minutes before the slump pressure pulse, the tubes may have been heated substantially by contact with superheated steam as well as by hot hydrogen, as the core uncovered. (Eventually this flow would be stopped by hydrogen buildup in the inverted U-tubes.) In addition, since there is no supply of liquid water going to the (dry) secondary side to be vaporized but there is likely a flow of superheated steam going from the steam generator to condense on cooler external piping, the secondary side pressure may drop somewhat below the secondary side relief pressure. If the secondary system steam relief valve fails in the open position, the pressure will drop to atmospheric. The primary system pressure required to cause steam generator tube failure at various temperatures with and without the secondary relief pressure of 7.0 MPa (1020 psia) is shown in Table 1. (See Appendix A for the calculations.)

Table 1. Primary system pressure required to fail the steam generator tubes at various temperatures T and secondary side pressures P_s.

P_s	T = 589 K (600°F)	T = 811 K (1000°F)
7.0 MPa (1020 psia)	25.8 MPa (3940 psia)	23.4 MPa (3400 psia)
0.1 MPa (15 psia)	20.2 MPa (2935 psia)	16.5 MPa (2395 psia)

As can be seen, the pressure pulse calculated by MARCH for Indian Point 3 (27.2 MPa (3939 psia)) would be sufficient to fail the steam generator tubes in all cases. MARCH calculates even higher pressures for Indian Point 2 (29.3 MPa (4243 psia)) and Zion (32.2 MPa (4665 psia)). Thus, steam generator tube failure with radioactive gas release to the atmosphere may be a strong possibility for a TMLB' accident depending on how well the MARCH code models the vaporization of water in the lower head. This problem is addressed more fully in reference 3.

2.2 S2D Accident

Just prior to the time of core slump in an S2D accident, MARCH calculates the reactor pressure to be 5.0 MPa (723 psia)[4] which implies that the accumulators have not yet dumped (since they inject only when the pressure drops below 4.0 MPa (600 psia)). After the core slumps into the lower plenum water, MARCH calculates the pressure to rise to 28.7 MPa (4159 psia). Thus, all the conclusions made in the TMLB' section concerning high pressure vessel failure, small-hole steam leakage, delay in core dumping, steam generator tube rupture, and the validity of the assumed slump pressure pulse apply also to the S2D accident.

Of additional interest is the dumping of the molten debris into the reactor cavity. In the initial MARCH calculations,[4] the cavity was assumed to be dry. Thus, the resulting steam generation from quenching was limited by the amount of water in the accumulators (i.e., the quenching was not complete). However, since the containment spray system is assumed functional, the cavity may have sufficient water in it for complete quenching of the debris. This could lead to a much higher pressure spike than indicated in either the original MARCH calculations or in a TMLB' accident.

If large-scale vessel failure is delayed (because of depressurization), the total hot debris available for quenching would be greater and the resulting pressure spike would be larger. In 15 minutes, enough decay heat is generated to vaporize an additional 10,000 kg (22,000 lb) of steam. This would cause an additional 0.028 MPa (4 psi) on a pressure spike. However, there is considerable uncertainty in this estimate. Any heat which raised the temperature of an anchored solid structure should be excluded from the dump since it would not drop into the cavity. Conversely, any heat which melted an already hot structure would be amplified since all the preheating of that structure would be included in the dump. In either event, delays on the order of an hour may have significant effects. (See next section for a discussion of low-pressure vessel failure times.)

2.3 ABG Accident

In an ABG accident, by the time the core melts and slumps, the reactor pressure has been reduced to very near that of the containment building. If a large fraction of the core and structure descends to the lower plenum, the major stress on the vessel will be from the weight of the debris itself. The debris could easily fill the entire lower hemisphere of the reactor vessel. Since the lowest point of the hemisphere would retain water the longest, it would be the coolest portion. If the debris were molten and convection currents were established, the greatest heat flux into the wall would be near the top edges of the pool. Thus, the hottest and most weakened part of the vessel lower hemisphere would be its sides. Once the sides were hot enough, failure would occur from the weight of the debris, and once a failure began, the resulting weakened head would continue to tear until it and its contents fell from the pressure vessel.

The two major obstacles to accurately predicting this type of failure are: (1) the lack of high-temperature strength data for the pressure vessel steel (A-533-B) above 560 K (550°F), and (2) the lack of knowledge about molten debris bed formation and heat transfer. However, since the tensile stress on the vessel from the debris and vessel bottom weight is only 1.64 MPa (240 psi) failure will likely not occur until almost total melt-through. A one-dimensional conduction model indicates that the vessel walls would approach 1000 K in about 25 minutes (see Appendix B). Using this value with the melt-through model of Hakim[5] indicates that melt-through would take about 60 minutes. The MARCH code predicts the time of failure to be 30 minutes.[2] Thus, MARCH predicts gross failure to occur about 30 minutes early. This difference in time to fail would allow time for additional heating of the debris and would result in a larger pressurization of containment than would be predicted by MARCH (if there were adequate reactor cavity water for complete quenching).

2.4 Other Accidents

The detailed assessment of pressure generation in containment for the particular accidents previously discussed (TMLB', S2D, ABC) may be misleading. It appears that in all accidents in which there is a core meltdown, the major release of molten materials into the reactor cavity may not occur until after the vessel has depressurized. In that case, large-scale vessel failure with nearly complete dropping of the molten debris into the cavity is possible (and perhaps likely). In the TMLB' and ABG accidents there is insufficient water in the cavity for complete quenching of the debris since there are no active injection systems operating. Thus the pressure spike from steam generation is limited. However, simple perturbations on these accidents could result in sufficient water in the cavity for complete quenching. One example is a TMLB' accident with AC power restoration and core injection after sufficient core damage to yield eventual vessel failure. Another is an ABG accident with containment heat exchanger failure but operable and plentiful containment spray. Thus one must consider the consequences of a complete quenching of the entire molten debris after low-pressure vessel failure.

The MARCH code maintains a reasonable heat balance during the accident calculation. Thus, although it may not accurately predict how much structure is molten at the time of head failure, it probably assesses fairly well how much energy would be in an assumed amount of debris. In the TMLB' accident, MARCH calculates that 223000 kg (492000 lbm) of debris would have an average temperature of 2560 K (4149°F) at vessel failure.[2] (It calculates a similar state for the ABG accident.) (That amount of mass is equal to all the UO_2 and Zr plus slightly more than an equal mass of structure.) The heat released in quenching that mass to 422 K (300°F) (including the latent heat of fusion of UO_2 since the assumed melting point of UO_2 is 2550 K (4130°F) in MARCH) is about 3.24×10^{11} J (3.06×10^{8} BTU). This is sufficient energy to cause a change in pressure in containment from 0.10 MPa

(15 psia) to 0.46 MPa (67 psia) (i.e., a 0.35 MPa (50 psi) rise). (This allows for heating of 373 K (212°F) liquid water to the 0.46 MPa boiling point, vaporization of the water, and heating of the initial containment gas.) The amount of water required is 134000 kg (296000 lb, or 4950 ft^3) which is about equal to that initially in the reactor vessel. This would fill the reactor cavity to a depth of 2 m (7 ft).

Features which could reduce the size of this 50-psi pressure rise include lower initial water temperature, incomplete core melting, early head failure, incomplete debris dumping, limited amounts of water, incomplete quenching from explosive scattering of the debris, and slow vaporization from poor mixing. Features which could increase the pressure rise include heat and hydrogen generated by oxidation of the molten structural materials, and the inclusion of loose hot (but non-molten) structural materials (not included in the MARCH debris estimates) in the debris. In addition, containment damage from the dynamic effects of a steam explosion must also be considered in association with this event.

The possibility of undesirable consequences from the mixing of core debris with reactor cavity water suggests limiting the amount of water allowed in the cavity. However, such an action may also have undesirable consequences and should not be considered lightly. For example, without water, core debris in the cavity would not be coolable (see Appendix D) and would begin penetrating into the basemat and generating vapors. (In general, it is risky to withhold coolant from heat-generating materials.) The decision to limit reactor cavity sump water involves considerations and calculations which are beyond the limited scope of the present study.

