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IDENTIFICATION AND ASSESSMENT OF BWR IN-VESSEL SEVERE ACCIDENT MITIGATION STRATEGIES*

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ABSTRACT

This paper briefly describes the results of work carried out in support of the U.S. Nuclear Regulatory Commission Accident Management Research Program to evaluate the effectiveness and feasibility of current and proposed strategies for BWR severe accident management. These results are described in detail in the just-released report *Identification and Assessment of BWR In-Vessel Severe Accident Mitigation Strategies,* NUREG/CR-5869, which comprises three categories of findings. First, an assessment of the current status of accident management strategies for the mitigation of in-vessel events for BWR severe accident sequences is combined with a review of the BWR Owners' Group Emergency Procedure Guidelines (EPGs) to determine the extent to which they currently address the characteristic events of an unmitigated severe accident. Second, where considered necessary, new candidate accident management strategies are proposed for mitigation of the late-phase (after core damage has occurred) events. Finally, two of the four candidate strategies identified by this effort are assessed in detail. These are (1) preparation of a boron solution for reactor vessel refill should control blade damage occur during a period of temporary core dryout and (2) containment flooding to maintain the core debris within the reactor vessel if the injection systems cannot be restored.

1. INTRODUCTION

Work sponsored by the Reactor and Plant Systems Branch of the Division of Systems Research, Office of Nuclear Regulatory Research, United States Nuclear Regulatory Commission (USNRC) to identify and assess BWR in-vessel accident management strategies was recently completed at Oak Ridge National Laboratory (ORNL). The purpose of this effort was the systematic development of new strategies for mitigation of the late phase events, that is, the events that would occur in-vessel after the onset of significant core damage. The methodology employed and the results of this effort are described in detail in the report *Identification and Assessment of BWR In-Vessel Severe Accident Mitigation Strategies*[1], NUREG/CR-5869. This paper briefly describes the contents of this recently published report.

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NUREG/CR-5869 addresses the subject of BWR severe accident management for in-vessel events in three successive categories. First, the current status of BWR accident management procedures is assessed from the standpoint of effectiveness for application to the mitigation of critical (dominant) severe accident sequences. Second, where considered necessary, new candidate accident management strategies are proposed for mitigation of the late-phase events* and briefly assessed. Third, for the two new candidate strategies for which the initial assessments are judged insufficient to adequately determine effectiveness and which are believed to have sufficient potential to justify additional consideration, detailed quantitative analyses are provided. The results and conclusions associated with each of these three topic categories are summarized in the following Sections of this paper.

2. EXISTING BWR ACCIDENT MANAGEMENT STRATEGIES

With respect to the current status of BWR accident management procedures, the BWR Owners' Group Emergency Procedure Guidelines[2] (EPGs) have been examined from the standpoint of their application to Station Blackout and Anticipated Transient Without Scram (ATWS). These accident sequences have been consistently identified by Probabilistic Risk Assessment (PRA) to be the predominant contributors to the overall calculated core damage frequency for BWR internally-initiated accidents. This examination was performed for two reasons. The first was to determine the extent to which the EPGs currently implement the intent of the BWR accident management strategies that have been suggested in the Brookhaven National Laboratory (BNL) report *Assessment of Candidate Accident Management Strategies*[3] (NUREG/CR-5474), published in March 1990. The second objective was to determine the extent to which the current operator actions specified by the EPGs would be effective in unmitigated severe accident situations. It was found that many of the candidate strategies discussed in NUREG/CR-5474 are included in the current version (Revision 4) of the EPGs and that with one exception, the remainder involve plant-specific considerations to the extent that they may be more appropriate for inclusion within local plant emergency procedures than within the generic symptom-oriented EPGs. The exception is a strategy for injection of boron following core damage and control blade relocation, which clearly is appropriate for the general applicability of the EPGs.

With respect to the second objective of this review, it has been determined that the EPGs do not provide guidelines for operator actions in response to the in-vessel events that would occur only after the onset of significant core damage. The general conclusion of this review is that additional guidance should be provided under these circumstances beyond the currently specified repetitive actions to restore reactor vessel injection capability, although restoration of vessel injection should retain first priority. Thus, the greatest potential for improvement of the existing BWR emergency procedure strategies lies in the area of severe accident management, both for determining the extent of ongoing damage to the in-vessel structures and for attempting to terminate the accident.

3. REQUIREMENT FOR ADDITIONAL STRATEGIES

The second main topic category of the recently published NUREG/CR-5869 addresses the identification of new candidate accident management strategies for mitigation of the late-phase in-vessel events of a BWR severe accident, including a discussion of the motivation for consideration

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* The late-phase events of a severe accident sequence are those events that would occur only after core damage including structural degradation and material relocation.

of these strategies and a general description of the methods by which they might be carried out. The identification of new candidate strategies was subject to the constraint that they should not require major equipment modifications or additions, but rather should be capable of implementation using only the existing equipment and water resources of the BWR facilities. Also, accident management strategies already included within the EPGs have not been addressed; the intention is to identify new candidate strategies that could enhance or extend the EPGs for the management of severe accidents.

In pursuing the goal of identifying strategies for coping with severe accidents, it is logical to first consider the vulnerabilities of the BWR to the challenges imposed. In general, BWRs are well protected against core damage because they have redundant reactor vessel injection systems to keep the core covered with water. Therefore, it is not surprising that probabilistic risk assessments have consistently identified the station blackout accident sequence as the leading contributor to the calculated core damage frequency for BWRs. The apparent vulnerability to Station Blackout arises simply because the majority of the reactor vessel injection systems are dependent upon the availability of AC power. While the detailed descriptions provided in the remainder of this paper are based upon the BWR-4 Mark I containment design, the associated conclusions are considered to have general applicability.

The steam turbine-driven reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) systems can operate during Station Blackout, but do require DC power for valve operation and turbine governor control and are susceptible to mechanical failure. These systems would, therefore, be lost if AC power is not restored before the unit batteries become exhausted. Loss of reactor vessel injection capability in this manner defines the "long-term" station blackout accident sequence, since a significant period of time (typically six to eight hours) would elapse before battery exhaustion. "Short-term Station Blackout," on the other hand, denotes the station blackout accident sequence in which all reactor vessel injection capability is lost at the inception of the accident, most probably by a combination of loss of electrical power and HPCI/RCIC turbine mechanical failures. In either case, core degradation follows the uncovering of the core, which occurs as the reactor vessel water inventory is boiled away without replacement.

Other dominant core damage accident sequences also involve failure of reactor vessel injection, since the core must be at least partially uncovered in order for structural degradation and melting to occur. The ATWS accident sequence is consistently identified as second in order of calculated core melt frequency. With the core at power while the Main Steam Isolation Valves (MSIVs) are closed, the dominant form of this accident sequence tends to maintain the reactor vessel at pressures somewhat higher than normal, sufficient for steam release through the safety/relief valves (SRVs) to the pressure suppression pool. Since the rate of energy deposition into the pool can greatly exceed the capacity of the pool cooling equipment, the primary containment would become overheated and pressurized in an unmitigated ATWS accident sequence.

Containment events are the basic cause of the loss of reactor vessel injection systems for ATWS. However, the various injection systems would be lost in different ways. Most of the vessel injection systems are low-pressure systems, requiring that the reactor vessel be depressurized for performance of function. The turbine-driven HPCI and RCIC systems are capable of high pressure injection, but are susceptible to elevated pressure suppression pool temperatures when taking suction from this source since their lubricating oil is cooled by the water being pumped. In addition, both of these systems have high turbine exhaust pressure trips so that high primary containment pressure can defeat their function. Steam-driven feedwater pumps would be lost at the inception of the accident sequence when MSIV closure cuts off their steam supply.

Review of the results of probabilistic risk assessment for other important accident sequences demonstrates again that the postulated scenarios leading to core damage always include means for failure of function of the vessel injection systems. As defined, the various severe accident sequences involve different pathways to and timing of loss of vessel injection capability but, in every case, the core must become uncovered before core damage can occur. Nevertheless, the detailed means by which vessel injection capability might be lost are highly plant-specific; the detailed nature of the threats to the injection systems and the optimum measures that should be taken to cope with these threats depends upon the equipment characteristics of the individual plants. Extension of the methodology of the recent NRC-sponsored assessment of severe accident risks[4] (NUREG-1150) to take into consideration the plant-specific features of individual facilities is the responsibility of the plant operators as part of the individual plant examination (IPE) process.[5,6]

It is also desirable for defense-in-depth to develop mitigative strategies for coping with the late-phase severe accident events that would occur in the unlikely event that adequate reactor vessel injection cannot be maintained. Current accident management procedures are derived from the EPGs, which provide effective guidance for preventative measures to avoid core damage, including numerous diverse methods of maintaining reactor vessel injection capability with the provision of backup methods for use in abnormal circumstances. Some recommendations for improvement of the preventative guidelines of the EPGs can be offered, primarily in the realm of ATWS, where it is believed that the scrutability of the guidelines would be improved if distinctly separate procedures were provided for this accident sequence. Based upon the arguments that the signatures of ATWS are unmistakable so that operators would know when to invoke the ATWS procedures and that the operator actions required to deal with ATWS do not fit within the envelope of actions required to deal with other accident sequences[7], it seems that the very complicated procedures required for coping with ATWS could be more concisely and effectively implemented as a separate document. This would also permit the remaining symptom-oriented guidelines to be greatly simplified.

Other recommendations with respect to the provisions of the EPGs from the standpoint of their application to ATWS are offered. These are first, that care be taken to avoid leading the operators to attempt manual depressurization of a critical reactor, second, that consideration be given to control the reactor vessel injection rate as a means for reduction of reactor power (as opposed to reactor vessel water level control as currently directed), and third, that removal of the rod sequence control system to facilitate the manual insertion of control blades under ATWS conditions be undertaken, as authorized by the NRC.

A final recommendation applicable to all accident sequences involving partial uncovering of the core has to do with the timing of opening of the automatic depressurization system valves for the steam cooling maneuver, which is intended to delay fuel heatup by cooling the uncovered upper regions of the core with a rapid flow of steam. It is believed that this maneuver would be more effective if performed at a lower reactor vessel water level, such as the level that was specified by Revision 3 of the EPGs. The current Revision 4 of the EPGs provides for steam cooling to be implemented with the water level near the top of the core; since the increase in temperature of the uncovered portion of the core would be small at this time, the amount of steam cooling achieved would be insignificant.

4. AVAILABILITY OF PLANT INSTRUMENTS

In considering new candidate severe accident mitigation strategies for use with existing plant equipment, it is important to first recognize any limitations imposed upon the plant accident management team by lack of information with respect to the plant status. The most restrictive limitation as to plant instrumentation would occur as a result of loss of all electrical power, including that provided by the unit battery. This occurs after battery failure in the long-term station blackout accident sequence and in the (less-probable) version of the short-term station blackout accident sequence for which common-mode failure of the battery systems is an initiating event. For these accident sequences, loss of reactor vessel injection and the subsequent core degradation occur only after loss of DC power[8,9].

For accident sequences such as Short-Term Station Blackout (with mechanical failure of HPCI and RCIC), ATWS, LOCA, or Loss of Decay Heat Removal, electrical power (DC and perhaps AC) is maintained after loss of reactor vessel injection capability. Therefore, the availability of information concerning plant status is much greater for these sequences. The more limiting case is that for which only DC power obtained directly from the installed batteries and the AC power indirectly obtained from these battery systems is available. The sources of AC power during Station Blackout include the feedwater inverter and the unit-preferred and plant-preferred systems for which single-phase 120-volt AC power is produced under emergency conditions by generators driven by battery-powered DC motors. Emergency control room lighting would be available.

5. CANDIDATE SEVERE ACCIDENT MITIGATION STRATEGIES

With respect to application of the EPGs to the late phase of a severe accident sequence, these guidelines are not intended to propose actions in response to the accident symptoms that would be created by events occurring only after the onset of significant core damage. The final guidance to the operators, should an accident proceed into severe core damage and beyond, is that reactor vessel injection should be restored by any means possible and that the reactor vessel should be depressurized. While these are certainly important and worthwhile endeavors, additional guidance can and should be provided for the extremely unlikely, but possible severe accident situations where reactor vessel injection cannot be restored before significant core damage and structural relocation have occurred.

While recognizing that the probability of a BWR severe accident involving significant core damage is extremely low, it remains desirable to seek effective yet inexpensive mitigation measures that could be implemented employing the existing plant equipment and requiring only additions to the plant emergency procedures. Based upon the considered need for additional guidelines for BWR severe accident management for in-vessel events, four candidate late accident mitigation strategies are identified. These are:

1. Keep the Reactor Vessel Depressurized. Reactor vessel depressurization is important should an accident sequence progress to the point of vessel bottom head penetration failure because it would preclude direct containment heating (DCH) and thereby reduce the initial threat to containment integrity. This candidate strategy would provide an alternate means of reactor vessel venting should the SRVs become inoperable because of loss of control air or DC power. PRAs consistently include accident sequences involving loss of DC power and control air among the dominant sequences leading to core melt for BWRs.

2. Restore Injection in a Controlled Manner. Late accident mitigation implies actions to be taken after core melting, which requires at least partial uncovering of the core, which occurs because of loss of reactor vessel injection capability. BWRs have so many electric motor-driven injection systems that loss of injection capability implies loss of electrical power. (This is why Station Blackout is consistently identified by PRAs to be the dominant core melt precursor for BWRs.) If electric power is restored while core damage is in progress, then the automatic injection by the low-pressure, high-capacity pumping systems could be at a rate more than two hundred times greater than that necessary to remove the decay heat. This strategy would provide for controlled restoration of injection and is particularly important if the control blades have melted and relocated from the core.

3. Inject Boron if Control Blade Damage has Occurred. This strategy would provide that the water used to fill the reactor vessel after vessel injection capability was restored would contain a concentration of the boron-10 isotope sufficient to preclude criticality, even if none of the control blade neutron poison remained in the core region. This candidate strategy is closely related to the previous proposal for control of reactor vessel injection.

4. Containment Flooding to Maintain Core and Structural Debris In-Vessel. This candidate strategy is proposed as a means to maintain the core residue within the reactor vessel in the event that vessel injection cannot be restored as necessary to terminate the severe accident sequence. Containment flooding to above the level of the core is currently incorporated within the EPGs as an alternative method of providing a water source to the vessel in the event of design-basis LOCA (the water would flow into the vessel from the containment through the break). Here it is proposed that containment flooding might also be effective in preventing the release of molten materials from the reactor vessel for the risk-dominant non-LOCA accident sequences such as Station Blackout.

As explained in the Introduction, the third category of NUREG/CR-5869 derives from a reconsideration of these four candidate late-phase, in-vessel strategies for the purpose of identifying any that require (and have sufficient potential to justify) detailed quantitative assessment. The candidate strategy to keep the reactor vessel depressurized is not recommended for further assessment at this time because it is believed far more practical to improve the reliability of the control air and DC power supplies for the SRVs than to invent alternative methods for venting of the reactor vessel into the secondary containment under severe accident conditions. Nevertheless, consideration of the reliability of control air and DC power should be an important part of the IPE process since loss of these systems is involved in the risk-dominant sequences leading to core melt consistently identified for BWRs by PRAs such as the recent NRC-sponsored risk assessment (NUREG-1150).

The candidate strategies for restoration of injection in a controlled manner and injection of boron if control blade damage has occurred are recommended to be combined into a single concept for "Prevention of BWR Criticality as a Late Accident Mitigation Strategy." As described in the following Section, this would provide a sodium borate solution for the injected flow being used to recover the core, in sufficient concentration to preclude criticality as the water level rises within the reactor vessel. (The proposal for containment flooding will be addressed in Section 7.)

6. REFLOOD WITH BORATED WATER

This strategy for prevention of inadvertent criticality induced by severe accident recovery efforts could be implemented using only the existing plant equipment but employing a different chemical form for the boron poison. Available information concerning the poison concentration required is derived from the recent Pacific Northwest Laboratory (PNL) study, *Recriticality in a BWR Following a Core Damage Event*[10], NUREG/CR-5653. This study indicates that much more boron would have to be injected than is available (as a solution of sodium pentaborate) in the Standby Liquid Control System (SLCS). Furthermore, the dominant BWR severe accident sequence is Station Blackout and without means for mechanical stirring or heating of the injection source, the question of being able to form the poisoned solution under accident conditions becomes of supreme importance. Hence the need for the alternate chemical form.

Polybor, produced by the U. S. Borax Company, seems to be an ideal means for creating the required sodium borate solution. It is formed of exactly the same chemical constituents (sodium, boron, oxygen, and water) as sodium pentaborate but has the advantages that for the same boron concentration, it requires about one-third less mass of powder addition and has a significantly greater solubility in water. Whereas sodium pentaborate solution is formed by adding Borax and boric acid crystals to water, which then react to form the sodium pentaborate, a solution of Polybor is formed simply by dissolving the Polybor powder in water. This attribute, that two separate compounds are not required to interact within the water, is a major reason for the greater solubility of Polybor.

The specific goal of the proposed strategy is to provide for the addition of the boron-10 isotope together with the flow being used to recover the core, in sufficient quantity to preclude criticality as the water level rises within the reactor vessel. NUREG/CR-5653 provides the estimate that a boron-10 concentration of between 700 and 1000 ppm would be required within the vessel to preclude criticality once control blade melting had occurred. This is much greater than the concentration (about 225 ppm) attainable by injection of the entire contents of the SLCS tank.

One means to achieve such a high reactor vessel boron concentration would be to mix the powder directly with the water in the plant condensate storage tank and then, upon restoration of electrical power, to take suction on this tank with the low-pressure system pump to be used for vessel injection. It is, however, not a simple matter to invoke this strategy and preplanning and training would be necessary.

During normal reactor operation, the condensate storage tank provides makeup flow to the main condenser hotwells via an internal tank standpipe, as illustrated in Figure 1. Any practical strategy for direct poisoning of the tank contents must provide for partial draining to reduce the initial water volume, particularly if boron-10 concentrations on the order of 700 ppm are to be achieved. The condensate storage tank could be gravity-drained through the standpipe to the main condenser hotwells under station blackout conditions.

Even with partial tank draining, however, the amount of powder required to obtain a boron-10 concentration of 700 ppm is large. Considering the Peach Bottom plant configuration, and assuming the use of Polybor to take advantage of its greater solubility, 19,300 lbs (8,750 kg) would have to be added to the partially drained tank. [If Borax/boric acid were used, the requirement would be 28,400 lbs (12,880 kg).] Clearly, this is too much to be manhandled [50-lb (23-kg) bags] to the top of the tank and poured in. The practical way to poison the tank contents would be to prepare a slurry of extremely high concentration in a smaller container at

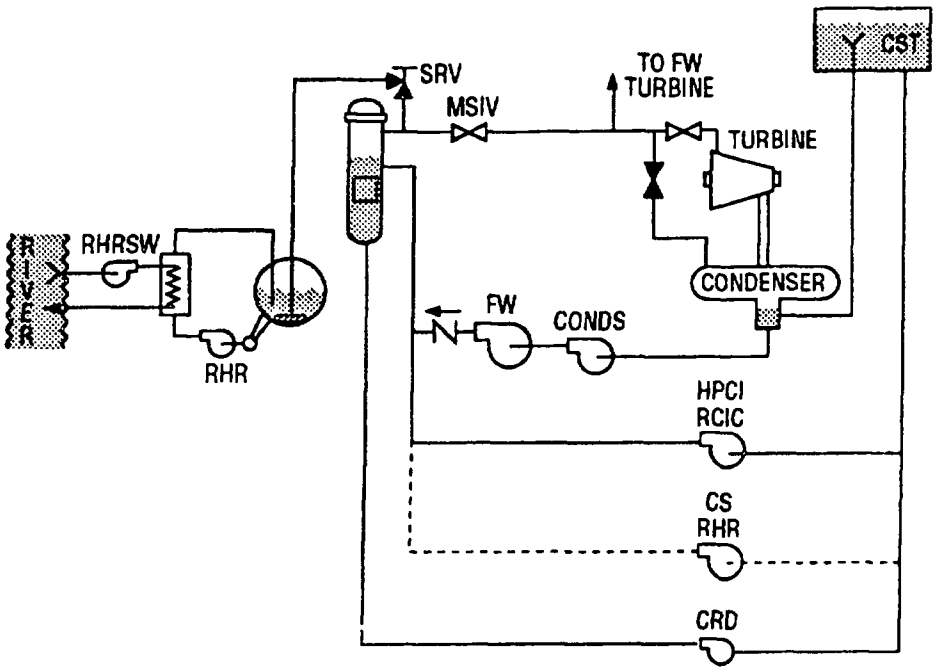

Fig. 1. Reactor vessel injection can be provided from the condensate storage tank by the motor-driven core spray (CS) or residual heat removal (RHR) system pumps.

ground level; then to pump the contents of this small container into the upper opening of the condensate storage tank. (Extremely high concentrations can be achieved with Polybor.) To avoid any requirement for procurement of additional plant equipment, a fire engine with its portable suction tank might be employed to perform the pumping function.

With the candidate accident management strategy identified, a simplified cost-benefit analysis was performed based upon the methodology described in NUREG-0933, *A Prioritization of Generic Safety Issues*[11] and following the guidelines of References 12 and 13. Implementation of the strategy was estimated to provide a reduction in the frequency of unmitigated core melting of 1.19E-06 per reactor-year (RY). The strategy proposed would, if implemented, affect the progression of severe accident events during the time window for recriticality, which is opened by the occasion of some core damage (the melting of the control blades). Thus, some core damage is associated even with successful implementation of the strategy. The goal of the strategy is to avert vessel breach and containment failure.

The estimated change in public risk associated with the proposed strategy is found to be 6.1 man-rem/RY. When applied to the present inventory of 38 BWR facilities with an average remaining lifetime of 21.1 years, the total potential risk reduction estimate is 4860 man-rem.

Implementation of the proposed strategy is estimated to involve per-plant expenditures (1982 dollars) of $70,000 for engineering analysis, preparation of procedures, personnel training, management review, and acquisition of material (sodium borate powder in the form of Polybor). In addition, it is estimated that 20 man-hr/RY would be required for periodic procedure review and team training (including drills). With a cost of $56.75 per man-hr (1982 dollars) and an average remaining plant life of 21.1 years, the average industry cost per reactor is estimated to be about $93,950.

NRC costs for implementation of the proposed strategy would be small since the general approach has already been developed by the Office of Research as a candidate accident management procedure. It is anticipated that the strategy would be implemented on a voluntary, plant-specific basis by the industry. Therefore, no additional NRC development costs would be incurred. Allowance is made, however, for the costs associated with oversight of the associated plant procedures and of the general readiness (status of personnel training) to successfully execute the plant-specific actions. These oversight activities are estimated to require an average NRC cost per reactor of about $7100.

Based upon an average industry cost of $94,000 per reactor and an NRC oversight cost of $7000 per reactor, the total cost (1982 dollars) associated with implementation of this strategy for the 38 BWR facilities is estimated to be $3.84M.

The value/impact assessment consistent with the procedures of NUREG-0933 for the proposed strategy is

$$S = \frac{4860 \text{ man-rem}}{\$3.84\text{M}}$$

$$= 1266 \text{ man-rem/\$M},$$

from which a priority ranking of MEDIUM is obtained for the proposed strategy.

Based upon this ranking, what further actions should be recommended? As pointed out in NUREG-0933, decisions should be tempered by the knowledge that the assessment uncertainties are generally large:

> "The criteria and estimating process on which the priority rankings are based are neither rigorous nor precise. Considerable application of professional judgment, sometimes guided by good information but often tenuously based, occurs at a number of stages in the process when numerical values are selected for use in the formula calculations and when other considerations are taken into account in corroborating or changing a priority ranking. What is important in the process is that it is systematic, that it is guided by analyses that are as quantitative as the situation reasonably permits, and that the bases and rationale are explicitly stated, providing a "visible" information base for decision. The impact of imprecision is blunted by the fact that only approximate rankings (in only four broad priority categories) are necessary and sought."

With these considerations in mind, it is recommended that each plant assess its need for the proposed strategy based upon the results of its Individual Plant Examination (IPE). By far, the most important aspect of this recommended plant-specific assessment of the need for this strategy is the frequency of station blackout events predicted to progress through the first stages of core damage (the melting of control blades). In the generic analysis of public risk reduction reported here, the probability of a recriticality event was taken to be 1.25E-06/py, based upon the recent PNL study (NUREG/CR-5653).

The PNL study is based upon the NUREG-1150 results for Peach Bottom, which includes a core-melt frequency of about 4.5E-06 derived from station blackout events. If individual plants discover in their IPE process that a much lower station blackout core damage frequency applies,

then correspondingly lower recriticality potential would also apply and implementation of the proposed strategy would probably not be practical for their facility.

As a final note with respect to the question of boration under severe accident conditions, it is important to recognize that many of the BWR facilities are currently implementing accident management strategies, on a voluntary basis, to provide back-up capability for the SLCS. These back-up strategies invoke such methods as modification of the HPCI or RCIC system pump suction piping to permit connection to the SLCS tank, or poisoning of the condensate storage tank. In all known cases, however, the effect of these plant-specific strategies is to provide a means to obtain a reactor vessel concentration of the boron-10 isotope similar to that attainable by use of the SLCS system itself. It seems highly desirable that these facilities should include information within their training programs and procedural notes that according to the analyses reported by PNL (NUREG/CR-5653), this concentration would be insufficient to preclude criticality associated with vessel reflood after control blade melting.

7. DRYWELL FLOODING

The basis for this paper, the recently published NUREG/CR-5869, also provides a detailed assessment of the proposed strategy for containment flooding to maintain the core and structural debris within the reactor vessel. This strategy would be invoked in the event that vessel injection could not be restored to terminate a severe accident sequence. Geometric effects of reactor vessel size dictate that the effectiveness of external cooling of the vessel bottom head as a means to remove decay heat from an internal debris pool would be least for the largest vessels. Considering also that the motivation for maintaining any core and structural debris within the reactor vessel is greatest for the Mark I drywells, the primary focus of the detailed assessment is upon the largest BWR Mark I containment facilities such as Peach Bottom or Browns Ferry.

The immediate goal of the considered strategy for containment flooding would be to surround the lower portion of the reactor vessel with water, thereby protecting both the instrument guide tube penetration assemblies and the vessel bottom head itself from failure by overtemperature[1, 14]. (The concept is illustrated in Figure 2.) The threat would be provided by the increasing temperature of the lower plenum debris bed after dryout. First, molten liquids forming within the bed would relocate downward into the instrument guide tubes challenging their continued integrity[15]. Subsequently, heating of the vessel bottom head by conduction from the debris would threaten global failure of the wall by creep rupture.

Nevertheless, it seems beyond question that all portions of the reactor vessel pressure boundary (including the instrument guide tubes) that are in contact with and cooled by water on their outer surfaces would survive any challenge imposed by a lower plenum debris bed or its relocated liquids. There is a problem, however, in that most of the upper portion of the reactor vessel could not be covered by water and, more significant in the short term, much of the outer surface of the vessel bottom head would be dry as well.

That the upper portion of the reactor vessel could not be covered is due to the location within the containment of the drywell vents. Since low-pressure pumping systems would be used for flooding, the drywell would have to be vented during filling and the water level could not rise above the elevation of the vents, at about two-thirds vessel height. That much of the outer surface of the reactor vessel bottom head would be dry is due to the gas pocket that would be trapped within the vessel support skirt during the process of raising the water level within the drywell. The situation immediately after lower plenum dryout is illustrated in Figure 3.

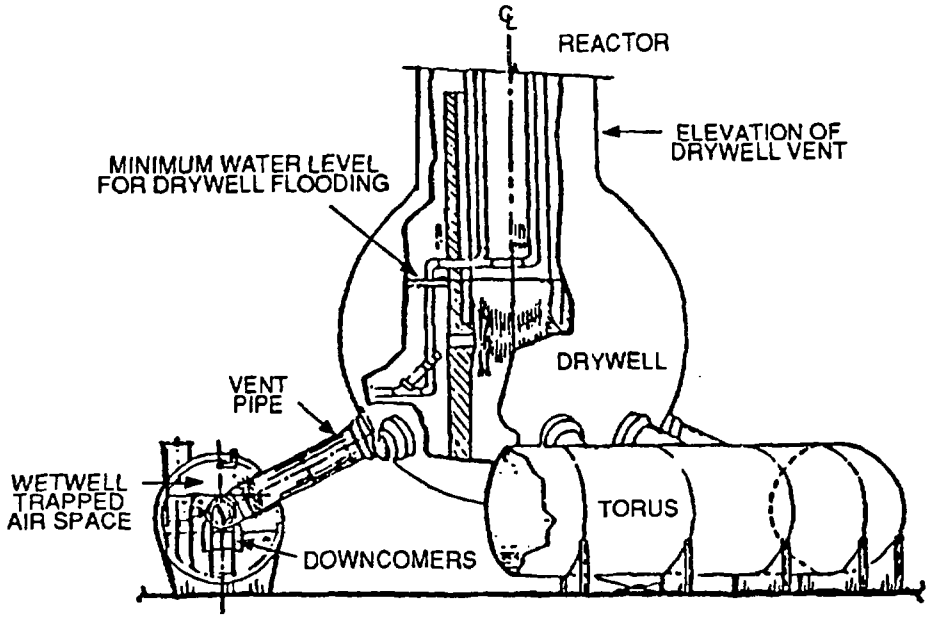

Fig. 2. Containment flooding to cover the reactor vessel bottom head in the BWR Mark I containment design.

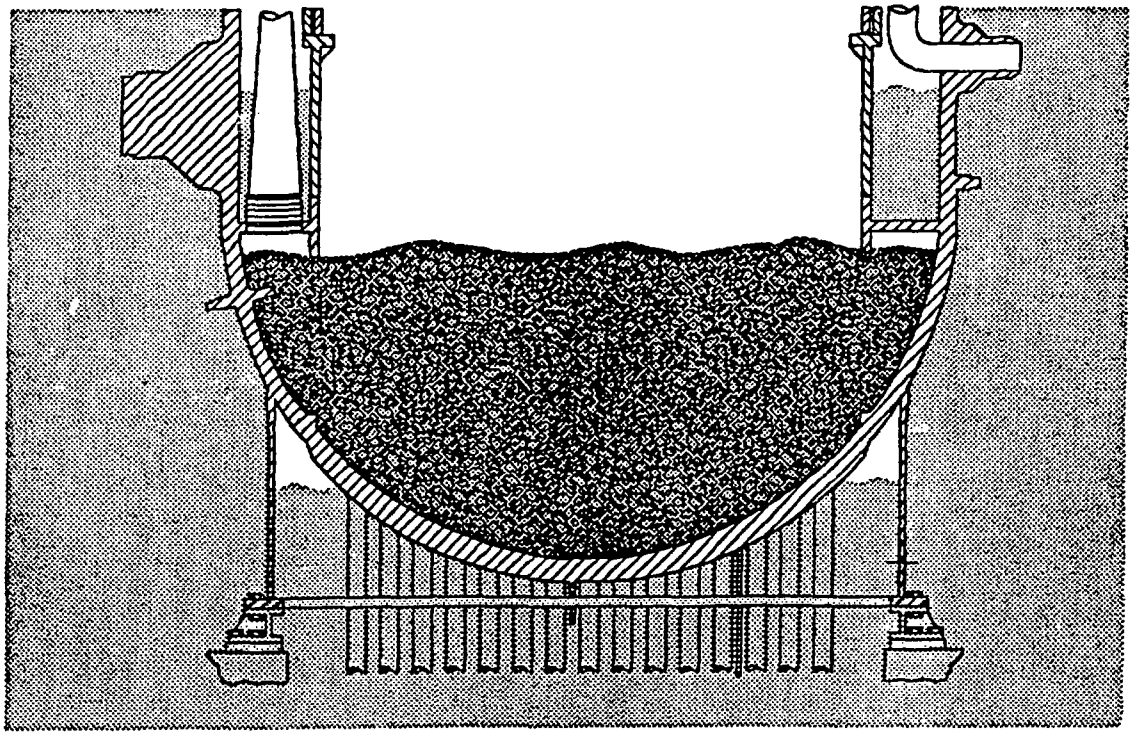

Fig. 3. An illustration of the quenched debris bed within the BWR lower plenum immediately after bed dryout.

The detailed assessment results demonstrate that the existence of a trapped gas pocket beneath the vessel skirt attachment would ultimately prove fatal to the integrity of the bottom head wall. Figure 4 illustrates the insulating crust of varying thickness that would remain adjacent to the wall after melting of the central portion of the debris. Nevertheless, the most important attribute of drywell flooding, that of preventing early failure of the instrument guide tube penetration assemblies, would be realized. These results are among those listed in Table 1 where it is shown (first entry) that in the absence of water, penetration assembly failures would be expected at about 250 minutes after scram. If penetration failures did not occur, then creep rupture of the bottom head would be expected after 10 hours if the bottom head is dry and after 13 hours if the drywell is flooded. The important contribution of drywell flooding is to shift the expected failure mode from penetration failures (Table 1 first entry) to bottom head creep rupture (Table 1 third entry).

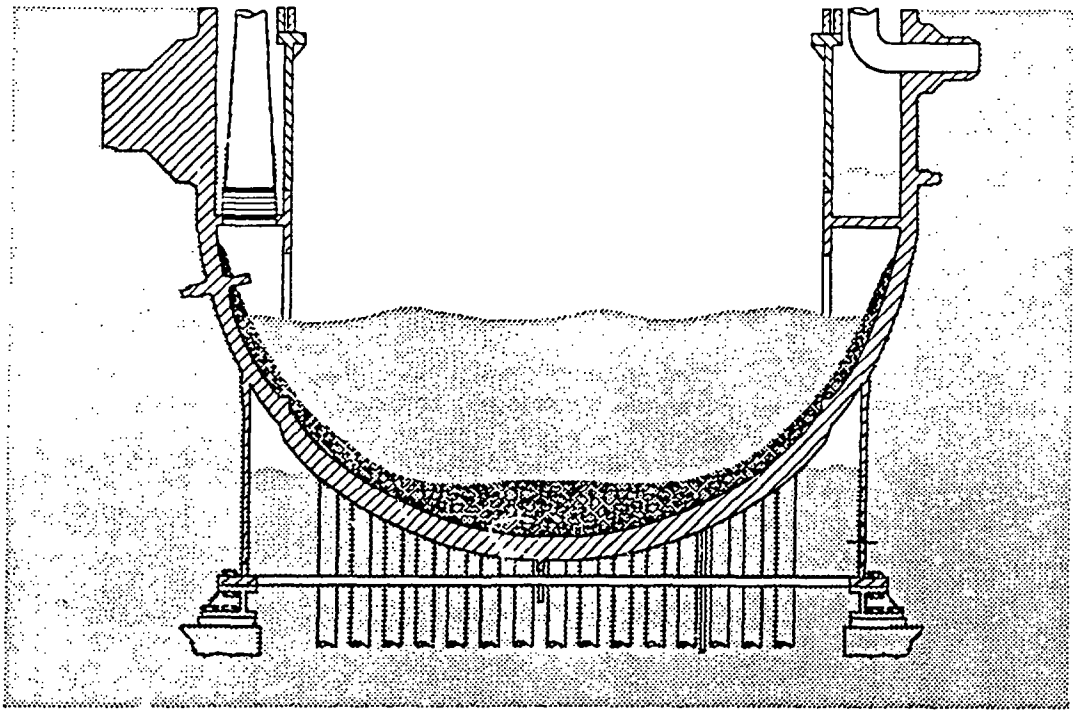

Fig. 4. A molten pool is predicted to form and spread radially from the upper center of the quenched debris.

