Issue 157: Containment Performance (Rev. 1)

The results of ongoing staff-sponsored research which culminated in the assessment of risk at five U.S. nuclear

reactors (NUREG-1150[1]) indicated that, for the Peach Bottom MARK I containment, the core-melt probability was relatively low. However, it also indicated that the containment could be severely challenged if a large

core-melt occurred. Therefore, it was decided to examine MARK I plants for potential plant and containment modifications to improve containment performance. Subsequently, this examination was expanded to include all other types of containment: MARK II, MARK III, Ice Condenser, and Dry. These studies were conducted under the Containment Performance Improvement (CPI) program, the results of which are documented by containment type below. Thus, based on the staff's evaluations of each of the five types of containments, this issue was

RESOLVED. In an RES evaluation,[2] it was concluded that consideration of a 20-year license renewal period did not affect the resolution.

(1) MARK I CONTAINMENTS DESCRIPTION

Historical Background

At the time of this evaluation in February 1992, there were 24 BWRs with MARK I containments at 17 sites. The

staff issued a plan in December 1987 to address the performance of MARK I containments (SECY-87-297).[3] This plan involved identification and narrowing of issues, focusing related research, and assessing whether improvements were justified.

Safety Significance

The containment represents the final barrier to the release of radioactive material to the environment. GDC 16 requires a leaktight containment and its failure, especially early in a severe accident, has the potential for a significant release of radioactivity to the environment.

Possible Solutions

Potential containment and plant improvements in the following six areas were evaluated to determine potential benefits in terms of reducing the core-melt frequency, containment failure probability, and offsite consequences:

(1) hydrogen control; (2) alternate water supply for reactor vessel injection and containment drywell sprays; (3) containment pressure relief capability (venting); (4) enhanced RPV depressurization system reliability; (5) core debris controls; and (6) emergency procedures and training.

Hydrogen Control: Although MARK I containments are required to be operated with an inerted atmosphere, plant TS permit deinerting to commence 24 hours prior to shutdown and do not require inerting to be completed until 24 hours after startup, in order to permit personnel access. Also, in the event of a severe accident such as a long-term station blackout, a concern was expressed that loss of control of

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[1] NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Vol. 1) December 1990, (Vol. 2) December 1990, (Vol. 3) January 1991.

[2] Memorandum for W. Russell from E. Beckjord, "License Renewal Implications of Generic Safety Issues (GSIs) Prioritized and/or Resolved Between October 1990 and March 1994," May 5, 1994. [9406170365]

[3] SECY-87-297, "MARK I Containment Performance Program Plan," U.S. Nuclear Regulatory Commission, December 8, 1987. [8803080354]

the valves and containment leakage could eventually lead to containment deinerting.

Two potential improvements with regard to H_2 control were evaluated: elimination of the the 24-hour deinerted periods and providing a backup supply of nitrogen. Since the probability of a severe accident occurring during either of the two 24-hour deinerted periods is small compared to the probability of accident occurrence during normal operations, eliminating this time of deinerting would not significantly reduce risk.

Reactor pressure is anticipated to increase during a severe accident, releasing steam and non-condensible gases into containment. This will increase containment pressure, preventing ingress of air. Therefore, the containment atmosphere would not become deinerted for an extended period of time. Since offsite supplies of nitrogen could be readily obtained during this period, an onsite backup supply of nitrogen would not significantly reduce risk.

Alternate Water Supply for Reactor Vessel Injection and Containment Drywell Sprays: An important proposed improvement was to employ a backup or alternate supply of water and a pumping capability independent of normal and emergency AC power. By connecting this source to the low pressure residual heat removal system as well as to the existing drywell sprays, water could be delivered either into the reactor vessel or to the drywell by use of an appropriate valve arrangement.

An alternate source of water injection into the reactor vessel would reduce the likelihood of core-melt due to station blackout or loss of long-term decay heat removal, as well as provide significant accident management capability.