3.0 AC POWER RESTORATION IN A TMLB' ACCIDENT

If AC power is restored early enough during a TMLB' accident, it will be possible to mitigate many of the consequences and even prevent significant fuel damage. If power is restored prior to steam generator dryout (about 83 minutes)[2], feedwater flow could be restored (the switches and valves would be set to do this). Natural circulation would remove the heat from the core and there would be minimal, if any, core damage.

From the time of steam generator dryout to the start of core uncovering is about 136 minutes.[2] If an adequate heat sink could be established during this period, heat removal from the core could proceed via natural circulation and little, if any, core damage would result. However, restoraton of feedwater to dry hot steam generators may not be desirable (if alternatives are available) because the thermal shock could crack the tubes. If any of the fuel pins had clad failure, radioactive fission gas could escape to the secondary side of the steam generator. Since the steam generators vent outside of containment, this would create a direct radioactive release to the atmosphere. In addition, the accident would subsequently assume some of the characteristics of a small-break LOCA. The engineered safety system operates in a TMLB' accident to set the switches and valves so as to direct auxiliary feedwater to the steam generators. Thus, if these switches are not manually reset, restoration of AC power after steam generator dryout would immediately direct cold water to the hot steam generators and could cause an inadvertant failure of containment. This possibility needs further consideration and action.

If restoration of feedwater is undesirable after steam generator dryout, decay heat must be removed in a different manner. One way is by injection of external water, vaporization, and vapor release through pressure relief valves (inside containment). For steady-state operation, the steam must be condensed and recirculated to the injection pumps. The charging pumps can deliver enough water to remove all the decay heat by vaporization. However, it may be desirable to remove the heat via the high pressure or low pressure injection pumps. This would require depressurization (to at least 11.7 MPa (1700 psi)). One could achieve this either by cooling with the charging pumps or by opening the power-operated relief valves on the pressurizer. (It would require both relief valves open to remove all the vapor produced by decay heat, and a manual override of the PORV normal setting may be necessary.) Thus, there are paths available to cool the core after steam generator dryout and before core uncovering without using the steam generators.

The time from the start of core uncovering to core slump is predicted to be about 54 minutes. During this time, clad oxidation and core damage will occur. Cooling the core upon power restoration will involve core reflooding. (Again this would require either the charging pumps or depressurization via the PORV's in order to use the HPIS or LPIS.) This could cause fracturing of the hot fuel and fragmentation of any (small amounts of) molten materials. This will probably result in debris particles with an average diameter in excess of several millimeters. An in-vessel particle bed involving the entire core at this time (~250 minutes) is calculated to be coolable via water boiling if the average particle diameter exceeds 2 mm. (See Appendix D.) Thus power restoration will probably prevent further melting and save the primary vessel at this time. (Evidence for this is found in the Three Mile Island accident in which the primary vessel survived partial core uncovery.)

MARCH calculates the time from core slump to vessel failure and dropping of the core into the reactor cavity to be about 8 minutes. The calculated time of head failure may indeed be that short since the pressures are high and not much vessel heatup is required to cause failure. However, the time of debris dump into the cavity could be considerably later. As discussed in Sections 2.1 and 2.3, since the initial failure relieves the internal pressure, large-scale vessel failure and debris drop would require heating the vessel wall to near the melting point. This would take about 60 minutes or more. Unfortunately, restoration of power at this time would probably be of little value since any water injected in the largely molten debris (via the downcomer) would probably cause fragmentation into sub-millimeter sized particles (due to the greater likelihood of energetic interaction at low pressure). A full-core in-vessel bed of this particle size most likely cannot be cooled. (See Appendix D.) In addition, the energetics of a steam explosion could itself cause significant damage. Thus, AC power restoration between core slump and head failure probably cannot prevent large-scale vessel failure.

When the core drops into the reactor cavity, there may be water there from condensation of steam on the containment walls (or operation of the containment spray from earlier power restoration). Upon contacting the water, fragmentation and energetic dispersal of the melt will most likely occur. This may leave the debris in a particle state which can be cooled continuously via water boiling and thus prevent remelting and penetration of the basemat. Unfortunately, this event also produces a pressurization of containment. Quenching the entire core (UO_2 and Zr) and an equal mass of structure to the boiling point of water requires about 2.0 m (7 ft) of water in the cavity and produces a pressure increase of about 0.35 MPa (50 psi). (See Section 2.4.) Less water will produce a smaller pressure increase but will leave the fragmented debris hot. The containment must be able to withstand this pressure in order to avoid radioactive release to the atmosphere.

These considerations may impact operator actions if power is restored after core slump but before the core drops into the cavity. One course of action is to adjust the water level in the sump (via the containment spray and cavity pump) to an amount which would not yield a damaging pressurization (say about 0.5 m) but which would cause fragmentation and particle dispersal. This would leave the debris in a state which might be coolable by water (which could be added more slowly later) and which (if it were coolable) would not attack the basemat. Radioactive contamination may be dispersed inside the containment, but this would not be an immediate safety hazard. However, one would want to be certain that the steam explosion energetics could not damage containment. Another course is to pump the cavity dry and let the debris land in the cavity without dispersal or cooling. The debris would then begin to melt into the basemat but may freeze before penetration and thus be contained in a (rather large) concrete encasement. Even if penetration occurred, the actual risk to the public would not be as great as with airborne penetration (although the perceived risk may be as large). A similar choice remains if power is restored after core dump with respect to putting water on the melt: a quenched and fragmented bed may be coolable without basemat penetration, but may also result in a steam explosion or large fuel dispersal.

In summary, in a TMLB' accident with power restoration, core damage most likely can be prevented if restoration occurs before the start of core uncovering (about 219 minutes). But if restoration occurs after steam generator dryout (about 83 minutes) cooling via the charging pumps or high pressure injection may be preferred since thermal shocking the steam generators with feedwater could cause breach of containment via venting. (Indeed, failure to turn off feedwater switches after steam generation dryout could cause an inadvertent breach.) Power restoration after the start of core uncovering (219 minutes) but before core slumping (273 minutes) may preserve the reactor vessel integrity but not core integrity. Power restoration after the melt of a large fraction of the core probably cannot prevent breach of the reactor vessel or prevent the movement of large amounts of core materials into the reactor cavity. However, it may allow cooling of ex-vessel debris and prevent basemat penetration.

4.0 PARTICLE BED COOLING

To prevent high temperatures and structural damage of the core following shutdown requires removal of the decay heat generated in the fuel to an ultimate heat sink. The loss of this heat removal capability in the TMLB', S2D, and ABG accidents is what causes the fuel damage. However, at some point the heat removal capability may be restored (e.g., AC power restoration in the TMLB' accident). It is then possible to remove the decay heat from the damaged fuel (debris). This is usually achieved via a recirculating coolant (water) which passes the heat from the debris to a sink. However, even though the coolant may have easy access to a good sink, the debris may be in a state in which it is difficult for the heat to reach the surrounding coolant. In that case excessive temperatures in the debris and nearby structure may occur. One such state of low coolability is a particle bed.

A particle bed may form in an accident sequence from fracturing of solid fuel pins (e.g., by thermal shock of hot pins by reentering cold water). One may also form from molten debris contacting cold water and fragmenting and quenching (e.g., in the lower plenum or in the reactor cavity). If the particles are large, the coolant can easily enter the bed and all the heat can be removed from the bed to the overlying pool of coolant by boiling. In this case the bed and supporting structure will remain at the relatively cool boiling temperature of the coolant.

(In the extreme case of very large particles, the temperature rise within an individual particle can be large. However, a very simple conduction calculation shows that to melt the center of a particle (with boiling water on its surface) requires a "particle" diameter of 0.24 m. For this reason, the temperature rise within an individual particle has never been a concern in particle bed coolability studies). If the particles are small, the liquid from the overlying pool cannot penetrate the particle bed swiftly enough to offset the vaporization. In this case, portions of the bed will dry out and begin heating to above the coolant boiling point. Because of the low thermal conductivity of dry particle beds (several times smaller than for non-porous UO_2), only a small dry zone (about 50 mm thick) is needed for a section of the bed to reach temperatures sufficient to remelt the particles or weaken the supporting structure. Thus even if there is a maintained coolant pool overlying a particle bed, it is still possible to have structural failure or particle remelt. For this reason, the conditions necessary to achieve dryout have been the major concern in particle bed coolability studies.