Table 1. Estimated failure times for the reactor vessel bottom head pressure boundary for Peach Bottom/Browns Ferry short-term station blackout

Drywell Flooded	Failure Mechanism	Time to Failure Minutes	Hours
No	Penetration Assemblies	250	4.2
No	Bottom Head Creep Rupture	600 – 640	10.0 – 10.7
Yes	Bottom Head Creep Rupture	780 – 840	13.0 – 14.0

The effectiveness of drywell flooding could be improved if the reactor vessel support skirt were vented in order to reduce the trapped gas volume and increase the fraction of bottom head surface area contacted by water. Partial venting could be achieved by loosening the cover on the support skirt manhole access hole. This would increase the wetted portion of the bottom head from 55% to 73% of the total outer surface area, which delays the predicted time of bottom head creep rupture by about one hour. The predicted failure times for the basic case without skirt venting and for the case of partial venting at the manhole access are indicated in the first two entries of Table 2.

Table 2. Effect of skirt venting upon time to failure of the bottom head pressure boundary for Peach Bottom/Browns Ferry short-term station blackout with drywell flooding

Skirt Vented	Failure Mechanism	Time to Failure Minutes	Hours
No	Bottom Head Creep Rupture	780 – 840	13.0 – 14.0
Partial	Bottom Head Creep Rupture	840 – 900	14.0 – 15.0
Complete	Melting of Upper Vessel Wall	>1200	>20.0

Complete venting of the reactor vessel support skirt would provide 100% water coverage of the vessel bottom head but would require special measures such as provision of a siphon tube or the drilling of small holes at the upper end of the skirt, just below the attachment weld. Because of the associated personnel radiation exposure penalty and the predicted low core melt frequencies for the existing plants, this is not considered to be a practical suggestion for the existing BWR facilities, but provision for complete venting is inherent for the SBWR design. As indicated by the last entry in Table 2, 100% water coverage of the vessel bottom head would convert the failure mechanism from bottom head creep rupture to melting of the upper vessel wall and would delay the predicted time of failure to more than 20 hours after scram.

In summary, all portions of the reactor vessel wall that are covered by water would be adequately protected against failure by melting or creep rupture. For the cases with no venting or partial venting of the support skirt, the creep rupture failure is predicted to occur in the portion of the vessel wall adjacent to the trapped gas pocket beneath the skirt. Partial venting would reduce the size of the gas pocket and delay the predicted time of failure, but the failure mechanism would still be creep rupture beneath the skirt attachment weld. With complete venting, however, there would be no gas pocket and this failure mechanism would be eliminated.

What cannot be eliminated, however, is the radiative heat transfer upward within the reactor vessel from the surface of the lower plenum debris bed. About one-half to two-thirds of all energy release within the bed would be radiated upward after bottom head dryout. Initially, the primary heat sink for this radiation would be the water trapped in the downcomer region between the core shroud and the vessel wall above the debris bed. It is the heating of this water that creates the only steam source within the reactor vessel after lower plenum dryout.

After the water in the downcomer region became exhausted, the upward radiative heat transfer from the debris surface would serve to increase the temperature of the upper reactor vessel

internal structures. For calculations with the existence of a gas pocket beneath the skirt, bottom head creep rupture is predicted to occur while the temperature of these internal stainless steel heat sinks remains below the melting point. If bottom head creep rupture did not occur, however, the debris would remain within the vessel, the upward radiation would continue, and the upper internal structures would melt.

The mass of the BWR internal structures (core shroud, steam separators, dryers) is large. Melting of these stainless steel structures under the impetus of the upward debris pool radiation more than 14 hours after scram would occur over a long period of time. Nevertheless, decay heating of the debris pool and the associated upward radiation would be relentless and, after exhaustion of the stainless steel, the only remaining internal heat sink above the pool surface would be the carbon steel of the upper vessel wall. All portions of the wall cooled by water on their outer surfaces would remain intact, but unless the water height within the drywell extended well above the surface of the debris pool, upper portions of the vessel exposed to the drywell atmosphere would ultimately reach failure temperatures. The calculated minimum flooding height required to preclude inner wall melting for the Peach Bottom reactor vessel is illustrated in Figure 5.

It should be obvious from this discussion of the effect of water upon cooling of the vessel wall that it would be desirable to have a drywell flooding strategy that would completely submerge the reactor vessel and thereby eliminate questions concerning the required water level within the containment. This could not be achieved in existing facilities because of the limitation that the height of water within the drywell cannot exceed the elevation of the drywell vents. Future designs, however, might provide for complete coverage of the reactor vessel as a severe accident mitigation technique.

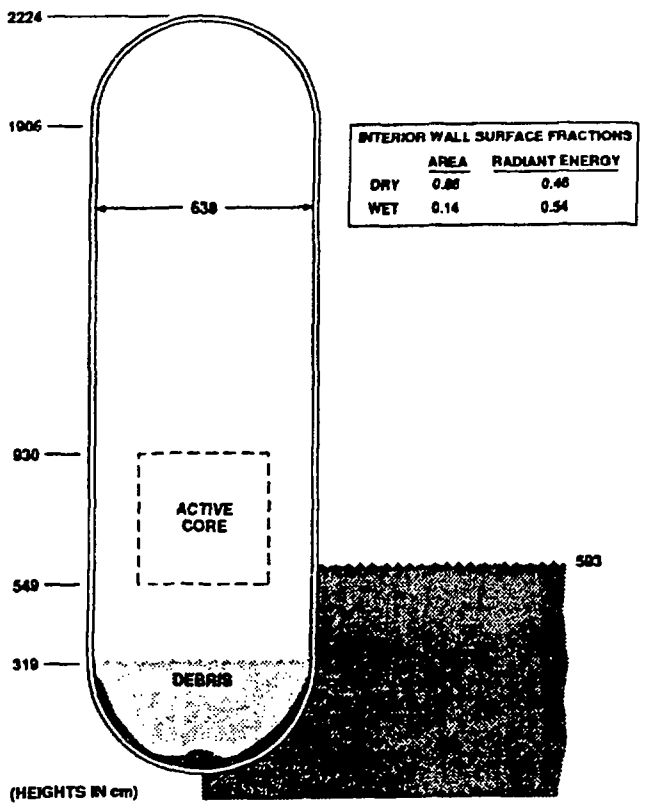

Fig. 5. The calculated minimum coverage of the upper reactor vessel necessary to preclude melting of the inner surface of the wall.

Table 3 provides a summary of the calculated failure times and release mechanisms for all of the cases considered in this study. These include the cases previously discussed in connection with Tables 1 and 2, plus one additional case (third entry) in which it is assumed that reactor vessel pressure control is lost at the time of drywell flooding, because of the submergence of the safety/relief valves. The increased wall tensile stress associated with this case would cause the wall creep rupture to occur at a lower temperature, advancing the time of failure by about two hours over the depressurized case (compare the third and fourth entries in Table 3).

Table 3. Effect of drywell flooding upon time of debris release from the reactor vessel for the short-term station blackout accident sequence based upon Peach Bottom/Browns Ferry

Drywell Flooded	Skirt Vented	Reactor Vessel Depressurized	Release Mechanism	Time to Failure Minutes	Hours
No	—	Yes	Penetration Failures	250	4.2
No	—	Yes	Bottom Head Creep Rupture	600 – 640	10.0 – 10.7
Yes	No	No	Bottom Head Creep Rupture	660 – 700	11.0 – 11.7
Yes	No	Yes	Bottom Head Creep Rupture	780 – 840	13.0 – 14.0
Yes	Partial	Yes	Bottom Head Creep Rupture	840 – 900	14.0 – 15.0
Yes	Complete	Yes	Melting of Upper Vessel Wall	>1200	>20.0

The most important disadvantage of a drywell flooding strategy for existing plants is the requirement for venting to the external atmosphere[16] while the containment is being filled by the low-pressure pumping systems and during the subsequent steaming from the water surrounding the reactor vessel bottom head. Because of this, implementation of the drywell flooding strategy would initiate a noble gas release to the surrounding atmosphere as well as a limited escape of fission product particulates. All particulate matter released from the reactor vessel prior to failure of the vessel wall would enter the pressure suppression pool via the safety/relief valve T-quenchers and would be scrubbed by passage through the water in both the wetwell and drywell. Therefore, the concentration of particulates in the drywell atmosphere and any release through the drywell vents would remain small as long as the reactor vessel wall remained intact.

Creep rupture of the vessel bottom head beneath the support skirt attachment would release debris into the water-filled pedestal region to fall downward onto the drywell floor. Since containment flooding would provide a water depth of more than 30 feet (9.144 m) over the drywell floor, the particulate matter released from the debris mass should be adequately scrubbed provided, of course, that violent steam explosions do not occur. Furthermore, the large volume of water in the drywell would protect the drywell shell from late failure in Mark I containment

facilities, since the accumulating debris would never reach a height sufficient to break the water surface.

The advantages and disadvantages of a drywell flooding strategy for existing BWR facilities are summarized in Table 4. The listed advantages involve significant contributions to accident mitigation, which have previously been discussed. The listed disadvantages, however, are also important and will be discussed in the following paragraphs.

Table 4. Advantages and disadvantages of a drywell flooding strategy for severe accident mitigation in existing BWR facilities

Advantages	1. Prevent failure of the bottom head penetrations and vessel drain 2. Increased scrubbing of fission product particulate matter 3. Delay creep rupture of the reactor vessel bottom head 4. Prevent failure of the Mark I drywell shell when core debris does leave the vessel
Disadvantages	1. Requires availability of power source and pump capable of filling the drywell to the level of the vessel bottom head within 150 minutes under station blackout conditions. 2. Requires that the drywell be vented.

First, implementation of the proposed strategy would require equipment modifications and additions. Although there may be plant-specific exceptions, containment flooding with the existing pumping systems would require too much time; furthermore, the existing systems would not be available for the dominant station blackout accident sequences. What is needed is a reliable ability to sufficiently flood the drywell within a short period of time, since it would be unrealistic to expect that emergency procedures would call for containment flooding (and the associated undesirable effects upon installed drywell equipment) until after core degradation has begun. If the water did not reach the vessel bottom head until after lower plenum debris bed dryout and the initial heating of the vessel wall, it would be too late to prevent penetration assembly failures.

The second disadvantage, that the drywell vents would have to be opened early in the accident sequence to permit flooding of the containment, is particularly undesirable since this in turn involves early release of the fission product noble gases, beginning soon after the onset of core degradation. After the water had contacted the vessel bottom head, a continuous steam generation would begin within the drywell that would be released to the outside atmosphere by means of the open vents. This would tend to sweep any particulate matter from the drywell atmosphere through the vents. The amount of particulate matter reaching the drywell atmosphere would, however, be limited by water scrubbing as long as the reactor vessel wall remained intact above the water level in the drywell. This is expected to be the case for the existing BWR facilities where the ultimate failure of the wall would occur by creep rupture beneath the skirt attachment weld.

It is interesting, however, to briefly consider the potential benefits of application of a drywell flooding strategy to future BWR facilities, where the disadvantages listed in Table 4 might be avoided by appropriate plant design. Much less water would be required since the reactor vessel would be located in a cavity instead of suspended high above a flat drywell floor. Provision could be made for complete venting of the reactor vessel support skirt so that all of the bottom head would be in contact with water. This would preclude creep rupture of the vessel bottom head, shifting the potential failure mode to melting of the upper vessel wall, above the water level in the drywell.

For the existing BWR facilities, failure of the upper reactor vessel wall would provide a direct path from the upper surface of the debris pool to the open drywell vents without the benefit of water scrubbing.* For future plant designs, this could be avoided in two ways. First, submergence of most, or all, of the reactor vessel wall above the debris pool surface would preclude failure of the upper vessel wall. Second, the requirement for containment venting could be eliminated by provision of an adequate water source within the containment and provision for condensation of the generated steam. Both of these approaches are within the scope of design features currently under consideration for the advanced passive design.

8. SUMMARY

The new report *Identification and Assessment of BWR In-Vessel Severe Accident Mitigation Strategies*[1] addresses the need for BWR accident management in the unlikely event that an accident sequence should proceed through core degradation into relocation of material debris into the reactor vessel lower plenum. Although the low predicted probability of such events does not demand remedial action for the existing BWR facilities, it seems that efficacious counter-measures might be established by a diligent utility on a cost-effective basis for (1) coping with vessel reflood after control blade melting and (2) maintaining core debris within the reactor vessel. The advanced SBWR equipment and structural design inherently supports implementation of both of these objectives.

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Abstract

This report provides the results of work carried out in support of the U.S. Nuclear Regulatory Commission Accident Management Research Program to develop a technical basis for evaluating the effectiveness and feasibility of current and proposed strategies for boiling water reactor (BWR) severe accident management. First, the findings of an assessment of the current status of accident management strategies for the mitigation of in-vessel events for BWR severe accident sequences are described. This includes a review of the BWR Owners' Group Emergency Procedure Guidelines (EPGs) to determine the extent to which they currently address the characteristic events of an unmitigated severe accident and to provide the basis for recommendations for enhancement of accident management procedures. Second, where considered necessary, new candidate accident management strategies are proposed for mitigation of the late-phase (after core damage has occurred) events. Finally, recommendations are made for consideration of additional strategies where warranted, and two of the four candidate strategies identified by this effort are assessed in detail: (1) preparation of a boron solution for reactor vessel refill should control blade damage occur during a period of temporary core dryout and (2) containment flooding to maintain the core debris within the reactor vessel if the injection systems cannot be restored.

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Nomenclature

ac	alternating current
ADS	automatic depressurization system
ARI	alternate rod insertion
ASEP	Accident Sequence Evaluation Program
ATWS	anticipated transient without scram
BNL	Brookhaven National Laboratory
BWR	boiling water reactor
CRDHS	control rod drive hydraulic system
CS	core spray system
dc	direct current
DCH	direct containment heating
ECCS	emergency core cooling system
EOP	Emergency Operating Procedure
EPG	Emergency Procedure Guideline
EPRI	Electric Power Research Institute
FSAR	Final Safety Analysis Report
ft	feet
FWCI	feedwater coolant injection system
gal/min	gallon(s) per minute
h	hour(s)
HPCI	high-pressure coolant injection
HPCS	high-pressure core spray
HVAC	heating, ventilation, and air conditioning
in.	inches
INEL	Idaho National Engineering Laboratory
IPE	individual plant examination
ISL	interfacing systems LOCA
$K	thousand (dollars)
ksi	thousand pounds per square inch
lb	pound(s)
LOCA	loss-of-coolant accident
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
$M	million (dollars)
min	minute(s)
MSIV	main steam isolation valve
MW	megawatt(s)
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PNL	Pacific Northwest Laboratory
PRA	probabilistic risk assessment
psia	pounds force per square inch absolute
psid	pounds force per square inch differential
psig	pounds force per square inch gauge
PSP	pressure suppression pool
py	plant-year as used in NUREG/CR-2800 (same as RY)
RCIC	reactor core isolation cooling
RHR	residual heat removal
RPS	reactor protection system
RSCS	rod sequence control system
RWCU	reactor water cleanup
RWM	rod worth minimizer
RY	reactor-year as used in NUREG-0933
SASA	Severe Accident Sequence Analysis
SER	Safety Evaluation Report
SGTS	standby gas treatment system
SLCS	standby liquid control system
SNL	Sandia National Laboratories
SRV	safety/relief valve

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This work to consider severe accident management procedures for boiling water reactor applications was sponsored under the direction of Dr. Louis M. Shotkin, Chief, Reactor and Plant Systems Branch of the U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research, Division of Systems Research. The Technical Monitor for this effort was Dr. James T. Han, who provided invaluable guidance and assistance. The authors would also like to express their sincere appreciation for the thoughtful and insightful comments of the external peer reviewers, Ms. Susan Dingman of Sandia National Laboratories and Professor William Kastenberg of the University of California-Los Angeles, with his graduate students Hyunjae Park, Leiming Xing, and Donghan Yu. Dr. Joy Rempe of Idaho National Engineering Laboratory reviewed our initial analyses of the external flooding strategy and provided her assessment of our results. Mr. Ted Fox of Oak Ridge National Laboratory reviewed our cost-benefit analysis for the boron injection strategy. There is no doubt that the clarity and completeness of this report have been improved as a result of the modifications implemented in response to the helpful and constructive comments of our reviewers.

Executive Summary

This report addresses the subject of boiling water reactor (BWR) severe accident management for in-vessel events in three successive categories. First, the current status of BWR accident management procedures is assessed from the standpoint of effectiveness for application to the mitigation of critical (dominant) severe accident sequences. Second, where considered necessary, new candidate accident management strategies are proposed for mitigation of the late-phase events* and are briefly assessed. Third, for the two new candidate strategies for which the initial assessments are judged insufficient to adequately determine effectiveness and which are believed to have sufficient potential to justify additional assessment, detailed quantitative analyses are provided. The results and conclusions associated with each of these three categories are summarized in the following paragraphs.

With respect to the current status of BWR accident management procedures, the BWR Owners' Group Emergency Procedure Guidelines (EPGs) have been examined from the standpoint of their application to station blackout and anticipated transient without scram (ATWS), which have been consistently identified by probabilistic risk assessment (PRA) to be the predominant contributors to the overall calculated core damage frequency for BWR internally initiated accidents. This review was performed for two reasons. The first was to determine the extent to which the EPGs currently implement the intent of the BWR accident management strategies that have been suggested in the Brookhaven National Laboratory (BNL) report *Assessment of Candidate Accident Management Strategies* (NUREG/CR-5474), published in March 1990. The second objective was to determine the extent to which the current operator actions specified by the EPGs would be effective in unmitigated severe accident situations. It was found that many of the candidate strategies discussed in NUREG/CR-5474 are included in the current version (Revision 4) of the EPGs and that, with one exception, the remainder involve plant-specific considerations to the extent that they may be more appropriate for inclusion within local plant emergency procedures than within the generic symptom-oriented EPGs. The exception is a strategy for injection of boron following core damage and control blade relocation, which clearly is appropriate for the general applicability of the EPGs.

With respect to the second objective of this review, it has been determined that the EPGs do not provide guidelines for operator actions in response to the in-vessel events that would occur only after the onset of significant core damage. The general conclusion of this review is that more can be done under these circumstances than the currently specified repetitive actions to restore reactor vessel injection capability, although restoration of vessel injection should retain first priority. Thus, the greatest potential for improvement of the existing BWR emergency procedure strategies lies in the area of severe accident management, both for determining the extent of ongoing damage to the in-vessel structures and for attempting to terminate the accident.

The second main category of this report addresses the identification of new candidate accident management strategies for mitigation of the late-phase in-vessel events of a BWR severe accident, including a discussion of the motivation for these strategies and a general description of the methods by which they might be carried out. The identification of new candidate strategies was subject to the constraint that they should not require major equipment modifications or additions, but rather should be capable of implementation using only the existing equipment and water resources of the BWR facilities. Also, accident management strategies already included within the EPGs are not addressed within this report; the intention is to identify candidate strategies that could enhance or extend the EPGs for the management of severe accidents.

In pursuing the goal of identifying strategies for coping with severe accidents, it is logical to first consider the vulnerabilities of the BWR to the challenges imposed. In general, BWRs are well protected against core damage because they have redundant reactor vessel injection systems to keep the core covered with water. Therefore, it is not surprising that probabilistic risk assessments have consistently identified the station blackout accident sequence as the leading contributor to the calculated core damage frequency for BWRs. The apparent vulnerability to station blackout arises simply because the majority of the reactor vessel injection systems are dependent upon the availability of ac power.

The steam turbine-driven reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems can operate during station blackout, but do require dc power for valve operation and turbine governor control and are susceptible to mechanical failure. These systems would, therefore, be lost if ac power is not restored before the unit batteries become exhausted. Loss of reactor vessel injection capability in this manner defines the "long-term"

*The late-phase events of a severe accident sequence are those events that would occur only after core damage, including structural degradation and material relocation.

station blackout accident sequence, because a significant period of time (typically 6 to 8 h) would elapse before battery exhaustion. "Short-term station blackout," on the other hand, denotes the station blackout accident sequence in which reactor vessel injection capability is lost at the inception of the accident, by a combination of loss of electrical power and HPCI/RCIC turbine mechanical failures. In either case, core degradation follows the uncovering of the core, which occurs as the reactor vessel water inventory is boiled away without replacement.

Other dominant core damage accident sequences also involve failure of reactor vessel injection, because the core must be at least partially uncovered for structural degradation and melting to occur. The ATWS accident sequence is consistently identified as second in order of calculated core melt frequency. By its very nature, with the core at power while the main steam isolation valves (MSIVs) are closed, the dominant form of this accident sequence tends to maintain the reactor vessel at pressures somewhat higher than normal, sufficient for steam release through the safety/relief valves (SRVs) to the pressure suppression pool. Because the rate of energy deposition into the pool can greatly exceed the capacity of the pool cooling equipment, the primary containment would become overheated and pressurized in an unmitigated ATWS accident sequence.

Containment events are the basic cause of the loss of reactor vessel injection systems for ATWS. However, the various injection systems would be lost in different ways. Most of the vessel injection systems are low-pressure systems, requiring that the reactor vessel be depressurized for performance of function. The turbine-driven HPCI and RCIC systems are capable of high-pressure injection but are susceptible to elevated pressure suppression pool temperatures when taking suction from this source because their lubricating oil is cooled by the water being pumped. In addition, both of these systems have high turbine exhaust pressure trips so that high primary containment pressure can defeat their function. Steam-driven feedwater pumps would be lost at the inception of the accident sequence when MSIV closure cuts off their steam supply.

Review of the results of probabilistic risk assessment for other important accident sequences demonstrates again that the postulated scenarios leading to core damage always include means for failure of function of the vessel injection systems. As defined, the various severe accident sequences involve different pathways to and timing of loss of vessel injection capability but, in every case, the core must become uncovered before core damage can occur. Nevertheless, the detailed means by which vessel injection capability might be lost are highly plant-specific; the detailed nature of the threats to the injection systems and the optimum measures that should be taken to cope with these threats depends upon the equipment characteristics of the individual plants. Extension of the methodology of the recent Nuclear Regulatory Commission (NRC)-sponsored assessment of severe accident risks (NUREG-1150) to take into consideration the plant-specific features of individual facilities is the responsibility of the plant operators as part of the individual plant examination (IPE) process.

It is also desirable for defense-in-depth to develop mitigative strategies for coping with the late-phase severe accident events that would occur in the unlikely event that adequate reactor vessel injection cannot be maintained. Current accident management procedures are derived from the EPGs, which provide effective guidance for preventative measures to avoid core damage, including numerous diverse methods of maintaining reactor vessel injection capability with the provision of backup methods for use in abnormal circumstances. Some recommendations for improvement of the preventative guidelines of the EPGs can be offered, primarily in the realm of ATWS, where it is believed that the scrutability of the guidelines would be improved if distinctly separate procedures were provided for this accident sequence. Based upon the arguments that the signatures of ATWS are unmistakable so that operators would know when to invoke the ATWS procedures and that the operator actions required to deal with ATWS do not fit within the envelope of actions required to deal with other accident sequences, it seems that the very complicated procedures required for coping with ATWS could be more concisely and effectively implemented as a separate document. This would also permit the remaining symptom-oriented guidelines to be greatly simplified.

Other recommendations with respect to the provisions of the EPGs from the standpoint of their application to ATWS are that care be taken to avoid leading the operators to attempt manual depressurization of a critical reactor, that consideration be given to control the reactor vessel injection rate as a means for reduction of reactor power (as opposed to reactor vessel water level control as currently directed), and that removal of the rod sequence control system to facilitate the manual insertion of control blades under ATWS conditions be undertaken, as authorized by the NRC.

A final recommendation applicable to all accident sequences involving partial uncovering of the core has to do with the timing of opening of the automatic depressurization system valves for the steam cooling maneuver, which is intended to delay fuel heatup by cooling the uncovered upper regions of the core with a rapid flow of steam. It is believed that this maneuver would be more

effective if performed at a lower reactor vessel water level, such as the level that was specified by Revision 3 of the EPGs. The current Revision 4 of the EPGs provides for steam cooling to be implemented with the water level near the top of the core; because the increase in temperature of the uncovered portion of the core would be small at this time, the amount of steam cooling achieved would be insignificant.

In considering new candidate severe accident mitigation strategies for use with existing plant equipment, it is important to first recognize any limitations imposed upon the plant accident management team by lack of information with respect to the plant status. The most restrictive limitation as to plant instrumentation would occur as a result of loss of all electrical power, including that provided by the unit battery. This occurs after battery failure in the long-term station blackout accident sequence and in the (less-probable) version of the short-term station blackout accident sequence for which common-mode failure of the battery systems is an initiating event. For these accident sequences, loss of reactor vessel injection and the subsequent core degradation occur only after loss of dc power.

For accident sequences such as short-term station blackout (with mechanical failure of HPCI and RCIC as an initiating event), ATWS, LOCA, or loss of decay heat removal, electrical power (dc and perhaps ac) is maintained after loss of reactor vessel injection capability. Therefore, the availability of information concerning plant status is much greater for these sequences. The more limiting case is that for which only dc power obtained directly from the installed batteries and the ac power indirectly obtained from these battery systems is available. The sources of ac power during station blackout include the feedwater inverter and the unit-preferred and plant-preferred systems for which single-phase 120-V ac power is produced under emergency conditions by generators driven by battery-powered dc motors. Emergency control room lighting would be available.

With respect to application of the EPGs to the late phase of a severe accident sequence, these guidelines are not intended to propose actions in response to the accident symptoms that would be created by events occurring only after the onset of significant core damage. The final guidance to the operators, should an accident proceed into severe core damage and beyond, is that reactor vessel injection should be restored by any means possible and that the reactor vessel should be depressurized. While these are certainly important and worthwhile endeavors, additional guidance can and should be provided for the extremely unlikely, but possible, severe accident situations where reactor vessel injection cannot be restored before significant core damage and structural relocation have occurred.

While the probability of a BWR severe accident involving significant core damage is extremely low, there may be effective yet inexpensive mitigation measures that could be implemented employing the existing plant equipment and requiring only additions to the plant emergency procedures. Based upon the considered need for additional guidelines for BWR severe accident management for in-vessel events, four candidate late accident mitigation strategies are identified.

1. Keep the reactor vessel depressurized. Reactor vessel depressurization is important should an accident sequence progress to the point of vessel bottom head penetration failure because it would preclude direct containment heating (DCH) and reduce the initial threat to containment integrity. This candidate strategy would provide an alternate means of reactor vessel venting should the SRVs become inoperable because of loss of control air or dc power. PRAs consistently include accident sequences involving loss of dc power and control air among the dominant sequences leading to core melt for BWRs.

2. Restore injection in a controlled manner. Late accident mitigation implies actions to be taken after core melting, which requires at least partial uncovering of the core, which occurs because of loss of reactor vessel injection capability. BWRs have so many electric motor-driven injection systems that loss of injection capability implies loss of electrical power. (This is why station blackout is consistently identified by PRAs to be the dominant core melt precursor for BWRs.) If electric power is restored while core damage is in progress, then the automatic injection by the low-pressure, high-capacity pumping systems could be more than 200 times greater than that necessary to remove the decay heat. This strategy would provide for controlled restoration of injection and is particularly important if the control blades have melted and relocated from the core.

3. Inject boron if control blade damage has occurred. This strategy would provide that the water used to fill the reactor vessel after vessel injection capability was restored would contain a concentration of the boron-10 isotope sufficient to preclude criticality, even if none of the control blade neutron poison remained in the core region. This candidate strategy is closely related to Item 2.

4. Containment flooding to maintain core and structural debris in-vessel. This candidate strategy is proposed as a means to maintain the core residue within the reactor

vessel if vessel injection cannot be restored as necessary to terminate the severe accident sequence. Containment flooding to above the level of the core is currently incorporated within the EPGs as an alternative method of providing a water source to the vessel in the event of design-basis LOCA (the water would flow into the vessel from the containment through the break). Here it is proposed that containment flooding might also be effective in preventing the release of molten materials from the reactor vessel for the risk-dominant non-LOCA accident sequences such as station blackout.

The third category of this report provides a reconsideration of these four candidate late-phase, in-vessel strategies for the purpose of identifying any that require (and have sufficient potential to justify) detailed quantitative assessment and for carrying out the additional analyses. The candidate strategy to keep the reactor vessel depressurized is not recommended for further assessment at this time because it is thought far more practical to improve the reliability of the control air and dc power supplies for the SRVs than to invent alternative methods for venting of the reactor vessel into the secondary containment under severe accident conditions. Nevertheless, consideration of the reliability of control air and dc power should be an important part of the IPE process because loss of these systems is involved in the risk-dominant sequences leading to core melt consistently identified for BWRs by PRAs such as the recent NRC-sponsored risk assessment (NUREG-1150).

The candidate strategies for restoration of injection in a controlled manner and injection of boron if control blade damage has occurred are recommended to be combined into a single concept for "Prevention of BWR Criticality as a Late Accident Mitigation Strategy." This would provide a sodium borate solution for the injected flow being used to recover the core, in sufficient concentration to preclude criticality as the water level rises within the reactor vessel. This strategy could be implemented using only the existing plant equipment but employing a different chemical form for the boron poison. Available information concerning the poison concentration required is derived from the recent Pacific Northwest Laboratory (PNL) study, *Recriticality in a BWR Following a Core Damage Event*, NUREG/CR-5653. This study indicates that much more boron would have to be injected than is available (as a solution of sodium pentaborate) in the standby liquid control system (SLCS). Furthermore, the dominant BWR severe accident sequence is station blackout, and without means for mechanical stirring or heating of the injection source, the question of being able to form the poisoned solution under accident conditions becomes of supreme importance. Hence the need for the alternate chemical form.

Polybor®, produced by the U.S. Borax Company, seems to be an ideal means for creating the required sodium borate solution. It is formed of exactly the same chemical constituents (sodium, boron, oxygen, and water) as sodium pentaborate but has the advantages that for the same boron concentration, it requires about one-third less mass of powder addition and has a significantly greater solubility in water. Whereas sodium pentaborate solution is formed by adding borax and boric acid crystals to water, which then react to form the sodium pentaborate, a solution of Polybor® is formed simply by dissolving the Polybor® powder in water. This attribute, that two separate compounds are not required to interact within the water, is a major reason for the greater solubility of Polybor®.

The specific goal of the proposed strategy is to provide for the addition of the boron-10 isotope together with the flow being used to recover the core, in sufficient quantity to preclude criticality as the water level rises within the reactor vessel. NUREG/CR-5653 provides the estimate that a boron-10 concentration of between 700 and 1000 ppm would be required within the vessel to preclude criticality once control blade melting had occurred. This is much greater than the concentration (about 225 ppm) attainable by injection of the entire contents of the SLCS tank.

One means to achieve such a high reactor vessel boron concentration would be to mix the powder directly with the water in the plant condensate storage tank and then take suction on this tank with the low-pressure system pump to be used for vessel injection. It is, however, not a simple matter to invoke this strategy, and preplanning and training would be necessary.

During normal reactor operation, the condensate storage tank provides makeup flow to the main condenser hotwells via an internal tank standpipe. Any practical strategy for direct poisoning of the tank contents must provide for partial draining to reduce the initial water volume, particularly if boron-10 concentrations on the order of 700 ppm are to be achieved. The condensate storage tank could be gravity-drained through the standpipe to the main condenser hotwells under station blackout conditions.

Even with partial tank draining, however, the amount of powder required to obtain a boron-10 concentration of 700 ppm is large. Considering the Peach Bottom plant configuration and assuming the use of Polybor® to take advantage of its greater solubility, 19,300 lb (8,750 kg) would have to be added to the partially drained tank. [If borax/boric acid were used, the requirement would be

28,400 lb (12,880 kg).] Clearly, this is too much to be manhandled [50-lb (23-kg) bags] to the top of the tank and poured in. The practical way to poison the tank contents would be to prepare a slurry of extremely high concentration in a smaller container at ground level, then to pump the contents of this small container into the upper opening of the condensate storage tank. (Extremely high concentrations can be achieved with Polybor®.) To avoid any requirement for procurement of additional plant equipment, a fire engine with its portable suction tank might be employed to perform the pumping function.

With the candidate accident management strategy identified, a simplified cost-benefit analysis based upon the methodology described in NUREG-0933, *A Prioritization of Generic Safety Issues*, was performed. Implementation of the strategy was estimated to provide a reduction in the frequency of unmitigated core melting of 1.19E-06 per reactor-year (RY). The strategy proposed would, if implemented, affect the progression of severe accident events during the time window for recriticality, which is opened by the occasion of some core damage (the melting of the control blades). Thus, some core damage is associated even with successful implementation of the strategy. The goal of the strategy is to avert vessel breach and containment failure.

The estimated change in public risk associated with the proposed strategy is found to be 6.1 man-rem/RY. When applied to the present inventory of 38 BWR facilities with an average remaining lifetime of 21.1 years, the total potential risk reduction estimate is 4860 man-rem.

Implementation of the proposed strategy is estimated to involve per-plant expenditures (1982 dollars) of $70,000 for engineering analysis, preparation of procedures, personnel training, management review, and acquisition of material (sodium borate powder in the form of Polybor®). In addition, it is estimated that 20 man-h/RY would be required for periodic procedure review and team training (including drills). With a cost of $56.75/man-h (1982 dollars) and an average remaining plant life of 21.1 years, the average industry cost per reactor is estimated to be about $93,950.

NRC costs for implementation of the proposed strategy would be small because the general approach has already been developed by the Office of Research as a candidate accident management procedure. It is anticipated that the strategy would be implemented on a voluntary, plant-specific basis by the industry. Therefore, no additional NRC development costs would be incurred. Allowance is made, however, for the costs associated with oversight of the associated plant procedures and of the general readiness (status of personnel training) to successfully execute the plant-specific actions. These oversight activities are estimated to require an average NRC cost per reactor of about $7100.

