3.6 In-Vessel Fuel-Coolant Interactions

When molten core material (fuel) comes into contact with liquid water (coolant), a variety of different fuel-coolant interactions (FCIs) can occur. The FCIs can range from quiescent boiling to explosive fragmentation of the fuel with rapid steam generation. An explosion caused by the rapid fragmentation of fuel and vaporization of water due to heat transfer from the fragmented fuel is called a steam explosion. If the melt contains unoxidized metals, exothermic metal-water reactions can accompany the fuel coolant interaction, resulting in enhanced energy release and the generation of hydrogen. The nature of the FCI determines the rates of steam and hydrogen production and the potential for damaging the reactor vessel or containment building. Much theoretical and experimental research has been devoted to FCIs over the last three decades. This research is summarized in several review articles.[1,2,3,4,5]

3.6.1 Steam Explosions

Steam explosions occur when heat is transferred from the melt to water on a very short time scale (approximately 1 msec.). Steam explosions have occurred ever since man began to work with molten metals. The first known written record of such an explosion appears in the Canterbury Tales of the 14th century.[6] Destructive steam explosions have occurred in aluminum, steel, and copper foundries; arc-melting facilities; paper mills; granulation plants; and (some believe) Chernobyl.[7,8,9,10,11]

The four major stages of a steam explosion are:

1. Initial *coarse mixing* of melt and water during which heat transfer is generally characterized by stable film boiling (Figure 3.6-1),
2. a *triggering* event that causes local destabilization of film boiling and local fragmentation of melt into small drops, on the order of 0.01 to 0.1 mm in diameter,
3. *propagation* of the region of rapid heat transfer through the coarse mixture, and
4. explosive *expansion* driven by steam at high pressure.

In the absence of a triggering event, a nonexplosive FCI would occur. Coarse mixing would result in some quenching of the melt with associated steam and hydrogen production.

3.6.2 Conditions Affecting Steam Explosions

The probability and magnitude of steam explosions depend on various initial and boundary conditions, including:

- mass, composition, and temp-erature of the molten material,
- water mass, depth, and temperature,
- vessel geometry, degree of confinement, and the presence and nature of flow restrictions and other structures,
- fuel-coolant contact mode, in particular, for melts poured into water, the melt entry velocity and pour diameter,
- the ambient pressure,

- the timing and strength of any external trigger that might be applied (e.g. in an experiment, not a reactor accident).

Intermediate conditions that strongly influence the probability and magnitude of steam explosions include:

- the extent of coarse mixing (drop sizes and surface areas),
- the rate of heat production by the exothermic oxidation of molten metals and partially oxidized materials by the surrounding coolant, and
- the occurrence, timing, and strength of a spontaneous trigger (see below).

During mixing, some of the molten drops may spontaneously fragment into much smaller drops, on the order of 0.01 to 0.1 mm in diameter. This local fragmentation event is generally called a trigger. It may be produced by natural oscillations in the vapor film about the drop leading to fuel-coolant contact, or it may be induced by shock waves from falling objects, contact of the fuel with the bottom surface, entrance of the fuel into a region of colder water, or by turbulence generated in part of the mixing region. If the fragmentation is rapid enough, local shock waves can be produced, which can cause neighboring drops to fragment. If such a chain reaction escalates, a steam explosion can result.

Steam explosions can occur for a variety of high-temperature molten materials including uranium and its oxides. Spontaneous (no external trigger) steam explosions have been observed for aluminum, iron, tin, and associated oxides in all possible contact modes including melt pours, stratified water over melt, and reflooding. High ambient pressure and low water subcooling have been shown to reduce the probability of spontaneous steam explosions at experimental scales; however, explosions can still occur if the necessary triggers are available.

Experimentally measured conversion ratios (the work done divided by the thermal energy available) range from zero to values approaching the thermodynamic limit. Explosion pressures have been measured over the range of tens of bars to 2 kilobars. Steam explosion computer codes have predicted that pressures of many kilobars are possible for strong steam explosions.

Significant rates of hydrogen production have been observed for both explosive and nonexplosive interactions. Much finer fragments produced in explosive interactions can potentially lead to more rapid production of steam and hydrogen. The actual hydrogen production rate, however, is a result of two competing processes. The large surface-to volume ratio of the molten drop tends to increase the rate of heat transfer from the drop to water, but it also tends to increase the rate of exothermic oxidation, which adds energy to the drop and hot hydrogen gas to the vapor film surrounding the drop. The occurrence of a steam explosion as opposed to a nonexplosive interaction is generally thought to favor increased hydrogen production, especially when the melt is metallic as in foundries.

3.6.3 Limitations on In-Vessel FCIs

A rough estimate of the potential for energy release from in-vessel FCIs (excluding Zr oxidation) can easily be computed by calculating the energy that would have to be transferred to water in order to quench the entire core. For example, a typical PWR core might contain 10^5 kg of UO_2 and 2×10^4 kg Zr. Assume that all of this material (plus 10^4 kg Fe to allow for structural material in the melt) is liquefied at 4712°F (2600°C = 2873K), below the UO_2 melt temperature of 5180°F (2860°C = 3133K). The decrease in sensible and latent heat required to quench this melt to 212°F (100°C = 373K), which is the saturation temperature for water at atmospheric pressure is approximately 170 GJ (a steam explosion of 1 to 1.5 GJ could fail the reactor vessel lower head). A 170 GJ steam explosion would require the evaporation of approximately 75,000 kg or 75 m^3 of saturated water at atmospheric pressure.

In reality, the energy transferred from core materials to residual water would be less than 170 GJ for two reasons:

1. The volume of residual in-vessel water would be limited, in the absence of ECC restoration, and

2. lower melt temperatures and/or higher in-vessel pressures, which would be anticipated in most severe accident scenarios, would reduce the temperature difference between molten core materials and residual in-vessel water.