Water for the drywell sprays would also provide significant mitigative capability to cool the containment steel shell to delay or prevent its failure and either to cool core debris or, if the debris configuration is not coolable, to scrub particulate fission products through an overlying water pool.

A review of some MARK I facilities indicated that most plants have one or more diesel-driven pumps which could be used to provide an alternate water supply. The flow rate using this backup water system may be significantly less than the design flow rate for the drywell sprays. The potential benefits of modifying the spray headers to ensure a spray were compared to having the water run out of the spray nozzles. The result of this comparison was that removal of airborne fission products in the small crowded volume in which the sprays would be effective did not change sufficiently to warrant modifications to the spray nozzles.

Containment Pressure Relief Capability (Venting): Venting of the containment is currently included in BWR emergency operating procedures. The vent path external to existing containment penetrations typically consists of a ductwork system which has a low design pressure of only a few pounds per square inch gauge (psig).

Venting under high pressure severe accident conditions would fail the ductwork, release the containment atmosphere into the reactor building, and potentially contaminate or damage equipment needed for accident recovery. In addition, with the existing hardware and procedures at some plants, it may not be possible to open or close the vent valves for some severe accident scenarios.

The staff concluded that venting, if properly implemented, could have a significant benefit on plant risk. However, venting via a sheet metal ductwork path, as currently implemented at some MARK I plants, would be likely to greatly hamper or complicate post-accident recovery activities, and was therefore viewed by the staff as yielding reduced improvements in safety. The capability to vent has been recognized as important in reducing risk

from operation of MARK I plants for loss of long-term decay heat removal (TW) events, provided the potential downsides of using existing hardware are corrected. Controlled venting can prevent the failure of ECCS pumps from both inadequate NPSH and re-closure of the ADS valves.

A hard pipe vent and vent valves capable of withstanding the anticipated severe accident pressure loadings would eliminate the problems with operating the vent system during a severe accident. The vent

isolation valves should be remotely operable from the control room and should be provided with a power supply independent of normal or emergency AC power. Other changes, such as raising the RCIC turbine back-pressure setpoint, may also be desirable and could be considered. Venting capability, in conjunction with proper operating procedures and other improvements discussed in this item, would greatly reduce the probability of core-melt due to TW and station blackout sequences.

Given a core-melt accident, venting of the wetwell would provide a scrubbed vent path to reduce release of particulate fission products to the environment. Venting has been estimated to reduce the likelihood of late containment overpressure failure and to reduce offsite consequences for severe accident scenarios in which the containment shell does not fail for other reasons. Failure of the shell due to core debris attack (shell melt- through) would reduce the benefits from venting in that it would release fission products directly into the reactor building.

Inadvertent venting could result in the release of normal coolant radioactivity to the environment even when core degradation is averted or vessel integrity maintained. Measures to reduce the probability of inadvertent venting should be considered in the vent design.

Enhanced RPV Depressurization System Reliability: The ADS consists of safety relief valves which can be remotely operated to depressurize the RCS. Actuation of the ADS valves requires DC power. In an extended station blackout after station batteries have been depleted, the ADS would not be available and the reactor would re-pressurize. With enhanced RPV depressurization system reliability, depressurization of the RCS would have a greater degree of assurance. A major benefit of enhanced RPV depressurization reliability would be to significantly reduce the likelihood of high pressure severe accidents, such as from short-term station blackout. Together with a low pressure alternate source of water injection into the reactor vessel, another benefit would be to reduce the likelihood of low pressure severe accidents such as from long-term station blackout.

An additional benefit is in the area of accident mitigation. Reduced reactor pressure would reduce the possibility of core debris being expelled under high pressure, given a core-melt and failure of the RPV. Enhanced RPV

depressurization system reliability would also delay containment failure and could reduce the quantity and type of fission products ultimately released to the environment.