Based upon an average industry cost of $94,000 per reactor and an NRC oversight cost of $7000 per reactor, the total cost (1982 dollars) associated with implementation of this strategy for the 38 BWR facilities is estimated to be $3.84M.

The value/impact assessment consistent with the procedures of NUREG-0933 for the proposed strategy is

$$S = \frac{4860 \text{ man-rem}}{\$3.84 \text{ M}}$$

$$= 1299 \text{ man-rem/\$M},$$

from which a priority ranking of MEDIUM is obtained for the proposed strategy.

Based upon this ranking, what further actions should be recommended? As pointed out in NUREG-0933, decisions should be tempered by the knowledge that the assessment uncertainties are generally large:

> The criteria and estimating process on which the priority rankings are based are neither rigorous nor precise. Considerable application of professional judgment, sometimes guided by good information but often tenuously based, occurs at a number of stages in the process when numerical values are selected for use in the formula calculations and when other considerations are taken into account in corroborating or changing a priority ranking. What is important in the process is that it is systematic, that it is guided by analyses that are as quantitative as the situation reasonably permits, and that the bases and rationale are explicitly stated, providing a "visible" information base for decision. The impact of imprecision is blunted by the fact that only approximate rankings (in only four broad priority categories) are necessary and sought.

With these considerations in mind, it is recommended that each plant assess its need for the proposed strategy based upon the results of its IPE. By far, the most important aspect of this recommended plant-specific assessment of the need for this strategy is the frequency of station

blackout events predicted to progress through the first stages of core damage (the melting of control blades). In the generic analysis of public risk reduction reported here, the probability of a recriticality event was taken to be 1.25E-06/py, based upon the recent PNL study (NUREG/CR-5653).

The PNL study is based upon the NUREG-1150 results for Peach Bottom, which include a core-melt frequency of about 4.5E-06 derived from station blackout events. If individual plants discover in their IPE process that a much lower station blackout core damage frequency applies, then correspondingly lower recriticality potential would also apply, and implementation of the proposed strategy would probably not be practical for their facility.

As a final note with respect to the question of boration under severe accident conditions, it is important to recognize that many of the BWR facilities are currently implementing accident management strategies, on a voluntary basis, to provide backup capability for the SLCS. These backup strategies invoke such methods as modification of the HPCI or RCIC system pump suction piping to permit connection to the SLCS tank, or poisoning of the condensate storage tank. In all known cases, however, the effect of these plant-specific strategies is to provide a means to obtain a reactor vessel concentration of the boron-10 isotope similar to that attainable by use of the SLCS system itself. It seems highly desirable that these facilities should include within their training programs and procedural notes the information that according to the analyses reported in NUREG/CR-5653, this concentration would be insufficient to preclude criticality associated with vessel reflood after control blade melting.

A detailed assessment has also been performed for the proposed strategy for "Containment Flooding to Maintain the Core and Structural Debris within the Reactor Vessel." This strategy, which is the subject of the remainder of this Executive Summary, would be invoked in the event that vessel injection cannot be restored to terminate a severe accident sequence. Geometric effects of reactor vessel size dictate that the effectiveness of external cooling of the vessel bottom head as a means to remove decay heat from an internal debris pool would be least for the largest vessels. Considering also that the motivation for maintaining any core and structural debris within the reactor vessel is greatest for the Mark I drywells, the primary focus of this assessment is upon the largest BWR Mark I containment facilities such as Peach Bottom or Browns Ferry.

The immediate goal of the considered strategy for containment flooding would be to surround the lower portion of the reactor vessel with water, thereby protecting both the instrument guide tube penetration assemblies and the vessel bottom head itself from failure by overtemperature. The threat would be provided by the increasing temperature of the lower plenum debris bed after dryout. First, molten liquids forming within the bed would relocate downward into the instrument guide tubes challenging their continued integrity. Subsequently, heating of the vessel bottom head by conduction from the debris would threaten global failure of the wall by creep rupture.

Nevertheless, it seems beyond question that all portions of the reactor vessel pressure boundary (including the instrument guide tubes) that are in contact with and cooled by water on their outer surfaces would survive any challenge imposed by a lower plenum debris bed or its relocated liquids. There is a problem, however, in that most of the upper portion of the reactor vessel could not be covered by water and, more significant in the short term, much of the outer surface of the vessel bottom head would be dry as well.

That the upper portion of the reactor vessel could not be covered is due to the location within the containment of the drywell vents. Because low-pressure pumping systems would be used for flooding, the drywell would have to be vented during filling and the water level could not rise above the elevation of the vents, at about two-thirds vessel height. That much of the outer surface of the reactor vessel bottom head would be dry is due to the gas pocket that would be trapped within the vessel support skirt during the process of raising the water level within the drywell.

The results of this assessment demonstrate that the existence of a trapped gas pocket beneath the vessel skirt attachment would ultimately prove fatal to the integrity of the bottom head wall. Nevertheless, the most important attribute of drywell flooding, that of preventing early failure of the instrument guide tube penetration assemblies, would be realized. These results are among those listed in Table ES-1 where it is shown (first entry) that in the absence of water, penetration assembly failures would be expected at ~250 min after scram. If penetration failures did not occur, then creep rupture of the bottom head would be expected after 10 h if the bottom head is dry and after 13 h if the drywell is flooded.* The important contribution of drywell flooding is to shift the expected failure mode from penetration failures (Table ES-1 first entry) to bottom head creep rupture (Table ES-1 third entry).

*Calculational uncertainties associated with creep rupture do not permit a determination of bottom head failure time more precise than the ranges indicated in Table ES-1.

Table ES.1 Estimated failure times for the reactor vessel bottom head pressure boundary for Peach Bottom/Browns Ferry short-term station blackout

Drywell flooded	Failure mechanism	Time to failure min	Time to failure h
No	Penetration assemblies	250	4.2
No	Bottom head creep rupture	600–640	10.–10.7
Yes	Bottom head creep rupture	780–840	13.0–14.0

The effectiveness of drywell flooding could be improved if the reactor vessel support skirt were vented to reduce the trapped gas volume and increase the fraction of bottom head surface area contacted by water. Partial venting could be achieved by loosening the cover on the support skirt manhole access hole. This would increase the covered portion of the bottom head from 55% to 73% of the total outer surface area, which delays the predicted time of bottom head creep rupture by ~1 h. The predicted failure times for the basic case without skirt venting and for the case of partial venting at the manhole access are indicated in the first two entries of Table ES-2.

Table ES.2 Effect of skirt venting upon time to failure of the bottom head pressure boundary for Peach Bottom/Browns Ferry short-term station blackout with drywell flooding

Skirt vented	Failure mechanism	Time to failure min	Time to failure h
No	Bottom head creep rupture	780–840	13.0–14.0
Partial	Bottom head creep rupture	840–900	14.0–15.0
Complete	Melting of upper vessel wall	>1200	>20.0

Complete venting of the reactor vessel support skirt would provide 100% water coverage of the vessel bottom head but would require special measures such as provision of a siphon tube or the drilling of small holes at the upper end of the skirt, just below the attachment weld. Because of the associated personnel radiation exposure penalty and the predicted low core melt frequencies for the existing plants, this is not considered to be a practical suggestion for the existing BWR facilities, but provision for complete venting might be easily implemented for the advanced BWR designs. As indicated by the last entry in Table ES-2, 100% water coverage of the vessel bottom head would convert the failure mechanism from bottom head creep rupture to melting of the upper vessel wall and would delay the predicted time of failure to more than 20 h after scram.

In summary, all portions of the reactor vessel wall that are covered by water would be adequately protected against failure by melting or creep rupture. For the cases with no venting or partial venting of the support skirt, the creep rupture failure is predicted to occur in the portion of the vessel wall adjacent to the trapped gas pocket beneath the skirt. Partial venting would reduce the size of the gas pocket and delay the predicted time of failure, but the failure mechanism would still be creep rupture beneath the skirt attachment weld. With complete venting, however, there would be no gas pocket and this failure mechanism would be eliminated.

What cannot be eliminated, however, is the radiative heat transfer upward within the reactor vessel from the surface of the lower plenum debris bed. About one-half to two-thirds of all energy release within the bed would be radiated upward after bottom head dryout. Initially, the primary heat sink for this radiation would be the water trapped in the downcomer region between the core shroud and the vessel wall above the debris bed. It is the heating of this water that creates the only steam source within the reactor vessel after lower plenum dryout.

After the water in the downcomer region became exhausted, the upward radiative heat transfer from the debris surface would serve to increase the temperature of the upper reactor vessel internal structures. For calculations with the existence of a gas pocket beneath the skirt, bottom head creep rupture is predicted to occur while the temperature of these internal stainless steel heat sinks remains below the melting point. If bottom head creep rupture did not occur, however, the debris would remain within the vessel, the upward radiation would continue, and the upper internal structures would melt.

The mass of the BWR internal structures (core shroud, steam separators, dryers) is large. Melting of these stainless steel structures under the impetus of the upward debris pool radiation more than 14 h after scram would occur over a long period of time. Nevertheless, decay heating of the debris pool and the associated upward radiation would be relentless and, after exhaustion of the stainless steel, the only remaining internal heat sink above the pool surface would be the carbon steel of the upper vessel wall. All portions of the wall cooled by water on their outer surfaces

would remain intact, but unless the water height within the drywell extended well above the surface of the debris pool, upper portions of the vessel exposed to the drywell atmosphere would ultimately reach failure temperatures.

It should be obvious from this discussion of the effect of water upon cooling of the vessel wall that it would be desirable to have a drywell flooding strategy that would completely submerge the reactor vessel. This could not be achieved in existing facilities because of the limitation that the height of water within the drywell cannot exceed the elevation of the drywell vents. Future designs, however, might provide for complete coverage of the reactor vessel as a severe accident mitigation technique.

Table ES-3 provides a summary of the calculated failure times and release mechanisms for all of the cases considered in this study. These include the cases previously discussed in connection with Tables ES-1 and ES-2, plus one additional case (third entry) in which it is assumed that reactor vessel pressure control is lost at the time of drywell flooding, because of the submergence of the safety/relief valves. The increased wall tensile stress associated with this case would cause the wall creep rupture to occur at a lower temperature, advancing the time of failure by about 2 h over the depressurized case (compare the third and fourth entries in Table ES-3).

The most important disadvantage of a drywell flooding strategy for existing plants is the requirement for venting to the external atmosphere while the containment is being filled by the low-pressure pumping systems and during the subsequent steaming from the water surrounding the reactor vessel bottom head. Because of this, implementation of the drywell flooding strategy would initiate a noble gas release to the surrounding atmosphere as well as a limited escape of fission product particulates. All particulate matter released from the reactor vessel before failure of the vessel wall would enter the pressure suppression pool via the SRV T-quenchers and would be scrubbed by passage through the water in both the wetwell and drywell. Therefore, the concentration of particulates in the drywell atmosphere and any release through the drywell vents would remain small as long as the reactor vessel wall remained intact.

Creep rupture of the vessel bottom head beneath the support skirt attachment would release debris into the water-filled pedestal region to fall downward onto the drywell floor. Because containment flooding would provide a water depth of more than 30 ft (9.144 m) over the drywell floor, the particulate matter released from the debris mass should be adequately scrubbed provided, of course, that violent steam explosions do not occur. Furthermore, the large volume of water in the drywell would protect the drywell shell from late failure in Mark I containment facilities, because the accumulating volume of debris would never break the water surface.

Table ES.3 Effect of drywell flooding upon time of debris release from the reactor vessel for the short-term station blackout accident sequence based upon Peach Bottom/Browns Ferry

				Time to failure	
Drywell flooded	Skirt vented	Reactor vessel depressurized	Release mechanism	min	h
No		Yes	Penetration failures	250	4.2
No		Yes	Bottom head creep rupture	600–640	10.0–10.7
Yes	No	No	Bottom head creep rupture	660–700	11.0–11.7
Yes	No	Yes	Bottom head creep rupture	780–840	13.0–14.0
Yes	Partial	Yes	Bottom head creep rupture	840–900	14.0–15.0
Yes	Complete	Yes	Melting of upper vessel wall	>1200	>20.0

The advantages and disadvantages of a drywell flooding strategy for existing BWR facilities are summarized in Table ES-4. The listed advantages involve significant contributions to accident mitigation, which have previously been discussed. The listed disadvantages, however, are also important and will be discussed in the following paragraphs.

Table ES.4 Advantages and disadvantages of a drywell flooding strategy for severe accident mitigation in existing BWR facilities

Advantages	Prevent failure of the bottom head penetrations and vessel drain
	Increased scrubbing of fission product particulate matter
	Delay creep rupture of the reactor vessel bottom head
	Prevent failure of the Mark I drywell shell when core debris does leave the vessel
Disadvantages	Requires availability of power source and pump capable of filling the drywell to the level of the vessel bottom head within 150 min under station blackout conditions
	Requires that the drywell be vented

First, implementation of the proposed strategy would require equipment modifications and additions. Although there may be plant-specific exceptions, containment flooding with the existing pumping systems would require too much time; furthermore, the existing systems would not be available for the dominant station blackout accident sequences. What is needed is a reliable ability to sufficiently flood the drywell within a short period of time, because it would be unrealistic to expect that emergency procedures would call for containment flooding (and the associated undesirable effects upon installed drywell equipment) until after core degradation has begun. If the water did not reach the vessel bottom head until after lower plenum debris bed dryout and the initial heating of the vessel wall, it would be too late to prevent penetration assembly failures.

The second disadvantage, that the drywell vents would have to be opened to permit flooding of the containment, is particularly undesirable because it would involve early release of the fission product noble gases, beginning soon after the onset of core degradation. (The vents would be opened before core degradation.) After the water had contacted the vessel bottom head, a continuous steam generation would begin within the drywell that would be released to the outside atmosphere by means of the open vents. This would tend to sweep any particulate matter from the drywell atmosphere through the vents. The amount of particulate matter reaching the drywell atmosphere would, however, be limited by water scrubbing as long as the reactor vessel wall remained intact above the water level in the drywell. This is expected to be the case for the existing BWR facilities where the ultimate failure of the wall would occur by creep rupture beneath the skirt attachment weld.

It is interesting, however, to briefly consider the potential benefits of application of a drywell flooding strategy to future BWR facilities, where the disadvantages listed in Table ES-4 might be avoided by appropriate plant design. Much less water would be required because the reactor vessel would be located in a cavity instead of suspended high above a flat drywell floor. Provision could be made for complete venting of the reactor vessel support skirt so that all of the bottom head would be in contact with water. This would preclude creep rupture of the vessel bottom head, shifting the failure mode to melting of the upper vessel wall, above the water level in the drywell.

For the existing BWR facilities, failure of the upper reactor vessel wall would provide a direct path from the upper surface of the debris pool to the open drywell vents without the benefit of water scrubbing.* For future plant designs, this could be avoided in two ways. First, submergence of most, or all, of the reactor vessel wall above the debris pool surface would preclude failure of the upper vessel wall. Second, the requirement for containment venting could be eliminated by provision of an adequate water source within the containment and provision for condensation of the generated steam. Both of these approaches are within the scope of design features currently under consideration for the advanced passive design.

*This case corresponds to the last entry in Table ES-3. The reader is reminded that it is based upon complete venting of the vessel support skirt, which is not considered practical for the existing facilities.

1 Introduction

Boiling water reactors (BWRs) have unique features that would cause their behavior under severe accident conditions to differ significantly from that expected for the pressurized water reactor design.[1–5] Consequently, it has been necessary to analyze BWR accident sequences separately, and the Nuclear Regulatory Commission (NRC) has sponsored programs at Oak Ridge National Laboratory (ORNL) for this purpose since 1980. The objective of these BWR severe accident programs has been to perform analyses of a spectrum of accident sequences beyond the design basis for typical specific U.S. BWR reactor designs. The accident sequences selected for analysis have been in general those identified as dominant in leading to core melt for BWRs by the methods of probabilistic risk assessment (PRA) as carried out by other programs. The specific plants modeled and the accident sequences considered were selected by the process of nomination by the ORNL program manager and approval by the NRC technical monitors.

The detailed analyses of the dominant BWR severe accident sequences initially identified by PRA have been performed in recognition that PRA, by the basic nature of its requirements to consider every possible accident sequence, cannot enter into matters of detail. The purpose of the detailed analyses has been either to confirm the adequacy of or to challenge the simplifying assumptions necessarily applied to the candidate dominant sequences in the PRA and to provide a realistic appraisal of the sequence of events and the aftermath. Further preventative measures that might be taken to decrease the probability and accident management procedures that can be implemented to reduce the consequences of each severe accident sequence studied have been addressed. Feedback of the results of the detailed analyses has always been provided to the other facilities performing the PRA; most recently, this has involved close cooperation with the NUREG-1150 effort.[6]

With the comprehensive information provided by NUREG-1150 concerning the relative probabilities of BWR severe accident sequences and with the knowledge and experience gained from the series of detailed accident analyses,[7–15] it is now logical and practical to consider the facets of BWR severe accident management in a structured process, with the goal of identifying potential new strategies and enhancements. Therefore, the first purpose of this report is to assess the current status of accident management procedures with respect to their potential for effective mitigation of the dominant BWR severe accident sequences. To this end, the BWR Owners' Group Emergency Procedure Guidelines (EPGs)[16] have been reviewed to determine the extent to which they currently address the characteristic events of an unmitigated severe accident and to provide the basis for recommendations for enhancement of accident management procedures.

1.1 Report Outline

Chapter 2 provides a discussion of the progression of events for the dominant BWR severe accident sequences identified by NUREG-1150 and other PRA studies. The importance of plant-specific considerations in the application of the lessons learned from PRAs is also addressed.

The current status of accident management procedures for coping with the dominant BWR severe accident sequences is discussed in Chap. 3. In particular, the BWR Owners' Group EPGs (Revision 4) are assessed to determine the extent to which they implement the intent of several candidate accident management strategies previously identified in a companion study at Brookhaven National Laboratory (BNL)[17] and for their effectiveness in unmitigated severe accident situations. Where the EPGs are currently effective, no further recommendations concerning proposed enhancements are necessary.

Based upon the dominant severe accident sequences described in Chap. 2 and the current status of severe accident management as discussed in Chap. 3, the potential for enhancement of current BWR accident management strategies is discussed in Chap. 4. It is found that the greatest potential for improvement of BWR emergency procedure strategies lies in the area of severe accident management. This potential is explored in Chaps. 5 through 8.

Specifically, Chap. 5 provides a discussion of BWR severe accident vulnerabilities. The current EPGs with respect to operator actions under severe accident conditions are briefly reviewed in Chap. 6. The information that might be available to the operators concerning plant status under severe accident conditions is described in Chap. 7. Finally, four new strategies for late mitigation of in-vessel events are proposed in Chap. 8 with assessments of their feasibility, effectiveness, and any associated adverse effects.

The remainder of this report provides additional information concerning two of the four strategies proposed in Chap. 8, for which additional assessments are considered justified. Chapters 9 through 13 provide a detailed quantitative analysis of the proposed strategy for prevention of BWR criticality upon reactor vessel flooding if control

blade damage has occurred. The motivation for this strategy is reviewed in Chap. 9.

The most simple and straightforward strategy for injection of a boron solution into the reactor vessel under severe accident conditions would be based upon use of the Standby Liquid Control System (SLCS). The capabilities of this system for such a purpose are discussed in Chap. 10.

The dominant set of BWR accident sequences leading to core damage involves station blackout, where simultaneous initiation of the SLCS would not be adequate to prevent criticality upon vessel reflooding if control blade damage has occurred. The only reliable strategy for prevention of criticality due to control blade damage for all recovered BWR severe accident sequences requires that the water used to recover the core contain a sufficient concentration of the boron-10 isotope to ensure that the reactor remains shutdown. Methods for accomplishing this are discussed in Chap. 11.

Chapter 12 provides a simplified cost-benefit analyses for the proposed strategy. This analysis is derived from and directly follows the methodology described in NUREG-0933, *A Prioritization of Generic Safety Issues.*[18] It provides an evaluation of the estimated risk reduction associated with the proposed strategy and the estimated costs to the NRC and the industry in implementing such a strategy. The priority ranking for the strategy is established in Chap. 13.

As indicated previously, two new strategies for late mitigation of in-vessel events were selected for additional assessment. The second of these considers containment flooding to maintain the core and structural debris within the reactor vessel if vessel injection cannot be restored as necessary to terminate a severe accident sequence. The motivation for this strategy is reviewed in Chap. 14 while the quantitative analysis is provided in Chaps. 15 through 22.

If drywell flooding is to be effective in maintaining reactor vessel integrity, this strategy must be capable of quick implementation, because release of molten materials from the lower plenum debris bed to the drywell floor by means of failed penetration assemblies would otherwise be expected to occur soon after bottom head dryout. The general topic of drywell flooding, the means to accomplish it, and the effectiveness of this maneuver in cooling the reactor vessel exterior surface are addressed in Chap. 15.

While it can be shown that the submerged portions of the reactor vessel wall can be effectively cooled by the presence of water, there are physical limitations to the fraction of the overall decay power that can be removed downward through the lower portion of the debris bed boundary. These unfortunate realities are discussed in Chap. 16.

Cooling of the bottom head can greatly delay any failure of the reactor vessel structural boundary, but the first requirement to accomplish this goal is that failures of the penetration assemblies, induced by dryout and the entry of molten materials, be precluded. The success of water surrounding the vessel exterior in achieving this is described in Chap. 17.

The stand-alone models for the response of the lower plenum debris bed and the reactor vessel bottom head wall are described in Chap. 18. The results obtained by application of these models to the large BWR facilities such as Peach Bottom or Browns Ferry are discussed in Chaps. 19 and 20 for cases with and without venting of the trapped atmosphere within the reactor vessel support skirt.

Current provisions for reactor vessel depressurization as specified by the BWR EPGs are intended to lessen the severity of any BWR severe accident sequence. Therefore, it is important to recognize that drywell flooding, which would submerge the safety/relief valves (SRVs), might lead to failure of the dc power supply and thereby induce vessel repressurization. The potential for failure of SRVs remote control due to drywell flooding and the effects of this eventuality are discussed in Chap. 21.

The general results and recommendations of the three main divisions of the report are summarized in Chap. 22.

Three appendixes provide additional information in detail. One conclusion of the assessment of the strategy for prevention of recriticality is that the use of Polybor® instead of a sodium pentaborate solution generated by a mixture of borax and boric acid crystals would facilitate the implementation of the strategy. Appendix A provides information concerning the characteristics of this special sodium borate product.

Perhaps the most desirable characteristic of Polybor® from the standpoint of the proposed strategy is its ability to readily dissolve in cool water. Appendix B provides a discussion of several simple tabletop experiments performed

at ORNL to investigate the limits and implications of this high solubility.

Finally, Appendix C provides the results of a calculation demonstrating the increased effectiveness of the proposed drywell flooding strategy when applied to the smaller BWR reactor vessels.

1.2 Selection of Units for Text

In preparing a report such as this, the authors are challenged to choose a primary system of units for use in the text that will be most in line with the experience of their intended readers. Because this report deals with current and proposed accident management procedures for existing BWR facilities in the United States, the choice seems simple—American Engineering Units. The Final Safety Analysis Reports (FSARs), operator training manuals, and control room instruments for these facilities employ only this system of units.

Nevertheless, among the chapters of this report, there are areas where the topic under discussion involves recent experiments or other considerations without direct application to any existing facility. In these cases, the acknowledged superiority of the International System of units (SI) and the general trend toward adoption of this system throughout the world dictates its use as the primary system within these portions of the text.

In summary, the authors have attempted to employ the optimum system of units for primary use within each of the various chapters and sections of this report. This has required exercise in judgement in proceeding from topic to topic. Where appropriate, quantities cited in the text are repeated (within parentheses) in the secondary system of units. This has not been practical, however, for many of the extensive tables of calculated information, where only the primary system of units is employed.

2 Dominant BWR Severe Accident Sequences

S. A. Hodge

This chapter provides a discussion of the risk-dominant boiling water reactor (BWR) severe accident sequences identified by the recent NRC-sponsored assessment of severe accident risks (NUREG-1150)[6] and other probabilistic risk assessment (PRA) studies. The importance of plant-specific considerations in the application of the lessons learned from PRA is also addressed.

2.1 Results from PRA

The severe accident risks report (NUREG-1150) considers the five representative plants listed in Table 2.1. The calculated mean core damage frequencies (internally initiated accidents) for these plants are provided in Table 2.2. As indicated, the calculated BWR core damage frequencies are approximately one order-of-magnitude less than the PWR frequencies. A breakdown of the major contributors to core damage frequency for the two BWR plants is shown in Fig. 2.1. Station blackout and anticipated transient without scram (ATWS) are the predominant contributors.

Table 2.1 Commercial nuclear power plants considered in the severe accident risks assessment (NUREG-1150)[6]

Pressurized water reactors (Westinghouse)	
Sequoyah	1148-MW(e) four-loop ice condenser (1981)
Surry	788-MW(e) three-loop subatmospheric (1972)
Zion	1100-MW(e) four-loop large dry (1978)
Boiling water reactors (General Electric)	
Peach Bottom	1150-MW(e) BWR-4 Mark I containment (1974)
Grand Gulf	1250-MW(e) BWR-6 Mark III containment (1985)

Table 2.2 Calculated mean core damage frequencies (internally initiated accidents) from NUREG-1150[6]

Plant type	Plant	Frequency per 10,000 reactor-years
PWR	Sequoyah	0.57
	Surry	0.41
	Zion	3.40
BWR	Peach Bottom	0.045
	Grand Gulf	0.040

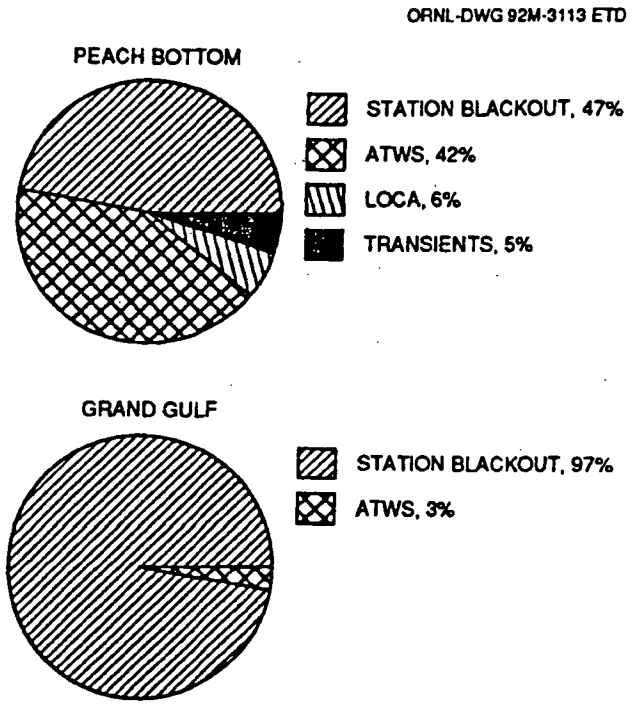

Figure 2.1 Dominant accident sequence contributors: station blackout and ATWS

Although the initial NUREG-1150 study did not include consideration of a BWR plant with Mark II containment, this is being rectified by additional work currently being performed at Sandia National Laboratories based upon the LaSalle design, a 1036-MW(e) BWR-5 with Mark II containment. In the meantime, results[19,20] from other PRAs for two Mark II containment plants are available; these are summarized in Table 2.3. As indicated, station blackout and ATWS are again identified as dominant contributors to the overall risk of BWR core melt.

2.2 Description of Accident Progression

This section provides background information concerning the progression of the dominant BWR severe accident sequences as necessary to support the subsequent discussion (Chap. 3) of the extent to which the current accident management strategies would be effective in coping with an unmitigated sequence. Events occurring after the onset of severe core damage and material relocation are not well understood at the present time, and other scenarios have been postulated. The accident progressions described here represent the opinions of the authors of this report.

Dominant

Table 2.3 Relative contribution of dominant accident sequences to core-melt frequency for two BWRs with Mark II containment

Plant	Accident sequence	Relative contribution (%)
Limerick	Station blackout	42
	ATWS	28
	Loss of injection	27
	Loss of decay heat removal	3
Susquehanna	Station blackout	62
	ATWS	32
	LOCA	4
	Transients	2

2.2.1 Station Blackout

Historically, station blackout has been considered to be the accident sequence initiated by loss of off-site power and the associated reactor scram, combined with failure of the station diesels (and gas turbines, if applicable) to start and load. However, with the advent of the accident classification methodology adopted for the recent severe accident risks study (NUREG-1150), this accident sequence is now classified as "long-term station blackout." In this accident sequence, water is injected into the reactor vessel by the steam turbine-driven reactor core isolation cooling (RCIC) or high-pressure coolant injection (HPCI) systems as necessary to keep the core covered for as long as dc (battery) power for turbine governor control remains available from the unit batteries, a period of about 6 h. The rationale for the "long-term" designation is that the definition of station blackout has been expanded to include two cases that heretofore would have been classified as loss of injection, or TQUV, in WASH-1400 parlance. In these "short-term" station blackout sequences, the capability for water injection to the reactor vessel is lost at the inception of the accident sequence. The general distinction between long-term and short-term station blackout is illustrated in Fig. 2.2. It is useful to remember that the short-term designation derives from the fact that the BWR core is uncovered relatively quickly for this category of station blackout.

The early total loss of injection initiating event for short-term station blackout might occur in either of two ways. First, there might be *independent* failures of both the RCIC and HPCI steam turbine systems when they are called upon to keep the core covered during the period while dc power remains available. (Because these are high-pressure injection systems, success of their function does not depend

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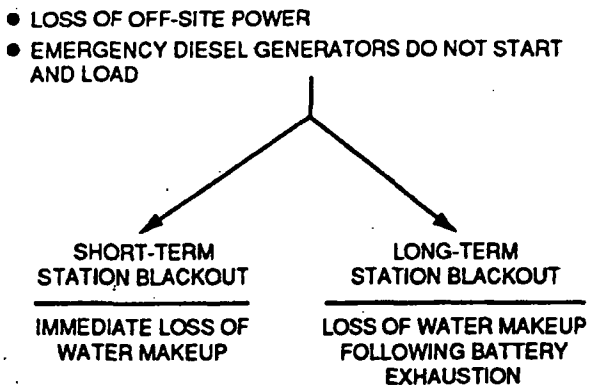

Figure 2.2 Station blackout involving loss of ac electrical power

upon reactor vessel depressurization.) In this connection, it should be recognized that the BWR-5 and BWR-6 designs have substituted an electric-motor driven high-pressure core spray (HPCS) system in lieu of HPCI so that these plants have only one steam-turbine driven injection system (RCIC). Similarly, the BWR-2 and early BWR-3 plants employ a feedwater coolant injection system (FWCI) instead of HPCI; operation of the FWCI system also requires ac power. However, as indicated in Table 2.4, most of the U.S. BWR plants (25 out of 38 in operation or under construction) have two independent systems that can keep the core covered without the availability of ac power.

It should be noted from Table 2.4 that the BWR-2 plants and three of the older BWR-3 plants employ an isolation condenser (IC) in lieu of RCIC. This emergency core cooling system employs natural circulation through the elevated isolation condenser tubes to remove decay heat from the reactor vessel if electrical power is unavailable. The shell volume (vented to the atmosphere) of the condenser contains a water volume that boils away to remove the heat transferred from the reactor. The shell water volume is sufficient to provide about 30 min of core cooling without the addition of makeup water (which is taken from the firemain system, or from the condensate transfer system).

The second way in which the early total loss of injection initiating event for short-term station blackout might occur is by a common-mode failure of the dc battery systems. The diesel generators at Peach Bottom are started from the unit batteries, and therefore failure of these batteries would, upon loss of off-site power, preclude starting of the diesel generators and thereby be a contributing cause of the station blackout. Furthermore, without dc power for valve operation and turbine governor control, the steam turbine-driven injection systems would not be operable. For plants other than Peach Bottom, this second version of short-term

Table 2.4 Availability of reactor vessel injection systems that do not require ac power to maintain the core covered

Plant	Class	RCIC or IC	HPCI
Big Rock Point	BWR-1		
Oyster Creek	BWR-2	IC	
Nine Mile Point 1	BWR-2	IC	
Millstone	BWR-3	IC	
Dresden 2 and 3	BWR-3	IC	HPCI
Monticello	BWR-3	RCIC	HPCI
Quad Cities 1 and 2	BWR-3	RCIC	HPCI
Pilgrim	BWR-3	RCIC	HPCI
Browns Ferry 1, 2, and 3	BWR-4	RCIC	HPCI
Vermont Yankee	BWR-4	RCIC	HPCI
Duane Arnold	BWR-4	RCIC	HPCI
Peach Bottom 2 and 3	BWR-4	RCIC	HPCI
Cooper	BWR-4	RCIC	HPCI
Hatch 1 and 2	BWR-4	RCIC	HPCI
Brunswick 1 and 2	BWR-4	RCIC	HPCI
Fitzpatrick	BWR-4	RCIC	HPCI
Fermi 2	BWR-4	RCIC	HPCI
Hope Creek 1	BWR-4	RCIC	HPCI
Susquehanna 1 and 2	BWR-4	RCIC	HPCI
Limerick 1 and 2	BWR-4	RCIC	HPCI
LaSalle 1 and 2	BWR-5	RCIC	
WNP-2	BWR-5	RCIC	
Nine Mile Point 2	BWR-5	RCIC	
Grand Gulf	BWR-6	RCIC	
Perry 1 and 2	BWR-6	RCIC	
Clinton	BWR-6	RCIC	
River Bend	BWR-6	RCIC	

station blackout is much less likely to occur than the first, simply because the diesel generators have independent starting batteries. It should be noted, however, that the loss of dc power associated with this second version would also render the safety/relief valves (SRVs) inoperable in the remote-manual mode; thus, the reactor vessel could not be depressurized.

With respect to the basic characteristics of the dominant forms of the severe accident sequence, the difference between long-term and short-term station blackout can be summarized as follows: dc power remains available during the period of core degradation for short-term station blackout, which is initiated by independent failures of HPCI and RCIC; the decay heat level is relatively high, and the reactor vessel is depressurized during the period of core degradation and material relocation within and from the vessel. For long-term station blackout, the core remains covered for more than 6 h so the decay heat level is ~50% less during the period of core degradation and, because the SRVs cannot be manually operated without dc power, the reactor vessel would be pressurized at the time of bottom head penetration failure and initial release of debris from the vessel.