Considerable research has been done on particle bed dryout in the LMFBR safety program. Measurements involving water, acetone, methanol, and sodium as fluids with steel, lead, sand, and urania as particles have been performed. Empirical correlations and phenomenological models have been developed. These models are discussed and applied to PWR accident cases in Appendix D. It is found that to avoid dryout in an in-vessel particle bed involving the entire core on an impermeable support* (assuming there is adequate water and an ultimate heat sink) requires a particle diameter of at least 2.4, 4.8, and 8.5 mm for the TMLB', S2D, and ABG accidents, respectively (at the pressures calculated by the MARCH code[4]). Fragmentation studies (also discussed in Appendix D) suggest that the expected average particle diameter from quenching would be several tenths or hundredths of a millimeter if the fragmentation is explosive and possibly several millimeters if it is not. Thus, a total-core in-vessel particle bed will probably dry out unless it is produced non-explosively at least an hour past shutdown with a high pressure maintained.

To avoid dryout of an ex-vessel particle bed involving the entire core and occupying a 5.2-meter (17-foot) square in the reactor cavity requires a particle diameter of at least 1.1, 1.6, and 2.3 mm for the TMLB', S2D, and ABG accidents, respectively, for 0.5 MPa (75 psia) of pressure (and about twice that for one atmosphere). (This again assumes there is adequate water in the cavity and an ultimate heat sink.) Thus a non-explosively-produced ex-vessel particle bed probably would not dry out. However, fragmentation studies indicate that spontaneous explosions are common with large-scale oxide melts dropped into water at one atmosphere. Explosively-produced debris would have a smaller average diameter; however, it also would be scattered throughout the reactor cavity (and perhaps farther). In this dispersed case particle diameters of 0.5, 0.7, and 0.9 mm, respectively, are required (for 0.5 MPa). The TMLB' requirement

*Whether or not the bed is on an impermeable surface has less than a 30% effect on the dryout flux with natural circulation.

is close to a large fraction of the diameters expected from large-scale explosive events. Thus an explosively-produced ex-vessel particle bed may be coolable, at least in the TMLB' case. (This is even more plausible if one considers that the whole core may not be involved.) Coolability of a reactor cavity debris bed would eliminate the concern for basemat penetration. Unfortunately, the present data base on fragmentation and debris dispersal is inadequate to clearly indicate whether or not dryout and remelt will occur.

5.0 IMPACT ON FILTERED-VENTING CONTAINMENT SYSTEMS

The manner of primary containment failure and the possibility of cooling particulate core debris has been considered. The former strongly influences the size of pressure spikes from steam generation. The latter may prevent gas generation during penetration of the basemat by molten debris. Both of these items affect the loading on the vent. The following conclusions with respect to filtered venting may be made:

(1) In the TMLB' and S2D accidents, initial primary vessel failure will most likely be a pressure rupture. A small-hole rupture could reduce the flow rate of steam enough to allow a filtered vent to reduce the peak containment pressure. The size of this effect cannot be calculated at this time.

(2) A small-hole rupture near the top of slumped debris in the lower plenum may allow accumu-

lator water to escape without vaporizing. If that occurred, the pressure spike in containment due to release of high-pressure primary steam (from vaporization of lower plenum water) would be separated in time from the spike due to accumulator water vaporization. The separation time would be that required to fail the vessel under low pressure and could be as much as an hour. This would reduce the peak load on the vent.

(3) In all three accidents (TMLB', S2D, ABG) analyzed, a major steam-generation event could result from a low-pressure failure of the reactor vessel. The entire contents of the lower plenum (molten core and structure) may drop into the reactor cavity at this time. Under the TMLB' and ABG accident assumptions, the amount of water in the cavity would be small enough that the interaction would be water starved (i.e., the quenching would not be complete). This would limit the steam generation. However, small perturbations on these accidents could easily lead to sufficient cavity water for complete quenching. In that case, a containment pressure increase of around 0.35 MPa (50 psi) is possible.

(4) Moderate pressurization of the primary system from a core slump into the lower plenum water and/or

moderate heating of the steam generator tubes could cause tube failure. This would open a direct path for a radioactive gas release to the atmosphere since the steam generator secondary side vents outside containment. Thermal shock of the secondary tubes from cold feedwater entering a dry steam generator upon AC power restoration in a TMLB' accident could achieve a similar result.

(5) In the TMLB' accident sequence, ex-vessel particulate debris may be sufficiently coolable by natural circulation to prevent dryout and particle melting if an overlying pool of water is maintained. This would keep the debris at very low temperatures and prevent basemat penetration as well as the associated gas generation from the concrete. Unfortunately, the route to such a coolable state may involve energetic quenching with a pressure pulse and wide dispersal of the radioactive debris (within the containment). Both the pressure pulse and the debris may load the vent.

6.0 ADDITIONAL RESEARCH NEEDS

There are many uncertainties in the previous analyses. Some of these uncertainties could be reduced by further research:

(1) The small-hole rupture of the primary vessel by high-pressure can reduce the magnitude of pressure spikes depending on the size and location of the

hole. A calculational effort using models for ductile crack propagation could be made to estimate these parameters. Some material property measurements might also be needed.

(2) The rate of accumulator water vaporization as it enters the core as well as the possibility that it escapes through the vessel failure point will affect the pressure spike size. A calculational effort (perhaps using the TRAC code[7]) could help scope these effects. However, the possibility of a vapor explosion could cause additional complications.

(3) The beneficial effects on filter design of items (1) and (2) may be nullified by consideration of the pressure pulse formed from mixing a whole molten core with sufficient reactor cavity water for total quenching. This event requires large-scale core and structural melt, large-scale vessel failure, and near total quenching of the debris by the cavity water. A program to assess the likelihood of these requirements could focus on the time and manner of low-pressure vessel failure, as well as on the amount of molten material at that time. This activity would be both analytical and experimental.

(4) The final state of a once-molten core may be a particulate bed submerged in a pool of water and cooled via natural circulation. The deter-

mination of this possibility requires more knowledge on fuel fragmentation and dispersal in large-scale interactions of core melt and water. In addition, particle bed dryout data for very deep beds and beds at high pressure would be required. Due to the complex phenomena involved, this program would need to be experimentally based, and would require much more time than is available in the present (filtered venting) study.

REFERENCES

1. R. O. Wooton and H. I. Avci, A User's Manual for MARCH, (Draft), Battelle Memorial Laboratories, Columbus (1979).

2. P. Cybulskis (Battelle Columbus Laboratories), letter to A. S. Benjamin, (Sandia National Laboratories), February 8, 1979.

3. J. B. Rivard, "Review of In-Vessel Meltdown Models," SAND80-0455 Sandia National Laboratories, 1980.

4. P. Cybulskis (Battelle Columbus Laboratories), letter to A. S. Benjamin, (Sandia National Laboratories), January 16, 1979.

5. S. J. Hakim, Nucl. Sci. and Eng., 72, 129 (1979).

6. L. D. Buxton and W. B. Benedick, Steam Explosion Efficiency Studies, SAND79-139 (1979), NUREG/CR-0947.

7. Safety Code Development Group, TRAC-P1A, An Advanced Best-Estimate Computer Program for PWR LOCA Analysis, Los Alamos Scientific Laboratories, NUREG/CR-0665, LA-7777-MS (1979).

APPENDIX A

STEAM GENERATOR TUBE PRESSURE FAILURE

Although failure of the primary vessel during meltdown will result in radioactive release into the containment building, the building is designed to prevent release of this radioactivity into the atmosphere. However, a ruptured steam generator tube may cause a bypass of the containment building retention features and the release of radioactive gases directly from the primary vessel to the atmosphere. This is because under accident conditions, the secondary side of the steam generators vent directly to the atmosphere outside the containment building.

Rupture of the steam generator tubes could come from some combination of high primary side pressure, low secondary side pressure, and high temperatures (although all of these features need not be present). High primary side pressures could come from rapid slumping of a molten core into lower plenum water. This would cause pressurization if the rate of steam generation exceeded the rate of steam loss through the pressurizer relief valves. A secondary side pressure lower than the vented relief valve setting could occur after steam generator dryout if superheated steam on the secondary side condensed on distant piping with no replenishing water flow. Alternatively, if the relief value failed in the open position, the secondary pressure would be atmospheric. High temperatures could occur during core uncovery when superheated steam would flow to and condense on the dry steam generator tubes. (This process could be terminated by hydrogen buildup and blockage once clad oxidation began.)