Figure 3.6-2 illustrates the limited capacity for in-vessel FCI energy releases at various pressures in a PWR if the residual water is limited to 29 m^3, which is approximately the volume below the lower core plate of a Westinghouse PWR. Table 3.6-1 shows the corresponding limitations of the mass of core material that could be quenched.[12] In general, BWR lower plenums are larger and hold more water relative to the mass of the core. Considerations with respect to the potential for debris quenching in a BWR lower plenum are discussed in Section 3.7.6.

Reactor vessel lower plenums, particularly in BWRs, contain significant quantities of structural materials as illustrated in Figures 3.1-5 and 3.1-9. Such structures could restrict the volumes of melt and/or water participating in FCIs at a given time. Table 3.6-2 provides some data on features and geometry that characterize these flow restrictions.[13]

It should be noted that the preceding estimates ignore the potential contribution to FCI energy releases associated with oxidizing metallic Zr contained in the melt. As noted in Subsection 3.3, quantities of unoxidized zirconium are likely to be involved in the core-liquefaction processes. Mixing of this metallic phase at high temperatures with the water in the lower plenum would promote rapid oxidation of the zirconium, depending primarily upon the degree to which fragmentation of the melt provides large increases in the interfacial surface area. The heat of reaction for Zr oxidation is approximately 6.5 MJ/kg of Zr reacted. If only 1% of the Zr typically contained in a PWR core (2×10^4 kg) were oxidized during in-vessel FCIs, an additional 1.3 GJ would be released. Regardless of the exact outcome, the addition of reaction energy and liberation of a quantity of hydrogen by the oxidation of zirconium during the melt-water interaction phase seems likely.

3.6.4 In-Vessel FCI Scenarios

In assessing the impact of in-vessel FCIs on accident progression, three alternative scenarios can be postulated:

1. No steam explosion but violent boiling, which may partially or totally quench the core debris, depending on the quantity of water available and the agglomeration of the debris.

2. One or more relatively low-yield steam explosions and nonexplosive quenching until the whole of the molten mass of fuel has been fragmented or all of the water evaporates.

3. A large steam explosion involving a significant fraction of the melt, triggered either spontaneously or by a low-yield steam explosion.

Because of the resultant disruption (and possible dispersal) of internal structures and residual core materials, the occurrence of even a relatively low-yield steam explosion could significantly alter the subsequent progression of damage.

3.6.5 Alpha Mode Containment Failure

Energetically, it is possible that a large in-vessel steam explosion could cause (a) breach of the reactor vessel,[14] or (b) breach of the reactor vessel and generation of containment-failing missiles.[15] Either event would completely alter the course of the accident by causing the immediate ejection of fuel and fission products from the reactor vessel. The second would result in nearly simultaneous venting of the containment. The possibility of these events accounts for the nil minimum duration for Stage 5 given in Table 3.1-1.

The Reactor Safety Study (RSS) first identified the possibility that a large-scale in-vessel steam explosion could result in containment failure. This is commonly referred to as the alpha mode of containment failure. The RSS took the alpha mode failure probability to be 0.01, although the uncertainty in this probability was acknowledged by also providing a pessimistic estimate of 0.1.[12]

Since the RSS, there has been considerable experimental research performed on fuel-coolant interactions at small to intermediate scales (50 mg to 157 kg). Early experiments investigated steam explosion efficiencies and various aspects of triggering in geometries that were open to the atmosphere. This early work is summarized in three review papers.[2,3,4]

A 1984 study showed that conversion ratios less than 5.3% and masses of actively participating molten core less than 5000 kg, as suggested by several mixing models,[16,17] imply an alpha mode failure probability of 0.0001 or less. However, some argued that the possibility of larger conversion ratios or larger masses actively participating could not be excluded and that the uncertainty in the alpha-mode containment failure probability was therefore large.[18]

In 1985 the first NRC-sponsored Steam Explosion Review Group (SERG-1) assessed the probability of alpha mode failure for NUREG-1150.[19] The SERG-1 pessimistic failure probability was 0.1, unchanged from the pessimistic estimate of the RSS. The NUREG-1150 alpha mode failure probabilities are listed in Table 3.6-3.

NRC-funded FCI research after the initial SERG-1 workshop sought to enhance the technical basis of the alpha mode failure estimates given by the experts, and reduce uncertainties in the estimates. Numerous experiments were conducted from 1985 through 1995 in both U.S. and European facilities. A review of these experiments is provided in a recent paper.[5] The experiments demonstrate that steam voiding around hot debris particles causes the mixing region to be depleted of water in part as a result of its vaporization due to rapid melt-to-coolant heat transfer, and, in part due to displacement of remaining water mass away from the interfacial region. Depletion is even more pronounced in the case of adjacent simultaneous pours as occurred through multiple holes in the elliptical flow distributor at TMI-2.

In June 1995 the NRC convened the SERG-2 workshop to reassess the alpha mode failure issue and to evaluate the current understanding of other FCI issues of potential risk significance. As illustrated in Table 3.6-4, all but two of the 11 SERG-2 experts concluded that the alpha mode failure issue is essentially resolved, meaning that this mode of failure is of very low probability, that it is of little or no significance to the overall risk from nuclear power plants, and that further research is not likely to change this conclusion.

The SERG-2 experts based their judgements regarding the likelihood of alpha mode failure largely on experimentally substantiated arguments favoring limits to mixing. There is a consensus among the experts that the triggering process is poorly understood due largely to its inherently random nature. Assumptions regarding triggering under accident conditions tend, therefore, to be conservative. Triggering is postulated at the worst time during premixing, leading to trigger amplification or shock wave propagation.

It should be emphasized, however, that in experiments performed with prototypic reactor melts interacting with saturated to subcooled water at an ambient pressure of nominally 0.1 MPa, only one or two cases exhibited weak steam explosions either at high melt-to-coolant volume ratios or at high subcooling, and only when an external trigger was used. In contrast, many more cases using iron-alumina thermite and iron oxide as melt simulants produced strong steam explosions at a wide range of melt-to-coolant volume ratios, much lower subcooling to almost saturated conditions, with or without trigger.