2.2.1.1 Event Sequence for Short-Term Station Blackout

The following summary of events is based upon calculations recently performed at Oak Ridge with the BWRSAR code[2,3] in support of the Task Group on the BWR Mark I Shell Melt-Through Issue. The reference plant is Peach Bottom.

If a Peach Bottom unit were operating at 100% power when station blackout occurs, and if both HPCI and RCIC were to fail upon demand, then the swollen reactor vessel water level would fall below the top of the core in ~40 min. This assessment is based upon an assumption that the operators follow procedure and manually operate the SRVs as necessary to maintain reactor vessel pressure in the range between 1100 and 950 psig (7.69 and 6.65 MPa). [A slight delay (~3 min) could be obtained if the SRVs were left to automatic actuation only.] The events after core uncovery would progress rapidly because the decay heat level is relatively high for the short-term branch of the station blackout event tree.

DC power from the unit battery would remain available during and after core uncovery so that the operators could take the actions regarding manual SRV operation that are directed by the Emergency Operating Instructions (EOIs). [These local emergency procedures are based upon the BWR Owners Group Emergency Procedure Guidelines (EPGs).] Specifically, EOI-1, reactor pressure vessel (RPV) control, directs the operators to open one SRV if the reactor vessel water level cannot be determined and to open all (five) automatic depressurization system (ADS) valves when the reactor vessel pressure falls below 700 psig (4.93 MPa). (Water level indication would be lost when the level drops below the indicating range at about one-third core height.) These actions provide temporary cooling of the partially uncovered core and are predicted to be taken at times 77 and 80 min after scram, respectively.

The high rate of flow through the open SRVs would cause rapid loss of reactor vessel water inventory, and core plate dryout would occur at about time 81 min. Heatup of the

Dominant

totally uncovered core would then lead to significant structural relocation (molten control blade and canister material), beginning at time 131 min. Because the core plate would be dry at this time, heatup and local core plate failure would occur immediately after debris relocation began. Subsequent core damage would proceed rapidly, with the fuel rods in the central regions of the core predicted to be relocated into the reactor vessel lower plenum at time 216 min.

Following the code prediction, the continually accumulating core debris in the reactor vessel lower plenum would transfer heat to the surrounding water over a period of ~30 min until initial bottom head dryout, and in the process, the lower layer of core debris would be cooled to an average temperature of 1310°F (983 K). After vessel dryout occurs at time 246 min, the temperature in the middle debris layer would be sufficient [2375°F (1575 K)] to cause immediate failure of the control rod guide tubes in the lower plenum. Failure of this supporting structure would immediately cause the remaining intact portions of the core to collapse into the lower plenum.

The temperatures in the central portions of the lower plenum debris bed would also be sufficient to cause failure of the reactor vessel bottom head penetrations. However, because the operators, by procedure, would have already depressurized the reactor vessel through the SRVs, no significant containment pressure increase would be associated with the failure of the bottom head pressure boundary.

The metallic components of the core debris in the reactor vessel bottom head are predicted to begin melting and to begin to flow onto the drywell floor at time 263 min. The containment response to the presence of core debris upon the drywell floor was calculated by the CONTAIN code; however, discussion of the predicted containment response is beyond the scope of this report.

The synopsis of the major events of the short-term station blackout accident sequence together with the calculated event timing for this illustrative example is provided in Table 2.5. Because of the relatively high decay heat levels during the period of core degradation, this is a fast-moving sequence. The core is uncovered in less than 1 h, and core debris begins to leave the reactor vessel in less than 4.5 h. It must be remembered that the initiation of short-term station blackout requires, in addition to loss of off-site power and failure of all station diesels to start, that there also be independent failures of both the HPCI and RCIC systems. Thus, the short-term station blackout accident sequence is much less likely to be initiated than is the long-term case.

Table 2.5 Calculated sequence of events for Peach Bottom short-term station blackout with ADS actuation

Event	Time (min)
Station blackout-initiated scram from 100% power. Independent loss of the steam turbine-driven HPCI and RCIC injection systems	0.0
Swollen water level falls below top of core	40.3
Open one SRV	77.0
ADS system actuation	80.0
Core plate dryout	80.7
Relocation of core debris begins	130.8
First local core plate failure	132.1
Collapse of fuel pellet stacks in central core	215.9
Reactor vessel bottom head dryout; structural support by control rod guide tubes fails; remainder of core falls into reactor vessel bottom head	245.8
Bottom head penetrations fail	255.0
Pour of molten debris from reactor vessel begins	263.1

Before proceeding to a discussion of the long-term station blackout accident sequence, it is necessary to briefly consider an important variation of the short-term case. If the reactor vessel depressurization initiated by manual actuation of the ADS does not occur, either by system malfunction (loss of dc power) or by failure of the operator to follow procedures, then liquid water would remain in the lower core region during the early portion of the core degradation phase of the short-term station blackout accident sequence. The associated steam generation would cause a much-higher degree of metal-water reaction within the core region and an accelerated core degradation rate by means of the associated energy release.

Table 2.6 provides a comparison of the timing of events for two calculations of the short-term station blackout accident sequence based upon the Susquehanna plant. In the first calculation, the operators manually actuate the ADS when the reactor vessel water level has fallen to one-third core height. For the second calculation, there is no manual SRV actuation of any kind. (This also delays the time that the core becomes uncovered, by ~1 min.) These are recent calculations with the BWRSAR code performed in support of the NRC Containment Performance Improvement (CPI) Program; a detailed discussion of results is provided in Ref. 21.

Table 2.6 Calculated timing of sequence events for two cases of the short-term station blackout accident sequence

Event	Time (min)	
	With ADS	Without ADS
Loss of offsite power-initiated scram from 100% power. Failure of on-site ac power. Independent loss of the steam turbine-driven HPCI and RCIC injection systems	0.0	0.0
Swollen water level falls below top of core	37.2	38.2
Open one SRV	78.0	
ADS system actuation	79.5	
Core plate dryout	81.2	
Relocation of core debris begins	124.1	90.6
First local core plate failure	129.2	
Core plate dryout		135.3
First local core plate failure		155.8
Collapse of fuel pellet stacks in central core	220.0	163.8
Reactor vessel bottom head dryout; structural support by control rod guide tubes fails; remainder of core falls into reactor vessel bottom head	263.2	193.8
Initial failure of bottom head penetrations	263.3	246.1

It is emphasized that the BWR Owners Group EPGs require unequivocally that the operators act to manually depressurize the reactor vessel should the core become partially uncovered under station blackout conditions. As indicated in Table 2.6, this delays the onset of core degradation, allowing more time for the restoration of reactor vessel injection capability before significant core damage has occurred. The rapid vessel depressurization causes flashing of the water in the core region and core plate dryout so that if the accident is not terminated, the subsequent core degradation would occur under steam-starved conditions. Finally, because the reactor vessel is depressurized at the time of bottom head penetration failure, there would be no large energy release within the bottom head debris bed by metal-water reactions during vessel blowdown. The advantages of reactor vessel depressurization will be addressed in more detail in Chap. 3.

2.2.1.2 Event Sequence for Long-Term Station Blackout

With the HPCI and RCIC systems available, the operators would act to maintain reactor vessel water level in the normal operating range by intermittent operation of the RCIC system, with the higher-capacity HPCI system as a backup. The operators would also, by procedure, take action during the initial phase of the accident sequence to control reactor vessel pressure by means of remote-manual operation of the reactor vessel relief valves. The SRVs would actuate automatically to prevent vessel overpressurization if the operator did not act; the purpose of pressure control by remote-manual operation is to reduce the total number of valve actuations by means of increasing the pressure reduction per valve operation and to permit the steam entering the pressure suppression pool to be passed by different SRVs in succession. This provides a more even spacial distribution of the transferred energy around the circumference of the pressure suppression pool.

The plant response during the initial phase (before battery failure) of a long-term station blackout can be summarized as an open cycle. Water would be pumped from the condensate storage tank into the reactor vessel by the RCIC system as necessary to maintain indicated water level in the normal operating range. The injected water mass would be heated by the reactor decay heat and subsequently passed to the pressure suppression pool as steam during the periods when the operator remote-manually opens different relief valves in succession as necessary to maintain the desired reactor vessel pressure. (Some of the steam is passed to the pressure suppression pool via the RCIC turbine.) Stable reactor vessel level and pressure control would be maintained during this period, while the condensate storage tank is being depleted and both the level and temperature of the pressure suppression pool are increasing. However, without question, the limiting factor for continued removal of decay heat and the prevention of core uncovery is the availability of dc power.

The results of the Oak Ridge Severe Accident Sequence Analysis (SASA) study[7] of station blackout based upon the Browns Ferry plant establish the reasons why it would be

beneficial for the operators to depressurize the reactor vessel early in the initial phase of a long-term station blackout, while dc power for SRV operation remained available. Briefly, this manually instigated depressurization would remove a great deal of steam and the associated stored energy from the reactor vessel during the period, while the steam turbine-driven HPCI and RCIC systems remained available for use in injecting replacement water from the condensate storage tank, thereby maintaining the reactor vessel level. Subsequently, when water injection capability is lost by battery exhaustion, remote-manual capability for relief valve operation is also lost; there would be no further steam discharge from the reactor vessel until its internal pressure is restored to the setpoint [1105 psig (7.72 MPa)] for automatic SRV actuation. Because of the large amount of water to be reheated and the reduced level of decay heat, this repressurization would require a significant period of time. Furthermore, the subsequent boiloff would begin from a very high vessel level because of the increase in the specific volume of the water as it is heated and repressurized. Thus, an early depressurization will provide a significant period (~2 h) of valuable additional time for preparative and possible corrective action before core uncovery after injection capability is lost.

Operator action to depressurize the reactor vessel is required by the BWR Owners Group EPGs whenever the "Heat Capacity Temperature Limit," based upon the temperature of the pressure suppression pool, is exceeded. The curve defining this limit for the Browns Ferry plant requires that reactor vessel depressurization begin when suppression pool temperature exceeds 160°F (344 K) and specifies that reactor vessel pressure must be less than 115 psia (0.79 MPa) whenever suppression pool temperature exceeds 200°F (366 K). Although these requirements are not based upon station blackout considerations, the suppression pool temperature would reach 160°F (344 K) ~4.5 h after initiation of the long-term station blackout accident sequence. Because dc power would still be available at this time, it is expected that the operators would take the required action and that the reactor vessel would be depressurized. The procedures also specify that the depressurization not lower the reactor vessel pressure below 100 psig (0.79 MPa), so that the RCIC or HPCI system steam turbines can continue to be operated. The depressurization would be accomplished by opening one SRV and leaving it open. Reactor vessel pressure would fall rapidly at first, then stabilize at about 140 psia (0.97 MPa).

A synopsis of the major events in the calculated long-term station blackout accident sequence and the event timing is provided in Table 2.7. The unit batteries would be expected to continue to provide dc power for a period of 6 to 10 h, depending upon operator actions to reduce unnecessary electrical loads. For these calculations, dc power was assumed to be lost after 6 h of demand. Subsequently, the operators could no longer manually actuate the SRVs or inject water into the reactor vessel, and the reactor vessel would repressurize over a period of ~2 h. Then would begin a monotonic decrease of the reactor vessel water level (boiloff) due to intermittent loss of fluid through a cycling SRV, which would be periodically actuated by high reactor vessel pressure, automatically.

Table 2.7 Calculated timing of sequence events for the long-term station blackout accident sequence

Event	Time after scram (min)
Loss of off-site power-initiated scram from 100% power. Failure of on-site ac power	0
Heat capacity temperature limit exceeded, reactor vessel depressurization begins	285
Loss of dc power; failure of the steam turbine-driven HPCI and RCIC systems; loss of remote-manual SRV capability	360
Reactor vessel at pressure; automatic SRV actuation begins	470
Swollen water level falls below top of core	631
Structural relocation begins	736
Core plate dryout	780
First local core plate failure	810
Collapse of fuel pellet stacks in central core	927
Reactor vessel bottom head dryout; structural support by control rod guide tubes fails; remainder of core falls into reactor vessel bottom head	938
Initial failure of bottom head penetrations	942

Without restoration of electrical power, the operators could do nothing to impede the further progression of the accident. The swollen water level is predicted to fall below the top of the core at 631 min after the inception of the accident sequence, and the core structures would then begin the process of heating, oxidizing, and melting. Significant core structural relocation (molten control blades and canisters) begins in this calculation at 736 min after scram. Downward relocation of the molten material immediately increases steaming, which lowers the water level above the core plate and increases the rate of automatic SRV actuations. Core plate dryout occurs at 780 min. Reactor vessel pressure slowly decreases after core plate dryout because steam production is temporarily halted, and there is continued leakage through the main steam isolation valves (MSIVs). Structural relocation of molten control blade and

canister material onto the dry core plate continues until portions of the core plate begin to fail due to elevated temperature and the accumulation of mass on the core plate upper surface; the first local core plate failure (at the center of the plate) occurs at 810 min.

The long-term station blackout accident sequence is characterized by relatively low decay heat levels during the period of core degradation. As indicated in Table 2.7, there is a significant period of time (~10-1/2 h) between reactor scram and the uncovering of the top of the core. Consequently, as portions of the core are relocated into the lower plenum, there is a relatively slow boiloff of the water in the reactor vessel bottom head over a period of about 2 h, and in the process, the core debris is quenched. After bottom head dryout at 938 min, the debris reheats, causing failure (by overtemperature) of the control rod guide tubes in the lower plenum soon thereafter. This causes collapse of all remaining portions of the core. Heatup of the accumulated debris in the bottom head leads to failure of the bottom head penetrations. The reactor vessel then depressurizes into the drywell. At this point, all of the core and structural debris in the reactor vessel lower plenum would still be frozen solid; individual components of this debris would subsequently leave the reactor vessel in the order in which they reached their liquid state. In this context, it should be noted that the debris temperature would be rapidly increased by the energy released by oxidation of zirconium metal within the debris during the blowdown of the pressurized reactor vessel into the drywell.

2.2.2 ATWS

This section provides a brief description of the sequence of events initiated by a postulated complete failure to scram following a transient event that has caused closure of all MSIVs. This accident sequence is the most severe of a class of sequences commonly denoted "ATWS," the acronym for "Anticipated Transient Without Scram." (Other types of ATWS are discussed in Chap. 2 of Ref. 13.) With the MSIVs closed, almost all of the steam exiting the reactor vessel would be passed into the pressure suppression pool through the SRVs; the remainder would be used to drive the HPCI or RCIC system turbines during their periods of operation and then, as turbine exhaust, would also enter the pressure suppression pool. Because the rate of energy deposition into the pool can greatly exceed the capacity of the pool cooling equipment, the possibility of excessive pressure suppression pool temperatures leading to primary containment failure by overpressurization is of major concern during ATWS accident sequences.

The MSIV-closure initiated ATWS accident sequence might be triggered by an event such as main steamline space high temperature or high main steamline radiation that directly causes MSIV closure. The reactor protection system logic is designed to recognize the beginning of MSIV closure and to produce an immediate scram, effective before the MSIVs have completely closed. (In actuality, the event of MSIV closure produces a series of four scram signals. In order of receipt these are MSIV position <90% open, high reactor power, high reactor vessel pressure, and low reactor vessel water level.) Alternatively, this ATWS accident sequence might be initiated by any transient that creates conditions calling for scram and MSIV closure such as turbine trip or loss of feedwater. In this case, the MSIV closure would be successful, but the scram would not.

The following discussion is based upon a version of MSIV closure-initiated ATWS in which there is a complete failure of the scram function; that is, the control rods remain in the withdrawal pattern that existed before the inception of the transient. Total failure of rod movement constitutes the most severe ATWS case, but is also the most improbable of the possible scram system failures. Thus, the purpose of this discussion is to provide an upper bounding estimate of the consequences of these very unlikely events. Where specific setpoints are given, the values appropriate to the Browns Ferry Plant are used for the purpose of illustration.

As in all reactor designs, the criticality of the BWR depends upon a complicated set of factors that simultaneously introduce positive or negative reactivity. Whether there is a power increase, constant power, or a power decrease at a given point in time depends upon the particular reactivity balance at that instant. In BWR studies, it is necessary to recognize the importance of the void coefficient of reactivity. In the BWR, boiling takes place within the core, and "voids" are created by the steam bubbles formed within the core volume. The moderation or slowing-down of neutrons is much less in steam than in liquid water, so increased voiding has the effect of reducing the supply of thermal neutrons. Therefore, an increase in voids introduces negative reactivity, and a decrease in voids introduces positive reactivity. Because the BWR operates with the water moderator at saturation conditions within the core, negative or positive reactivity insertions caused by the creation or elimination of voids are a natural, important, and immediate result of reactor vessel pressure changes.

Provision is made for rapid reactor shutdown under emergency conditions by neutron-absorbing control blades that can quickly and automatically be inserted (scrammed) into the core upon the demand of the reactor protection system logic. When inserted, the control blades introduce enough negative reactivity to ensure that the reactor is maintained subcritical even with the moderator at room temperature

and with zero voids in the core. (This is true even with as many as five control blades stuck in the fully withdrawn position.) It is easy to imagine that there must be many dangerous situations that might arise during reactor power operation that would require instantaneous shutdown by reactor scram. However, careful review reveals that only one transient might actually require control blade scram to prevent the occurrence of a severe accident, which by definition involves extensive fuel damage, melting, and fission product release.

The one transient for which it is possible that only the rapid shutdown from power operation that is provided by scram could preclude severe fuel damage and melting is a closure of all MSIVs compounded by failure of automatic recirculation pump trip. This is an "unanticipated" transient; in other words, it is not expected to occur during the operating lifetime of the plant. Before considering the ramifications of failure of recirculation pump trip, it is instructive to examine the progression of the accident without scram but *with* recirculation pump trip.

During the 3- to 5-s period while the MSIVs are closing, the reactor vessel is progressively isolated, and, because the reactor is at power, the reactor vessel pressure rapidly increases. The pressure increase causes the collapse of some of the voids in the core, inserting positive reactivity and increasing reactor power, which in turn causes increased steam generation and further increases pressure. All of this happens in a matter of seconds. The cycle is interrupted when the reactor vessel pressure reaches the level of the SRV setpoints; the SRVs open to reduce the rate of pressure increase and the recirculation pumps are automatically tripped.* With the tripping of the recirculation pumps, the core flow is reduced to ~25% of its former value as the driving mechanism is shifted from forced circulation to natural circulation. With reduced flow, the temperature of the moderator in the core region is increased, producing voids, and introducing a significant amount of negative reactivity. The rapid increase of reactor power is terminated, and the power then rapidly decreases to ~30% of that at normal full-power operation.

If failure of the installed automatic protection logic caused the recirculation pumps to continue operation after the reactor vessel pressure had exceeded their trip setpoint (highly improbable), then two possible outcomes must be considered. Because the total relief capacity of the SRVs is ~85% of normal full-power steam generation, an increasing spiral of reactor power and reactor vessel pressure might continue to the point of overpressurization failure of the primary system, inducing a LOCA. On the other hand, with all of the SRVs open and with no makeup water being added to the reactor vessel, the loss of coolant through these valves could cause uncovering of the core and a concomitant reactor shutdown by loss of moderator before the pressure became sufficiently high to cause rupture of the vessel pressure boundary.

In considering the possibility of primary system overpressurization to the point of pressure boundary breech, it should be recognized that two independent protection system failures would be required to develop the power-pressure spiral postulated here: failure of scram upon MSIV closure or high reactor vessel pressure [setpoint 1070 psia (7.38 MPa)] combined with failure of recirculation pump trip [setpoint 1133 psia (7.81 MPa) or upon low reactor vessel water level at 8-2/3 ft (2.64 m) above the top of the core]. At any rate, recent calculations with the RAMONA code at Brookhaven National Laboratory[22] have indicated a peak reactor vessel pressure of 1340 psia (9.24 MPa) for the case of ATWS without recirculation pump trip, which is below the design pressure of the reactor vessel. Thus, the results of these RAMONA calculations indicate that the loss of coolant from the vessel would effectively terminate the power-pressure spiral.

Assuming that the recirculation pump trip does function as designed, it is axiomatic that although all transient-initiated accident sequences can most easily and quickly be brought under control and terminated by scram, they can also be brought under control and terminated by appropriate other operator-initiated actions. In other words, given *properly trained operators and properly functioning equipment*, a failure-to-scram can be considered to be merely a nuisance requiring more complicated and time-consuming methods of achieving shutdown. The real difficulty for the ATWS accident sequence is that improper actions by the operator might create an unstable and threatening situation.

ATWS, or failure of the automatic scram function, requires that the operators manually take the actions necessary to introduce enough negative reactivity into the core to produce shutdown. The operators might do this by manual scram, in case the ATWS was caused by failure of the protective system logic. Otherwise, the operators could manually drive in the control blades, one at a time. This procedure, for the most part, involves different piping and valves than are used for scram and, therefore, although relatively slow, has a significant probability of success. In the meantime, the most important recourse of the operators is to

*Normal operating pressure is 1020 psia (7.03 MPa). The 13 SRVs have setpoints between 1120 and 1140 psia (7.72 and 7.86 MPa). Automatic recirculation pump trip occurs when the reactor vessel pressure reaches 1133 psia (7.81 MPa).

initiate the standby liquid control system (SLCS); this system injects a neutron-absorbing solution of sodium pentaborate solution into the reactor vessel by means of positive displacement pumps.

In general, the MSIV-closure initiated ATWS accident sequence can be brought under control with no short-term actions by the plant operators other than initiation of the SLCS within 5 min of the inception of the accident. This is true because the capacity of the pressure suppression pool is sufficient to ensure that the temperature increase sustained by the pool during the period of reactor shutdown does not cause containment-threatening pressures. (Operator-provided pressure suppression pool cooling would be essential over the long term.) It is the compounded case of ATWS with failure of the SLCS system that requires special accident management strategies; the effectiveness of the current procedures in dealing with this situation is discussed in Chap. 3.

2.3 Plant-Specific Considerations

The NRC-sponsored severe accident risks report (NUREG-1150) provides the results of detailed probabilistic risk assessments for two U.S. commercial BWRs, Peach Bottom and Grand Gulf. The question arises as to the extent to which the study results for these plants can be extended to other U.S. BWRs of similar designs.

Experience with severe accident evaluations has demonstrated that plant-specific differences preclude any simple extension of the results obtained by a detailed analysis of one BWR plant to other plants of the same classification. This is true even for plants of supposedly similar design such as Peach Bottom and Browns Ferry. These BWR-4 Mark I 1065-MW(e) reactors were constructed during the same period, Peach Bottom 2 being placed in commercial operation in July 1974 with Browns Ferry 1 following the next month. Tables 2.8 and 2.9 provide listings of the recognized plant-specific design differences that must be considered in attempting to determine the progression of events for these plants under severe accident conditions.

Although Peach Bottom and Browns Ferry have the same source (General Electric) for their nuclear steam supply systems, these plants were constructed by different architect-engineering firms (Bechtel and the Tennessee Valley Authority, respectively), and this is the reason for most of the design differences listed on Tables 2.8 and 2.9. Other important differences, however, stem from backfitting activities conducted after plant construction such as the replacement of the three-stage Target Rock SRVs at Browns Ferry with valves of the more advanced two-stage design. As indicated in Table 2.8, the two-stage valves behave differently under accident conditions involving reduced availability of control air or increasing drywell pressure. Detailed information concerning valve requirements for control air is available in Ref. 15 and in Chap. 4 of Ref. 5.

The most important design difference from the standpoint of PRA derives from the dc power source used for starting the site diesel generators upon loss of off-site power. At Browns Ferry, each diesel has its own starting battery, whereas at Peach Bottom, the diesels are started from the unit batteries. This gives rise at Peach Bottom to a risk-significant short-term station blackout accident sequence initiated by a loss of off-site power with common-mode failure of the dc systems; loss of these battery systems precludes starting of the diesel generators and thereby initiates a station blackout. This case is not applicable to Browns Ferry or other BWRs where power for diesel generator starting and loading is independent of the unit batteries and not susceptible to common-mode failure. Obviously, considerations such as these are extremely important in determining the extent to which the NUREG-1150 results can be extended to other plants of similar design.

Table 2.8 Plant differences affecting primary system and primary containment response to accident conditions

Item	Peach Bottom	Browns Ferry
DC power for diesel start, field flashing, and shutdown board control	From unit 125-V dc systems (two per unit)	Independent 125-V battery for each diesel; independent 250-V battery for each shutdown board
Safety/relief valves (SRVs)	11 three-stage Target Rock	13 two-stage Target Rock
Control air requirement for SRV opening as function of reactor vessel drywell pressure differential	26 psi at 150 psid 5 psi at 50 psid	0 psi at 1120 psid 25 psi at 0 psid
Control air requirement to hold an open valve open	5 psi	0 psi at 1120 psid 25 psi at 0 psid
Number of SRVs assigned to the automatic depressurization system (ADS)	Five	Six
Spring-loaded safety valves (discharge into drywell)	Two	None
Drywell control air system reliability	Long-term assured supply	Compressors lost at 2.45-psig drywell pressure; accumulators bleed down at 10 psi/h
RCIC system pump suction	Automatically shifted to pressure suppression pool on low CST level	No automatic shift
Size of feedpump startup bypass piping	3-in. (RFP A only)	8-in.
Condensate system pumps	Condensate pumps only	Condensate pumps and condensate booster pumps
Location of control rod drive hydraulic system pumps	Turbine Building (116 level)	Reactor Building (Basement Center Room)
Condensate storage tank volume	200,000 gal	375,000 gal
Depth and combined volume of drywell sumps	1.42 ft 207 ft^3	4.00 ft 200 ft^3

Table 2.9 Plant differences affecting secondary containment response to accident conditions

Item	Peach Bottom	Browns Ferry
Area of refueling bay-atmosphere blowout panels (0.35 psi)	240 ft^2	3200 ft^2
Refueling bay free volume	1.10×10^6 ft^3 (per unit)	2.62×10^6 ft^3 (common to all thee units)
Refueling bay-reactor building separation	None. Equipment shaft is open within reactor building and to refueling bay	Blowout panels (0.25 psi) Unit 1: On vertical walls enclosing equipment shaft in reactor building Units 2 & 3: On horizontal equipment shaft hatch cover at refueling floor. Equipment shaft is open within reactor building
Reactor building free volume (one unit)	1.3×10^6 ft^3	1.8×10^6 ft^3
Reactor building compartmentalization	Torus room + thee floors, basement corner rooms isolated	Torus room + four floors, basement corner rooms open to torus room and first floor above
Stairwells within reactor building	Enclosed	Open
Fire protection system sprays	None except limited-area spray curtain on 135 level (168 gal/min)	Overhead and cable tray
Reactor building basement drains	Corner room drains isolated from torus room and from each other	All basement drains interconnected
Location of interface between high-pressure and low-pressure 18-in. wetwell vent ducting	Torus room	565 level (first floor above torus room)
Alternate high-pressure venting path	6-in. line direct to atmosphere	None
Location of reactor building-wetwell vacuum breakers	Basement corner room	565 level
Estimated drywell pressure to initiate head flange leakage	140 psia	225 psia

3 Status of Strategies for BWR Severe Accident Management

S. A. Hodge

The Nuclear Regulatory Commission (NRC) has defined the concept of Accident Management with respect to commercial operation of nuclear power plants as follows:

> Accident Management encompasses those actions taken during the course of an accident by the plant operating staff to:
>
> 1. prevent core damage,
> 2. terminate the progress of core damage if it begins and retain the core within the reactor vessel,
> 3. maintain containment integrity as long as possible, and
> 4. minimize offsite releases.
>
> Accident Management, in effect, extends the defense-in-depth principle to plant operating staff by extending the operating procedures well beyond the plant design basis into severe fuel damage regimes, with the goal of taking advantage of existing plant equipment and operator skills and creativity to find ways to terminate accidents beyond the design basis or to limit offsite releases.[23]

A significant portion of the NRC-sponsored research activities in support of the accident management concept is concerned with assessment of the feasibility of various strategies that might be implemented to prevent or mitigate severe accidents. This report was developed as part of one of these research activities, for the purpose of determining the current status of general accident management for the boiling water reactor (BWR) severe accident sequences defined in the previous chapter.

In this chapter, the candidate accident management strategies proposed for BWR applications in a previous study[17] (Brookhaven National Laboratory) are briefly reviewed in Sect. 3.1. Then, Sect. 3.2 provides an assessment of the BWR Owners Group Emergency Procedure Guidelines (EPGs) to determine the extent to which they currently implement the intent of the candidate accident management strategies and for their effectiveness in the unmitigated severe accident situations described in Chap. 2. The rationale for this review is that no further recommendations are necessary in the areas where the EPGs are currently completely effective.

3.1 Candidate Accident Management Strategies

The report "Assessment of Candidate Accident Management Strategies" (NUREG/CR-5474)[17] was published in March 1990. The report provides assessments of a set of accident management strategies derived from a review of various NRC and industry reports on the subject of prevention or mitigation of core damage [both pressurized water reactor (PWR) and BWR]. Each assessment describes the strategy, considers its relationship to existing requirements and practices, and identifies possible associated adverse effects.

The candidate accident management strategies for BWR applications identified by NUREG/CR-5474 are briefly summarized in the following paragraphs. The reader should refer to the basic report[17] for additional description of and information concerning these strategies. Where the need for or the effect of the actions invoked by these strategies has been evaluated in an existing BWR accident sequence analysis, this is indicated. The extent to which these strategies are already incorporated in the EPGs for the critical BWR severe accident sequences will be addressed in Sect. 3.2.

3.1.1 Coping with an Interfacing Systems Loss-of-Coolant Accident (LOCA)

This strategy is to limit the effects of an interfacing systems LOCA (ISL) by early detection and isolation or take other actions to mitigate the consequences if isolation cannot be achieved. An ISL would be indicated by high temperatures and radiation levels outside of the BWR primary containment. The condensate storage tank would be drained at a rate higher than usual and proportional to the rate at which leakage was occurring into the secondary containment. Focused training may improve the ability of the operator to rapidly detect the system involved and to isolate a break.

If isolation of the break cannot be rapidly achieved, then reactor vessel depressurization would reduce the rate at

which mass is lost through the break. Flooding of the break location in the secondary containment might be effective for the low-pressure sections of the emergency core cooling system (ECCS) piping located in the reactor building basement rooms. With the break under water, fission product releases would be effectively scrubbed. Another approach to the same effect for elevated breaks would be to use the reactor building fire protection system sprays (to the extent that their existing design permits) to wash the building atmosphere above the break.

3.1.2 Maintaining the Condensate Storage Tank as an Injection Source

Under severe accident conditions, the pressure suppression pool can become overheated and radioactive and, therefore, unsuitable as an injection source for the pumping systems available to supply water to the reactor vessel. This strategy is to augment the original supply of water in the condensate storage tank and thereby to maintain this source of cool water for reactor vessel injection. The object is that plant management should prepare in advance for the use of unusual methods of supplying makeup (including untreated systems as a last resort) to the condensate storage tank.

3.1.3 Alternate Sources for Reactor Vessel Injection

If all higher-priority systems and water supplies cannot be used, this strategy calls for advance planning and consideration of available river, lake, municipal water system, ocean, or other supplies and clever methods for the introduction of such sources, including temporary hose connections.

3.1.4 Maintain Pump Suction upon the Condensate System

This strategy is based upon the general intent to maintain reactor vessel injection capability (keep the core covered) and, as such, is closely related to the strategies previously discussed. Here the specific objective is to switch any available ECCS pump suctions away from the pressure suppression pool, should that source be overheated to the point that its use might induce pump failure.

This strategy receives separate classification here because existing plants have automatic plant protection logic that removes HPCI (and in some cases RCIC) system pump suction from the condensate storage tank and places it upon the pressure suppression pool, the reverse of what is desired under many plant accident situations. The existing logic is based upon consideration of large-break LOCA, where it is necessary to convert an open cycle (condensate storage tank to reactor vessel to pressure suppression pool) to a closed cycle (pool to vessel to pool) to prevent excessive pressure suppression pool water level. Nevertheless, in non-LOCA situations the injection rate from the condensate storage tank would be much less, and this automatic shift of pump suction may be detrimental to plant protection (for station blackout, see Ref. 7).

3.1.5 Operator Override of Injection Pump Trips

This strategy is to maintain operation of reactor vessel injection systems beyond the point at which they would normally trip. Preparatory planning for use of this strategy involves selection of the trips suitable for bypassing under emergency conditions by assessments performed as part of the strategy evaluation process. The assessment should consider the design bases for each trip and should include analyses of the potential accident sequences for which each trip might be bypassed.

3.1.6 Maintain RCIC System Availability

This strategy is an extension of that summarized in Subsect. 3.1.5, but it is treated separately here because the risk reduction potential associated with special procedures to maintain the operability of RCIC under accident conditions is perceived to be greater than for the other injection systems. This is because RCIC is a steam turbine-driven system, operable without ac power, but incorporating several turbine-protective trips.

It is particularly recommended that this strategy be considered for use in ATWS or station blackout accident sequences where RCIC operation is needed to maintain reactor vessel water level, but elevated pressure suppression pool temperature and reduced vessel pressure may cause turbine trip. (The turbine exhausts to the pressure suppression pool and is subject to a high exhaust pressure trip. System isolation, involving turbine trip and shutting of steam supply valves, occurs upon space high temperature, low reactor vessel pressure, and other signals.) Use of the RCIC system for station blackout and ATWS accident sequences based upon the Browns Ferry Plant is discussed in Refs. 7 and 13.

3.1.7 Use of Control Rod Drive Hydraulic System (CRDHS) Pumps for Decay Heat Removal

The CRDHS pumps inject cooling water through the CRD mechanism assemblies during normal reactor operation.

This normal injection is taken from the condensate storage tank at a low rate [~0.3 gal/min (1.9×10^{-5} m^3/s) per control blade] and would be insufficient, by itself, to provide significant core cooling under accident conditions. However, this injection rate is increased (approximately doubled) whenever a scram is in effect, because the throttle valve that limits the flow under normal conditions is automatically bypassed. Additional flow increase will occur if the vessel is depressurized.

This strategy involves advance planning to establish the maximum potential for effective use of the CRDHS under accident conditions. At many BWRs, these are the only ac motor-driven pumps capable of injection with the reactor vessel at pressure, and therefore their use in conjunction with RCIC for level control under ATWS conditions may be desirable. Additional information on the use of these pumps under accident conditions is available in Refs. 12, 13, and 15.

3.1.8 Load-Shedding to Conserve Battery Power

Without ac power under station blackout conditions, continued reactor vessel injection as necessary to maintain the core covered may depend upon the availability of dc power for RCIC and HPCI turbine governor and valve control. This strategy calls for establishment of a procedure for shedding of nonessential dc loads under accident conditions as necessary to prolong the period before battery exhaustion.