The steam generator tubes are Inconel 600 with an outer diameter of 22.2 mm (0.875 in.) and a thickness of 1.27 mm (0.050 in.). The primary membrane stress for thin shells is

$$s = P(r + t/2)/7 = 9.25P$$

where P is the differential pressure across the tube. Failure will occur when the membrane stress exceeds the yield stress. The differential pressures which cause failure at various temperatures are shown in Table 1. The pressures are close to those which could be present in accident situations.

Table 1. Differential Pressure Causing Failure of the Steam Generator Tubes

Temperature (K)	(°F)	Yield Stress (MPa)	(ksi)	Failure Pressure (MPa)	(ksi)
294	70	248	36	26.8	3.89
589	600	186	27	20.1	2.92
700	800	193	28	20.9	3.03
811	1000	152	22	16.4	2.38
922	1200	152	22	16.4	2.38

Reference

1. Standard Handbook for Mechanical Engineers, 8th Ed. (1978).

APPENDIX B

HEATING OF THE PRIMARY VESSEL

As core material begins to slump into the lower head region, partially molten core material will be quenched by water (which may not be present for some scenarios) and colder structure material. It may be assumed that core debris which initially comes in contact with the lower head will be solid. After contact the debris will lose sensible heat to the vessel wall. Decay heat will begin to heat up the debris but a solid crust will remain on the wall until the bulk of the debris has changed to molten material. When the transition from solid to liquid occurs, the heat transfer rate from the molten debris will increase through convective heat transfer. During this time, the solid crust near the wall will melt exposing the vessel surface which could be molten. Further attack of the vessel wall is dominated by convection. If the amount of quenching is large, the time to establish a molten pool with convection is large. The vessel wall heating will then be dominated by conduction.

In order to estimate the vessel heatup time after core slump for an ABG accident, a one-dimensional conduction calculation was made. It included decay power and phase change in the debris. A decay heat value of 2.6 MW/m^3 was used, which corresponds to the decay power in a non-porous mixture of equal volumes of UO_2 and structure at 23 minutes past scram (which is the calculated time of core slump). The thermal conductivity used for the debris was 3.0 W/m·K, which presumes non-porous debris. The initial debris temperature was 2500 K. The resulting temperature profiles are shown in Figure 1. Since the debris melts from 1500 sec to 2000 sec, the results are less valid beyond 2000 sec.

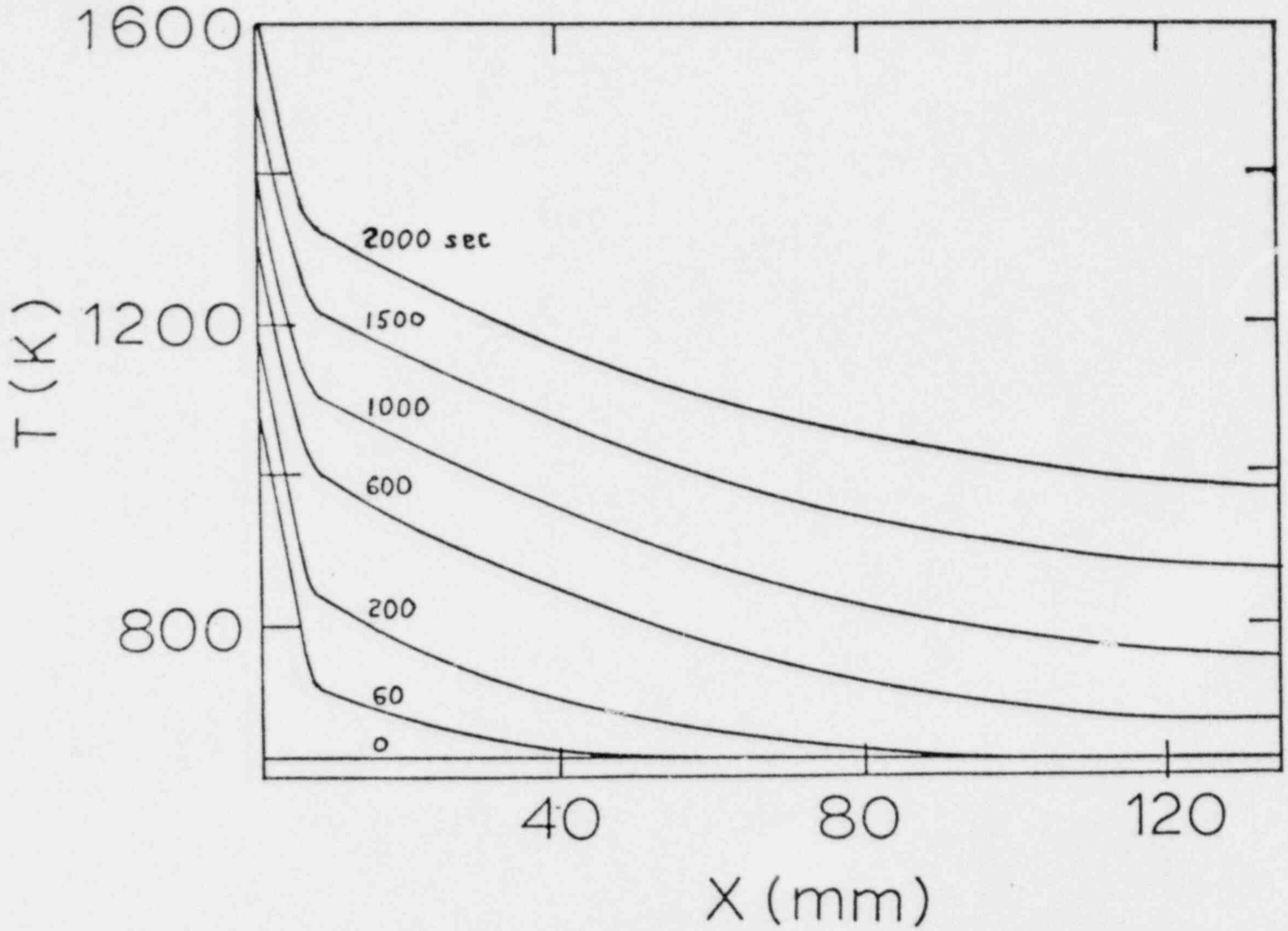


Figure 1. Temperature of Vessel Wall vs. Penetration for Various Times after Contact by Debris

APPENDIX C

TRAC CALCULATION OF ACCUMULATOR LIQUID DELIVERY

One of the assumptions of the MARCH code calculations of the TMLB' sequence in Zion/Indian Point (ZIP) is that after vessel melt-through the primary system pressure drops almost instantaneously from about 2550-3400 psia to the containment pressure of about 80 psia. Since the accumulators in ZIP-type plants are passive systems, their operation depends only on the primary system pressure decreasing below the setpoint ($\sim$ 600 psia). Once the pressure decreases below the setpoint, the accumulators will begin delivering highly subcooled liquid (125°F). MARCH assumes that 100% of this liquid reaches the molten debris bed where it is then instantaneously vaporized. This accounts for a large increase in containment pressure. The purpose of the calculation to be described is to determine how much accumulator liquid can reach the bottom of the downcomer without boiling on the hot piping and downcomer walls, and how long it takes the accumulators to empty by using a mode which more accurately describes the heat transfer and fluid flow phenomena than the MARCH code. In other words, is the MARCH assumption conservative or is it close to best-estimate? The code chosen to be used for this study is the Transient Reactor Analysis Code (TRAC)[1] developed by the Los Alamos Scientific Laboratory as a best-estimate, thermal-hydraulic code to calculate PWR transients.

Brief Description of TRAC and the Model

TRAC is a best-estimate, PWR transient computer code which is capable of simulating the thermal-hydraulics of the entire primary system. It contains a three-dimensional representation of the flow and heat transfer in the reactor vessel and a one-dimensional representation of the remainder of the system. A two-fluid, nonequilibrium treatment is used in the vessel and a drift-flux treatment is used in the one-dimensional components. One can simply use TRAC with any number and type of component such as a pipe connected to a tee connected to a pump or an entire PWR system can be modeled.