3.6.6 Vessel Breach by an In-Vessel Steam Explosion and Related Issues

The steam-explosion energy required to fail the bottom head of a PWR has been estimated to be between 1 GJ and 1.5 GJ. That is, a steam explosion need not involve large quantities of melt or water in order to yield such energies. In one study of PWR in-vessel steam explosions, failing the bottom head by an in-vessel steam explosion was found to be much more likely (probability of 0.2 versus 0.0001) than alpha mode failure.[20] Figure 3.6-4 illustrates this mode of vessel breach, which has the potential for driving particulate debris from the reactor cavity, resuspending radioactive aerosols previously plated out within the reactor coolant system, and forming additional aerosols during the explosion.

Steam explosion research has been conducted at several research facilities to address several issues including the possibility of lower head failure due to an in-vessel steam explosion, the potential for significant structural damage due to a steam explosion in the reactor cavity (see Section 4.3), pressure suppression effects on triggering, and effects of melt composition and melt-coolant-confinement geometry on both triggering and energetics of steam explosions. Table 3.6-5 provides summary information on four current steam explosion research facilities.[5]

The current level of understanding of the propagation phase of a steam explosion is adequate for estimating the net energy transfer to the coolant and hence, estimating the alpha mode failure probability. Understanding of shock loading of lower head and reactor cavity structures requires more rigorous treatment for which detailed two or even three-dimensional propagation phase models may be required.

3.6.7 Impact of Melt Discharge from Vessel

Four modes of discharge of core materials from the vessel can be postulated:

1. Massive failure of the vessel by an in-vessel explosion,
2. a pressure-driven melt jet,
3. gravity-driven pour of a large molten mass,
4. continuous dripping of core materials not involved in the initial release.

These modes of melt discharge are depicted in Figures 3.6-4 through 3.6-7.

The mode of vessel breach can strongly influence the timing and nature of potential loads imposed on containment. In 1984, the NRC sponsored Containment Loads Working Group identified the fact that pressurized dispersal of high-temperature melt into containment at the time of vessel breach (Figure 3.6-5), could result in rapid direct heating and exothermic chemical reactions within the containment atmosphere and pose a severe threat to containment integrity. On the other hand, if the vessel is depressurized, molten material would simply flow into the reactor cavity by gravity (Figure 3.6-6), although if water were present in the reactor cavity significant loads on containment could result from ex-vessel fuel coolant interactions or from the additional hydrogen generated in such interactions. In general, BWR containment drywells are relatively small, and, hence, special procedures are provided to assure that the reactor vessel would be depressurized under severe accident conditions.

The initial geometry and potential for cooling of ex-vessel debris, as well as the nature of interactions between core materials and concrete, are strongly influenced by the mode of vessel breach. The mode of melt discharge into containment also has a strong influence on the resulting concentrations of fission products, particularly in aerosol form, in the containment. Ex-vessel phenomena are discussed in Chapter 4.

Following either a pressurized ejection or a gravity-driven pour of melt from the vessel, a significant fraction of core materials may remain unmelted in the core region. Without coolant, much of this material may subsequently melt and drop out of the vessel in small amounts over a period of hours. This mode of discharge is illustrated in Figure 3.6-7. If there is water below the vessel, the dripping mass may prolong ex-vessel fuel-coolant or core-concrete interactions. If the hot leg or surge line had failed earlier natural circulation could be established with flow from the reactor cavity up through the reactor vessel and out the failed pipe. The ingress of air from containment following vessel breach could cause additional expthermic oxidation of hot in-vessel debris. This would, in turn, lead to additional releases of radionuclides to containment. All such possibilities would affect the magnitude of the radiological release given late containment failure.

Table 3.6-1 Fractions of core mixture* that can be quenched in below-core water for a typical PWR**

	Saturated Water Pressure			
	Atmospheric	800 psia (5.5 MPa)	1595 psia (11 MPa)	2465 psi (17 MPa)
ΔT = 2700°F (1500°C) No Freeze	0.79	0.44	0.31	0.17
ΔT = 3600°F (2000°C) No Freeze	0.59	0.33	0.23	0.13
ΔT = 4500°F (2500°C) Freeze	0.37	0.21	0.14	0.08

*10^5 kg UO_2 + 2×10^4 kg Zr + 10^4 kg steel
**in 29 m^3 of water

Table 3.6-2 Lower plenum features of a Westinghouse PWR

Feature	Approx. Thickness (mm)	Water Volume to Next Feature (m^3)	Energy to Evaporate Water (GJ)**
Lower Core Plate	50	6.6	4.6
Diffuser Plate	37	14.1*	9.8
Bottom Support Plate	220	7.7*	5.4
Reactor Vessel Bottom	132	0	--

* Ratio of these two volumes approximate; sum (21.8 m^3) is volume of lower hemisphere.
** Based on a pressure of 2500 psia (17.2 MPa).

Table 3.6-3 NUREG-1150 alpha mode failure probabilities

	Plant	System Pressure	Lower Bound	Mean	Upper Bound
BWRs	Grand Gulf	High Low	0 0	1.0×10^{-3} 1.0×10^{-2}	0.1 1.0
	Peach Bottom	High Low	1.0×10^{-8} 1.0×10^{-7}	1.0×10^{-3} 1.0×10^{-2}	0.1 1.0
PWRs	Sequoyah	High Low	0 0	8.5×10^{-4} 8.5×10^{-3}	0.1 1.0
	Surry	High Low	0 0	9.1×10^{-4} 9.1×10^{-3}	0.1 1.0

Table 3.6-4 Alpha mode failure probability estimates (given a core melt accident)

Participant	SERG-1[a] (1985)	SERG-2 (1995)	View on Status of Alpha Mode Failure Issue
Bankoff	$< 10^{-4}$	$< 10^{-5}$	Resolved from risk perspective
Berthoud	--	$< 10^{-3}$	No statement on resolution
Cho	< RSS[a]	$< 10^{-3}$	Resolved from risk perspective
Corradini	10^{-4} - 10^{-2}	$< 10^{-4}$	Resolved from risk perspective
Fauske	Vanishingly small	Vanishingly small	Resolved from risk perspective
Fletcher	--	$< 10^{-4}$	Resolved from risk perspective
Henry	--	Vanishingly small	Resolved from risk perspective
Jacobs	--	Probably low likelihood	Not resolved from risk perspective
Sehgal	--	$< 10^{-2}$	Resolved from risk perspective
Theofanous	$< 10^{-4}$	Physically unreasonable	Resolved from risk perspective
Turland	--	$< 10^{-3}$	Resolved from risk perspective

[a] Reactor Safety Study (RSS) best estimate 10^{-2}; NUREG-1150 consensus estimate 10^{-2} at low reactor coolant system pressure, 10^{-3} at high reactor coolant system pressure.