3.1.9 Battery Recharging Under Station Blackout Conditions

This strategy would provide a station procedure for charging the unit batteries under emergency conditions when the installed battery chargers are not available. A portable, engine-powered charger with a practical arrangement for hookup to the dc power system would be made available under this strategy to maintain power to the essential dc loads. (These include emergency lighting, SRV remote-manual operation, HPCI and RCIC controls, and the vital ac bus loads supplied by a dc motor–ac generator combination.)

3.1.10 Replenish Pneumatic Supply for Safety-Related Air-Operated Components

This strategy involves preplanning for backup supplies of control air (or nitrogen) under emergency conditions. In this connection, the continued availability of control air at sufficient pressure to permit remote-manual SRV actuation for reactor vessel depressurization is of particular importance. Accident sequence analyses for loss of control air based upon the Browns Ferry Plant are provided in Ref. 15.

3.1.11 Bypass or Setpoint Adjustment for Diesel Generator Protective Trips

This strategy is to enable continued operation of the emergency diesel generators by either overriding certain designated protective trips or by adjusting the trip setpoints. (Automatic bypass of some protective trips is normally provided during emergency diesel start.) Advance planning for this strategy involves selection of the protective trips suitable for bypass or adjustment by consideration of the detailed design basis of each trip and the need for the power supplied by the diesel generator under various accident conditions.

3.1.12 Emergency Crosstie of ac Power Sources

The strategy would provide an emergency crosstie capability between independent sources of ac power at a plant site. Possible sources include equivalent ac systems at a multiunit site and gas-turbine generators where available.

3.1.13 Alternate Power Supply for Reactor Vessel Injection

This strategy involves advance planning for the use of emergency ac power from a mobile diesel generator or a gas turbine generator to drive a CRDHS pump or other appropriate pump for reactor vessel injection. While the primary purpose of such a strategy would be for mitigation of station blackout, an alternate ac generating unit for driving a CRDHS pump equipped for boron injection would also be beneficial in accident sequences involving significant core damage. (See Subsect. 3.1.16.)

3.1.14 Use of Diesel-Driven Fire Protection System Pumps for Vessel Injection or Containment Spray

Strategies to employ plant fire protection systems as backup water sources for reactor vessel injection or containment spray are generally attractive because the dedicated diesel-driven pumps are independent of the plant internal ac system, the fire protection water sources are typically unlimited or very large, and means to provide cross-connection with the reactor vessel injection/containment spray piping under emergency conditions are typically relatively easy to install. Provision for cross-

connection of the fire protection system is already in place at several BWRs and could be accomplished at other plants by a strategy based upon use of a temporary spool piece or hose connection arrangement that could be implemented quickly in emergency situations.

3.1.15 Regaining the Main Condenser as a Heat Sink

For BWR accident sequences in which the power associated with the steam flow leaving the reactor vessel exceeds the capacity of the available pressure suppression pool cooling equipment, it is obviously desirable to restore the main condenser as heat sink. This is subject to the restrictions that the accident in progress does not involve a piping break downstream of the main steam isolation valves (MSIVs), that significant fuel damage and fission product release have not occurred, and that main condenser vacuum can be restored. The ability to invoke this strategy quickly would be of particular value in dealing with an anticipated transient without scram (ATWS) in which the MSIVs were automatically tripped closed on a condition such as reactor vessel low level that has subsequently cleared. If the isolation signal remained in effect, however, then this strategy would require procedures for bypassing the valve opening interlocks.

It is emphasized that this strategy is not applicable to severe accident situations in which significant fission product release from fuel has occurred. The escape of fission products from the reactor vessel through the MSIVs to the main condenser would constitute bypass of the primary containment.

3.1.16 Injection of Boron Under Accident Conditions with Core Damage

For BWR accident sequences involving prolonged uncovering of the core with inadequate steam cooling of the uncovered region, severe accident calculations predict the metallic structural components (control blades and channel box walls) to melt and relocate downward while the higher-melting fuel and zirconium oxide remnants of the cladding remain in place. This raises the question of recriticality should reactor vessel water injection capability be restored after partial core degradation has occurred.

The BWR control blade neutron poison is B_4C powder, stored within the neutron absorber rods located within the control blade sheaths. The early relocation of the control blade structure is aggravated by the tendency of the B_4C powder to form a lower-melting-temperature mixture with the surrounding stainless steel of the absorber rods and sheath. This effect has been experimentally observed.[24]

The candidate strategy for dealing with the early relocation of control blades (should an accident progress into the severe core damage phase) calls for advance planning to ensure that proper concentrations of boron can be maintained in the reactor vessel to ensure reactor shutdown. Application of this strategy is not limited to ATWS accident sequences; boron injection would be required in the event of control blade relocation from the core region regardless of the accident sequence.

Boron injection under accident conditions would normally be accomplished by use of the standby liquid control system (SLCS). This requires the availability of ac power, as do most alternate boron injection methods that employ the CRDHS or the reactor water cleanup (RWCU) system. Advance planning for implementation of this strategy should include consideration of means for boron injection under station blackout conditions. For this purpose, this strategy might employ the injection system (having an alternate power supply) covered by the candidate strategy described in Subsect. 3.1.13. This question of injection of boron under accident conditions will be discussed in detail in Chaps. 9 through 13.

3.2 BWR Owners Group EPGs

The BWR Owners Group EPGs[16] are generic to the General Electric BWR plant designs with the exception that they do not address systems for the control of hydrogen in the Mark III containment. They are intended to be adapted for application to individual plants by the deletion of irrelevant material and the substitution, where necessary, of plant-specific information. The development of the Emergency Operating Procedures (EOPs), based upon the EPGs, for use at a particular plant is the responsibility of the local plant management.

The current version of the EPGs is Revision 4, issued as General Electric topical report NEDO-31331[16] in March 1987. The NRC has provided a Safety Evaluation Report[25] (SER) for this Revision, finding it "generally acceptable for implementation." Nevertheless, in forwarding this SER, the NRC staff has noted

> We expect that the BWR Owners will continue to improve the EPGs. Since the guidelines do not provide comprehensive severe accident mitigation strategies, we expect the Owners to upgrade the EPGs in parallel with resolution of severe accident issues.

It is the purpose of this section to discuss the application of the current version (Revision 4) of the EPGs to the dominant BWR severe accident sequences defined in Sect. 2.2.

In considering the application of the EPGs to specific accident sequences, it is important to recognize that these guidelines identify symptomatic operator actions. In other words, given a symptom requiring entry into the emergency procedures developed from these guidelines, it is intended that the operators take action in response to the symptom without any requirement to first diagnose the cause. Their development has been based upon realistic analyses, rather than upon licensing basis calculational methods, and they consider utilization of all plant systems, not just the safety systems. They are intended to address all mechanistically possible abnormal plant conditions, without consideration of the probability of the abnormal event or condition.

The basic functional goal is to establish the prudent actions to be taken by the operators in response to the symptoms observed by them at any point in time. Once entry into the EPGs has occurred, the operators are expected to take the specified actions regardless of equipment design bases limitations or licensing commitments. The guidelines use multiple mitigation strategies where possible so that recovery from an abnormal situation does not require successful operation of any one system or component.

The EPGs are comprised of four Control Guidelines, each with its own set of entry conditions: Reactor Vessel Control, Primary Containment Control, Secondary Containment Control, and Radioactivity Release Control. Because the scope of this report is limited to consideration of the in-vessel accident management strategies, only the Reactor Vessel Control Guideline plus that portion of the Primary Containment Control Guideline that directly affects (via feedback through the SRV interface) the in-vessel events are discussed in this section.

3.2.1 Station Blackout

BWRs are well protected against core uncovering through the provision of several diverse systems for the injection of water into the reactor vessel. Furthermore, only a small fraction of the normally available injection capacity is sufficient to remove the decay heat and prevent the water level from dropping below the top of the core in the event of scram from full power. As an example, an average injection of 225 gal/min (0.014 m^3/s) will prevent core uncovering (for a non-LOCA situation) at a plant the size of Peach Bottom or Browns Ferry,[12] whereas the total installed injection capacity (not counting backup systems) is more than 50,000 gal/min (3.155 m^3/s).

Most of the installed reactor vessel injection capability at a BWR plant is dependent upon the availability of ac power as demonstrated by the example of Table 3.1. (In addition to the pumping systems listed, the condensate/condensate booster pumps are electric motor-driven and can be employed for injection if the reactor vessel is depressurized, using the feedwater pump bypass piping.) Even with severely degraded availability of ac power, it is possible to inject sufficient water into the reactor vessel to keep the core covered. A demonstration of what can be done in a degraded electrical power situation is provided by the operator actions during the cable fire that occurred at the

Table 3.1 Reactor vessel injection system capacities at the Browns Ferry Nuclear Plant

System	Total capacity (gal/min)	Power requirement	Number of pumps	Vessel depressurization required?
Residual heat removal (RHR)	40,000	ac	4	Yes
Core spray	12,500	ac	4	Yes
Control rod drive hydraulic system[a]	120–518	ac	2	No
Standby liquid control system (SLCS)[b]	56	ac	1	No
High-pressure coolant injection (HPCI)	5,000	Steam turbine, dc	1	No
Reactor core isolation cooling (RCIC)	600	Steam turbine, dc	1	No

[a] The injection rate is determined by operator actions taken to enhance the flow (see Table 3.1 of Ref. 12).
[b] Two SLCS pumps are installed, but system interlocks permit only one to be operated at a time.

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[b] Two SLCS pumps are installed, but system interlocks permit only one to be operated at a time.

It is the purpose of this section to discuss the application of the current version (Revision 4) of the EPGs to the dominant BWR severe accident sequences defined in Sect. 2.2.

In considering the application of the EPGs to specific accident sequences, it is important to recognize that these guidelines identify symptomatic operator actions. In other words, given a symptom requiring entry into the emergency procedures developed from these guidelines, it is intended that the operators take action in response to the symptom without any requirement to first diagnose the cause. Their development has been based upon realistic analyses, rather than upon licensing basis calculational methods, and they consider utilization of all plant systems, not just the safety systems. They are intended to address all mechanistically possible abnormal plant conditions, without consideration of the probability of the abnormal event or condition.

The basic functional goal is to establish the prudent actions to be taken by the operators in response to the symptoms observed by them at any point in time. Once entry into the EPGs has occurred, the operators are expected to take the specified actions regardless of equipment design bases limitations or licensing commitments. The guidelines use multiple mitigation strategies where possible so that recovery from an abnormal situation does not require successful operation of any one system or component.

The EPGs are comprised of four Control Guidelines, each with its own set of entry conditions: Reactor Vessel Control, Primary Containment Control, Secondary Containment Control, and Radioactivity Release Control. Because the scope of this report is limited to consideration of the in-vessel accident management strategies, only the Reactor Vessel Control Guideline plus that portion of the Primary Containment Control Guideline that directly affects (via feedback through the SRV interface) the in-vessel events are discussed in this section.

3.2.1 Station Blackout

BWRs are well protected against core uncovering through the provision of several diverse systems for the injection of water into the reactor vessel. Furthermore, only a small fraction of the normally available injection capacity is sufficient to remove the decay heat and prevent the water level from dropping below the top of the core in the event of scram from full power. As an example, an average injection of 225 gal/min (0.014 m^3/s) will prevent core uncovering (for a non-LOCA situation) at a plant the size of Peach Bottom or Browns Ferry,[12] whereas the total installed injection capacity (not counting backup systems) is more than 50,000 gal/min (3.155 m^3/s).

Most of the installed reactor vessel injection capability at a BWR plant is dependent upon the availability of ac power as demonstrated by the example of Table 3.1. (In addition to the pumping systems listed, the condensate/condensate booster pumps are electric motor-driven and can be employed for injection if the reactor vessel is depressurized, using the feedwater pump bypass piping.) Even with severely degraded availability of ac power, it is possible to inject sufficient water into the reactor vessel to keep the core covered. A demonstration of what can be done in a degraded electrical power situation is provided by the operator actions during the cable fire that occurred at the

Table 3.1 Reactor vessel injection system capacities at the Browns Ferry Nuclear Plant

System	Total capacity (gal/min)	Power requirement	Number of pumps	Vessel depressurization required?
Residual heat removal (RHR)	40,000	ac	4	Yes
Core spray	12,500	ac	4	Yes
Control rod drive hydraulic system[a]	120–518	ac	2	No
Standby liquid control system (SLCS)[b]	56	ac	1	No
High-pressure coolant injection (HPCI)	5,000	Steam turbine, dc	1	No
Reactor core isolation cooling (RCIC)	600	Steam turbine, dc	1	No

[a] The injection rate is determined by operator actions taken to enhance the flow (see Table 3.1 of Ref. 12).
[b] Two SLCS pumps are installed, but system interlocks permit only one to be operated at a time.

Browns Ferry Plant on March 22, 1975, involving progressive and multiple failures of electrical power to plant systems.[26]

The reason that station blackout accident sequences are consistently reported as dominant in BWR probabilistic risk assessments (as discussed in Sect. 2.1) is simply that loss of ac power eliminates most of the installed plant pumping capability, leaving only the steam turbine-driven systems. If these systems are operable (long-term station blackout), then the core can be kept covered while dc power remains for turbine governor control. If, however, these systems suffer independent failure (short-term station blackout), then no reactor vessel injection capability remains and the operators can only act to delay the onset of core degradation while making every endeavor to restore electric power. The operator actions specified by the EPGs for these two cases of station blackout will now be addressed.

3.2.1.1 Operator Actions for Short-Term Station Blackout

This accident sequence would be initiated by loss of offsite and on-site ac power combined with independent failure of the steam turbine-driven reactor vessel injection systems. Unfortunately, the demonstrated reliability of the steam turbine systems is such that the short-term version of station blackout contributes a significant portion of the overall station blackout risk.

For the case of independent mechanical failure of the steam turbine systems, this accident sequence is identified as one of "the more probable combinations of failures leading to core damage" by the recent severe accident risk assessment (NUREG-1150). For Grand Gulf [which does not have a HPCI system (Table 2.4)], the sequence is summarized as follows:

> Loss of offsite power occurs followed by the successful cycling of the safety relief valves (SRVs). Onsite ac power fails because all three diesel generators fail to start and run as a result of either hardware or common-cause faults. The loss of all ac power (i.e., station blackout) results in the loss of all core cooling systems [except for the reactor core isolation cooling (RCIC) system] and all containment heat removal systems. The RCIC system, which is ac independent, independently fails to start and run. All core cooling is lost, and core damage occurs in approximately 1 hour after offsite power is lost.[6]

The actions that would be taken by the operators to cope with the symptoms of this accident sequence, assuming strict adherence with the EPGs, are described in the following paragraphs.

Entry to the Reactor Vessel Control Guideline would be triggered by vessel pressure above the high-pressure scram setpoint, vessel water level below the low-level scram setpoint, and drywell pressure above the high-drywell-pressure scram setpoint. Each of these triggers is by itself sufficient for entry; however, it is expected that all three would occur in the order listed for short-term station blackout. The high drywell pressure would be the result of loss of the drywell coolers and the consequent heating and expansion of the drywell atmosphere and would occur more quickly for the Mark I containment design than for the larger Mark II and Mark III designs.

The operators would attempt to establish effective reactor vessel injection; nevertheless, by the definition of this accident sequence, these efforts would fail.

The operators would act to terminate the SRV cycling initiated by the MSIV closure at the inception of the station blackout. This would be accomplished by initiation of the isolation condenser at plants so equipped (Table 2.4) and by control of the SRVs in the remote-manual mode. Operator control increases the pressure reduction per valve opening and reduces the total number of valve actuations. The operators would begin a controlled reactor vessel depressurization, remaining within the reactor vessel cooldown rate [~100°F/h (0.015 K/s)] allowed by the plant Technical Specifications.

With no reactor vessel injection, these operator actions to manually control pressure would to some extent hasten the onset of core uncovering. (The timing of events for an example accident sequence calculation based upon Peach Bottom is discussed in Subsect. 2.2.1.1.) Recognizing that the vessel water level cannot be maintained above the top of the core, the operators would implement the plant EOP developed from Contingency #1 of the EPGs, "Alternate Level Control." This contingency provides instructions for use of alternate or low water-quality backup systems for vessel injection, but again, the definition of this accident sequence precludes success of this step.

When the reactor vessel water level dropped to the top of the core and with no injection pump running, the operators would be directed [Contingency #1 (Step C1-3.2)] to implement the plant EOP developed from Contingency #3, "Steam Cooling." For plants with isolation condensers, this contingency merely provides that the operators should confirm that the isolation condenser has been placed in operation. For plants without isolation condensers, or if the isolation condenser is inoperable, then use of steam cooling is specified. Briefly, the concept is to delay fuel heatup by cooling the uncovered upper regions of the core by a rapid

flow of steam. Because the source of the steam is the remaining inventory of water in the reactor vessel, however, the steam cooling maneuver can provide only a temporary delay.

Steam cooling would be placed into effect when the reactor vessel water level dropped to the "Minimum Zero-Injection RPV Water Level." In Revision 4 of the EPGs, this is defined as the lowest vessel level at which the average steam generation rate within the covered portion of the core is sufficient to prevent the maximum clad temperature in the uncovered region of the core from exceeding 1800°F (1255 K). The Minimum Zero-Injection RPV Water Level is plant-specific; the basis for its determination is described in Appendix A of the EPGs, while the calculational procedure for plant use is provided in the EPGs' Appendix C.

With the core partially uncovered and the vessel water level decreased to the Minimum Zero-Injection RPV Water Level, the operators would be directed [Contingency #3 (Step C3.1)] to enter the plant EOP developed from Contingency #2, "Emergency RPV Depressurization." This directs the opening of all ADS valves. (The number of valves assigned to the automatic depressurization system is plant-specific, five at Peach Bottom.) This action provides flashing of the water in the core region, providing the desired temporary cooling of the uncovered region of the core. If sufficient ADS valves cannot be opened, then use of other SRVs or other means of rapid vessel depressurization is specified.

At this point, it is worthwhile to mention that the Minimum Zero-Injection RPV Water Level, as calculated in accordance with Appendix C of Revision 4 of the EPGs, provides that the emergency vessel depressurization would occur much earlier than was the case with Revision 3. It seems that the conservatisms built into the calculational procedure to ensure that under no conditions would the hottest point of the exposed cladding exceed 1800°F (1255 K) are such that the Minimum Zero-Injection RPV Water Level, while plant-specific, is typically at about 71% of core height. In other words, only the upper 29% of the core would be uncovered at the time the emergency vessel depressurization is specified. It is easy to show that the actual temperature of the clad in the uncovered region under station blackout conditions with the water level at this height would be much less than 1800°F (1255 K) and that the amount of steam cooling actually achieved would therefore be insignificant. Revision 3 of the EPGs had provided that the emergency vessel depressurization should occur when water level indication was lost, which occurs at about one-third core height (two-thirds of the core uncovered). This matter is currently under review.[27]

The emergency depressurization causes all of the water in the core region to be flashed and the reactor vessel water level to fall beneath the core plate into the lower plenum. (This occurs regardless of whether the vessel water level triggering the action to open the ADS valves is that specified by EPGs Revision 3 or 4). With the depressurization, reactor vessel water level indication would be lost and the operators would be directed [Contingency #2 (Step C2-1.4)] to the plant EOP developed from Contingency #4, "RPV Flooding." This contingency is intended to provide for adequate core cooling when vessel water level cannot be determined and calls for injection by any available means to fill the vessel until overflow into the main steam lines. Once again, however, the definition of short-term station blackout provides that no injection systems would be available, and therefore the actions specified by this contingency would be unsuccessful.

At this point, the remaining reactor vessel water inventory would be confined to the lower plenum, the core would be completely uncovered and heating up, the vessel pressure would be equalized with the containment pressure (through the open ADS valves), and the operators would be continuing to try to obtain vessel injection capability, by repairing RCIC (or HPCI) or by restoring ac power. No additional actions are specified by the EPGs should the accident sequence proceed into the severe core damage phase. The potential for additional mitigative actions is discussed in Chap. 4 of this report.

A second significant way in which failure-upon-demand of the steam turbine-driven injection systems might initiate the short-term station blackout accident sequence is by a common-mode failure of the dc battery systems. The NUREG-1150 study identifies this as among the "more probable combinations of failures leading to core damage" for Peach Bottom, describing this version of the short-term station blackout accident sequence as follows:

> Loss of offsite power occurs followed by a subsequent failure of all onsite ac power. The diesel generators fail to start because of failure of all the vital batteries. Without AC and DC power, all core cooling systems (including HPCI and RCIC) and all containment heat removal systems fail. Core damage begins in approximately 1 hour as a result of coolant boiloff.[6]

The ability of the operators to cope with this (battery failure) variation of short-term station blackout is less than for the version with mechanical failure of HPCI and RCIC because without battery power, the SRVs cannot be operated in the remote-manual mode; hence, the reactor vessel could not be depressurized. It should be recalled, however, that the susceptibility of Peach Bottom to initiation of this accident sequence derives from the use of the unit batteries

for starting of the diesel generators. This starting arrangement is plant-specific. For example, the diesel generators at the Browns Ferry Plant each have their own starting battery, as indicated in Table 2.8.

Without the emergency reactor vessel depressurization called for by the EPGs, water would remain above the core plate during the early phase of relocation of core structural materials (stainless steel and zirconium metal). As the relocating molten metals fell into this water, the concomitant steam generation would fuel the zirconium oxidation reaction in the uncovered region of the core. The associated energy release would in turn accelerate the degradation of the core structure. A comparison of the timing of events for the short-term station blackout accident sequence with and without emergency reactor vessel depressurization for an example calculation based upon the Susquehanna plant is provided in Table 2.6. Significant core damage occurs earlier for the steam-rich environment created by the case without ADS than for the steam-starved situation that occurs when the emergency vessel depressurization specified by the EPGs is successful.

It should be noted that the increased severity of this accident sequence with common-mode battery failure has connotations far beyond the acceleration of core damage in the early phase. For example, the emergency lighting systems within a plant depend upon battery power and, if the reactor vessel were to remain pressurized at the time of bottom head penetration failure, then a significant portion of the core and structural debris would melt and spew into the drywell with the vessel blowdown rather than slowly melt (decay heat only, without chemical energy release) and run out under the impetus of gravity.

3.2.1.2 Operator Actions for Long-Term Station Blackout

This accident sequence provides much more opportunity for recovery of ac power before the onset of core degradation than does the short-term case. This is because at least one steam turbine-driven system (HPCI or RCIC) functions to keep the core covered for as long as dc power remains available. The NUREG-1150 study includes two variations of this accident sequence among the "more probable combinations of failure leading to core damage" for Peach Bottom:

> Loss of onsite and offsite ac power results in the loss of all core cooling systems (except high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC), both of which are ac independent in the short term) and all containment heat removal systems. HPCI or RCIC (or both) systems function but ultimately fail at approximately 10 hours because of battery depletion or other late failure modes (e.g., loss of room cooling effects). Core damage results in approximately 13 hours as a result of coolant boiloff.
>
> Loss of offsite power occurs followed by a subsequent failure of a safety relief valve to reclose. All onsite ac power fails because the diesel generators fail to start and run from a variety of faults. The loss of all ac power fails most of the core cooling systems and all the containment heat removal systems. HPCI and RCIC (which are ac independent) are available and either or both initially function but ultimately fail at approximately 10 hours because of battery depletion or other late failure modes (e.g., loss of room cooling effects). Core damage results in 10 to 13 hours as a result of coolant boiloff.[6]

NUREG-1150 also identifies one long-term sequence for Grand Gulf [which does not have a HPCI system (Table 2.4)]:

> ... loss of offsite power occurs and all three diesel generators fail to start or run. The safety relief valves cycle successfully and RCIC starts and maintains proper coolant level within the reactor vessel. However, ac power is not restored in these long-term scenarios, and RCIC eventually fails because of high turbine exhaust pressure, battery depletion, or other long-term effects. Core damage occurs approximately 12 hours after offsite power is lost.

It should be noted that all considered versions of this accident sequence predict core damage to occur only after 10 h or more. During this period, aggressive actions would be taken by the plant operating staff to restore ac power. That this accident sequence appears among the dominant sequences leading to core damage is perhaps a testimony to the low core damage frequency associated with most other BWR accident sequences.

With the RCIC (or HPCI for plants so equipped) steam turbine-driven reactor vessel injection available, the operators would stave off core uncovering until after battery failure. The EPGs specify (Step RC/L-2) that the vessel water level should be maintained between the low-level and high-level scram setpoints and that the RCIC (or HPCI) pump suction should be maintained on the condensate storage tank. Furthermore, as specified by the EPGs (Step RC/P3), the reactor vessel would be depressurized at the rate [about 100°F/h (0.015 K/s)] allowed by the plant Technical Specifications. Depressurization at this rate would require ~2 h to reduce the vessel pressure to 140 psia (0.97 MPa).

During the controlled depressurization, it is probable that the pressure suppression pool temperature would exceed the "Heat Capacity Temperature Limit." (There is no pressure suppression pool cooling under station blackout

conditions.) The Heat Capacity Temperature Limit is defined as the highest pressure suppression pool temperature for which a subsequent release to the pool associated with reactor vessel depressurization would not cause either the suppression chamber design temperature (for the Mark III containment design) or the Primary Containment Pressure Limit to be exceeded. The Heat Capacity Temperature Limit is a function of reactor vessel pressure and is plant-specific. [At the Browns Ferry Plant, for example, this limit is 160°F (344 K) with the reactor vessel at 1100 psig (7.69 MPa), 170°F (350 K) with the vessel at 600 psig (4.24 MPa), and 189°F (360 K) with the vessel at 160 psig (1.205 MPa).] The basis for the determination of the Heat Capacity Temperature Limit is described in Appendix A of the EPGs, while the calculational procedure for establishing the plant-specific curve of suppression pool temperature vs reactor vessel pressure is provided in the EPGs' Appendix C.

If the pressure suppression pool temperature does exceed the Heat Capacity Temperature Limit as defined by the plant-specific curve, then the EPGs require (Step RC/P-1) that the rate of vessel depressurization be increased as necessary to remain within the limits of the curve. This is to be done even if it requires exceeding the cooldown rate allowed by the plant Technical Specifications. (EPGs Caution #6.)

When the installed capacity of the plant battery system finally becomes exhausted in this accident sequence, there are two important effects. First, reactor vessel injection capability would be lost, and second, the SRVs, which require dc power for actuation in the remote-manual mode, could no longer be held open for reactor vessel depressurization. Continued steam generation would increase the reactor vessel pressure while water loss from the vessel would be temporarily delayed until the automatic SRV cycling on high vessel pressure was resumed.

In this accident sequence, the reactor vessel is depressurized during the period, while dc power remains available, then repressurizes after battery power is lost. It is important to recognize the benefit of the temporary depressurization. Because reactor vessel injection is available, the reactor vessel water level is kept in the normal range during the period that the reactor vessel is depressurized. Then, when battery power is lost, the level swell associated with the heating of the vessel water inventory during repressurization ensures that when SRV cycling is resumed, the resulting boiloff of vessel inventory begins from a water level much higher than normal. This, plus the time required for vessel repressurization, significantly delays the uncovering of the core.

The events of this unmitigated long-term station blackout accident sequence that would occur after the core becomes uncovered are similar to the events of the short-term station blackout sequence with common-mode failure of batteries. In the long-term sequence, however, these events are greatly delayed, and their progression would be driven by a decay heat approximately one-half the magnitude of that for the short-term case.

With water remaining above the core plate during the relocation of core structural materials (stainless steel and zirconium metal), the steam generated as the molten material entered this water would provide a steam-rich environment for the metal oxidation reactions in the uncovered region of the core. The associated energy release would accelerate the degradation of the core structure until core plate dryout, when a steam-starved situation would exist in the core region. Subsequent to core plate failure, however, movement of heated material into the water remaining in the reactor vessel lower plenum would again create a steam-rich environment in the core region.

With the remaining reactor vessel water inventory limited to the vessel lower plenum, with the core completely uncovered, and with the reactor vessel at pressure, periodically discharging through the SRVs to the pressure suppression pool, the operators would be attempting to restore electrical power. If electrical power can be restored, the EPG guidance with respect to injecting water into the reactor vessel and reactor vessel depressurization can be followed. Otherwise, the EPGs specify no additional actions for this situation.* The potential for additional mitigative actions is discussed in Chap. 4.

3.2.2 ATWS

ATWS accident sequences involve failure of the scram function and, if not successfully brought under control, can lead to a severe accident situation at BWR plants. These accident sequences are characterized by an early threat to containment integrity because the energy release from the reactor vessel into the pressure suppression pool can greatly exceed the capacity of the pool cooling equipment. Automatic recirculation pump trip reduces the reactor power, and the operators can act to reduce power further by initiating the injection of liquid neutron poison (a few plants have automatic provision for this) and by manual insertion of control blades.

*The guidance provided in EPG Contingency #4 (Step C4-1.3) regarding entry to Contingency #6, "Primary Containment Flooding" might be construed as applicable in this case with the reactor vessel at pressure and one SRV cycling (but not continuously open). However, without electrical power the provisions of Contingency #6 could not be carried out.

Status

There are several versions of ATWS, but the chief distinction is between ATWS accident sequences with the reactor vessel isolated and ATWS accident sequences where the MSIVs remain open (but the main turbine is tripped so that steam flow into the main condenser is via the turbine bypass valves). The recent severe accident risk assessment (NUREG-1150) identifies two variations of MSIV-closure ATWS as among "the more probable combinations of failures leading to core damage" for Peach Bottom.

> Transient (e.g., loss of feedwater) occurs followed by a failure to trip the reactor because of mechanical faults in the reactor protection system (RPS) and closure of the main steam isolation valves (MSIVs). The standby liquid control system (SLCS) does not function (primarily because of operator failure to actuate), but the HPCI does start. However, increased suppression pool temperatures fail the HPCI. Low-pressure injection is unavailable and all core cooling is lost. Core damage occurs in approximately 20 minutes to several hours, depending on the LPCI failure mode.
>
> Transient occurs followed by a failure to scram (mechanical faults in the RPS) and closure of the MSIVs. SLCS is initiated but HPCI fails to function because of random faults. The operator fails to depressurize after HPCI failure and therefore the low-pressure core cooling systems cannot inject. Core damage occurs in approximately 15 minutes.[6]

The NUREG-1150 study also identifies one MSIV-closure sequence as "the most probable combination of failures leading to core damage" within the general class of ATWS sequences for Grand Gulf.

> Transient-initiating event occurs followed by a failure to trip the reactor because of mechanical faults in the reactor protection system (RPS). The standby liquid control system (SLCS) is not actuated and the high-pressure core spray (HPCS) system fails to start and run because of random hardware faults. The reactor is not depressurized and therefore the low-pressure core cooling system cannot inject. All core cooling is lost; core damage occurs in approximately 20 to 30 minutes after the transient-initiating event occurs.

It should, however, be recalled that by far the major portion of the overall threat of core damage identified by the NUREG-1150 study for Grand Gulf derives from station blackout (97%), as indicated on Fig. 2.1.

The dominant ATWS sequences identified by the NUREG-1150 study all include MSIV closure as an initiating event. It is also important to note that core damage occurs only after failure of adequate reactor vessel injection. If sufficient water can be injected under ATWS conditions to maintain a lower portion of the core critical, then the resulting steam generation would provide adequate steam cooling of the uncovered (subcritical) region. Severe core damage occurs in an ATWS accident sequence only after injection has been lost with the subsequent core heatup under the impetus of decay heat.

The control room operators would recognize the initiation of an ATWS by the existence of a combination of scram signals, continued indication of reactor power on the average power range monitors (APRMs), and continued indication that multiple control blades remained in their fully withdrawn positions. (Control blade positions are prominently displayed upon a large core mockup on the front panel of the control room.) For a case in which the reactor did not scram automatically in conjunction with an MSIV closure event, entry into the Reactor Vessel Control Guideline of the EPGs would be triggered by vessel pressure above the high pressure scram setpoint and "a condition which requires reactor scram, and reactor power above APRM downscale trip or cannot be determined."[16] Either of these triggers is by itself sufficient for entry; only the second, however, is a unique signature of ATWS. The high reactor vessel pressure would also cause tripping of the recirculation pumps and initiation of the isolation condenser at plants so equipped (Table 2.4).

The Reactor Vessel Control Guideline calls for simultaneous efforts to control reactor vessel water level, vessel pressure, and reactor power. Initial measures would be taken to induce reactor shutdown by manual scram and by repositioning of the reactor mode switch. (Typically, placing this switch in the SHUTDOWN position will provide a diverse scram signal.) The alternate rod insertion (ARI) system would be initiated, which vents the reactor scram air header and closes the scram discharge volume vent and drain valves. Each of these actions has the potential to induce scram, but by the definition of this unmitigated accident sequence, these efforts would not be successful.

With the MSIVs closed and the recirculation pumps tripped, several SRVs would be continuously open (the number depending on the reactor power) while one valve cycled open and closed. In accordance with Step RC/P-1 of the EPGs, the operators would attempt to terminate the valve cycling by taking remote-manual control of the SRVs and reducing reactor vessel pressure.

Attempts to reduce reactor vessel pressure by manual SRV actuation under ATWS conditions would be extremely difficult.[13] If the operator attempted to open a valve that was already open, nothing would happen. If the operator opened a previously closed valve, the reactor vessel pressure would only drop slightly until one of the previously

open valves went shut. Thus, there would be only a negligible response to operator SRV control until the operator had manually opened as many valves as had previously been automatically open. Upon manual opening of the next valve (the valve previously cycling, now to be held continuously open), however, vessel pressure would rapidly decrease because of the power reduction (caused by increasing voids) occurring while several relief valves remained manually held open. The operator would have to be extremely quick to avert a complete vessel depressurization by closing the SRVs in this situation, thereby causing a rapid vessel repressurization as void collapse increased reactor power. Under these rapidly changing conditions involving power and pressure oscillations, it could not be claimed that the operator had control of reactor vessel pressure.

If the main condenser is available and there has been no indication of gross fuel failures or of a main steam line break, then the EPGs direct action to open the MSIVs and establish the main condenser as a heat sink. If this maneuver were successful, steam flow into the main condenser would be via the turbine bypass valves. Typically, these bypass valves can pass about 25% of the normal full-power steam flow from the reactor vessel. Therefore, the steam flow into the pressure suppression pool and the pool heatup rate would be greatly reduced. By the definition of this dominant severe accident sequence, however, this maneuver would not be successful.

Implementation of all available pressure suppression pool cooling is directed by the primary containment control guideline of the EPGs (Step SP/T-1).