A schematic of the physical situation to be modeled is shown in Figure 1. Shown in the figure is the vessel, which is assumed to have been breached, the vessel internals, and two cold legs. ZIP-type reactors contain four loops with an accumulator on each loop. Because of the time constraints the situation to be modeled was simulated with TRAC in one dimension, as shown in Figure 2. Here the downcomer has been replaced with a pipe, with one cold leg and accumulator connected to it. The pipe diameter is equal to the downcomer gap size. Some of the important assumptions used to model the system are:

1. Assume all the accumulator liquid flow is directed at the vessel; therefore, there is no flow away from the cold-leg/vessel connection.
2. Assume the downcomer wall thickness is approximately the thickness of the thermal shield.

2. Assume, initially, that $T_{walls} = T_{sat}$ and $P_{sat} = 2550$ psia.

The simulation shown in Figure 2 is fairly representative of the true system and is certainly a better simulation than found in the MARCH code. We would like to know how much of the accumulator liquid reaches the bottom of the downcomer and how long it takes the accumulator to empty. Table 1 shows some of the important initial conditions used for the calculation.

Results and Discussion

Figure 3 shows both the void fraction at the downcomer exit and the accumulator liquid level as a function of time. As can be seen, it takes on the order of 35-40 seconds for the accumulator to empty in this calculation. Thus, the MARCH assumption of instantaneous liquid delivery is probably reasonable. The initial liquid reaches the bottom of the downcomer in about 3-5 seconds and completely fills the downcomer until about 35 seconds. Thus, a substantial amount of liquid does escape through the bottom of the vessel. The initial liquid mass in the accumulator and accumulator line is about 27,500 kg. The TRAC calculation shows that about 25,800 kg of this liquid (initially at 325 K) leaves the downcomer (at about 328 K). Therefore, only 1700 kg or 6.2% of the liquid is vaporized by the walls and the remainder is only heated 3 K. If we assume a fairly symmetric distribution, then a real plant with four accumulators could deliver 103,200 kg of subcooled liquid to the molten debris bed. Thus, the MARCH code assumption that all of the accumulator liquid is delivered to the debris bed is also probably reasonable.

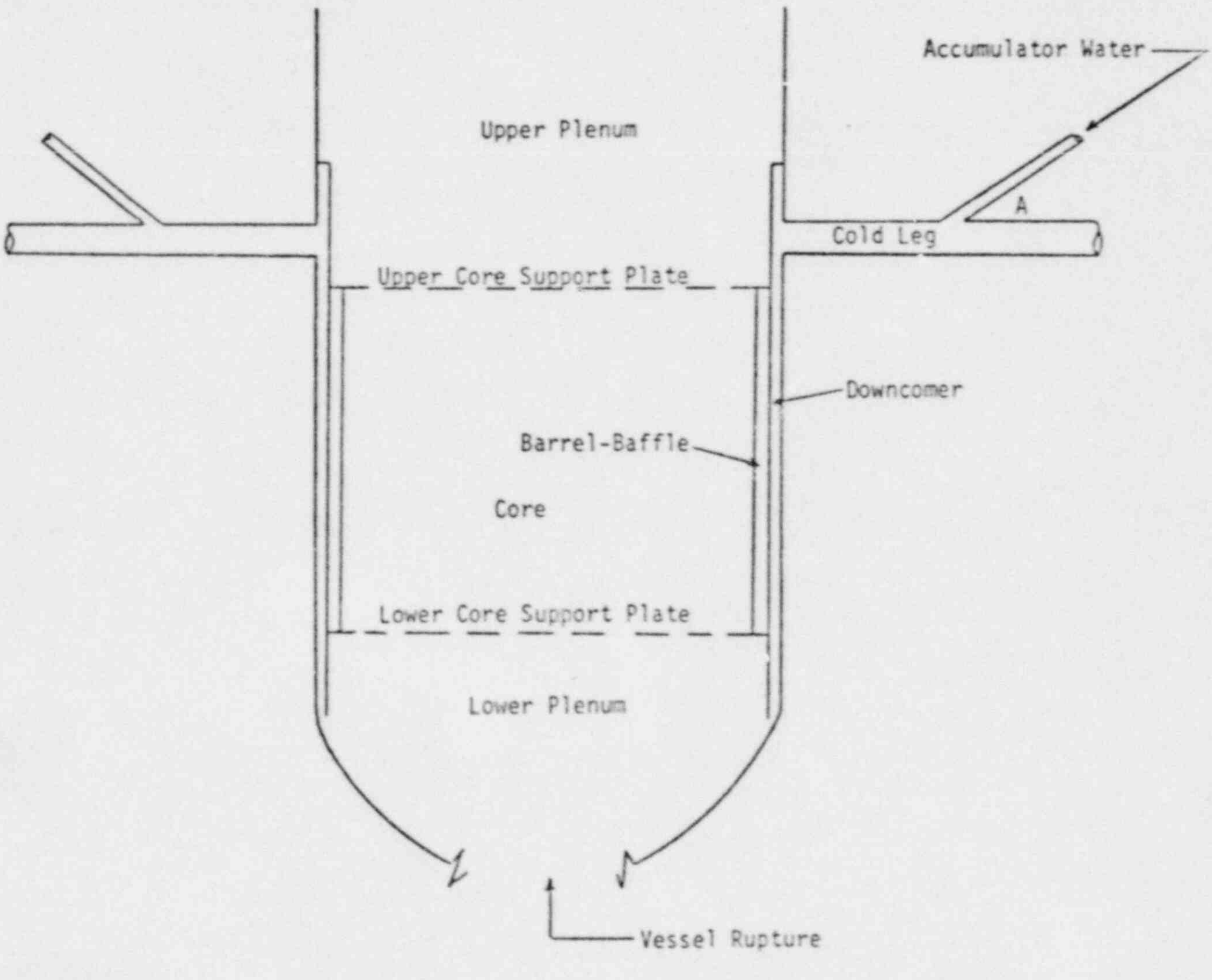
Reference

1. Safety Code Development Group, "TRAC-P1A, An Advanced Best-Estimate Computer Program for PWR LOCA Analysis," Los Alamos Scientific Laboratory, NUREG/CR-0665, LA-7777-MS (May 1979).

Table 1

Parameter	TRAC Value
1. Containment pressure	5.516×10^5 Pa
2. Downcomer wall temperature	627.4 K (T_{sat})
3. Downcomer wall thickness	0.0254 M
4. Downcomer gap size	0.256 M
5. Cold-leg wall thickness	0.0603 M
6. Accumulator pressure	4.08×10^6 Pa
7. Accumulator liquid temperature	325 K