Table 3.6-5 Fuel coolant interaction experimental facility characteristics

Facility	FARO	KROTOS	WFCI	ZREX
Location	Joint Research Center, Ispra		The University of Wisconsin	Argonne National Laboratory
Areas of Interest	Premixing, quenching, propagation, energetics, and debris coolability		Conditions favoring and suppressing energetic FCI	Chemical augmentation of FCI due to metals in the melt
Test Section Diameter (cm)	4.7 - 15	0.95 - 2.0	0.87 - 2.0	1.0
Melt Jet Diameter (cm)	1	0.3 - 0.5	0.3	0.25 - 0.5
Water Depth (cm)	50 - 200	1000	1000	1000
Pressure (MPa)	0.1 - 5.0	0.1 - 1.0	0.1	0.1
Melt	UO_2-ZrO_2 w/ & w/o Zr and stainless steel (SS)	UO_2-ZrO_2 or Al_2O_3	Sn, FeO, or Fe_3O_4	Zr w/ or w/o ZrO_2
Melt Mass (kg)	18 - 250	1.4 - 6.0	0.8 - 4.5	0.2 - 1.0

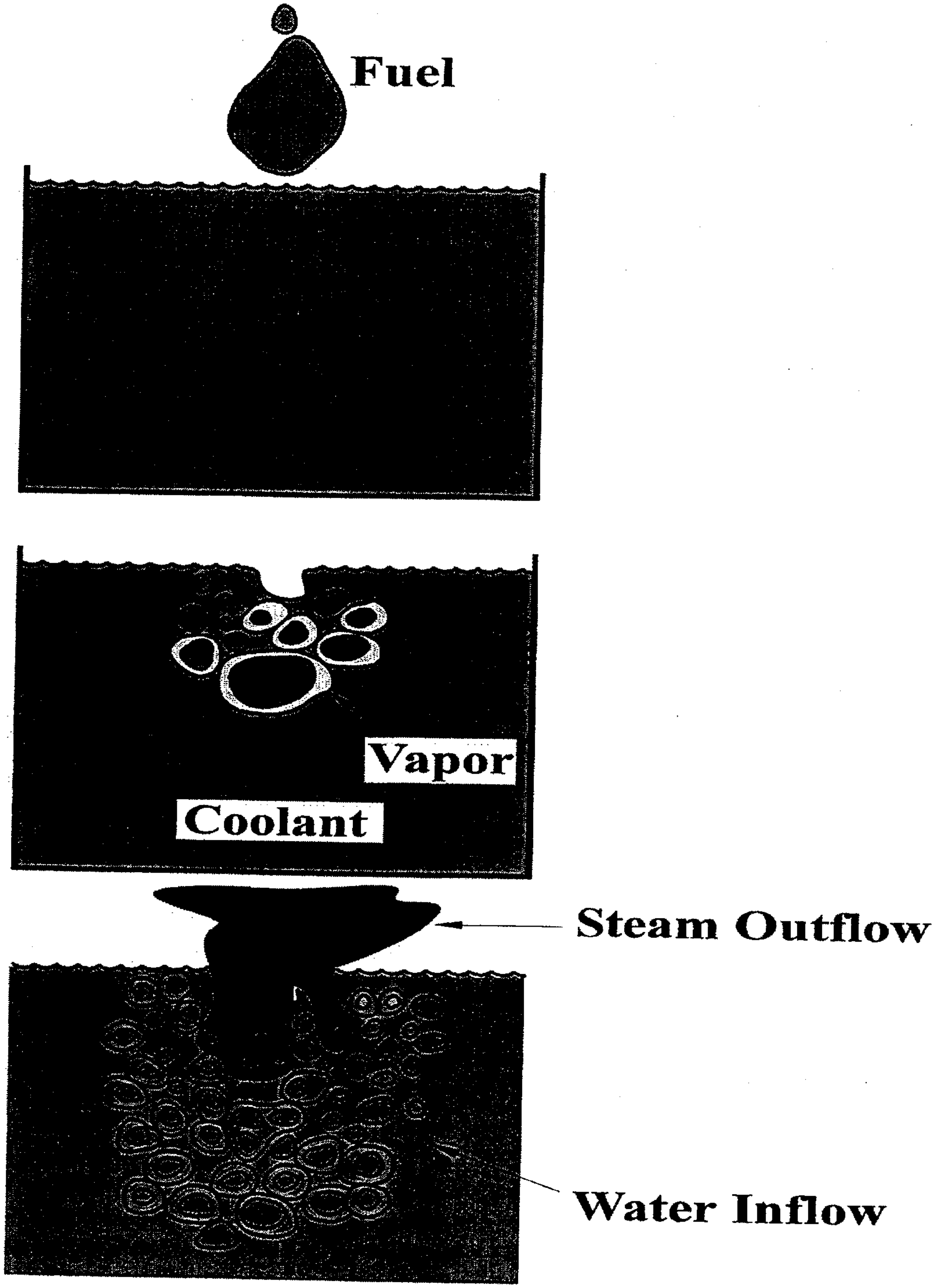

Figure 3.6-1 Progression of fuel-coolant mixing

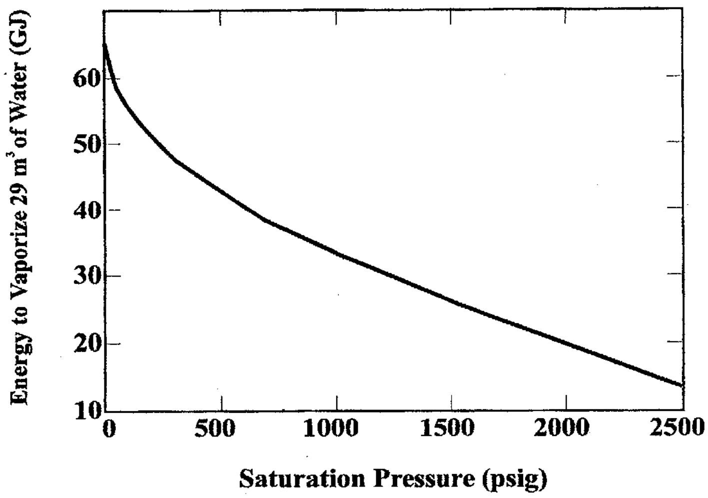

Figure 3.6-2 Energy required to vaporize 29 m^3 for water versus saturation pressure