In order to increase reliability of the RPV depressurization system, assurance of electrical power beyond the requirements of existing regulations may be necessary. In addition, performance of the cables needs to be reviewed for temperature capability during a severe accident.

Core Debris Controls: Core debris controls, in the form of curbs in the drywell or curbs or weir walls in the torus room under the wetwell, have been proposed in the past to prevent containment shell melt-through or to retain sufficient water to permit fission product scrubbing. However, the technical feasibility of such controls has not been established and the design and installation costs as well as ORE during installation could be significant.

There is a growing consensus that water in the containment (from an alternate supply to the drywell sprays) may help mitigate risk by fission product scrubbing and possibly by preventing or delaying containment shell melt-

through by core debris. Research is continuing under the SARP[4,5] to confirm and help quantify these initial conclusions.

Emergency Procedures and Training: A major element of the MARK I containment performance improvement evaluation involves emergency procedures and training. Current EOPs are symptom-based procedures that originated from requirements of TMI Action Plan Item I.C.1. Plant-specific EOPs are generally implemented based on generic Emergency Procedure Guidelines (EPG) developed by the BWR Owners' Group. As part

of the balanced approach to examining potential MARK I plant improvements, both the generic EPGs and the plant-specific implementation of EOPs and training have been examined.

NRC reviewed and approved Revision 4 of the BWR Owners' Group EPGs (NEDO-31331, Rev. 4, March 1987) which represented a significant improvement over earlier versions in that they continued to be based on symptoms, were simplified, and all open items from previous versions were resolved. The BWR EPGs extend well beyond the design bases and include many actions appropriate for severe accident management.

The improvement to EPGs is only as good as the plant-specific EOP implementation and the training that operators receive on use of the improved procedures. The staff's SER[6] encouraged licensees to

implement Revision 4 of the EPGs and reiterated the need for proper implementation and training of operators. Implementation of the EPGs has been voluntary, but was strongly recommended in the SER.

Accelerated Rule Implementation (ARI) for Station Blackout: In addition to the above six areas examined by the staff for containment and plant improvements, the staff evaluated accelerated implementation of the Station Blackout Rule (10 CFR 50.63). As indicated below, the staff's cost-benefit analysis found accelerated implementation to be cost-effective.

PRIORITY DETERMINATION

Table 3.157-1 gives a summary of the staff's cost-benefit analysis. The ranges for Alternatives 3 through 6 are due to the effects of TW frequency and two installation cost estimates. The detailed cost-benefit

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[4] NUREG-1365, "Revised Severe Accident Research Program Plan," U.S. Nuclear Regulatory Commission, August 1989, (Rev. 1) December 1992.
[5] SECY-91-316, "Status of Severe Accident Research," U.S. Nuclear Regulatory Commission, October 7, 1991. [9110160271]
[6] Letter to D. Grace (BWR Owners Group) from A. Thadani (U.S. Nuclear Regulatory Commission), "Safety Evaluation of "BWR Owners" Group"Emergency Procedure Guidelines, Revision 4," NEDO-31331, March 1987," September 12, 1988. [8809190198]

analysis is given

in the enclosures to SECY-89-017.[7]

Based on the cost-benefit analysis, the staff recommended that the Commission issue an order to accomplish

Alternative 6. In response to the staff's recommendation, the Commission requested[8] that the staff do the following: (1) expedite rule implementation for Station Blackout; (2) request licensees to consider the ADS and alternate water supply modifications as part of their IPEs; and (3) approve hardened vents for licensees who propose to install them and perform a backfit analysis for licensees who do not propose to install them.