Figure 1. Vessel geometry



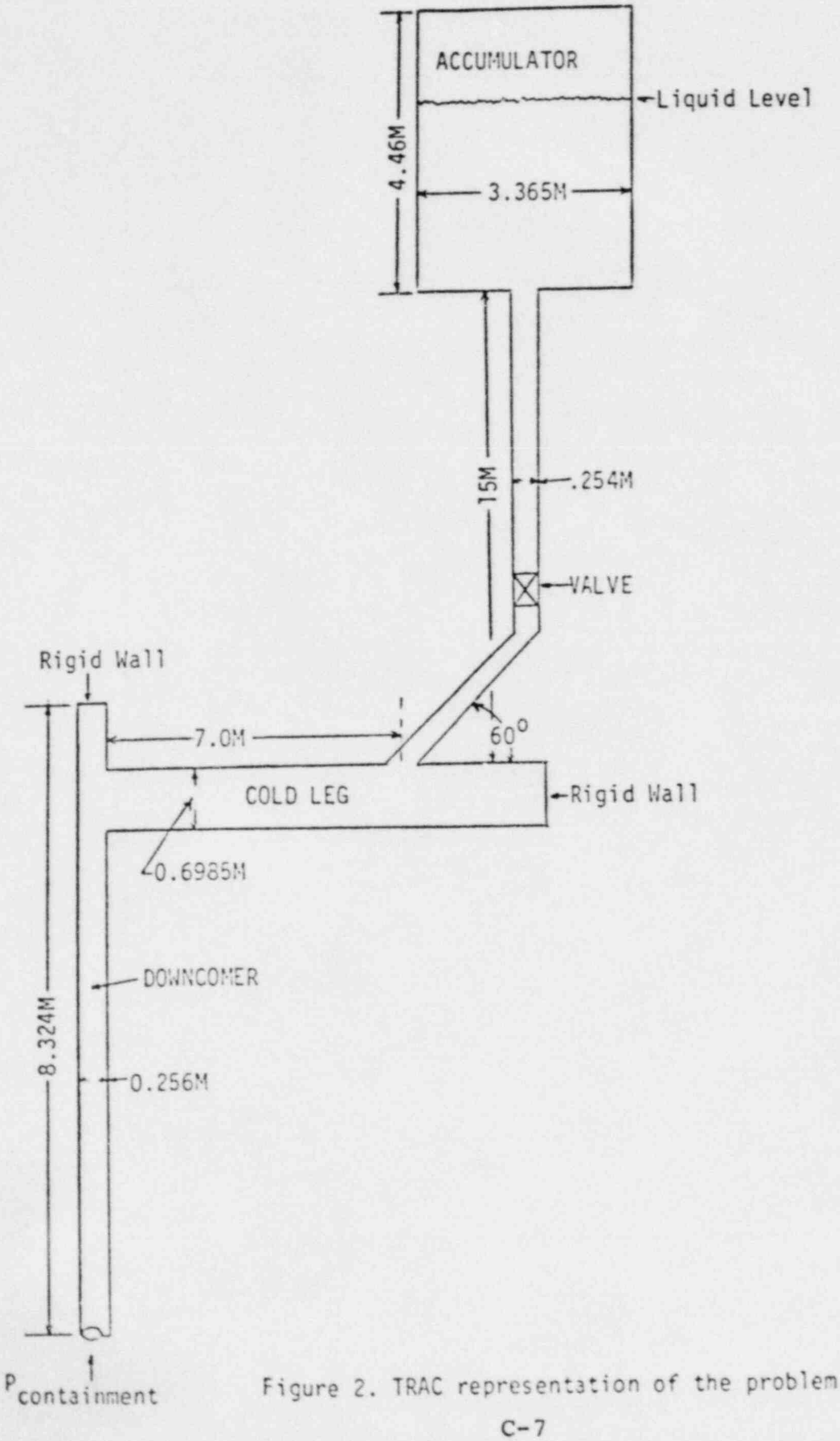
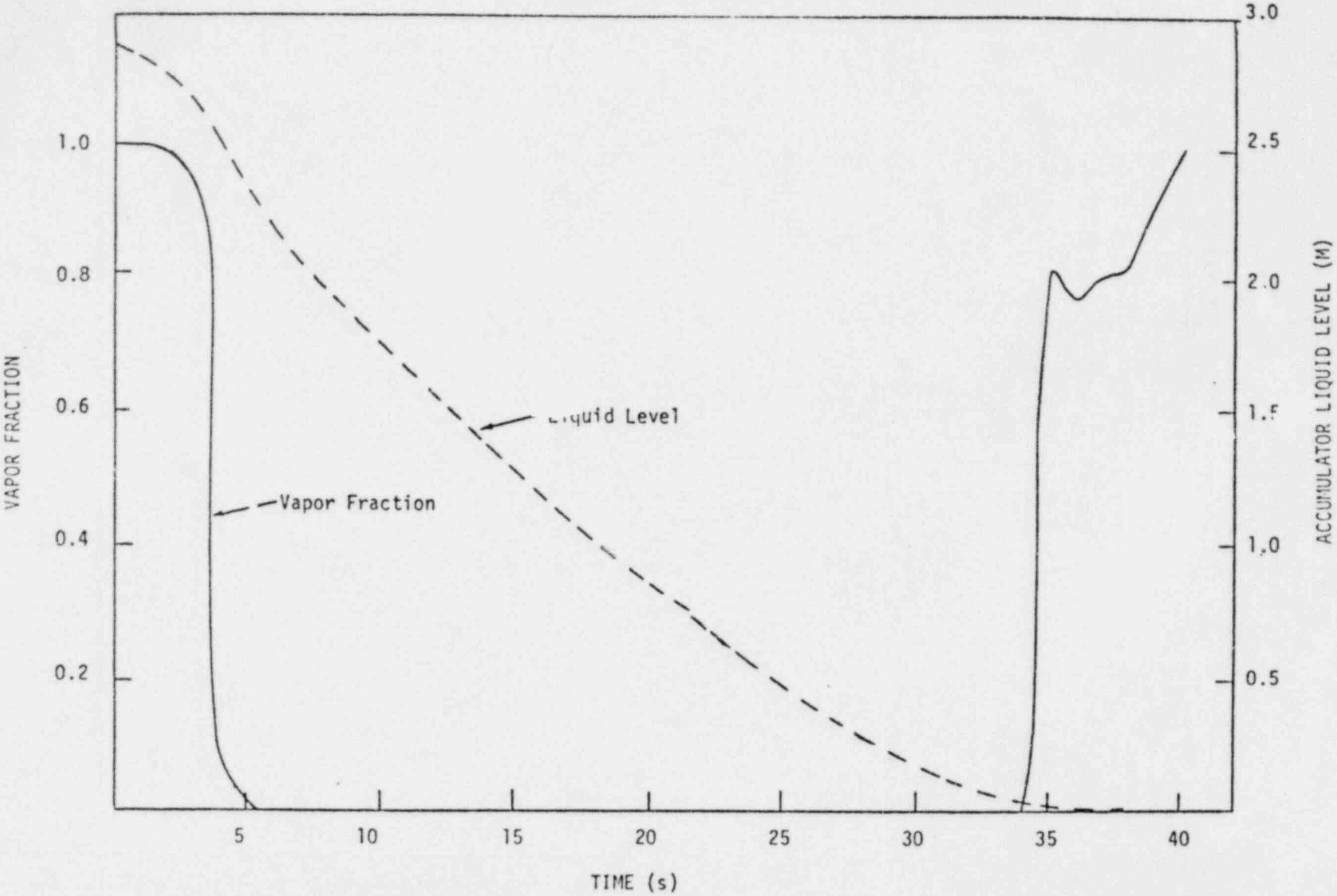


Figure 2. TRAC representation of the problem

Figure 3. Vapor fraction (downcomer exit) and accumulator liquid level versus time



APPENDIX D

COOLABILITY OF A PARTICLE BED

When molten core materials encounter liquid coolant, freezing and fragmentation normally occurs. The frozen particles may settle into horizontal beds on the structure below. If there is not an adequate heat sink or coolant replenishment, the coolant will eventually all boil off and the bed will heat, remelt, and attack the supporting structure. However, if there is an adequate heat sink or coolant source to maintain the overlying coolant pool, and if the decay power is sufficiently low, all decay heat produced can be removed by the boiling process. In this case, the bed remains at the boiling temperature of the coolant and the supporting structure remains intact. (The temperature difference between the center and the surface of an individual particle is negligible.) For sufficiently high decay power, though, the liquid from the overlying pool cannot penetrate the particle bed swiftly enough to offset the vaporization. In this case, portions of the bed dry out and begin heating to above the coolant boiling point. Because of the low thermal conductivity of dry particle beds, (several times smaller than for non-porous UO_2), only a small dry zone (about 50 mm thick) is needed for a section of the bed to reach temperatures sufficient to remelt the particles or weaken the supporting structure. Thus even if there is a maintained coolant pool overlying a particle bed, it is still possible to have structure failure or particle remelt.

With a loss of all AC power and failure of the turbine-driven feedwater (TMLB'), there is no heat sink and no water injection either inside or outside the pressure vessel. Thus, the question of steady-state particle-bed coolability by coolant boiling either in- or ex-vessel is not applicable since all the coolant quickly boils off. However, if there is power restoration at some time into the accident, the question is applicable. Similarly, in a small-break LOCA with a failure of emergency core cooling injection (S2D), in-vessel beds will boil dry, but ex-vessel beds could be cooled by recirculating containment building water. A large-break LOCA with failure of emergency coolant injection would also leave in-vessel and ex-vessel beds boiled dry, but again, later restoration of power could change this. Thus, the coolability of in- and ex-vessel particle beds by coolant boiling must be investigated.