Initiation of the standby liquid control system (SLCS) to inject liquid neutron poison (sodium pentaborate solution) into the reactor vessel is directed by the EPGs "before suppression pool temperature reaches the Boron Injection Initiation Temperature" (Step RC/Q-6). Simultaneous action to manually drive the control blades into the core is also directed (Step RC/Q-7). Several backup methods are specified for each endeavor should the primary means of accomplishment fail.

The Boron Injection Initiation Temperature is defined by the EPGs to be the greater of the pressure suppression pool temperature at which scram is required by the plant Technical Specifications or the pool temperature at which SLCS initiation would result in reactor (hot) shutdown under ATWS conditions before the Heat Capacity Temperature Limit is exceeded. It should be recalled that if the pool temperature exceeds the Heat Capacity Temperature Limit, then rapid depressurization of the reactor vessel is required by the EPGs; clearly the intent here is to avoid a requirement for rapid depressurization of a critical reactor by achieving hot shutdown before the pool temperature reached the limit. In some plants, however, this may not be possible, and the only way to avoid a rapid depressurization with the reactor critical would be to impose a higher Heat Capacity Temperature Limit under ATWS conditions. This is a subject undergoing current review.[28]

Instructions for control of reactor vessel water level under ATWS conditions are provided by Contingency 5 "Level/Power Control" of the EPGs. With the reactor remaining at power while sodium pentaborate solution is being injected, Step C5-2 of this contingency directs that the reactor vessel water level should be lowered to the top of the core. [Operation of the Automatic Depressurization System (ADS) while the water level is reduced is to be manually prevented.] Water level reduction is accomplished by restricting injection to the relatively small amounts provided by the boron injection system and the CRDHS. Once the water level has been reduced, it is to be maintained (by a controlled increased injection rate) within a range between the top of the core and a Minimum Steam Cooling RPV Water Level, which (employing several very conservative assumptions) ensures adequate cooling of a partially uncovered core.

Once the Hot Shutdown Boron Weight has been injected into the reactor vessel, the EPGs specify that the vessel water level should be restored to the normal range. Raising the vessel water level of course implies increased flow at the core inlet; this is intended to sweep the sodium pentaborate solution from the lower plenum up into the core region.

The strategy provided by the EPGs for dealing with an MSIV-closure ATWS can be summarized as follows: initiate injection of sodium pentaborate solution and lower the reactor vessel water level to the vicinity of the top of the core; when sufficient boron has been injected to achieve hot shutdown, restore the vessel level to the normal range. These actions should terminate the accident sequence without core damage. The principal challenge to this desired conclusion is that the operator actions undertaken while attempting to achieve the pressure control directed by the EPGs might unintentionally create an unstable situation.

If, however, all means of injection of sodium pentaborate solution into the reactor vessel fail, then temporary, partial measures to reduce core power such as lowering the reactor vessel water level can only delay the progression of events

into a severe accident. Manual control blade insertion can bring about permanent reactor shutdown, but this is a very slow process. Failure of the boron injection systems is a premise of the ATWS accident sequences leading to severe core damage identified by NUREG-1150. (The one exception involves early total loss of injection.)

Severe core damage resulting from ATWS occurs because the reactor vessel injection systems become failed and sufficient water cannot be kept in the core region. This is identical to the way in which core damage would occur for station blackout. Core damage and relocation of molten materials in BWR severe accident sequences would be driven by the impetus of decay heat. The guidance provided by the EPGs leaves the operator making every effort to restore reactor vessel injection.

4 Potential for Enhancement of Current BWR Strategies

S. A. Hodge

This chapter provides a discussion of the extent to which the candidate accident management strategies identified by the report "Assessment of Candidate Accident Management Strategies" (NUREG/CR-5474)[17] are currently incorporated within the BWR Emergency Procedure Guidelines (EPGs).[16] The potential for enhancement of the current strategies for application to situations involving severe core damage is also addressed. The purpose here is to identify the general accident management areas in which enhancement seems to be warranted.

4.1 Incorporation of Candidate Accident Management Strategies

Several of the candidate boiling water reactor (BWR) accident management strategies described in Sect. 3.1 are currently addressed in the BWR EPGs. It should be recalled, however, that these guidelines are generic to the several BWR plant designs and are intended to be adapted for application to individual plants as necessary to incorporate plant-specific considerations. Therefore, the EPGs do not provide detailed information as to how the strategies should be carried out. Because many of the strategies involve complicated actions not normally performed by the plant operating staff, effective implementation at a BWR facility cannot be accomplished without advance planning for use of the strategies, including preparation of detailed procedures and operator training.

The BWR accident management strategies described in Sect. 3.1 that are included in the guidelines of Revision 4 of the EPGs would be invoked at some point in either the station blackout or ATWS severe accident sequences. The application of these strategies is described in the following paragraphs, in the order in which these strategies are addressed in Sect. 3.1.

1. Alternate sources for reactor vessel injection (Sect. 3.1.3). Contingency #1 "Alternate Level Control" of the EPGs differentiates between "injection subsystems" and "alternate injection subsystems." The alternate injection subsystems are defined as systems and system interconnections capable of injecting water into the reactor vessel but not normally utilized for this purpose because of low water quality, the relative difficulty of establishing the injection line-up, or because the line-up is not permitted during normal plant operation. Several candidate alternate injection systems [residual heat removal (RHR) service water crosstie, fire system, etc.)] are suggested, but the identification of the systems for a particular plant is left to be accomplished with the generation of the plant-specific Emergency Operating Procedures (EOPs).

Contingency #1 provides guidance for reactor vessel water level control whenever the operators have concluded that the water level cannot be maintained above the top of the core using the normal water level control section of the Reactor Vessel Control Guideline. (This would occur in the station blackout accident sequences.) Step RC/L-2 permits (but does not require) use of the alternate injection systems and, if the water level cannot be maintained above the top of the core, directs the operators to enter the procedure developed from Contingency #1. Steps C1-2 through C14 then direct that the alternate systems be lined up, started, and employed for injection, if the preferred injection systems are not available.

For the special case of ATWS, reactor vessel level control is specified by Contingency #5, "Level/Power Control." Use of the alternate injection systems is specified at Step C5-3.2, provided that adequate level control cannot be achieved with the preferred systems.

If reactor vessel water level cannot be determined, then directions for vessel injection are taken from Contingency #4, "RPV Flooding." (The vessel is filled until the main steam lines are flooded or, for the case of ATWS, the core is adequately cooled by submergence and steam cooling.) Use of the alternate injection systems is specified at Step C4-1.3, if the preferred injection systems are not available.

2. Maintain pump suction upon the condensate system (Sect. 3.1.4). This strategy is implemented within the EPGs by specific direction for "suction from the condensate storage tank, defeating high suppression pool water level suction transfer logic if necessary" for every case where injection by the high-pressure coolant injection (HPCI) or reactor core isolation cooling (RCIC)systems is specified.

3. Operator override of injection pump trips (Sect. 3.1.5). In cases where reactor vessel water level cannot be adequately controlled by use of the preferred injection systems, use of pumps "irrespective of pump NPSH and vortex limits" is specified in Contingencies #1 (Alternate Level Control), #4 (RPV Flooding), and #5 (Level/Power Control). Although this does not strictly fall

under the category of "override of injection pump trips," it does provide clear guidance for the use of injection systems beyond their normal design limits when required.

Contingency #4 (RPV Flooding) specifies "defeating high RPV water level isolation interlocks if necessary" in Step C4-3.1 for use of high-pressure core spray (HPCS) and in Steps C41.3 and C4-3.1 for the motor-driven feedwater pumps. It should be recalled that this contingency is applicable for situations where reactor vessel water level cannot be determined. Specific guidance for override of high-level isolation interlocks if necessary is provided in recognition that the loss of water level indication that caused entry into this contingency may be due to instrument failure (off-scale high).

4. Maintain reactor core isolation cooling system availability (Sect. 3.1.6). Defeat of the low reactor vessel pressure RCIC system isolation interlocks, if necessary, is directed for every case where injection with this system is specified by the EPGs. In addition, the attention of the operators is directed to Caution #2, which warns of unstable system operation and equipment damage if the turbine is operated at speeds below the minimum recommended by the vendor, and to Caution #3, which warns that elevated suppression chamber pressure may cause RCIC turbine trip on high exhaust pressure. However, guidance to override this high exhaust pressure trip is not provided.

5. Use of control rod drive hydraulic pumps for decay heat removal (Sect. 3.1.7). The control rod drive hydraulic system (CRDHS) is listed among the preferred reactor vessel injection systems in the water level control guidelines (Step RC/L-2), in Contingency #4 "RPV Flooding" (Steps C4-1 and C4-3) and in Contingency #5 "Level/Power Control" (Step C5-3). However, the EPGs do not provide information with respect to the potential for operator actions to increase the injection rate that can be provided by this system.

6. Replenish safety-related air-operated components pneumatic supply (Sect. 3.1.10). When controlled reactor vessel depressurization is initiated in accordance with the EPGs at Step RC/P-3, the following direction is included as part of this step:

> If one or more SRVs are being used to depressurize the RPV and the continuous SRV pneumatic supply is or becomes unavailable, depressurize with sustained SRV opening.

The rationale for this guidance is provided in Sect. 6 of Appendix B of the EPGs as follows:

> Sustained SRV opening conserves accumulator pressure when the source of pressure to the SRV pneumatic supply system is isolated or otherwise out of service. Such action prolongs SRV availability should more degraded plant conditions later require SRVs be opened for rapid depressurization of the RPV.
>
> The term "continuous" encompasses any backup or alternate means of pressurizing the SRV pneumatic supply system in addition to the permanent (e.g., normal) SRV pneumatic source.

The EPGs also note in Step RC/P-2 that reactor vessel pressure control during the period before controlled depressurization is required can be augmented by several means, including remote-manual actuation of the safety/relief valves (SRVs). This, however, is subject to the proviso that the valve control switches must be placed in the position that signals valve closing if the continuous pneumatic supply is or becomes unavailable. Appendix B, Sect. 6 of the EPGs provides the following basis for this proviso:

> Loss of the continuous pneumatic supply to the SRVs limits the number of times that an SRV can be cycled manually since pneumatic pressure is required for this mode of valve operation. Even though the SRV accumulators contain a reserve pneumatic supply, leakage through in-line valves, fittings and actuators may deplete the reserve capacity. Thus, subsequent to the loss of the continuous SRV pneumatic supply, there is no assurance as to the number of SRV operating cycles remaining. For these reasons, if SRVs must be used to augment RPV pressure control and if the continuous SRV pneumatic supply is or becomes unavailable, the valve should be closed to limit the number of cycles on the valve and conserve pneumatic pressure so that, if emergency RPV depressurization is subsequently required, the valve will be available for this purpose. If other pressure control systems are not capable of maintaining RPV pressure below the lowest SRV lifting pressure, SRVs will still open when the lifting pressure is reached.

7. Use of diesel-driven fire protection system pumps for vessel injection or containment sprays (Sect. 3.1.14). The EPGs include the "Fire system" among the suggested alternate reactor vessel injection systems listed in the water level control guidelines (Step RC/L-2), in Contingency #1 "Alternate Level Control," in Contingency #4, "RPV Flooding," and in Contingency #5, "Level/Power Control," which would be invoked for ATWS accident sequences. The EPGs do not indicate any particular power source for this system. It is the responsibility of the individual plant developing the local EOPs from the guidance provided by

the EPGs to identify the alternate injection systems appropriate for that plant.

8. Regaining the main condenser as a heat sink (Sect. 3.1.15). Step RC/P-1 of the EPG reactor vessel control guideline directs that if boron injection is required, the main condenser is available, and there has been no indication of gross fuel failure or steam line break, then the main steam isolation valves (MSIVs) should be opened to establish the main condenser as a heat sink. In addition, it is specified that bypass of the pneumatic system and low reactor vessel water level MSIV closure interlocks should be performed if necessary to accomplish this step.

Clearly, this action is intended to be taken in response to an ATWS situation. The rationale for opening the MSIVs after they have tripped shut under accident conditions is provided in Sect. 6 of Appendix B of the EPGs, as follows:

> To stabilize and control RPV pressure, the reactor steam generation rate must remain within the capacity of systems designed to remove the steam from the RPV. With the reactor not shutdown, the amount of steam that may have to be released to effect RPV pressure control could be substantial. If this total heat energy is discharged to the suppression pool, the Heat Capacity Temperature Limit could be reached in a very short time. Therefore, utilization of the main condenser as a heat sink for this energy is of sufficient importance to warrant opening the MSIVs even if the valves have automatically closed. Such action may be the principal contributor to successful mitigation of a failure-to-scram condition.
>
> This override permits bypassing the low RPV water level portion of the MSIV isolation logic. Other MSIV isolation interlocks (i.e., main steam line high radiation) are not bypassed because they provide automatic protection for conditions where reopening the MSIVs is not appropriate. In addition, this override authorizes bypassing any interlocks which inhibit restoration of the pneumatic supply to the MSIV actuators since accumulator pressure alone may not be sufficient to open and hold open the MSIVs.

MSIVs may be reopened if all of the following conditions exist:

- Boron injection is required. This condition is described in Step RC/Q6 as "... the reactor cannot be shutdown before suppression pool temperature reaches the Boron Injection Initiation Temperature ..."
- The main condenser is available. The only reason for opening the MSIVs is to utilize the main condenser as the heat sink.
- There is no indication of "gross" fuel failure. Opening the MSIVs with failed fuel could result in a significant release of fission products to the environment. The means for detecting fuel failure are plant unique, and thus no further details are specified in this override. "Gross" fuel failure is specified to distinguish from small cladding leaks. The judgement is subjective, based on operator assessment of available indications. If it is concluded that no gross fuel failure exists but in actuality core damage has occurred, high radiation should be detected when the MSIVs are opened and the MSIV isolation logic should automatically reclose the valves. It is for this reason that bypassing the high radiation portion of the MSIV isolation logic is not authorized.
- There is no indication of a steam line break. Opening MSIVs with a break in the downstream piping could result in an uncontrolled loss of reactor coolant inventory, release fission products to the environment, and cause personnel injury or significant damage to plant equipment. It is difficult, however, to determine whether a steam line break exists with the MSIVs closed, other than by visual inspection of system piping. If there is reasonable assurance that no break existed before MSIV closure (i.e., there were no indications of high steam flow, high area temperatures, etc.), an operator may conclude that no break developed subsequent to valve closure. Still, the judgement is subjective, based on an operator's assessment of all available indications. If it is concluded that no steam line break exists but, in actuality, one does exist, high steam line flow and high steam tunnel temperature should be detected when the MSIVs are opened and the MSIV isolation logic should automatically reclose the valves. It is for this reason that bypassing the high steam flow or high steam tunnel temperature portions of the MSIV isolation logic is not authorized.

4.2 Candidate Strategies Not Currently Addressed

The following candidate BWR accident management strategies identified by the NUREG/CR-5474 report[17] and briefly described in Sect. 3.1 are not represented within the guidelines of Revision 4 of the EPGs:

1. Maintaining the condensate storage tank as an injection source (Sect. 3.1.2)
2. Load-shedding to conserve battery power (Sect. 3.1.8)
3. Battery recharging under station blackout conditions (Sect. 3.1.9)
4. Bypass or setpoint adjustment for diesel generator protective trips (Sect. 3.1.11)
5. Emergency crosstie of ac power sources (Sect. 3.1.12)

Potential

6 Alternate power supply for reactor vessel injection (Sect. 3.1.13)
7. Injection of boron under accident conditions (Sect. 3.1.16)

The candidate strategy for coping with an interfacing systems LOCA briefly discussed in Sect. 3.1.1 is primarily associated with the Secondary Containment Control Guideline of the EPGs and, being outside the scope of this report, will not be addressed further here.

Identification of the strategies listed as items 1–7 above as not being included within the EPGs is not intended to imply that they should all be so included. Of these seven items, one has to do with refilling the condensate storage tank, and five have to do with the electric power distribution system, all highly dependent upon plant-specific arrangements. Therefore, implementation of these six items within the plant EOPs (rather than the generic EPGs) is probably a more practical approach.

The seventh item, however, has to do with the injection of boron under severe accident conditions where the control blades have melted and relocated from the core, the fuel remains in a critical configuration, and reactor vessel injection capability has been restored. This item does have generic applicability to unmitigated BWR accident sequences and therefore is appropriate for inclusion either as an extension of the EPGs or as part of a separate set of procedures providing generic guidance for severe accident applications.

4.3 Management of BWR Severe Accident Sequences

As indicated in Sect. 3.2, the EPGs do not provide mitigation strategies proposing actions in response to the symptoms created by events that would occur only after the onset of significant core damage. As demonstrated for the station blackout and ATWS severe accident sequences, the final guidance to the operators should the accident proceed into the severe core damage phase and beyond is to restore injection to the reactor vessel by any means possible and to maintain the vessel depressurized. While these are certainly important endeavors, it seems that additional advance planning for coping with severe accident situations will suggest additional guidance for the core damage phase and beyond.

It is not within the scope of this chapter to propose specific new accident management strategies or enhancements. The purpose here has been to examine the current status of the accident management strategies provided by the EPGs and the candidate accident management strategies proposed for BWR applications by the NUREG/CR-5474 report[17] and to indicate the general accident management areas in which enhancement seems to be warranted for in-vessel events. Although several questions remain with respect to the overall response to the ATWS accident sequence, most of the potential benefit that could be attained by enhancement of the existing strategies lies in the realm of severe accident management, for the extremely unlikely, but possible, events associated with significant core damage and structural relocation.

Potential areas for enhancement include removal of the rod sequence control system to facilitate the manual insertion of control blades under ATWS conditions, advance planning for the use of the available instrumentation under station blackout conditions (including after-loss of dc power), advance consideration of the status of emergency lighting and the plant security system for all loss-of-power situations, and the provision of measures to maintain the reactor vessel depressurized without dc power and without a pneumatic source.

For coping with events beyond severe core damage, measures to ensure boron injection capability after control blade relocation have already been suggested by NUREG/CR-5474 (and summarized in Sect. 3.1.16). Clearly, means for the operating staff to determine the status of the in-vessel structures would be beneficial in severe accident situations, and the reactor vessel thermocouples might provide useful information in this regard. Should core and structural material relocate into the reactor vessel lower plenum inducing boiloff of all water remaining there, flooding of the primary containment and the presence of water surrounding the vessel might provide sufficient cooling of the vessel bottom head to maintain the core and structural debris in-vessel. (Primary containment flooding is already treated by Contingency #6 of the EPGs, but the concept is for LOCA situations where the water within the containment could enter the reactor vessel through the break.)

These ideas and other severe accident mitigation concepts to be developed for in-vessel applications will be addressed in the following chapters.

5 BWR Severe Accident Vulnerabilities

S. A. Hodge

It is a major purpose of the current study to propose candidate accident management strategies to mitigate the effects of in-vessel events during the late phase (after core degradation) of a boiling water reactor (BWR) severe accident sequence. In pursuing this goal, it is logical to first review the susceptibilities of the BWR to the challenges imposed under accident conditions. This chapter provides a brief summary of the recent results of probabilistic risk assessment (PRA), the challenges imposed by the dominant accident sequences, and the importance of plant-specific considerations in assessing accident sequence events.

5.1 Lessons of PRA

The station blackout accident sequence has consistently been identified as the leading contributor to the calculated core damage frequency in recent PRAs for plants of the BWR design. As described in Sect. 2.1, the recent Nuclear Regulatory Commission (NRC)-sponsored assessment of severe accident risks (NUREG-1150)[6] provides the estimates that 47% of the risk of core damage (internally initiated accidents) for Peach Bottom and 97% of the core damage risk for Grand Gulf is attributable to station blackout. Other recent PRAs based upon Limerick[19] and Susquehanna[20] have similar findings, with 42% and 62%, respectively, of the overall core damage frequency for these BWR plants calculated to derive from station blackout.

BWRs are well protected against core damage because they have redundant reactor vessel injection systems to keep the core covered with water. [Fuel damage cannot occur in covered regions of the core, even for anticipated transient without scram (ATWS) sequences.[13]] The reason that station blackout is the leading contributor to BWR core damage frequency is simply that the majority of the reactor vessel injection systems are dependent upon the availability of ac power and BWRs are vulnerable to loss of injection.

Other dominant core damage accident sequences also involve failure of reactor vessel injection, because the core must be at least partially uncovered for structural degradation and melting to occur. The manner in which the injection systems are lost for the dominant severe accident sequences is briefly reviewed in the following sections.

5.2 Station Blackout

Station blackout is the accident sequence initiated by loss of off-site power and the associated scram and closure of the main steam isolation valves (MSIVs) combined with failure of the station diesels (and gas turbines, if applicable) to start and load. Therefore, by the definition of the accident initiating events, all electric-motor-driven reactor vessel injection systems become unavailable at the inception of the accident sequence.

Most of the 37 operating BWR facilities in the United States are protected against loss of the motor-driven reactor vessel injection systems by having steam turbine-driven reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems. These injection systems take steam from the isolated reactor vessel, passing the turbine exhaust to the pressure suppression pool and pumping replacement water into the vessel. The newer plants of the BWR-5 (four U.S. units) and BWR-6 (four U.S. units) design have only one steam turbine-driven injection system, the RCIC. Three of the oldest facilities (the two BWR-2 plants and one of the BWR-3 plants) have neither RCIC nor HPCI, but instead rely upon an isolation condenser for station blackout protection. This emergency core cooling system employs natural circulation through the elevated isolation condenser tubes to remove decay heat from the reactor vessel in the event that electrical power is unavailable. The shell volume (vented to the atmosphere) of the condenser contains a water volume that boils away to remove the heat transferred from the reactor. The shell water volume is sufficient to provide ~30 min of core cooling without the addition of makeup water (which is taken from the firemain system or from the condensate transfer system).

As in other accident sequences, core degradation occurs in station blackout only after reactor vessel injection capability is lost. Because the RCIC and HPCI systems rely upon dc power for valve operation and turbine governor control, these systems will be lost if ac power is not restored before the unit batteries become exhausted. In the accident classification methodology adopted for the recent severe accident risks study (NUREG-1150), this form of the severe accident sequence initiated by loss of off-site and on-site ac power is classified as "long-term station blackout," because a significant period of time (typically 6 to 8 h)

would elapse before battery exhaustion caused loss of reactor vessel injection capability.

The second form of the severe accident sequence associated with loss of all ac power is termed "short-term station blackout" because, for this sequence, the RCIC (and HPCI, for plants so equipped) system fails to start upon demand. This might occur because of turbine mechanical failure or, less probably, by common-mode failure of the dc battery systems at the inception of the accident. (Failure of the batteries as an initiating event would also preclude starting of the diesel-generators upon loss of off-site power.)

The relative probabilities of the long-term and short-term versions of the station blackout accident sequence depend upon the plant-specific configuration of the battery systems (some plants have independent starting batteries for the diesels) and whether the plant has one or two steam turbine-driven injection systems. Additional description of the station blackout accident sequences is provided in Sect. 2.2.1. The main points to be made here are that reactor vessel injection capability is lost at the inception of the accident sequence for short-term station blackout, but is retained until the unit batteries become exhausted (typically 6 to 8 h later) for long-term station blackout. Core degradation follows the uncovering of the core, which occurs as the vessel water inventory is boiled away without replacement.

5.3 Anticipated Transient Without Scram

ATWS is the accident sequence initiated by a complete failure of control rod insertion following a transient event for which the plant protection system normally provides a scram. The ATWS initiated by a transient event that causes closure of the MSIVs is the most severe of this class of accident sequences. (Other types of ATWS are discussed in Chap. 2 of Ref. 13.) With the MSIVs closed, almost all of the steam exiting the reactor vessel would be passed through the safety/relief valves (SRVs) into the pressure suppression pool. (The remainder would be used to drive the HPCI or RCIC turbines during their periods of operation and then enter the pressure suppression pool as turbine exhaust.) Because the rate of energy deposition into the pool can greatly exceed the capacity of the pool cooling equipment, excessive pool temperatures leading to primary containment failure by overpressurization is of major concern during ATWS accident sequences. (A more detailed description of the ATWS accident sequence for BWRs is provided in Sect. 2.2.2 and in Ref. 13.)

As discussed in Sect. 5.1, all recent BWR PRAs have identified station blackout as the leading contributor to the overall risk of core melt (internally initiated accidents). It is now important to note that in every PRA cited,[6,19,20] the ATWS accident sequence (initiated by MSIV closure) is identified as second in order to station blackout, contributing 42% of the overall calculated risk for Peach Bottom, 3% for Grand Gulf, 28% for Limerick, and 32% of the calculated Susquehanna risk.

For the ATWS accident sequences, as for all other BWR severe accident sequences, core degradation can occur only after failure of adequate reactor vessel injection. If sufficient water were injected under ATWS conditions to maintain a lower portion of the core critical, then the resulting steam generation would be sufficient to provide adequate steam cooling of the uncovered (subcritical) upper region of the core. Structural degradation and melting would occur in an ATWS severe accident sequence only after reactor vessel injection had been lost, with the subsequent heatup of the uncovered core under the impetus of decay heating.

Reactor vessel injection capability would be lost in an unmitigated ATWS accident sequence because of events occurring in the overheated and pressurized primary containment. One way that this could happen is that the primary containment structure might actually lose its integrity by overpressure; the concomitant release of steam into the secondary containment (the surrounding reactor building) might then cause failure of essential reactor vessel injection system components (control panels, switchboards) located therein. However, it is not necessary for the primary containment pressure boundary to fail in order to lose reactor vessel injection capability under ATWS conditions.

Most reactor vessel injection systems are low-pressure systems, requiring that the reactor vessel be depressurized for successful fulfillment of their functions. By its very nature, with the core at power while the MSIVs are closed, the dominant form of the ATWS accident sequence tends to maintain the reactor vessel at pressures somewhat higher than normal (sufficient for steam release through the SRVs).

The steam turbine-driven HPCI and RCIC systems are capable of high-pressure injection but are susceptible to elevated pressure suppression pool temperatures when taking suction from this source because their lubricating oil is cooled by the water being pumped. In addition, both of

these systems have high turbine exhaust pressure trips so that high primary containment pressure can cause their failure. Steam-driven feedwater pumps would be lost at the inception of the accident sequence when MSIV closure cuts off their steam supply.

For Peach Bottom, which has both HPCI and RCIC, the recent NRC-sponsored study of severe accident risks[6] found that loss of reactor vessel injection capability for ATWS would be caused by MSIV closure (loss of feedwater) combined with either random faults for the turbine-driven systems or failure induced for these systems by high-pressure suppression pool temperature.

The Grand Gulf Plant has an electric-motor-driven high-pressure core spray (HPCS) system in lieu of HPCI. Here the NUREG-1150 study found that loss of reactor vessel injection would occur, for the dominant form of ATWS, because the HPCS "system fails to start and run because of random hardware faults. The reactor is not depressurized and therefore the low-pressure cooling systems cannot inject."

5.4 Other Important Accident Sequences

Review of the PRA results demonstrates conclusively that the BWR is vulnerable only to loss of reactor vessel injection and that the postulated accident sequence scenarios leading to core damage always include means for failure of function of the vessel injection systems. As defined, the various severe accident sequences involve different pathways to and timing of loss of vessel injection capability, but, in every case, the core must become uncovered before core damage can occur.

An example of an alternate pathway to loss of reactor vessel injection capability is provided by the accident sequence identified as third in order of calculated frequency leading to core melt for Peach Bottom by the NUREG-1150 risk assessment.[6] This is a medium-size LOCA, which is estimated to provide ~6% of the overall mean core damage frequency. For this accident sequence, injection by the smaller [600-gal/min (0.038-m^3/s)] RCIC system is insufficient to replace the water loss through the break. The HPCI system [5000 gal/min (0.315 m^3/s)] provides reactor vessel injection initially, but subsequently fails because of low turbine steam supply (reactor vessel) pressure. The low-pressure core cooling systems fail to activate primarily because of miscalibration faults of the vessel pressure sensors so that the open-permissive signals necessary for opening of the injection valves are never received. With all core cooling lost, core damage occurs within 2 h of the initiating event.

Another pathway to loss of vessel injection is demonstrated by the loss of decay heat removal (DHR) accident sequence, fourth in order of calculated risk of core melt for the Limerick PRA.[19] Pressure suppression pool cooling is lost following a transient-initiated scram in this accident sequence, which is estimated to contribute 3% of the overall core melt frequency for Limerick. Reactor vessel injection is maintained during the first phase of this accident sequence, while the steam generated by decay heating is passed to the pressure suppression pool. Without pool cooling, the water temperature increases over a period of several hours while the associated evaporation increases the containment pressure. Reactor vessel injection capability is ultimately lost because of loss of net positive suction head (NPSH) to the low-pressure pumps (which take suction on the pressure suppression pool) and failure of the HPCI and RCIC on high lubricating oil temperature or high containment backpressure. A more detailed description of the loss of DHR accident sequence is provided in Ref. 11.

5.5 Plant-Specific Considerations

While it is true in every case that core degradation must be preceded by loss of adequate reactor vessel injection and partial core uncovering, the detailed means by which vessel injection capability might be lost are highly plant-specific. While the overall goal of preventative strategies for accident management is clearly that adequate reactor vessel injection capability should be maintained, the detailed nature of the threats to the injection systems and the optimum measures that should be taken to cope with these threats depend upon the equipment characteristics of the individual plants. (For examples of important differences among supposedly similar BWR facilities, see Sect. 2.3.)

The recent NRC-sponsored assessment of severe accident risks (NUREG-1150) was based upon three pressurized water reactor (PWR) and two BWR plants, Peach Bottom and Grand Gulf. This study has recently been extended to include a third BWR, LaSalle, so that one BWR of each containment type has been assessed. Extension of the methodology to take into consideration the plant-specific differences at other BWR (and PWR) facilities is the responsibility of the plant operators as part of the individual plant examination process.[29,30] The results of this process should include plant procedures incorporating specific preventative measures for avoiding loss of adequate reactor vessel injection capability under accident conditions.

BWR

It is also desirable for defense-in-depth to develop mitigative strategies for coping with the late-phase severe accident events that would occur in the unlikely event that adequate vessel injection cannot be maintained. Candidate strategies for this purpose will be proposed in Chap. 8.

6 Status of Current BWR Accident Management Procedures

S. A. Hodge

This chapter provides a brief description of the current guidance for accident management at boiling water reactor (BWR) facilities together with an evaluation of the preventative and mitigative effectiveness of the current guidelines with respect to operator actions under severe accident conditions.

6.1 BWR Owners Group Emergency Procedure Guidelines (EPGs)

The BWR Owners Group EPGs[16] are generic to the General Electric BWR plant designs and are intended to be adapted for application to individual plants by the deletion of irrelevant material and the substitution, where necessary, of plant-specific information. The development of a set of Emergency Operating Procedures (EOPs), based upon the EPGs, for use at a particular plant is the responsibility of the local plant management.

The current version of the EPGs is Revision 4, issued as General Electric topical report NEDO-31331 in March 1987.[16] In considering their application to specific accident sequences, it is important to recognize that these guidelines identify symptomatic operator actions. In other words, given a symptom requiring entry into the EOPs developed from these guidelines, it is intended that the operators take action in response to the symptom without any requirement to first diagnose the cause. Development of these guidelines has been based upon realistic analyses, rather than upon licensing basis calculational methods, and considers utilization of all plant systems, not just the safety systems. The guidelines are intended to address all mechanistically possible abnormal plant conditions, without consideration of the probability of the abnormal event or condition.

The basic functional goal of the EPGs is to establish the prudent actions to be taken by the operators in response to the symptoms observed by them at any point in time. Once entry into the EPGs has occurred, the operators are expected to take the specified actions regardless of equipment design bases limitations or licensing commitments. The guidelines use multiple mitigation strategies wherever possible so that recovery from an abnormal situation does not require successful operation of any one system or component.

The application of the EPGs to accident management of in-vessel events for the risk-dominant station blackout and anticipated transient without scram (ATWS) accident sequences is discussed in detail in Sect. 3.2. The EPGs were found to provide effective guidance for preventative measures to be taken in response to the challenges imposed by these accident sequences. As appropriate, these preventative measures invoke numerous diverse methods of maintaining reactor vessel injection capability, including backup methods for use in abnormal circumstances. Some recommendations for improvement of the preventative guidelines for accident sequences that can be brought under control are suggested in the following section. However, the greatest potential for the improvement of BWR emergency procedure strategies lies in the area of severe accident mitigative management, as will be described in Sect. 6.3.

6.2 Recommendations Concerning Preventative Measures

One of the primary purposes of this report is to propose new candidate mitigative strategies for the late phase of a severe accident, after core damage has occurred. Before proceeding, however, it seems appropriate to note in passing certain recommendations with respect to the preventative guidelines of the current EPGs. These recommendations for reconsideration of a few of the provisions of the current guidelines are presented in this section.

The first recommendation is applicable to all accident sequences involving partial uncovering of the core and has to do with the provision of the EPGs for "steam cooling." Briefly, the purpose of steam cooling is to delay fuel heat-up by cooling the uncovered upper regions of the core by a rapid flow of steam. Because the source of the steam is the remaining inventory of water in the reactor vessel, however, the steam cooling maneuver can provide only a temporary delay.

Steam cooling would be placed into effect when the reactor vessel water level dropped to the "Minimum Zero-Injection RPV Water Level." In Revision 4 of the EPGs, this is defined as the lowest vessel level at which the average steam generation rate within the covered portion of the core will produce a sufficient flow of steam to prevent the maximum clad temperature in the uncovered region of the core from exceeding 1800°F (1255 K). (Runaway oxidation of the zirconium cladding is precluded for temperatures below this value.)

Status

With the core partially uncovered and the vessel water level decreased to the Minimum Zero-Injection RPV Water Level, the operators are directed by the EPGs to open all safety/relief valves (SRVs) associated with the automatic depressurization system (ADS). This action provides flashing of the water in the core region, providing the desired temporary cooling of the uncovered region of the core.