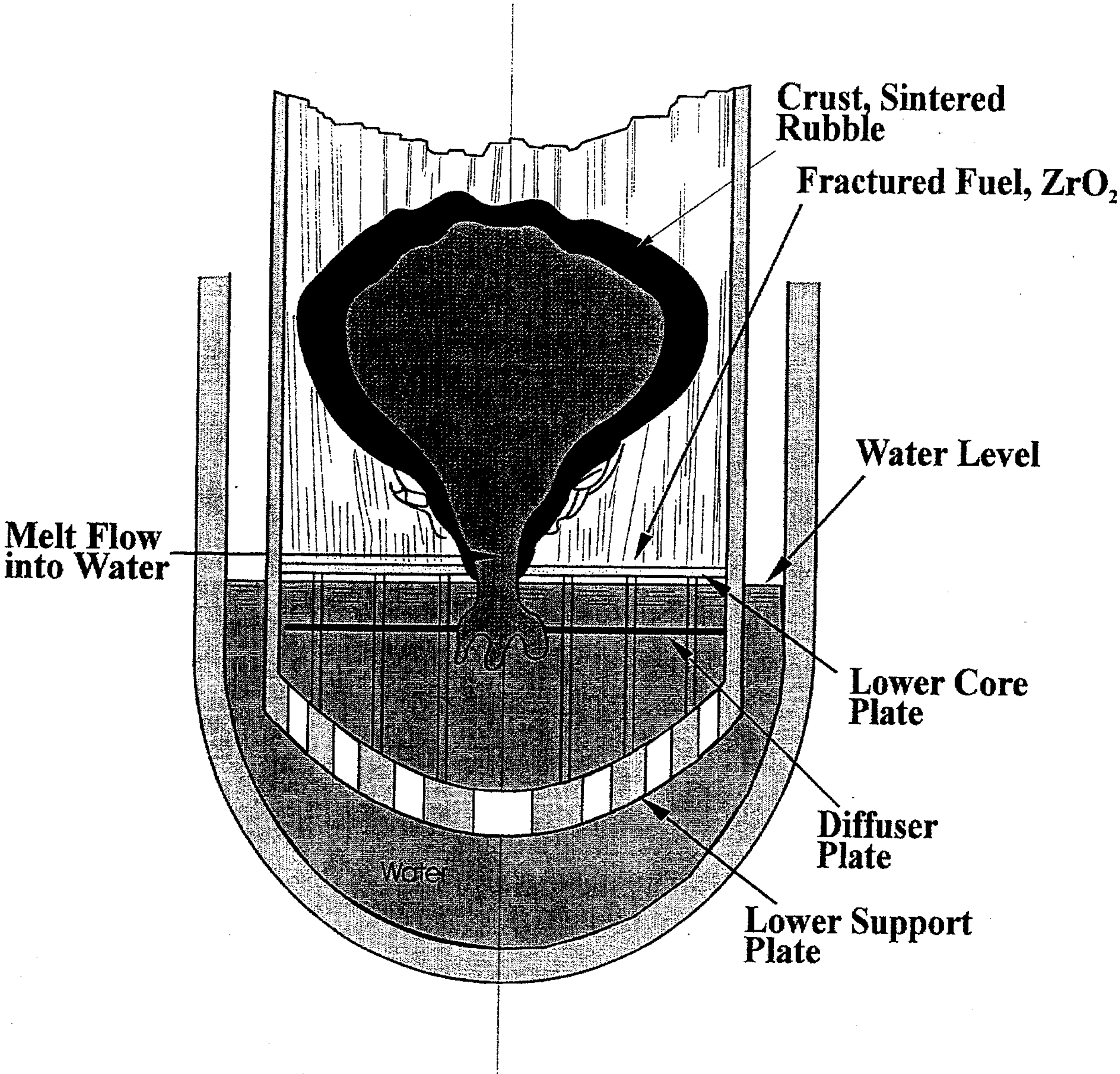

Figure 3.6-3 Melt pour into lower plenum by failure of core plate

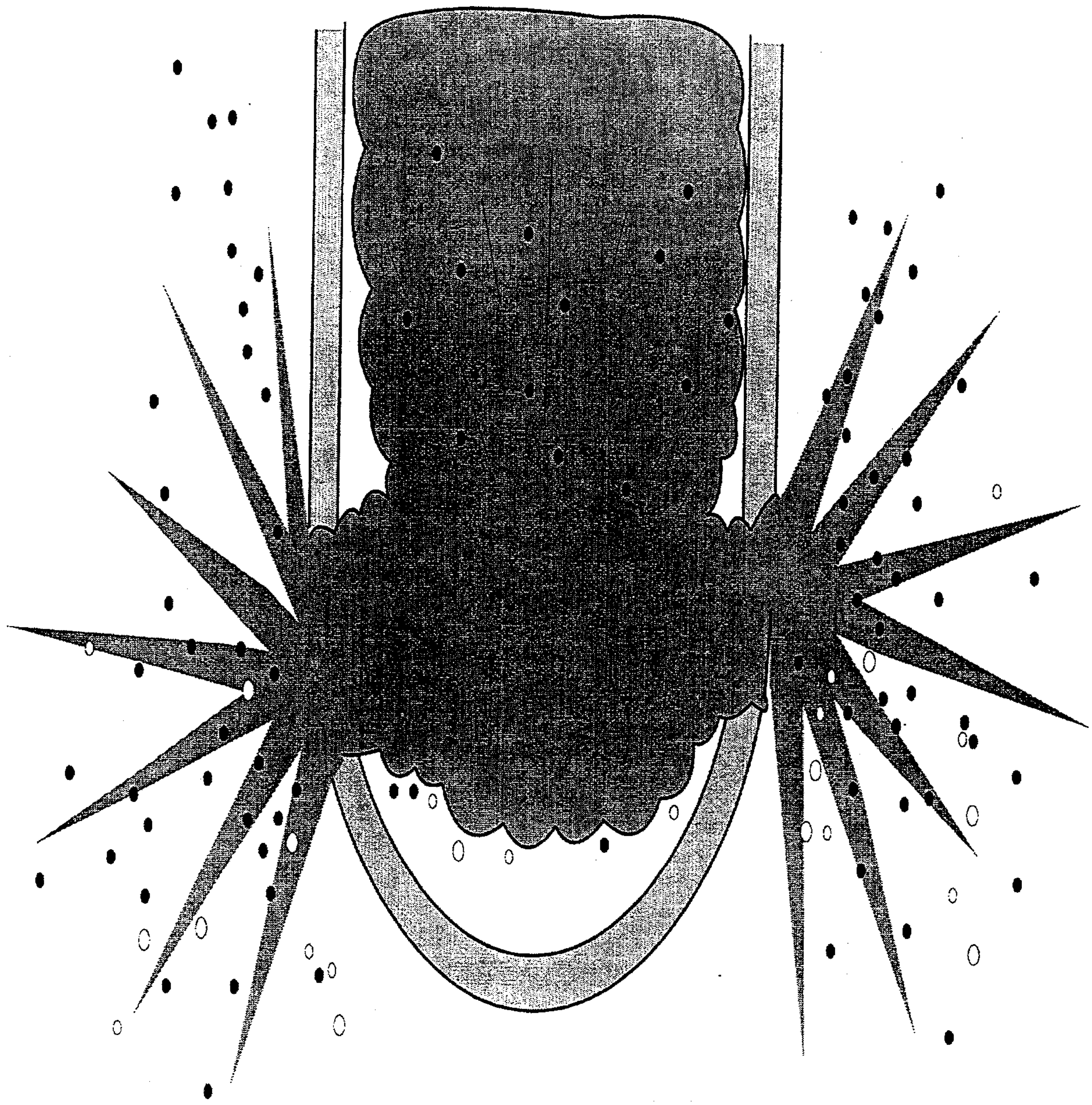

Figure 3.6-4 Vessel failure from steam explosion

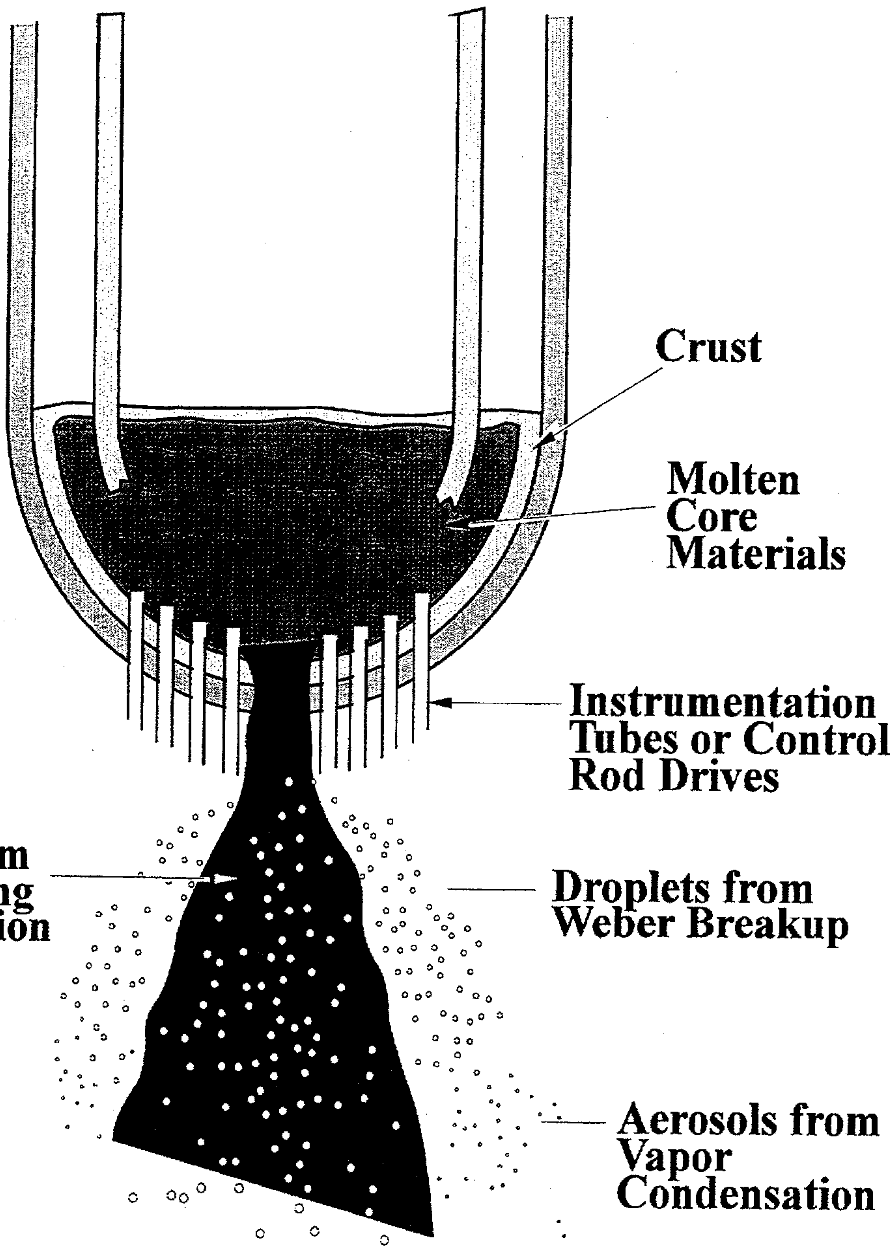

Figure 3.6-5 High pressure melt release from bottom of reactor vessel

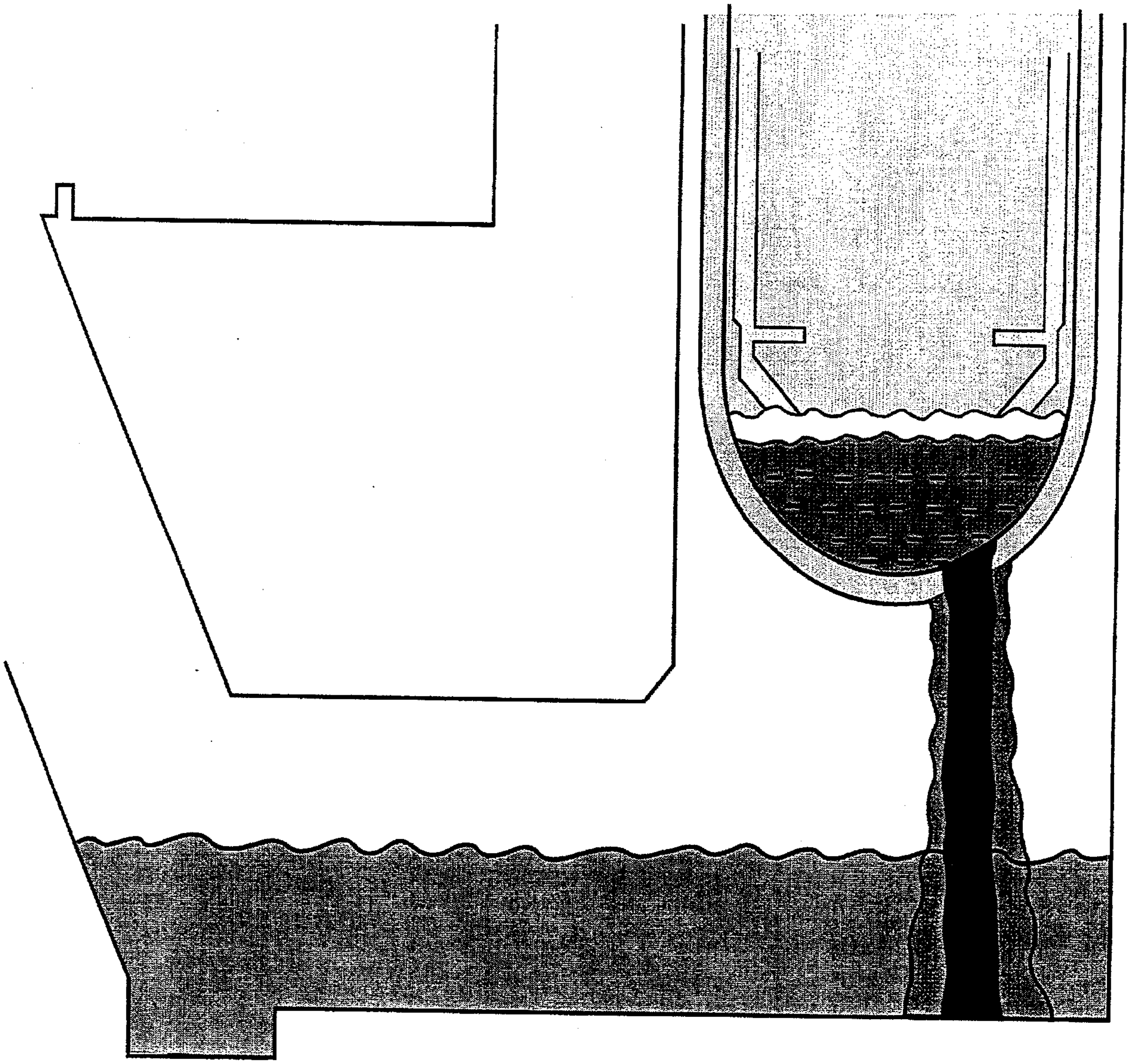

Figure 3.6-6 Low pressure melt release from bottom of reactor vessel

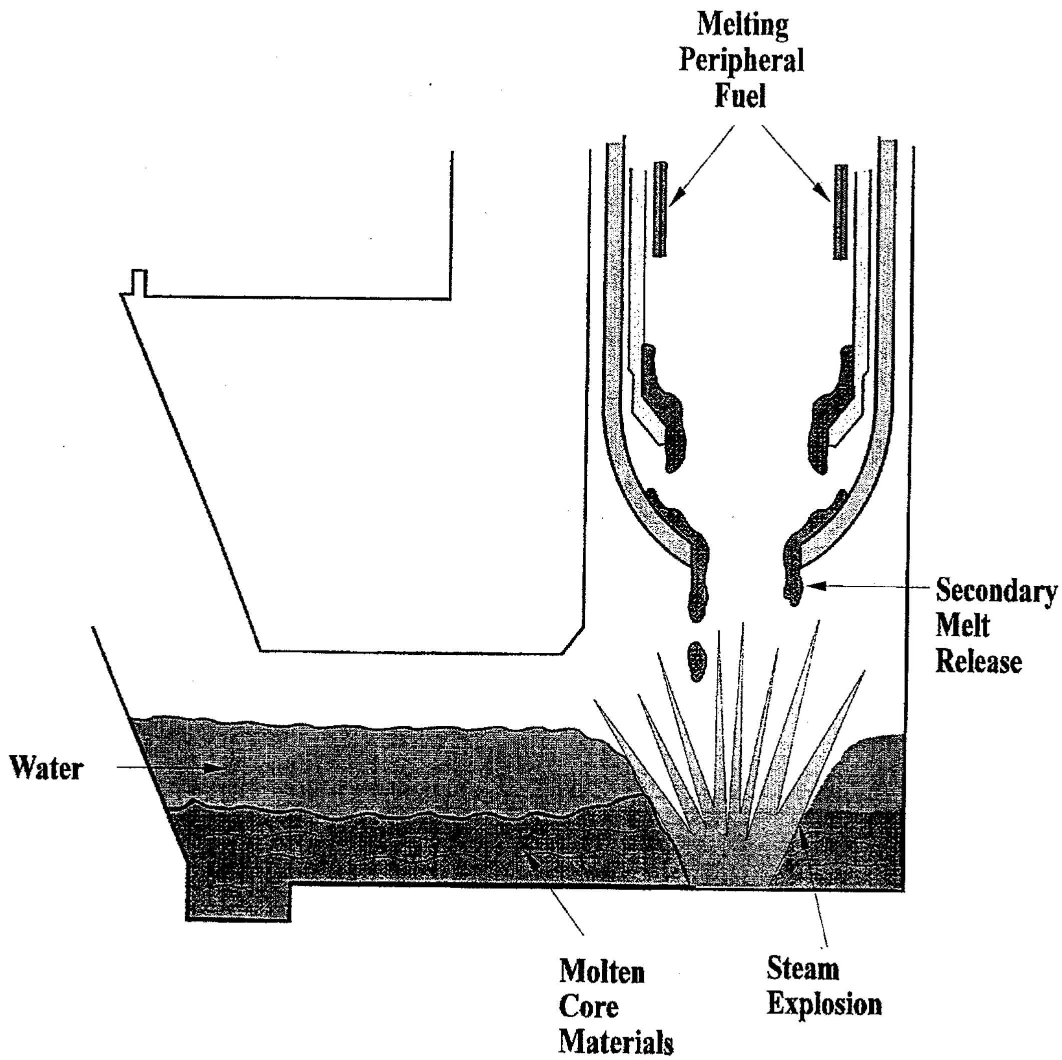

Figure 3.6-7 Secondary melt release in a Zion-type PWR reactor cavity

References for Section 3.6

1. R. C. Reid, "Rapid Phase Transitions from Liquid to Vapor," *Advances in Chemical Engineering*, 12, pp. 105-208, 1983.

2. M. L. Corradini, et al., "Vapor Explosions in Light Water Reactors: A Review Theory and Modeling," *Progress in Nuclear Energy*, 22(1), pp.1-117, 1988.