TABLE 3.157-1

Alternative	Action	Man-Rem/$M
	Cost-Benefit of Alternatives	
1	Do Nothing	0
2	ARI	990
3	ARI and Venting	4,060 to 9,630
4	ARI and ADS	72 to 290
5	ARI and Alternate Water Supply	62 to 190
6	ARI, Venting, ADS, and Alternate Water Supply	1,810 to 14,400

CONCLUSION

The staff issued Supplement 1 to Generic Letter 88-20[9] to request that licensees with MARK I containments consider the ADS and alternate water supply modifications as part of their IPEs. Also, the staff issued Generic

Letter 89-16[10] to indicate that the staff would approve hardened vents for MARK I licensees who proposed to install them. Thus, this item was RESOLVED.

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[7] SECY-89-017, "MARK I Containment Performance Improvement Program," U.S. Nuclear Regulatory Commission, January 23, 1989. [8903090205]
[8] Memorandum for V. Stello from S. Chilk, "SECY-89-017"MARK I Containment Performance Improvement Program," July 11, 1989. [8907270013]
[9] Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities from U.S. Nuclear Regulatory Commission, "Individual Plant Examination for Severe Accident Vulnerabilities—10 CFR § 50.54(f), (Generic Letter No. 88-20)," November 23, 1988 [ML031150465], (Supplement 1) August 29, 1989 [8908300001], (Supplement 2) April 4, 1990 [ML031200551], (Supplement 3) July 6, 1990 [ML031210418], (Supplement 4) June 28, 1991 [ML031150485], (Supplement 5) September 8, 1995.
[10] Letter to All Holders of Operating Licenses for Nuclear Power Reactors with Mark I Containments from U.S. Nuclear Regulatory Commission, "Installation of a Hardened Wetwell Vent (Generic Letter No. 89-16)," September 1, 1989. [8909010375]

(2) MARK II CONTAINMENTS DESCRIPTION

Historical Background

At the time of this evaluation in February 1992, there were 9 BWRs with MARK II containments located at 6 sites. The geometry of the reactor pedestal support and downcomers has a major impact on the performance of the containment, if the reactor vessel fails in a severe accident. Three major variations in geometry exist.

Safety Significance

The containment represents the final barrier to the release of radioactive material to the environment. GDC 16 requires a leak-tight containment and its failure, especially early in a severe accident, has the potential for a significant release of radioactivity to the environment.

Possible Solutions

Based on PRA insights, MARK II containment vulnerabilities and potential improvements that have been investigated are similar to those investigated for MARK I containments.

Staff-recommended MARK I improvements, other than the hardened vent, are being investigated under the IPE Program for MARK I plants. It is appropriate that MARK II licensees also investigate these same improvements on a plant-specific basis under the IPE Program, since they may also be generally applicable to these plants.

However, less definitive conclusions have been reached regarding the need for improved venting of MARK II containments. The primary benefit from venting remains the same as for MARK I plants,i.e., the prevention of core-melt for loss of long-term decay heat removal (TW) sequence. However, some additional considerations come into play for MARK II plants. Some MARK II plants may already have the capability to vent through a hardened pipe. In addition, due to the larger volume of the MARK II containments, the time available for operator recovery actions during a TW sequence may be longer. Thus, the likelihood that venting will be required is expected to be less in a MARK II plant compared with a MARK I plant. For these reasons, the risk reduction to be gained from improvements to the vent system for MARK II plants may be less than for MARK I plants.

For less probable cases in which venting is initiated after core-melt has occurred, there is uncertainty in the incremental benefit of scrubbing the fission products by the suppression pool during venting at a Mark II. This is because molten core material on the floor of the Mark II containment may fail downcomers or drain lines and result in suppression pool bypass. Performing analysis of severe accidents for Mark II plants required use of Revision 4 of the EPG entitled "Steam Cooling." The analysis indicated that using Revision 4 could result in less operator response time (up to 30 minutes less) until the onset of core degradation as compared with using Revision 3. The staff's findings were presented to the BWR Owners Group. Subsequently, the BWR Owners' Group issued an interim change to Revision 4 to provide additional operator response time.