In a volumetrically-heated particle bed cooled with a boiling fluid, there is a counter flow of downward-moving liquid replacing the upward streaming vapor. Incipient dryout will occur in the bed when the vapor generation rate is sufficiently large that it prevents the adequate flow of replenishing liquid. Incipient dryout thus depends on the fluid material properties (heat of vaporization, density, viscosity, etc.), particle sizes and shapes, bed depth, bed packing (space between particles), volumetric bed power, etc. Considerable research has been performed on particle bed dryout in the LMFBR safety program.[1-7] Experiments involving water, acetone, methanol, and sodium as fluids with steel, lead, sand, and urania as particles have been performed. Empirical correlations and phenomenological models have been developed. A semi-empirical model base partly on an assumption of Darcy flow and partly on a correlational fit to the measured dryout data is that of Dhir and Catton:[1]

$$q_d = .0177 \frac{\rho_1 g K h_{fg}}{\nu_1} (1 - \rho_v/\rho_1) \tag{1}$$

where q_d is the heat flux exiting the bed at incipient dryout, ρ_1 and ρ_v are the liquid and vapor densities, g is gravitational acceleration, K is permeability, h_{fg} is the heat of vaporization, and ν_1 is the liquid kinematic viscosity. The permeability is taken from the Kozeny Carmen relation:

$$K = \frac{d^2}{180} \frac{\varepsilon^3}{(1-\varepsilon)^2} \tag{2}$$

where d is the particle diameter, and ε is the inter-particle volume fraction.

A mechanistic model based on Darcy flow and optimizing liquid and vapor viscous drags is that of Hardee and Nilson:[2]

$$q_d = \frac{\rho_1 g K h_{fg}}{\left(\sqrt{\nu_v} + \sqrt{\nu_1}\right)^2} \tag{3}$$

where ν_v is the vapor kinematic viscosity. A semi-empirical model based on flooding correlations in packed beds is that of Ostensen:[3]

$$q_d = \frac{.90 h_{fg} \sqrt{\rho_1 \rho_v g} \sqrt[4]{K/\varepsilon}}{\left(1 + \sqrt[4]{\rho_v/\rho_1}\right)^2} \tag{4}$$

A mechanistic model (similar to the Hardee-Nilson model) which includes the effects of capillary forces is that of Shires and Stevens:[4]

$$q_d = D \frac{(\rho_l - \rho_v)\ g\ \kappa h_{fg}}{(\sqrt{\nu_v} + \sqrt{\nu_l})^2} \left(1 + \frac{0.90\ \lambda c}{CL}\right) \tag{5}$$

where

$$\lambda_c = \frac{\sigma\sqrt{\varepsilon/\kappa}}{2(\rho_l - \rho_v)g} \tag{6}$$

and where L is the bed depth, σ is the fluid surface tension, $C \approx 1$, and D is empirically determined from dryout measurements.

A very general mechanistic model has been developed by Lipinski.[5] It allows for both laminar and turbulent flow in both the liquid and vapor phases of the fluid. In addition, it includes both gravitational and capillary forces. In the laminar limit it is similar to the model of Shires and Stevens.[4] In the laminar limit and neglecting capillary forces, the model is similar to the Hardee-Nilson[2] model. In the turbulent limit (and again neglecting capillary forces) the model is similar to the flooding model of Ostensen.[3] The equation for dryout is[5]

$$qd = \rho_v h_{fg}\left(\sqrt{V_L^2 + V_T^2} - V_T\right) \tag{7}$$

where

$$V_L = \frac{n}{2\kappa\rho_v}\left(\frac{\nu_v}{(1-1.11\gamma)} + \frac{\nu_l}{\gamma^3}\right)\Bigg/\left(\frac{1}{\rho_v(1-\gamma)^3} + \frac{1}{\rho_l\gamma^3}\right) \tag{8}$$

$$v_T = \sqrt{\frac{\eta(\rho_l - \rho_v)g}{\rho_v^2}(1 + \lambda_c/L) \Bigg/ \left(\frac{1}{\rho_v(1-\gamma)^3} + \frac{1}{\rho_l \gamma^3}\right)} \qquad (9)$$

$$\eta = \frac{d}{1.75}\frac{\varepsilon^3}{1-\varepsilon} \qquad (10)$$

and where γ is varied from 0 to 1 until q_d is maximized.

When the models are compared with published dryout measurements (126 points involving five different fluids and many different particle diameters and bed heights) the Lipinski model appears to fit the data best. This is partly due to the fact that the other models have only a limited range of applicability.

The Lipinski model (Equation 7) and the Dhir-Catton model (Equation 1) will be used to predict dryout conditions for various accident states. Both in-vessel and ex-vessel beds will be considered. As a limiting case, the whole core will be assumed to be in the bed. (For other cases, the results may be scaled according to Equations 1 or 7.) MARCH calculations predict core slump for TMLB' at 273 minutes and vessel failure not much later.[8] For S2D, core slump is predicted at 88 minutes with head failure soon after, and for ABG core slump is predicted at 23 minutes. After an hour past shutdown, the decay heat decreases with the fourth root of time so perturbations in time about these values are insignificant. For the in-vessel case, the debris will be assumed to extend out to the core barrel and the non-uniformity in depth caused by the hemispherical bottom will be neglected. For the ex-vessel case, the debris will be assumed to cover a

5.2-meter (17-foot) square corresponding to the width of the reactor cavity sump (which is 5.2 m by 12 m). In both cases, the bed will be assumed to rest on an adiabatic impermeable base. Interparticle volume fractions in random beds normally range from 0.4 to 0.5, although more extreme packings are certainly possible. A value of $\varepsilon = 0.4$ will be used in the calculations. Under these conditions, a bed with equal parts by volume of urania and structure would have a depth of 2.9 m (9.6 ft) in-vessel and 1.22 m (4.0 ft) ex-vessel. The average particle diameter in a bed can vary widely (from less than 10 μm to over 10 mm) depending on how the fragmentation occurred. This variation can have a significant effect on the dryout heat flux. Thus, the particle diameter required for dryout under the above conditions will be calculated using the three models. Discussion of the likelihood of having those diameters will then follow.

The minimum particle diameters required to prevent dryout are given in Tables 1 and 2. There is large disagreement in the models for the high-power cases, but there is reasonable agreement for later times. The higher-power disagreement is expected since Equation 1 was fit to low-power data and does not possess the proper mechanism for high-power (turbulent) modeling. Thus Equation 1 is believed to be more accurate in those cases. Particle diameters of several millimeters are required for most of the cases.

There are some data on fragmentation of molten core materials in water from steam-explosion experiments.[8-13] Small-scale experiments (involving less than 100 g of molten material) required triggers in order to obtain explosions. The average size after an explosion was usually less than 100 μm.[8] Molten materials which quenched without an explosion sometimes did not fragment.[8] Other experiments typically yielded fragment sizes on the order of a millimeter.[10-12] Larger-scale tests involving about 20 kg of molten core materials are being performed.[13] Explosions have frequently occurred without a trigger. In alumina-thermite tests involving around 13 kg of melt dropping into several hundred kilograms of water at one atmosphere, explosions were obtained in 37 out of 48 tests.[9] (Spontaneous explosions were sometimes suppressed by coating the vessel walls with lard.) Although the explosions tended to scatter the debris, analysis of what remained indicated that the more energetic explosions produced the finer particles with a typical diameter of about 200 μm. This suggests the possibility that even with explosions, larger-scale interactions will produce larger average particle sizes. The non-explosive events produced average particle diameters of around 1.5 mm.

Comparing the fragmentation data with Table 1 suggests that in-vessel beds involving the entire core will most likely dry out unless the particles are produced non-explosively, over an hour has passed since shutdown, and the pressures are high.* Comparing the fragmentation data with Table 2 suggests that total-core ex-vessel beds also will dry out unless they are produced non-explosively and over an hour past shutdown. The fragmentation

*If the particle bed is resting on a highly permeable plate such as the grid support plate, the coolability will be somewhat greater. The increase in the dryout heat flux would be less than 30% with natural circulation.[14] However, with forced convection from below, the dryout heat flux would be greatly increased.

data indicate that explosions (from spontaneous triggers) are likely in large melt drops. However, an explosive event would disperse the fragments over a large area in the ex-vessel case. If the debris were spread over the 65 m^2 (700 ft^2) reactor cavity, the required particle diameter for coolability would be smaller, as shown in Table 3. These diameters are still larger than those expected to be produced. However, in the TMLB' case, the predicted dryout diameter is not too far from that expected, especially if larger melts yield larger particles. This suggests that an ex-vessel particle bed may possibly be coolable in the TMLB' case, either dispersed with small particles or localized with large particles. (This is even more plausible if one considers that the whole core may not be involved.) This would eliminate the concern for basemat penetration in that case.