3. M. L. Corradini, "Vapor Explosions: A Review of Experiments for Accident Analysis," *Nuclear Safety*, 32(3), pp. 337-362, 1991.

4. D. F. Fletcher, "A Review of the Available Information on the Triggering Stage of a Steam Explosion," *Nuclear Safety*, 35(1), pp. 36-57, 1994.

5. S. Basu and T. P. Speis, "An overview of fuel-coolant interactions (FCI) research at NRC," *Proceedings of the 23rd Water Reactor Safety Information Meeting*, Vol. 2, pp. 187-210, October 23-25, 1995.

6. G. Chaucer, "The Canon's Yeoman's Tale," in *Canterbury Tales*, Garden City Publishing Company, Inc., Garden City, NewYork, 1934.

7. M. Berman, "Thermodynamic and Fluid-Dynamic Modelling of Two-Phase Propagating Explosions," *Workshop on the Causes and Prevention of Melt-Water Interactions*, Sandia National Laboratories, Albuquerque, New Mexico, July 29, 1988.

8. M. Berman et al., "Chernobyl: Where Do We Go From Here," Discussions of Steam Explosions at Chernobyl, *Proceedings of the Conference -by-Computer*, Nuclear Publications, McGraw-Hill, New York, New York, September 29-October 17, 1986.

9. W. Sweet, "Chernobyl, What Really Happened," *Technology Review*, pp. 43-52, July, 1989.

10. F. Reisch, "Chernobyl - the Initiating Event?" *Nuclear News*, December 1987.

11. A. W. Cronenberg, "Recent Developments in the Understanding of Energetic Molten Fuel-Coolant Interactions," *Nuclear Safety*, 22(3), pp. 319-337, May-June 1980.

12. J. B. Rivard et al., "Identification of Severe Accident Uncertainties," NUREG/CR-3440, SAND83-1689, September 1984.

13. J. B. Rivard, "Review of In-Vessel Meltdown Models," NUREG/CR-1493, SAND80-0455, July 1980.

14. J. H. Gittus et al., "PWR Degraded Core Analysis," ND-R-610(S), United Kingdom Atomic Energy Authority, Springfields, UK, 1982.

15. U.S. Nuclear Regulatory Commission, "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, NUREG-75/014, October 1975.

16. T. G. Theofanous and M. Saito, "An Assessment of Class 9 (Core-Melt) Accidents for PWR Dry Containment Systems," *Nuclear Engineering Design*, pp. 301-332, 1981.

17. M. L. Corradini and G. A. Moses, "A Dynamic Model for Fuel-Coolant Mixing," *Proceedings from the International Meeting on LWR Severe Accident Evaluation*, Cambridge, Massachusetts, August 1983.

18. M. Berman, "Molten-Core Coolant Interactions Program," *Proceedingsfrom the 12th Water Reactor Safety Research Information Meeting*, 1984.

19. U.S. Nuclear Regulatory Commission, "A Review of the Current Understanding of the Potential for Containment Failure Arising from In-Vessel Steam Explosions," NUREG-1116, 1985.

20. M. Berman, D. V. Swenson, and A. J. Wickett, "An Uncertainty Study of PWR Steam Explosions," NUREG/CR-3369, SAND83-1438, May 1984.