CONCLUSION

Aside from a change to Revision 4 of the EPGs, the staff did not identify any generic improvements that would be applicable to all Mark II containments. Therefore, the staff issued Supplement 3 to Generic Letter 88-20[11] which requested that each licensee with a MARK II containment consider, as part of its IPE, the insights and improvements identified in the CPI Program. Thus, this item was RESOLVED.

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[11] Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities from U.S. Nuclear Regulatory Commission, "Individual Plant Examination for Severe Accident Vulnerabilities—10 CFR § 50.54(f), (Generic Letter No. 88-20)," November 23, 1988 [ML031150465], (Supplement 1) August 29, 1989 [8908300001], (Supplement 2) April 4, 1990 [ML031200551], (Supplement 3) July 6, 1990 [ML031210418], (Supplement 4) June 28, 1991 [ML031150485], (Supplement 5) September 8, 1995.

(3) MARK III CONTAINMENTS DESCRIPTION

Historical Background

At the time of this evaluation in February 1992, there were 4 operating BWRs with MARK III containments located at 4 sites. The MARK III containment is approximately five times the volume of the MARK I containment and 65% to 85% of the volume of a large, dry PWR containment. The containment design pressure is 15 psig (25% of a MARK I, 30% of a large dry).

Unlike MARK I and II containments, the MARK III containment is not inerted and has igniters for H_2 control. The

only PRA performed on a BWR with a MARK III containment was for Grand Gulf as part of the NUREG-1150[12] studies.

The low core-melt frequency (on the order of $10^{(6)}$/year) estimated by the NUREG- 1150[13] studies was primarily due to the fact that Grand Gulf was a modern BWR design with a diversity of ways to provide water to the core. Grand Gulf uses a motor-driven High Pressure Core Spray with a dedicated diesel generator which improves the reliability of this system for mitigation of transients and small LOCAs. In addition, Grand Gulf

has a number of low pressure coolant injection systems. Thus, Grand Gulf already has a number of diverse systems such as those studied for other BWR plants (e.g., diverse alternative supplies of water to the reactor). At least one such system (fire water connection) was implemented by the licensee as a direct result of the

NUREG-1150[14] studies.

Safety Significance

The containment represents the final barrier to the release of radioactive material to the environment. GDC 16 requires a leak-tight containment and its failure, especially early in a severe accident, has the potential for a significant release of radioactivity to the environment.

Possible Solutions

Based on PRA insights, potential plant improvements that were investigated are similar to those investigated for MARK I plants and thus would also be appropriate for evaluation on a plant-specific basis. Because of the relatively large volume of the MARK III containment, the need for venting may also be less likely than for the MARK I containment. Also, some MARK III plants may already have the capability to vent through a hardened system. In addition, the staff's analysis indicates that if the containment is vented the following could occur: (1) more radioactive material may be released to the environment than if the containment is not vented; (2) there is a potential increase in the probability of an H_2 deflagration or detonation; and (3) a vented containment will not

prevent containment failure if an H_2 detonation were to occur. The analysis further indicates that for a short-term station blackout there is a very small amount of time when an H_2 deflagration or detonation could occur in the drywell and, in the absence of an H_2 deflagration or detonation, there is a high probability that there will be no containment failure during the accident. Thus, no generic conclusion could

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[12] NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Vol. 1) December 1990, (Vol. 2) December 1990, (Vol. 3) January 1991.

[13] NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Vol. 1) December 1990, (Vol. 2) December 1990, (Vol. 3) January 1991.

[14] NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Vol. 1) December 1990, (Vol. 2) December 1990, (Vol. 3) January 1991.

be reached regarding the need for venting for MARK III plants.

A potential vulnerability for MARK III plants involves station blackout, during which the igniters would be inoperable. Under these conditions, a detonable mixture of H_2 could develop which could be ignited upon restoration of power. A potential improvement considered for MARK III containments is a backup power supply in order to be able to use igniters during a station blackout. However, no generic conclusions could be reached. (At least one licensee has taken the position that use of igniters is not necessary during a station blackout.)