This recommendation for reconsideration of the steam cooling maneuver as currently implemented arises because the Minimum Zero-Injection RPV Water Level calculated in accordance with Revision 4 of the EPGs provides that the emergency vessel depressurization would occur much earlier than was the case with Revision 3. It seems that the conservatisms built into the calculational procedure to ensure that under no conditions would the hottest point of the exposed cladding exceed 1800°F (1255 K) are such that the Minimum Zero-Injection RPV Water Level, while plant-specific, is typically at ~71% of core height. In other words, only the upper 29% of the core would be uncovered at the time the emergency vessel depressurization is specified. However, the actual temperature of the clad in the uncovered region under station blackout conditions with the water level at this height would be much less than 1800°F (1255 K), and the amount of steam cooling actually achieved would therefore be insignificant. Revision 3 of the EPGs had provided that the emergency vessel depressurization should occur when water level indication was lost, which occurs at about one-third core height (two-thirds of the core uncovered). This matter is currently under review.[27]

The other recommendations in connection with the preventative guidelines of the EPGs all pertain to the ATWS accident sequence. First and foremost, it is recommended that consideration be given to providing distinctly separate procedures for dealing with ATWS. The reasoning derives from the symptom-oriented nature of the EPGs. As explained in Sect. 6.1, the operator is not required to recognize or understand the accident sequence, but is expected to respond to plant symptoms. There is no question that this approach can be highly successful when applied to a group of accidents that have similar symptoms and require similar corrective actions by the operators. However, operator actions in response to the ATWS accident sequence include measures such as reduction of core inlet flow and intentional lowering of the reactor vessel water level to the vicinity of the top of the core. This is to increase the voids in the core, and thereby reduce core power and the rate of pressure suppression pool heatup, and is the proper thing to do when confronted with ATWS, but no other accident sequence would require these actions.

It is recommended that consideration be given to the separation of the ATWS guidelines from the symptom-oriented guidance for all other accident sequences. The bases for this recommendation are, first, that the signatures of ATWS are unmistakable so that operators would know when to invoke the ATWS procedures and, second, that the operator actions required to deal with ATWS do not fit within the envelope of operator actions required to deal with other accident sequences. The advantages to be gained by following this recommendation are that the very complicated procedures required for coping with ATWS would be more concisely and effectively presented while the remaining symptom-oriented guidelines would be greatly simplified.

A second recommendation with respect to the preventative guidelines for ATWS is that care be taken to avoid leading the operators to attempt manual depressurization of a critical reactor. Attempts to reduce reactor vessel pressure by manual SRV actuation under ATWS conditions would be extremely difficult. [See Sect. 4.1.3 of Ref. 13.] If the operator attempted to open a valve that was already open in the automatic mode, nothing would happen. If the operator opened a previously closed valve, the reactor vessel pressure would only drop slightly until one of the valves previously open in the automatic mode went shut. Thus, there would be only a negligible response to operator SRV control until the operator had manually opened as many valves as had previously been automatically open. However, upon manual opening of the next valve (the valve previously cycling, now to be held continuously open), vessel pressure would rapidly decrease because of the power reduction caused by increasing voids (while several relief valves remained manually held open). The operator would have to be extremely quick to avert a complete vessel depressurization by closing the SRVs in this situation; this action, however, would cause a rapid vessel repressurization as void collapse increased reactor power. Under these rapidly changing conditions involving power and pressure oscillations, it could not be claimed that the operator had control of reactor vessel pressure. If pressure reduction is required to reduce reactor power (in the extremely unlikely event that sodium pentaborate cannot be injected), then no attempt should be made to control the pressure at a lower range by reclosing the manually opened SRVs.

A third recommendation with respect to the preventative guidelines for ATWS is that consideration be given to control of the reactor vessel injection rate as a means for reduction of reactor power as opposed to the reactor vessel water level control directed by the current guidelines. Control of the vessel injection rate provides definitive control of the average reactor power [see Appendix B of

Ref. 13] whereas the reactor power associated with maintaining the water level in the vicinity of the top of the core is uncertain. Furthermore, directions to maintain a water level near the top of the core are unusual, require shifting to a different vessel level instrument calibrated for a different set of conditions, and are much more difficult than injection flow control to carry out with oscillating water levels under stressful conditions. It is recognized that injection flow control would not be successful in the extremely unlikely event that the ATWS were combined with a small-break loss-of-coolant accident (LOCA) (some of the injected flow intended to be boiled away by core power would instead be lost through the break); this eventuality, however, could be easily handled by procedure if the first recommendation above concerning separation of the ATWS instructions were adopted.

Finally, it is recommended that the guidance to the operators regarding manual insertion of control blades be expanded beyond the current "drive control rods, defeating RSCS and RWM interlocks if necessary." Manual control blade insertion may be essential to avoid containment pressures sufficient to threaten structural integrity in the unlikely event that the liquid neutron poison cannot be injected. However, manual blade insertion requires that the operators bypass the rod worth minimizer (RWM) and the rod sequence control system (RSCS), for the BWR-4 and BWR-5 plants that have these systems. (BWR-2 and BWR-3 plants have only the RWM, while the BWR-6 plants have neither the RWM nor the RSCS.)

Typically, the RWM can be quickly bypassed from the control room, but the RSCS can only be bypassed by the installation of jumpers in the relay room, an action that can reasonably be expected to take ~15 min once the decision to initiate the bypass is made. Because manual blade insertion for these plants is a slow process anyway (only one blade can be moved at a time, at a speed requiring about one minute for travel from fully withdrawn to fully inserted), the additional time required to effect bypass of the RSCS may be unacceptable from the standpoint of effective preplanning for severe accident management.

The RSCS was originally intended to eliminate the potential for local core damage from a high-worth control rod drop accident at low power. However, recent analysis by General Electric has demonstrated that such damage would not occur because local voiding would limit the associated power excursion. The Nuclear Regulatory Commission has approved this analysis and has issued a Safety Evaluation Report [31] that concludes that it is acceptable to remove the plant Technical Specification requirements for the RSCS. From the standpoint of enhancement of the ability of the operators to successfully respond to the ATWS accident sequence, it is desirable that this system be removed from the affected plants.

6.3 Sufficiency for Severe Accident Mitigation

It has never been a purpose of the current EPGs to address severe accidents or to invoke mitigation strategies involving proposed actions in response to the symptoms that would be created by events occurring only after the onset of significant core damage. As has been demonstrated (Sect. 3.2) for the risk-dominant station blackout and ATWS accident sequences, the final guidance to the operators, should an accident proceed into severe core damage and beyond, is that reactor vessel injection should be restored by any means possible and that the reactor vessel should be depressurized. While these are certainly important and worthwhile endeavors, additional guidance can and should be provided for the extremely unlikely, but possible,[6] severe accident situations that have occurred where reactor vessel injection cannot be restored before significant core damage and structural relocation. Four new candidate severe accident mitigation strategies for in-vessel events will be proposed in Chap. 8.

7 Information Available to Operators in Late Accident Mitigation Phase

S. A. Hodge

In considering new candidate severe accident mitigation strategies for use with existing plant equipment, it is important to first recognize any limitations imposed upon the plant accident management team by lack of information with respect to the plant status. This chapter describes instrumentation expected to be available to monitor in-vessel events during the period of core degradation that would occur following a loss of reactor vessel injection capability.

7.1 Station Blackout

The most restrictive limitation as to information available from plant instrumentation would occur as a result of loss of all electrical power, including that provided by the unit battery. This occurs after battery failure in the long-term station blackout accident sequence and in the (less-probable) version of the short-term station blackout accident sequence for which common-mode failure of the battery systems is an initiating event. For these accident sequences (described in Sect. 2.2.1), loss of reactor vessel injection and the subsequent core degradation occur only after loss of dc power.

The availability of the various plant instruments under accident conditions is a plant-specific consideration, and this should be an important part of each individual plant examination for severe accident vulnerabilities.[29,30] The following discussion, based upon the Browns Ferry (BWR-4) Plant, is intended to provide information with respect to the instruments expected to remain operational after loss of the 250-V unit battery at a typical BWR facility. These include instruments operated by plant-preferred power, instruments operated by relatively small independent battery systems, and instruments not requiring electrical power.

Plant-preferred power is 120-V single-phase ac power obtained by means of a dc-motor/ ac-generator combination from the station battery when normal ac power sources are not available. The station battery is similar in construction to each of the unit batteries, but would be more lightly loaded under station blackout conditions. The major loads on the station battery are the turbo-generator and seal oil pumps, which could be stopped when the turbo-generators have stopped rolling, because there would be no turbine jacking capability with station blackout in effect. Thus, the station battery should remain available for a significant period of time after loss of the unit batteries (as can be established for individual plants by battery capacity calculations).

Plant-preferred power would make available two types of information concerning plant status. The first is an indication of the temperatures at various points within the containment drywell as provided by both a recorder and an indicator-meter; the specific drywell location for which temperature is to be displayed is selected by manipulation of a set of toggle-switches. (This special equipment is intended for use during containment integrated leak rate tests.)

The second indication of plant status that could be monitored while plant-preferred power remained available is the temperature at various points on the reactor vessel surface as provided by 46 copper-constantan thermocouples. Probe-type thermocouples are used to measure temperature inside the vessel head studs, while magnetically attached thermocouples measure the surface temperature of the vessel and top head. These are intended to provide the information necessary to determine thermal stresses during vessel heatup or cooldown, but could also provide valuable information under severe accident conditions. Although the upper limit of the indicating range for the instruments used to display the thermocouple response is only 600°F (589 K), the accident management staff can infer that the core is uncovered and that the internal reactor vessel atmosphere is superheated if these thermocouple readouts are pegged high.

Two small independent battery systems provide power to control room instrumentation and alarm circuits. The first of these is a 24-V dc system, which supplies power to the source-range and intermediate-range neutron flux monitors as well as to radiation monitors for the off-gas, residual heat removal (RHR) service water, liquid radwaste, reactor building closed cooling water, and raw cooling water systems, none of which would be operational during station blackout. The battery chargers for the 24-V dc batteries are powered by the 120-V ac instrument and control buses, which would be de-energized during station blackout. Therefore, because the 24-V batteries are designed to supply the connected loads for 3 h without recharging, the 24-V dc system should become exhausted in long-term station blackout before the 250-V dc system powered by the unit batteries, which are expected to last between 6 and 8 h.

Information

For the version of short-term station blackout initiated by common-mode failure of the unit batteries, however, the 24-V dc system should remain available during the uncovering of the core and the period of core degradation and structural relocation. Because the neutron flux indicated by the source-range monitors would rapidly decrease as the fission chamber detectors became uncovered, observation of this event might be used to establish the time at which the reactor vessel water level reached the vicinity of the detector. As indicated in Fig. 7.1, the detector is between the core midplane and two-thirds core height when fully inserted. When fully withdrawn, the detector is about 1-1/2 ft below the core plate. The detector is withdrawn during power operation and must be reinserted before startup; this requires the availability of the same source of 120-V ac instrument and control power as is used for the 24-V dc battery chargers. (Whether the source-range detectors could be inserted with only battery power available is another of the many plant-specific considerations. At Browns Ferry, they could not, but it is known that at least one plant does have provision for this.)

The second of the small battery systems is a 48-V dc power supply and distribution system for the operation of the plant communication and annunciator systems. This system comprises three batteries, one of which supplies the plant communication system while the remaining two batteries are for the annunciator system. The 48-V dc system batteries are capable of supplying the connected loads for a period of 8 h without recharging. This exceeds the period of expected operation of the 250-V dc unit battery systems and ensures continued availability of the plant communications systems. The efficacy of the annunciator system would depend upon the availability of the power supplies to the signal transmission systems of the various sensors as well as the 48-V dc system. For all practical purposes, the alarm annunciator capability would be lost when the unit battery became exhausted.

The most important instruments not requiring electrical power are the mechanical Yarways, located in the reactor building outside of the primary containment. These would offer direct indication of the reactor vessel water level over a range from ~1/2 ft above the top of the core to near the top of the steam separators, ~18 ft (5.5 m) higher. While this would be very beneficial in allowing the accident management staff to monitor the approach of the water level to the top of the core, it should be recognized (as pointed out by the EPGs) that the abnormally high drywell temperature associated with loss of the drywell coolers would cause these instruments to read erroneously high. Furthermore, as the actual water level fell below the lower end of the indicating range, the mechanical Yarway instruments would continue to indicate a false on-scale water level of ~1 ft above the top of the core.

There would be no indication of reactor vessel pressure after loss of the unit battery and the associated loss of capability for remote control of the vessel relief valves. However, it can be expected that the reactor vessel would repressurize and that pressure would subsequently be maintained in the range of 1105 to 1055 psig (7.72 to 7.38 MPa) by repeated automatic actuation of the lowest-set safety/relief valve (SRV) for as long as the reactor vessel remained intact.

It should be noted that the emergency lighting for the control room is supplied from the unit 250-V dc system; after failure of this system, hand-held portable lighting for the control room would be necessary. The door security system is supplied by plant-preferred power and would remain operable as long as the plant 250-V dc system is functional. The area radiation monitors located throughout the plant are powered from the instrumentation and control buses and would not be operational from the inception of a station blackout.

7.2 Other Important Accident Sequences

For accident sequences such as short-term station blackout [with mechanical failure of high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) as an initiating event], anticipated transient without scram, loss-of-coolant accident, or loss of decay heat removal, electrical power (dc and perhaps ac) is maintained after loss of reactor vessel injection capability. Therefore, the availability of information concerning plant status is much greater for these sequences than for the sequences addressed in Sect. 7.1. The following discussion, based upon the Browns Ferry Plant, pertains to the more limiting case for which only dc power obtained directly from the installed batteries and the ac power indirectly obtained from these battery systems are available. The sources of ac power during station blackout include the feedwater inverter and the unit-preferred and plant-preferred systems for which single-phase 120-V ac power is produced under emergency conditions by generators driven by battery-powered dc motors. Emergency control room lighting would be available.

Two channels of control room instruments would provide indication of the reactor vessel water level over a range between 528 and 588 in. (13.411 and 14.935 m) above the vessel zero (the bottom of the vessel). The narrow range of this indication brackets the normal operating level of 561 in. (14.249 m) and covers the upper portion of the steam separators over a distance of 14 to 19 ft (4.267 to 5.791 m) above the top of the active fuel. Mechanical Yarway indication available outside of the control room covers an

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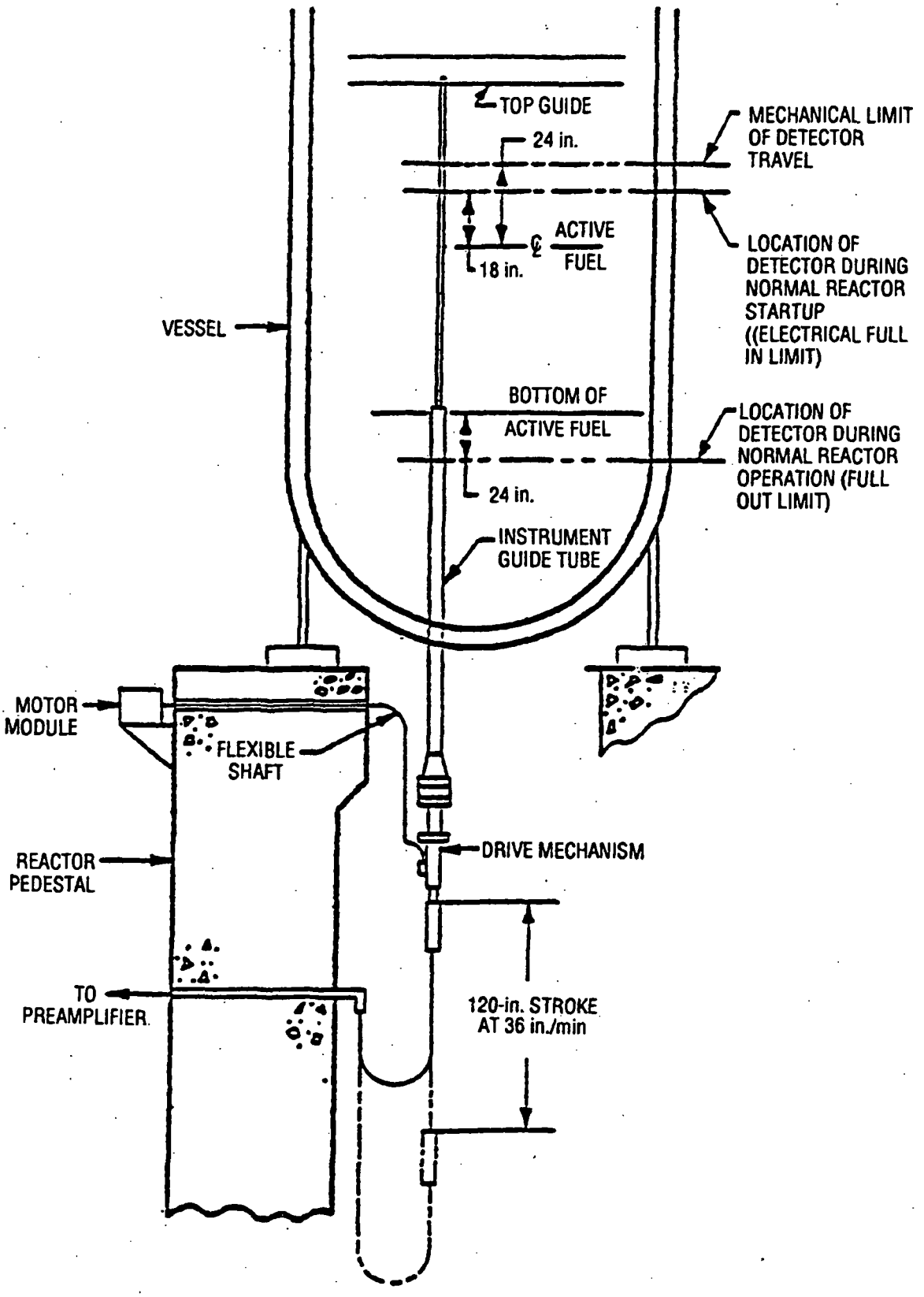

Figure 7.1 Source-range detector drive unit and locations of detector for startup and during power operation (from Browns Ferry Nuclear Plant Hot License Training Program)

additional range that extends from the low point of the control room indication [528 in. (13.411 m) above vessel zero] down to a point 373 in. (9.474 m) above vessel zero, which is 7 in. (0.178 m) above the top of the active fuel.

The level instrumentation derives the reactor vessel water level by comparing the head of water within the downcomer region of the vessel to the head of water within a reference leg installed in the drywell. With the loss of the drywell coolers attendant to station blackout, the drywell ambient temperature would increase significantly, heating the reference leg water to above-normal temperatures. This would reduce the density of the water in the reference leg, causing an error in the indicated water level, which would be too high. As an example, the drywell ambient temperature would increase from its normal range of 135 to 150°F (330.4 to 338.7 K) to ~280°F (410.9 K) during station blackout, so that the level indicated on the narrow range scale would be ~8 in. (0.20 m) too high. This is not a significant error when the actual water level is many feet above the top of the core. However, the nonconservative error due to high drywell temperature can have serious consequences when the apparent level is near the bottom of the Yarway indicating range. As mentioned in Sect. 7.1, the Yarway instrument can provide a low, but on-scale indication of water level for any actual water level below the lower instrument tap, if the drywell temperature is sufficiently high.

Reactor vessel pressure would be indicated by two instrument channels, extending over a range from 0 to 1200 psig (8.38 MPa).

With respect to measurement of neutron flux, the power range instruments would fail upon loss of the direct sources of ac power (i.e., the normal off-site power sources and the on-site diesel generators). The source- and intermediate-range monitors would remain operational, but the detectors for these systems are withdrawn from the reactor core into the reactor vessel lower plenum during power operation and could not be reinserted unless a direct source of ac power were available. (It is worth noting that insertion of the source-range detectors, if possible, would provide information concerning the uncovering of the lower portion of the core, as mentioned in Sect. 7.1).

The control rod position indication system would remain operational so that the operator could verify that the control rods had inserted with the scram.

With respect to reactor vessel SRV actuation, the operator has no indication of automatic actuation of a specific relief valve other than the recorded tailpipe temperatures available on charts behind the control room panels or the relief valve acoustic monitors, all of which would be inoperable without the direct sources of ac power. Nevertheless, remote-manual actuation of a relief valve is accomplished by energizing its dc solenoid operator; lights on the control panel for each valve indicate whether these solenoids are energized, and this capability would be maintained.

As discussed in Chap. 5, the steam turbine-driven RCIC and HPCI systems provide an important defense against the loss of ac power. All components normally required for initiating operation of the RCIC system are completely independent of ac power, plant service air, and external cooling water systems, requiring only dc power from a unit battery to operate the valves, vacuum pump, and condensate pump. On loss of control air, the RCIC steam supply line drains to the main condensers will fail closed; this is their normal position when the RCIC system is in operation. The drain functions of these valves is transferred to overseat drain ports in the turbine stop valves. Thus, the RCIC system can be operated and monitored without the availability of ac power.

Similarly, all components required for operation of the HPCI system are completely independent of ac power, control air systems, or external cooling water systems, requiring only dc power from the unit battery. On loss of control air, the HPCI steam line drains to the main condensers would fail closed (their normal position when the HPCI system is in operation). Furthermore, the condensate storage tank level indication would remain operational without any direct source of ac power so that the accident management staff could determine the amount of water remaining available for reactor vessel injection via the turbine-driven systems.

This discussion with respect to the information expected to be available to the plant accident management staff under accident conditions with no direct source of ac power is based upon a review of the specific instrument and control systems at the Browns Ferry Nuclear Plant. These findings are expected to be typical for all BWR facilities, but the details of instrumentation and control are highly plant-specific. [Additional information is available in Ref. 32.] Therefore, the IPE process[29,30] should include careful consideration of the availability of plant instruments under accident conditions. Accident management strategies cannot be effective without access of the management staff to the necessary information concerning plant status.

8 Candidate Strategies for Late Mitigation of In-Vessel Events

S. A. Hodge

This chapter provides a qualitative assessment of four candidate accident management strategies for control of in-vessel events during the late phase (after core melting has occurred) of postulated boiling water reactor (BWR) severe accidents. Each candidate strategy is required to be capable of implementation using the existing equipment and water resources of the BWR facilities. Strategies meeting this requirement were identified by a review of existing documented information relevant to BWR accident sequence analyses and accident management. Selection for further qualitative assessment was on the basis of potential for enhancement or extension of the existing BWR Owner's Group Emergency Procedure Guidelines (EPGs) for severe accident applications. Each of the selected candidates is assessed separately in the following sections according to feasibility and effectiveness; the potential of each strategy for the introduction of associated adverse effects is also addressed.

8.1 Keep the Reactor Vessel Depressurized

Reactor vessel depressurization is normally accomplished rather simply by manually induced actuation of the vessel safety/relief valves (SRVs) or by operation of the reactor core isolation cooling (RCIC) system turbine or, for plants so equipped, the isolation condenser or high-pressure coolant injection (HPCI) system turbine. Each of these methods relies to some extent, however, upon the availability of dc power or control air, which may not be available under accident conditions.

The BWR-2 units (Oyster Creek and Nine Mile Point 1) and three of the BWR-3 units (Millstone plus Dresden 2 and 3) incorporate isolation condensers as decay heat removal systems.* These systems employ natural circulation of steam from the reactor vessel to the elevated condenser, where the steam is condensed by heat transfer to water within the shell and the condensate is returned to the vessel. The shell water inventory (vented to the atmosphere) is sufficient to remove decay heat for at least 30 min without the addition of makeup water.

The isolation condenser system has the least dependence upon outside support, being completely independent of station normal or emergency on-site ac power and requiring dc power only for valve operation when the system is initially placed into operation. Makeup water for the shell side of the condenser can be provided from the plant fire protection system by means of an independent diesel-driven fire pump.

All BWR units except Oyster Creek, Nine Mile Point 1, and Millstone incorporate either the RCIC or HPCI steam turbine-driven reactor vessel injection system; the later BWR-3 plants and all BWR-4 plants have both. These systems can be used for reactor vessel pressure control when run continuously in the recirculation mode, pumping water from the condensate storage tank back to the condensate storage tank and periodically diverting a small portion of the flow into the reactor vessel as necessary to maintain the desired water level. The steam taken from the reactor vessel by the turbine is passed to the pressure suppression pool as turbine exhaust, which provides a slower rate of pool temperature increase than if the vessel pressure control were obtained by direct passage of steam from the vessel to the pool via the SRVs. Plants having both HPCI and RCIC systems can employ the HPCI turbine exclusively for pressure control while the RCIC system is used to maintain the reactor vessel water level. The HPCI turbine is larger than the RCIC turbine and, therefore, is more effective for pressure control. [Typically, HPCI takes 48 lb/s (21.8 kg/s) of steam when pumping into the pressurized reactor vessel at full capacity [5000 gal/min (0.315 m^3/s)] whereas RCIC takes 9 lb/s (4.1 kg/s) of steam at 600 gal/min (0.038 m^3/s).]

The HPCI and RCIC systems require dc power for valve and turbine governor control, but have no requirement for control air.

The most direct means of reactor vessel pressure control is by use of the SRVs, which require no outside energy source for operation in the automatic mode (as a safety valve) but do require both control air and dc power when used as a remotely operated relief valve. This dependence upon the availability of control air and dc power pertains both to remote-manual opening of the valves by the control room operators and to the valve-opening logic of the automatic depressurization system (ADS).

The purpose of the ADS is to rapidly depressurize the reactor vessel so that the low-pressure emergency core cooling

*The Oyster Creek unit has two isolation condenser loops; the other units each have one loop.

systems (ECCSs) can inject water to mitigate the consequences of a small or intermediate loss-of-coolant accident should the high-pressure injection systems prove inadequate. The ADS consists of redundant signal logics arranged in two channels that control separate solenoid-operated air pilot valves on each SRV assigned to the ADS function. The number of ADS-associated SRVs is plant-specific; these valves open automatically if required to provide reactor vessel depressurization for events involving small breaks. The ADS is initiated by coincidence of low reactor vessel water level and high drywell pressure, provided that at least one of the low-pressure pumps is operating.

All of the SRVs are located between the reactor vessel and the inboard main steam isolation valves (MSIVs) on a horizontal run of the main steam lines within the drywell. The discharge from each valve is piped to the pressure suppression pool, with the line terminating below the pool water level to permit the steam to condense in the pool. Vacuum breakers are installed on the SRV tailpipes within the drywell to relieve the vacuum created by condensation following actuation of a valve. The number of SRVs varies from plant to plant (i.e., 11 at Limerick; 24 at Nine Mile Point 2), as do the rated relief valve flows.

Some operating BWRs are equipped with three-stage Target Rock valves, which have exhibited a greater tendency to stick open in the past than have other types of valves. Many BWR-4 utilities, however, have replaced the original three-stage valves with the newer two-stage Target Rock valves (Fig. 8.1). Some operating BWRs are equipped with Dresser Electromatic relief valves. BWR-5 and BWR-6 plants are equipped with Crosby and Dikkers dual function SRVs (Fig. 8.2).

The differences in SRV operation in the automatic and remote-manual or ADS modes can be demonstrated with reference to the two-stage Target Rock design shown in Fig. 8.1. During normal reactor operation, a small piston orifice serves to equalize the steam pressure above and below the main valve piston, and the main valve disk remains seated. The reactor vessel pressure (valve inlet

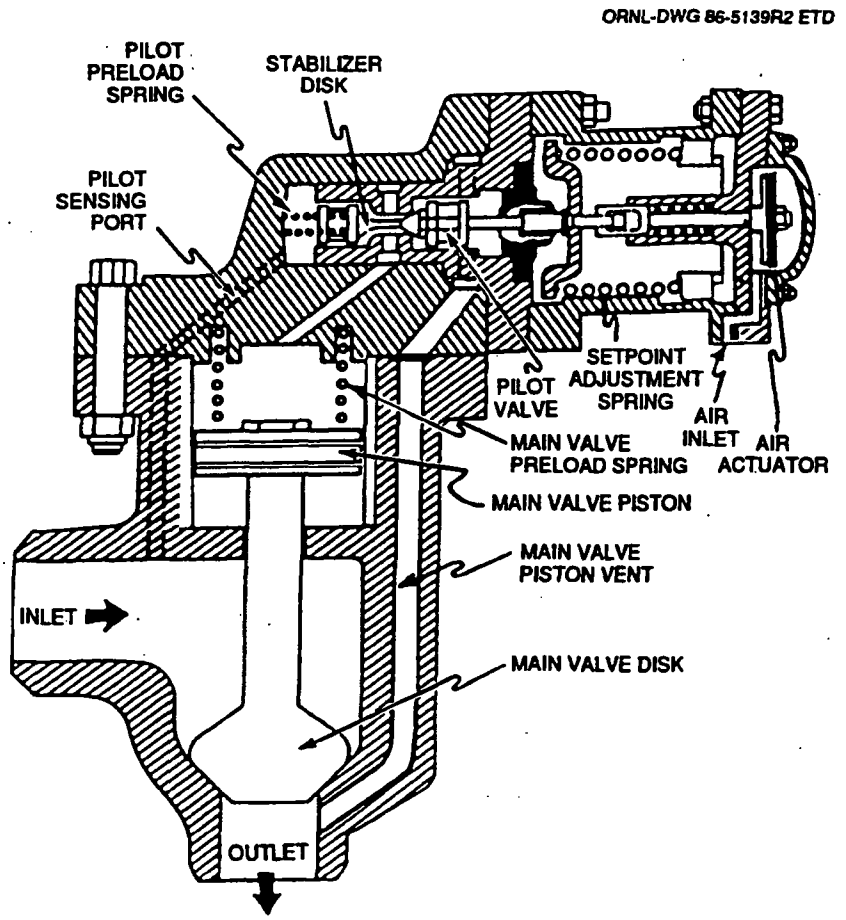

Figure 8.1 Control air and reactor vessel pressure act in concert to move the pilot valve in the two-stage Target Rock SRV design

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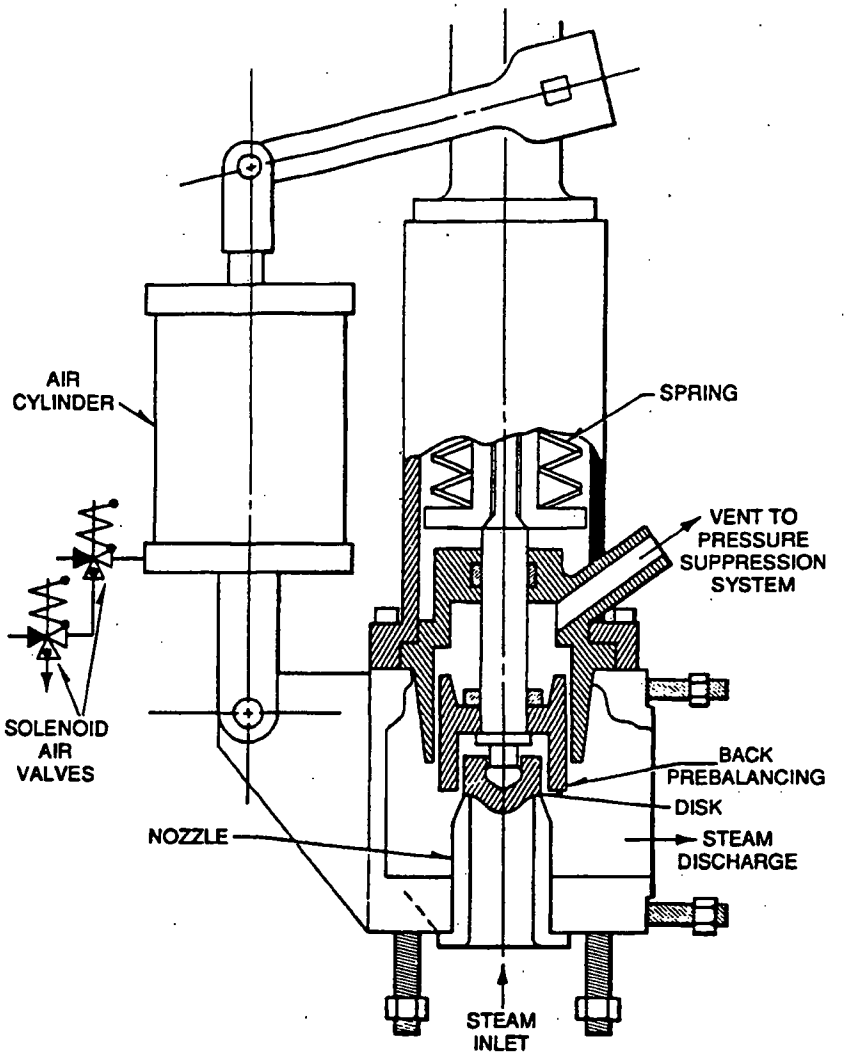

Figure 8.2 Schematic drawing of dual-function spring-loaded direct-acting SRV

pressure) is ported via the pilot sensing port to tend to push the pilot valve to the right. When the reactor vessel pressure exceeds the setpoint established by the setpoint adjustment spring, the pilot valve is moved to the right, the stabilizer disk is seated, and the volume above the main valve piston is vented to the valve outlet via the main valve piston vent. The sudden pressure differential causes the main valve piston to lift, opening the valve.

For the remote-manual or ADS modes, the SRV is opened by control air, which is admitted via a dc solenoid-operated valve (not shown) to the air inlet at the right of the setpoint adjustment spring. The control air moves the air actuator to the right (against drywell pressure), which compresses the setpoint adjustment spring and pulls the pilot valve open, seating the stabilizer disk and venting the space above the main valve piston. Because the control air pressure and the reactor vessel pressure work in tandem to move the pilot valve to the right, the amount of control air pressure required to open the SRV will depend upon the reactor vessel-to-drywell pressure differential.* Also, because the control air acts to move the air actuator against drywell pressure, the required control air pressure will increase with drywell pressure.

The spring-loaded direct-acting SRV shown in Fig. 8.2 is opened in the spring mode of operation by direct action of the reactor vessel pressure against the disk, which will pop open when the valve inlet pressure exceeds the setpoint

*It should be noted that the three-stage Target Rock valves behave differently with respect to the effect of the reactor vessel-to-drywell pressure differential. A good description of the operation of this older valve design is provided in TVA's *Browns Ferry Final Safety Analysis Report*, Vol. 2, Sect. 4.4.5.

value. In the power-actuated mode, a pneumatic piston within the air cylinder moves a mechanical linkage to compress the spring and open the valve. As in the case of the two-stage Target Rock valve, the control air is provided via dc solenoid-operated valves, and the air pressure required for valve opening decreases with reactor vessel pressure and increases with drywell pressure.

All SRVs associated with the ADS are fitted with pneumatic accumulators (located within the drywell) to ensure that these valves can be opened and held open following failure of the normal supply system.* In some plants, the other SRVs are also fitted with (smaller) accumulators to ensure some degree of operability following failure of the pneumatic supply system. For severe accident considerations, it is important to recall that remote operation of the SRVs is possible only as long as the pneumatic supply pressure exceeds the containment pressure by some minimum amount.