In summary, whether or not the steady-state coolability of fragmented core materials can be achieved via coolant boiling is of concern only when there is a means of removing the heat from the coolant. (Otherwise the particle bed will boil dry.) These cases include ex-vessel beds in an S2D accident and restoration of AC power in TMLB' and ABG accidents. The particle size required to prevent dryout and probable remelt in a full-core in-vessel bed is about 2 mm (in the TMLB' case) or greater. For ex-vessel beds, it is about one millimeter or greater for undispersed debris and about 0.5 mm or greater for dispersed debris. At present, there is an inadequate data base on large-scale core material fragmentation to determine accurately what size particles and what amount of dispersal would be expected in accident situations.

Table 1. MINIMUM PARTICLE DIAMETERS REQUIRED TO PREVENT DRYOUT IN-VESSEL AT VARIOUS TIMES PAST SCRAM

Time Past SCRAM (min)	Model Equation	Pressure: 1 bar	5 bars	25 bars	100 bars	175 bars
		Particle Diameter (mm)				
23 (ABG)	7	22.6	8.5	4.0	3.6	8.0
	1	1.8	1.8	1.4	1.8	2.8
88 (S2D)	7	11.8	4.8	2.3	2.1	4.2
	1	1.5	1.5	1.2	1.5	2.3
273 (TMLB')	7	6.0	2.9	1.5	1.3	2.4
	1	1.2	1.3	1.0	1.3	1.9

Table 2. MINIMUM PARTICLE DIAMETERS REQUIRED TO PREVENT DRYOUT EX-VESSEL AT VARIOUS TIMES PAST SCRAM (NON-DISPERSED)

Time Past SCRAM (min)	Model Equation	Particle Diameter (mm) Pressure: 1 bar	5 bars
23 (ABG)	7	5.2	2.3
	1	1.1	1.2
88 (S2D)	7	3.2	1.6
	1	1.0	1.0
273 (TMLB')	7	2.1	1.1
	1	0.8	0.8

Table 3. MININUM PARTICLE DIAMETERS REQUIRED TO PREVENT DRYOUT EX-VESSEL (DISPERSED)

Time Past SCRAM (Min)	Model Equation	Pressures:	Particle Diameter (mm) 1 bar	5 bars
23 (ABG)	7		1.76	0.90
	1		0.73	0.73
88 (S2D)	7		1.25	0.66
	1		0.61	0.61
273 (TMLB')	7		0.93	0.52
	1		0.51	0.51

References

1. V. K. Dhir and I. Catton, Study of Dryout Heat Fluxes in Beds of Inductively Heated Particles, NUREG-0262, NRC-7 (1977).

2. H. C. Hardee and R. H. Nilson, Nucl. Sci. and Eng., 63, 119 (1977).

3. R. W. Ostensen, Memo, Sandia National Laboratories, February 12, 1979.

4. G. L. Shires and G. F. Stevens, Dryout During Boiling in Heated Particulate Beds, Atomic Energy Establishment, Winfrith, United Kingdom, AEEM-M1779 (1980).

5. R. J. Lipinski as reported in Advanced Reactor Safety Research Quarterly Report, January-March 1980, Sandia National Laboratories, Albuquerque (1980).

6. J. B. Rivard, Nucl. Tech., 46, 344 (1979).

7. J. D. Gabor, E. S. Sowa, L. Baker, Jr., and J. C. Cassulo, Proc. of ANS Fast Reactor Safety Mtg., Beverly Hills, CA, CONF-740401, 823 (1974).

8. L. D. Buxton and L. S. Nelson, ANS Trans., 27, 663 (1977).

9. L. D. Buxton and W. B. Benedict, Steam Explosion Efficiency Studies, SAND79-1399 (1979), NUREG/CR-0947.

10. M. Amblard, G. Berthoud, C. Lackme, E. Scott, CSNI Specialists' Meeting on Fuel-Coolant Interactions, Norway, September 1970.

11. R. L. Gibby, Reaction of Molten UO_2 with Water, BNWL-362 (1967).

12. Argonne National Laboratories authors, ANL 7176, pp. 90-93 (1966).

13. M. Berman, Light Water Reactor Safety Research Program Quarterly Report, January-March 1979, SAND79-1542 NUREG/CR-0972.

14. R. J. Lipinski as reported in Advanced Reactor Safety Research Quarterly Report, October-December 1979, Sandia National Laboratories, Albuquerque (1980).

U. S. Department of Energy
Office of Nuclear Safety Coordination
Washington, DC 20545
Attn: R. W. Barber

U. S. Department of Energy (2)
Albuquerque Operations Office
P. O. Box 5400
Albuquerque, NM 87185
Attn: J. R. Roeder, Director
Operational Safety Division
D. K. Nowlin, Director
Special Programs Division
For: C. B. Quinn
D. Plymale

Science Applications, Inc. (6)
200 Lomas NW, Suite 600
Albuquerque, NM 87102
Attn: P. B. Bleiweis
W. L. Ferrell

Civil Systems, Inc. (3)
2201 San Pedro Dr. NE
Albuquerque, NM 87110
Attn: J. Tills

4000	A. Narath
4231	J. H. Renken
4231	J. A. Halbleib
4231	J. E. Morel
4400	A. W. Snyder
4410	D. J. McCloskey
4413	W. B. Murfin
4420	J. V. Walker (5)
4421	R. L. Coats
4421	J. E. Gronager
4421	G. W. Mitchell
4421	J. B. Rivard
4422	D. A. Powers
4422	D. W. Varela
4423	P. S. Pickard
4424	M. J. Clauser
4424	M. E. Senglaub
4424	J. A. Sholtis
4424	D. C. Williams
4425	W. J. Camp
4425	W. Breitung
4425	F. Briscoe
4425	R. J. Lipinski (10)
4425	M. Schwarz
4440	G. R. Otey
4441	M. Berman
4441	R. K. Cole, Jr.
4441	M. L. Corradini

DISTRIBUTION:

U. S. Nuclear Regulatory Commission
(380 copies for R7)
Division of Document Control
Distribution Services Branch
7920 Norfolk Avenue
Bethesda, MD 20014

U. S. Nuclear Regulatory Commission (4)
Division of Reactor Safety Research
Office of Nuclear Regulatory Research
Washington, DC 20555
Attn: C. N. Kelber, Assistant Director,
Advanced Reactor Safety Research
R. T. Curtis, Chief
Analytical Advanced Reactor Safety Research, ARSR
M. Silberberg, Chief
Experimental Fast Reactor Safety
R. W. Wright
Experimental Fast Reactor Safety

U. S. Nuclear Regulatory Commission
Division of Project Management
Office of Nuclear Reactor Regulation
Washington, DC 20555
Attn: T. P. Speis
J. K. Long
J. F. Meyer

U. S. Nuclear Regulatory Commission
Division of Site Safety and Environmental Analysis
Office of Nuclear Reactor Regulation
Washington, DC 20555
Attn: R. R. Bellamy

U. S. Nuclear Regulatory Commission
Probabilistic Analysis Staff
Office of Nuclear Regulatory Research
Washington, DC 20555
Attn: R. DiSalvo

Battelle Columbus Laboratories
505 King Avenue
Columbus, OH 43201
Attn: R. S. Penning
R. O. Wooton
P. Cybulskis

Los Alamos National Scientific Laboratory
P. O. Box 1663
Los Alamos, NM 87545
Attn: C. R. Bell
M. Stevenson

DISTRIBUTION: (con't)

4441	J. F. Muir
4442	W. A. Von Riesemann
4443	D. A. Dahlgren
4444	L. D. Buxton
4450	J. A. Reuscher
4451	T. R. Schmidt
4452	M. Aker
5836	L. S. Nelson
8266	E. A. Aas
3141	T. L. Werner (5)
3151	W. L. Garner (3)
	For: DOE/TIC (unlimited Release)
3154-3	R. P. Campbell (25)
	For: NRC Distribution to NTIS