CONCLUSION

The staff did not identify any generic improvements that would be applicable to all MARK III containments.

Therefore, the staff issued Supplement 3 to Generic Letter 88-20[15] which requested that each licensee with a MARK III containment consider, as part of its IPE, the insights and improvements identified in the CPI program. Thus, this item was RESOLVED.

(4) ICE CONDENSER CONTAINMENTS

DESCRIPTION

Historical Background

At the time of this evaluation in February 1992, there were 10 reactors with ice condenser containments located at 5 sites. The ice condenser/containment houses a four-loop **W** PWR, is typically about 1.2 million cubic feet

in volume, and has a design pressure of 12 to 15 psig. The NUREG-1150[16] evaluation of the Sequoyah plant provided the most up-to-date insights into the important contributors to core damage and containment challenges that face an ice condenser plant. In using these results care must be taken, however, to ensure that they are representative of other ice condenser containment plants. For instance, the loss of offsite power initiating event frequency is lower for Sequoyah than for other ice condenser plants because of the offsite grid reliability. Also, because of design differences, significant variation has been reported in calculations of the ultimate failure capability of containments. Estimates have ranged from 60 psig for Sequoyah to 120 psig for Watts Bar.

Safety Significance

The containment represents the final barrier to the release of radioactive material to the environment. GDC 16 requires a leak-tight containment and its failure, especially early in a severe accident, has the potential for a significant release of radioactivity to the environment.

Possible Solutions

The Sequoyah risk analysis indicated that containment bypass sequences dominate early fatality risk. Timing is a key factor in these sequences as is a lack of any mitigating systems to scrub the release. The CPI Program did not address this area, because there was a joint NRR/RES program on interfacing

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[15] Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities from U.S. Nuclear Regulatory Commission, "Individual Plant Examination for Severe Accident Vulnerabilities—10 CFR § 50.54(f), (Generic Letter No. 88-20)," November 23, 1988 [ML031150465], (Supplement 1) August 29, 1989 [8908300001], (Supplement 2) April 4, 1990 [ML031200551], (Supplement 3) July 6, 1990 [ML031210418], (Supplement 4) June 28, 1991 [ML031150485], (Supplement 5) September 8, 1995.

[16] NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Vol. 1) December 1990, (Vol. 2) December 1990, (Vol. 3) January 1991.

system LOCAs that was underway to develop guidance and possible additional requirements for interfacing system LOCAs, including those that could bypass the containment. In addition, interfacing system LOCAs were to be considered as part of the IPE.

Direct containment heating (DCH) is a phenomenon that has a great deal of uncertainty associated with it. Risk assessments have varied considerably in their characterizations of its contribution to containment failure by

overpressurization. Because of this uncertainty, research is being performed under the SARP[17] to reduce the uncertainty in risk due to DCH. In addition, the staff is investigating, as part of the Accident Management Program, possible means to prevent or decrease the severity of DCH events. The principal strategy being investigated was that of full or partial depressurization of the RCS. The CPI Program made use of the ongoing Accident Management Program work on depressurization and evaluated the impact of depressurization on the ice condenser containment. An important finding was that depressurization to prevent DCH for ice condenser

plants is not sufficient to prevent containment failure unless the igniters are operating to control the large amount of H_2 that may be produced.

Containment failure resulting from uncontrolled H_2 burns or detonations is a potentially important failure mode for ice condenser containments. This could occur in station blackout events if power to the H_2 igniter system is lost, high concentrations of H_2 are produced as a result of core degradation, and power is then restored at a later time.

CONCLUSION

The staff did not identify any generic improvements that would be applicable to all ice condenser containments.

Therefore, the staff issued Supplement 3 to Generic Letter 88-20[18] which requested each licensee with an ice condenser containment consider, as part of its IPE, the insights and improvements identified in the CPI Program. Thus, this item was RESOLVED.

(5) DRY CONTAINMENTS DESCRIPTION

Historical Background

At the time of this evaluation in February 1992, there were 63 PWR plants with large, dry containments located at 40 sites. There were also 7 PWRs located at 4 sites that used subatmospheric containments. The volume and design pressure of a large, dry containment are typically about 2.5 million cubic feet and 60 psig, respectively.

The containment volume and design pressure of a subatmospheric containment are typically about 2 million cubic feet and 45 psig, respectively.

Safety Significance

The containment represents the final barrier to the release of radioactive material to the environment. GDC 16 requires a leak-tight containment and its failure, especially early in a severe accident, has the

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[17] NUREG-1365, "Revised Severe Accident Research Program Plan," U.S. Nuclear Regulatory Commission, August 1989, (Rev. 1) December 1992.

[18] Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities from U.S. Nuclear Regulatory Commission, "Individual Plant Examination for Severe Accident Vulnerabilities—10 CFR § 50.54(f), (Generic Letter No. 88-20)," November 23, 1988 [ML031150465], (Supplement 1) August 29, 1989 [8908300001], (Supplement 2) April 4, 1990 [ML031200551], (Supplement 3) July 6, 1990 [ML031210418], (Supplement 4) June 28, 1991 [ML031150485], (Supplement 5) September 8, 1995.

potential for a significant release of radioactivity to the environment.

Possible Solutions

NUREG-1150[19] indicated that, given a core-melt, the conditional probability of early containment failure for a large, dry containment is low. For the Surry plant, containment bypass was found to be the dominant contributor to risk. However, containment bypass was not investigated further under the CPI Program because of the ongoing activities in this area discussed under Ice Condenser Containments above.

DCH was not important for the dry containments studied in NUREG-1150.[20] However, as noted above, this is an area of large uncertainty and the importance of DCH to risk may vary for plants with dry containments

other than those studied in NUREG-1150.[21] Research is ongoing to reduce the uncertainty in risk due to DCH. Depressurization to avoid DCH is being investigated as part of the Accident Management Program.

NUREG-1150[22] did not identify H_2 combustion as a significant threat to the containment for the two PWR plants investigated. However, the staff does not know whether this conclusion can be extended to all PWR containments.

H_2 combustion on a global basis is not believed to be a significant threat to large, dry containments. However, less firm conclusions have been reached for the smaller subatmospheric containments. It may also be possible for detonable mixtures of H_2 to buildup in localized compartments of both types of dry containments and damage equipment. Therefore, the potential and effects of local H_2 burns should be evaluated on a plant-specific basis.

CONCLUSION

The staff did not identify any generic improvements that would be applicable to all dry containments. Therefore,

the staff issued Supplement 3 to Generic Letter 88-20[23] which requested each licensee with a dry containment to consider, as part of its IPE, the insights and improvements identified in the CPI Program. Thus, this item was RESOLVED.

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[19] NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Vol. 1) December 1990, (Vol. 2) December 1990, (Vol. 3) January 1991.

[20] NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Vol. 1) December 1990, (Vol. 2) December 1990, (Vol. 3) January 1991.

[21] NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Vol. 1) December 1990, (Vol. 2) December 1990, (Vol. 3) January 1991.

[22] NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Vol. 1) December 1990, (Vol. 2) December 1990, (Vol. 3) January 1991.

[23] Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities from U.S. Nuclear Regulatory Commission, "Individual Plant Examination for Severe Accident Vulnerabilities—10 CFR § 50.54(f), (Generic Letter No. 88-20)," November 23, 1988 [ML031150465], (Supplement 1) August 29, 1989 [8908300001], (Supplement 2) April 4, 1990 [ML031200551], (Supplement 3) July 6, 1990 [ML031210418], (Supplement 4) June 28, 1991 [ML031150485], (Supplement 5) September 8, 1995.