In considering the late phase of a severe accident where core melting has occurred, it must be concluded that because all available forms of reactor vessel injection have failed, then the dc power and control air necessary for reactor vessel pressure control by remote-manual SRV actuation (or HPCI/RCIC steam turbine operation) may not be available. Thus, there is a potential need for another (backup and last-resort) method to keep the reactor vessel depressurized after the normal methods can no longer be used for this purpose.

The motivation for keeping the reactor vessel depressurized under severe accident conditions is, first, that the potential for quenching of the debris relocating from the core region into the lower plenum is enhanced and, second, that relocation of molten debris into the relatively small BWR drywell would then be, should bottom head penetration failure occur, by gravity-induced flow and not by rapid vessel blowdown. Although the BWR Mark I and Mark II containment designs are inerted, direct containment heating is not precluded *per se* because fine particles of zirconium metal spewed under pressure into the drywell would readily react with the steam-rich atmosphere created by pressure suppression pool heating under accident conditions. Therefore, keeping the reactor vessel depressurized eliminates direct containment heating concerns and would greatly reduce the initial challenge to the primary containment.

Unfortunately, there seems to be no way to keep the reactor vessel depressurized in the event of failure of the SRVs and other normal methods of pressure control other than by venting the vessel into the turbine building or the reactor building. Venting into the turbine building would be preferred for this purpose because the reactor vessel injection systems are located in the reactor building and access would be required for attempts to restore these systems. A typical arrangement of these separate buildings is shown in Fig. 8.3. The deliberate release of some fission products beyond the confines of the primary containment is the clearly undesirable aspect of this maneuver; however, the release would be not directly to atmosphere but rather to the turbine building and would be solely for the purpose of preventing or delaying a much more serious rupture of the primary containment pressure boundary.

With respect to diagnostic concerns, it would obviously be desirable in consideration of this strategy for the plant accident management staff to know the reactor vessel pressure. This information would be available as long as dc power remains, but loss of dc power is one of the ways in which the capability to manually control the SRVs or otherwise depressurize the reactor vessel would be lost. Given proper training, the operator should recognize that when dc power or control air is lost, the SRVs will shut and the reactor vessel pressure will increase. Thus, proper training could obviate the need for special instrumentation in support of this strategy, although the operators would obviously be highly stressed should an accident sequence progress to the point where core melting had occurred. Here the candidate strategy involves venting of the reactor vessel to the turbine building, releasing considerable fission products, while the primary containment remains intact. It is a proposed trade-off of accepting a small, avoidable, penalty (venting release) in the short term for the purpose of avoiding a much greater penalty (uncontrolled rupture of containment) in the long term. This choice among undesirable alternatives offered under stressful conditions has the characteristics of a classic human factors problem.

The proposed last-resort venting of the BWR reactor vessel into the turbine building could be carried out by manually initiated opening of the motor-operated bypass valves around the MSIVs. A typical arrangement of main steam lines and drains is shown in Fig. 8.4. Main steam lines A, B, C, and D are cross-connected upstream of the inboard MSIVs 1-14, 1-26, 1-37, and 1-51. A 3-in. bypass line passes from the drywell to the reactor building through penetration X-8. Bypass valves 1-55 and 1-56 would be automatically closed by the primary containment isolation system under accident conditions, while drain valves 1-57 and 1-58 would be open. If the bypass valves were opened

*The normal supply is the drywell control air system, which provides control air (actually nitrogen) from outside the drywell.

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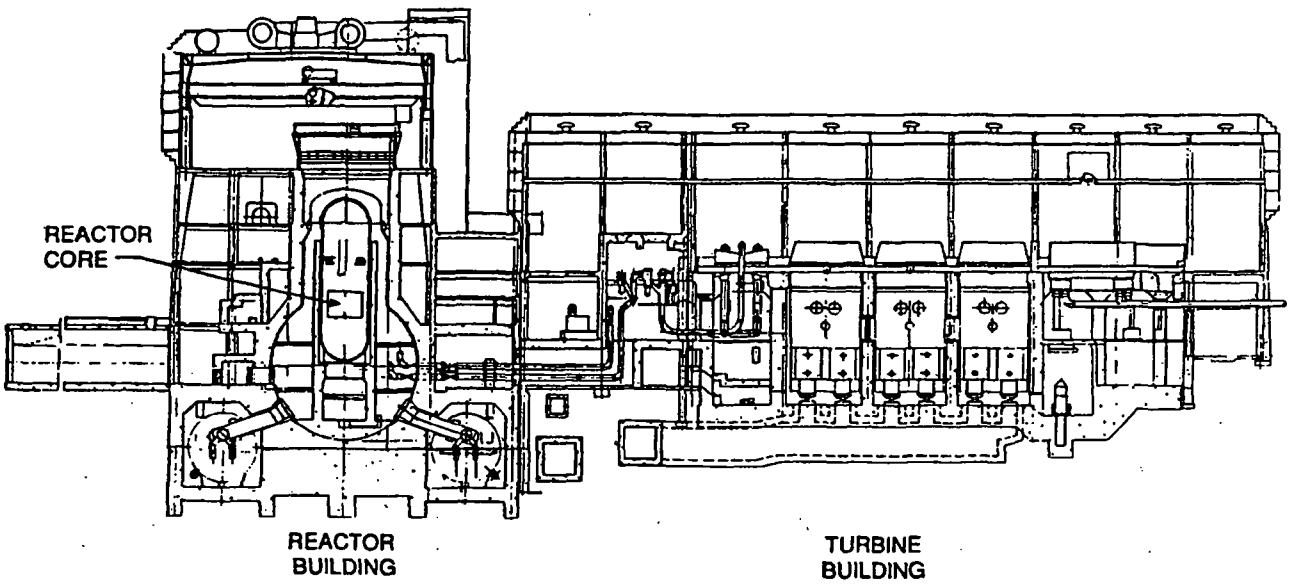

Figure 8.3 Typical arrangement of reactor building and turbine building for BWR plant of Mark I containment design (from Browns Ferry FSAR)

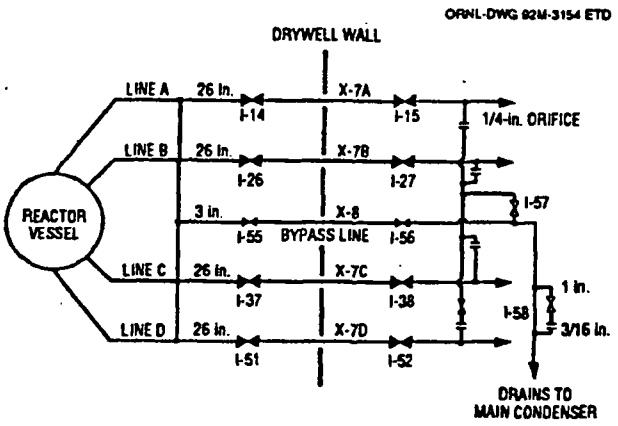

Figure 8.4 Arrangement of main steam lines and bypass line for Browns Ferry Nuclear Plant

during the late phase of a severe accident, pathways would be established for the flow of steam and hydrogen from the reactor vessel into the downstream portion of the main steam lines and through the drains into the main condenser.

Whether use of these small pathways would be effective in maintaining the reactor vessel depressurized would have to be established by additional analyses. Because the valves would not be opened until after reactor vessel bottom head dryout (as determined from the vessel thermocouples), the required flow would be small. If effectiveness can be shown, then provision of special means such as an alternate power supply to permit opening of the bypass valves under severe accident conditions should be relatively simple. Typically, the inboard bypass valve (1-55 on Fig. 8.4) is an ac motor-operated gate valve with power from the standby ac source (unit diesel generators), while the outboard bypass valve (1-56) is a dc motor-operated gate valve with power from the unit battery. A strategy involving opening of these valves in the late phase of a severe accident when the normal sources of opening power would be expected to be unavailable would not be successful without extensive preplanning and training of plant accident management staff.

The only other possible method of venting the reactor vessel to the main condenser during the late phase would be through the reactor water cleanup (RWCU) system. This system, illustrated in Fig. 8.5, maintains reactor water quality during normal reactor operation by removing fission products, corrosion products, and other soluble and insoluble impurities, provides a path for removal of reactor coolant from the reactor vessel in case of excess coolant inventory, and maintains circulation in the reactor vessel bottom head to minimize thermal stratification.

The RWCU consists of a pumping system that takes suction on both recirculation loop suction lines and the reactor vessel bottom head, pumps the water through heat exchange and ion exchange facilities, and pumps the water

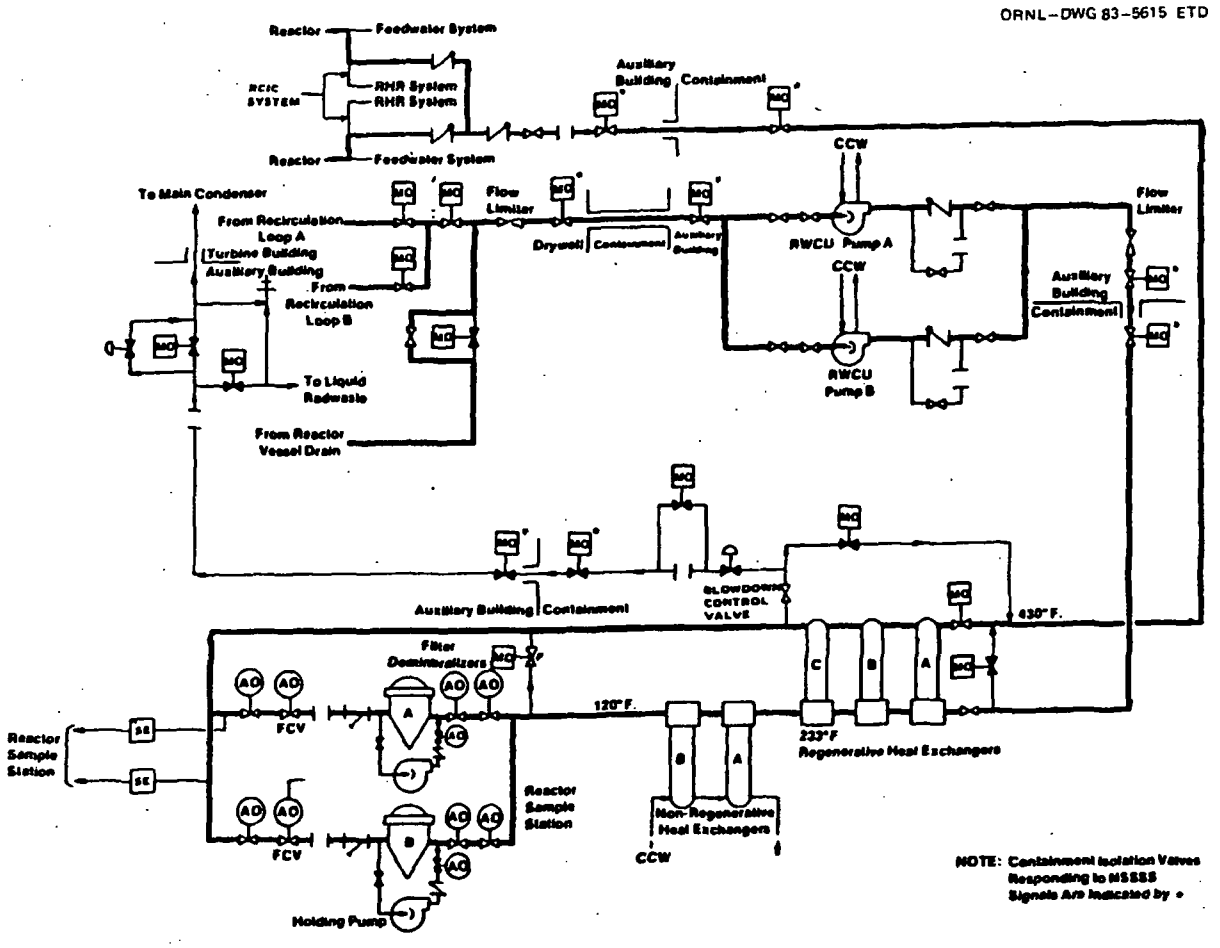

Figure 8.5 Typical arrangement of RWCU

back to the reactor vessel via the feedwater piping. The high-pressure flow passes through two 50% capacity pumps, three regenerative heat exchangers, and two non-regenerative heat exchangers. Depending on desired system operation, flow can be routed through two 50% capacity filter demineralizers. The RWCU has the capability to direct flow to the main condenser, the liquid radwaste system, or to the reactor vessel via the feedwater lines. Flow through the filter demineralizers and/or heat exchangers can be bypassed as desired depending on plant operating conditions.

The RWCU system is normally operated continuously during all phases of reactor operation, startup, shutdown, and refueling. However, under accident conditions the inboard ac motor-operated gate valve and outboard dc motor-operated gate valve are automatically shut and the pumps are stopped. It is thought that attempts to use this system for reactor vessel venting to the main condenser under late-phase accident conditions would involve more complicated maneuvering and be more difficult than use of the MSIV bypass lines. Other means of reactor vessel venting into secondary containment (the reactor building) would interfere with mechanical or electrical operators attempting local measures to restore reactor vessel injection under extremely abnormal situations.

Although it is without question highly desirable that the reactor vessel be depressurized should a severe accident proceed to the point of reactor vessel bottom head penetration failure, it seems preferable to ensure that the reactor vessel can be depressurized by improving the reliability of the SRVs rather than by providing an alternative venting strategy with the attendant undesirable fission product release beyond the confines of the primary containment. Because multiple SRVs are installed and operation of any one valve is sufficient for depressurization under severe accident conditions, improved reliability of SRV operation can be attained simply by ensuring that dc power and control air will be available (to at least one valve) when required. Therefore, consideration of the reliability of the dc power and control air supply to the SRVs under accident conditions should be an important part of each individual plant examination for severe accident

vulnerabilities.[29,30] In other words, it should be recognized that the recent Nuclear Regulatory Commission (NRC)-sponsored assessment of severe accident risks (NUREG-1150) and other probabilistic risk assessment (PRA) studies have consistently identified sequences involving loss of the unit battery and drywell control air systems as among the dominant accident sequences (internal events) leading to core melt for BWRs.

8.2 Restore Injection in a Controlled Manner

Late accident mitigation strategies are intended for use in the extremely unlikely event that core melting is in progress. This means that the core is at least partially uncovered, which in turn signifies that reactor vessel injection capability has been lost. Considering the plethora of reactor vessel water injection systems at a BWR facility, the most probable cause is that electrical power is not available. Brief descriptions of the electric-motor-driven injection systems are provided in the following paragraphs.

BWR-5 and -6 plants are equipped with a high-pressure core spray (HPCS) system rather than a turbine-driven HPCI system. The purpose of the HPCS is to maintain reactor vessel inventory after small breaks that do not depressurize the vessel, to provide spray cooling for line breaks that result in the reactor core becoming uncovered, and to backup the RCIC system during situations in which the reactor vessel is isolated.

The HPCS, shown in Fig. 8.6, is a single-loop system comprised of a suction shutoff valve, one motor-driven pump, a discharge check valve, a motor-operated injection valve, a minimum flow valve, a full flow test valve to the suppression pool, two high-pressure flow test valves to the condensate storage tank, a HPCS spray sparger (inside the vessel above the core shroud), and associated piping and instrumentation. The HPCS pump takes suction from the condensate storage tank and delivers the flow into a sparger mounted within the core shroud. Spray nozzles mounted on the spargers are directed at the fuel bundles. The suppression pool is an alternate source of water; the HPCS logic switches the pump suction from the condensate storage tank to the suppression pool upon either pool high level or low condensate storage tank level.

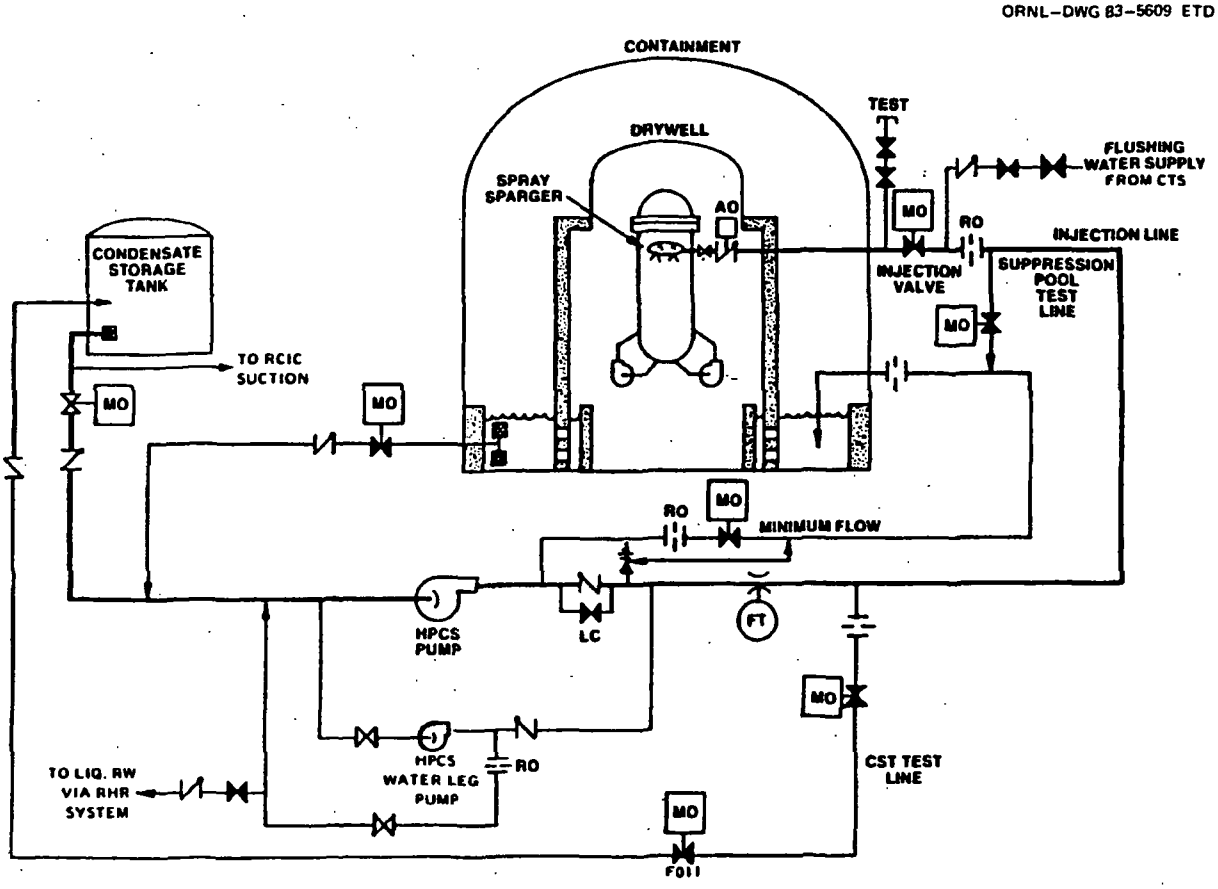

Figure 8.6 Arrangement of HPCS system installed at BWR-5 and BWR-6 facilities

Candidate

The HPCS pump is a vertical, centrifugal, motor-driven pump capable of delivering at least 1550 gal/min (0.098 m^3/s) at 1147 psi (7.908 MPa) reactor pressure, 6110 gal/min (0.385 m^3/s) at 200-psi (1.37-MPa) reactor pressure, and a maximum of 7800 gal/min (0.492 m^3/s) at runout flow conditions. The HPCS can be powered from either the normal or standby ac power systems. Major HPCS system components are located in the auxiliary building.

All BWR facilities employ the low-pressure coolant injection (LPCI) mode of the residual heat removal (RHR) system as the dominant operating mode and normal valve lineup configuration; the RHR system will automatically align to the LPCI mode whenever ECCS initiation signals such as low reactor vessel water level or high drywell pressure plus low reactor vessel pressure are sensed. During operation in the LPCI mode, the RHR motor-driven centrifugal pumps take suction from the suppression pool and discharge into the reactor vessel via the recirculation loops (BWR-4, Fig. 8.7) or directly into the region between the outermost fuel assemblies and the inside of the core shroud (BWR-5 and BWR-6, Fig. 8.8). BWR-4 plants have the capability for the RHR pumps to draw suction from the condensate storage tank, and in some plants (BWR 6), the RHR pumps can be realigned to take suction on the fuel pool cooling and cleanup system. BWR-5 and BWR-6

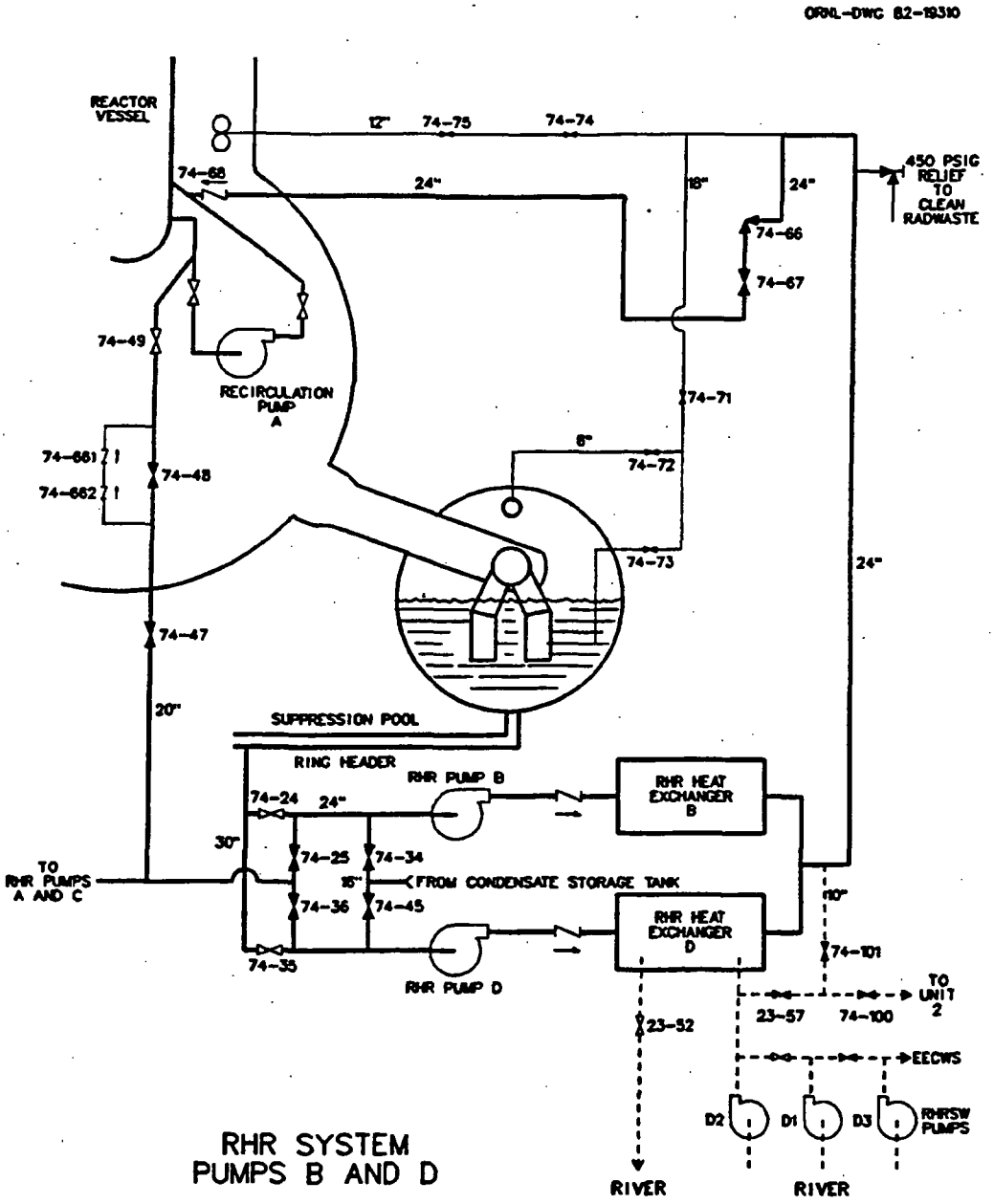

Figure 8.7 Arrangement of one loop of RHR system at Browns Ferry Nuclear Plant

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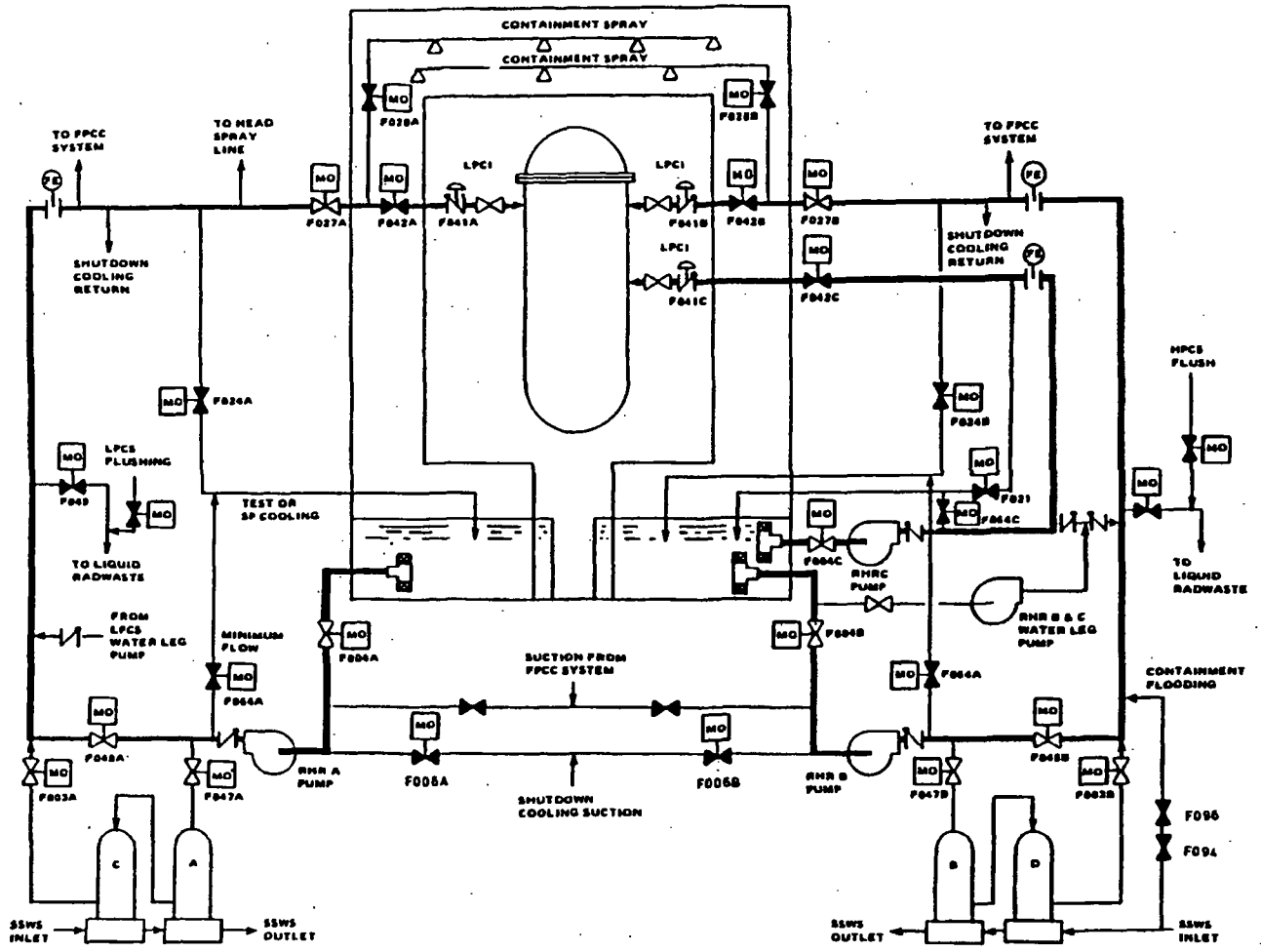

Figure 8.8 Typical arrangement of the RHR system for BWR-6 facility

plants have three-pump/three-loop LPCI systems that bypass the RHR heat exchangers; BWR-4 plants have four-pump/two-loop systems.

Operation in the LPCI mode is intended to restore and maintain the reactor vessel coolant inventory after a loss-of-coolant accident in which the reactor is depressurized or after ADS actuation. The LPCI mode provides a low-head, high-flow injection [290 psid (1.999 MPa) shutoff head with 20,000 gal/min (1.262 m^3/s)—BWR-6 or 40,000 gal/min (2.524 m^3/s)—BWR-4 rated flow at 20 psid (0.138 MPa) typical]. The LPCI flow is sufficient to completely fill an intact reactor vessel in <5 min.

All BWR facilities also employ a low-pressure core spray (LPCS) system to protect against overheating of the fuel in the event that the core is uncovered by a loss of primary coolant following a break or rupture of the primary system. This cooling effect is accomplished by directing spray jets of cooling water directly onto the fuel assemblies from spray nozzles mounted in sparger rings located within the shroud just above the reactor core.

Each loop of the core spray system consists of one or more motor-driven centrifugal pumps; a spray sparger in the reactor vessel above the core; and such piping, valves, and control logic as are necessary to convey water from the pressure suppression pool to the reactor vessel. BWR-4 facilities employ two 50% capacity core spray loops, each with its own sparger, and have the capability to take LPCS suction from the condensate storage tank as an alternative to the pressure suppression pool. A typical arrangement of one core spray loop is shown in Fig. 8.9. BWR-5 and BWR-6 plants utilize a single-pump, single-loop system.

Typical total rated LPCS injection rates (all pumps operating) are 6000 gal/min (0.379 m^3/s)—BWR-6 or 12,500 gal/min (0.789 m^3/s)—BWR-4 at suppression pool-to-

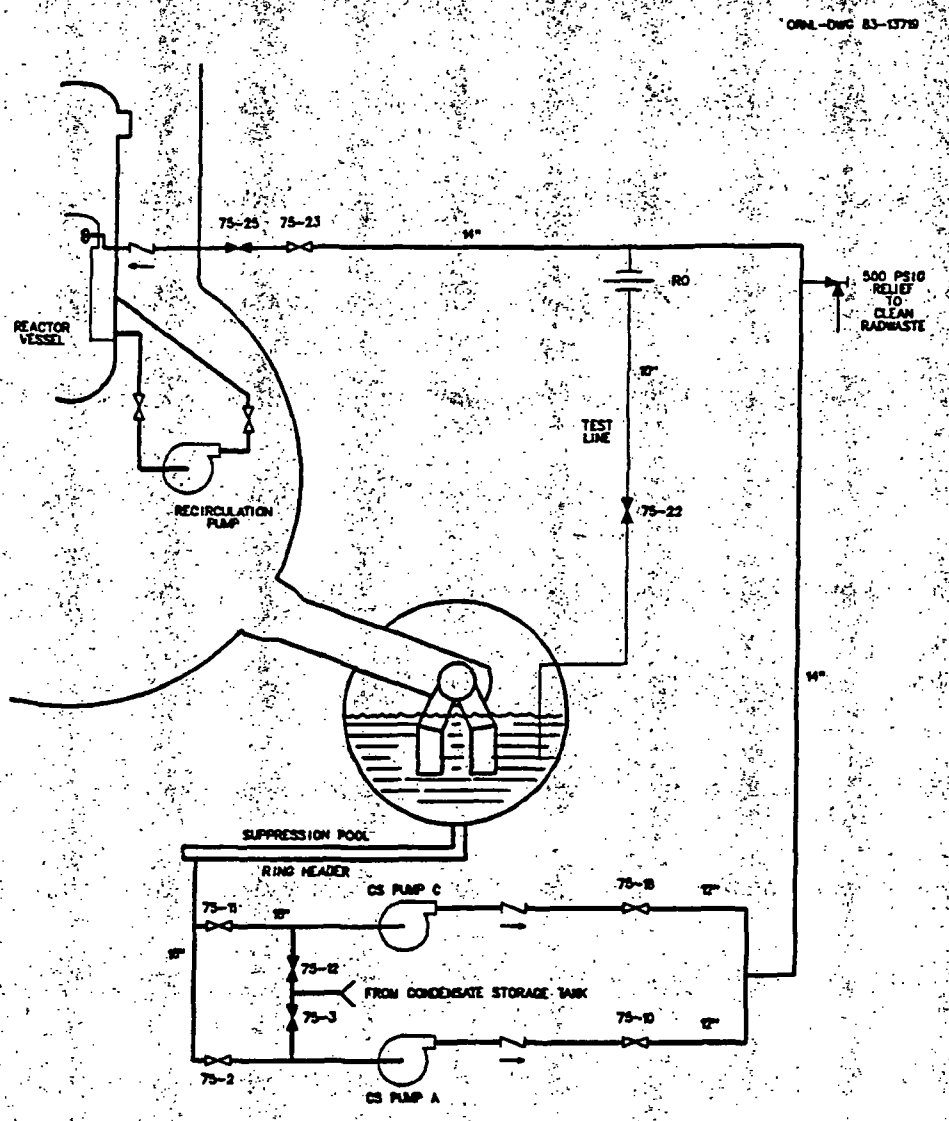

Figure 8.9 Arrangement of one loop of CS at Browns Ferry Nuclear Plant

reactor vessel pressure differentials of 150 psi (1.034 MPa). The shutoff pressure of the LPCS pumps is typically 300 psig (2.170 MPa).

If electrical power were restored while core melting was in progress, then the combined capacity of the RHR and core spray systems, which would be in the automatic injection mode, would be more than 50,000 gal/min (3.155 m^3/s). The amount of vessel injection necessary to remove decay heat, however, is only about 200 gal/min (0.013 m^3/s). [Where specific quantities are given as examples in the following discussion, the numbers are based upon a 1065-MW(e) BWR-4 facility such as Browns Ferry or Peach Bottom.] Figure 8.10 illustrates the disparity between injection capacity and required injection for a non-LOCA accident sequence. For the RHR and core spray systems, the indicated injection capability is for one pump, whereas four pumps are installed for each system. The required injection curve was calculated on the basis of constant reactor vessel pressure, scram from 100% power, and decay heat in accordance with the 1979 ANS standard with consideration of actinide decay. The required water flow is assumed to be injected at a temperature of 90°F (305.4 K) and removed from the vessel as dry saturated steam.

With electrical power restored, the operators should initiate reactor vessel injection quickly, but in a controlled manner. This strategy is related to the requirement for keeping the reactor vessel depressurized (Sect. 8.1), because the reactor vessel pressure must be below the shutoff head [~300 psid (2.068 MPa) between reactor vessel and pressure suppression pool] of the core spray or RHR system pumps. Given a choice of systems, the bottom-flooding RHR system should be used in preference to the core spray. The RHR system supplies water to the lower core via the vessel

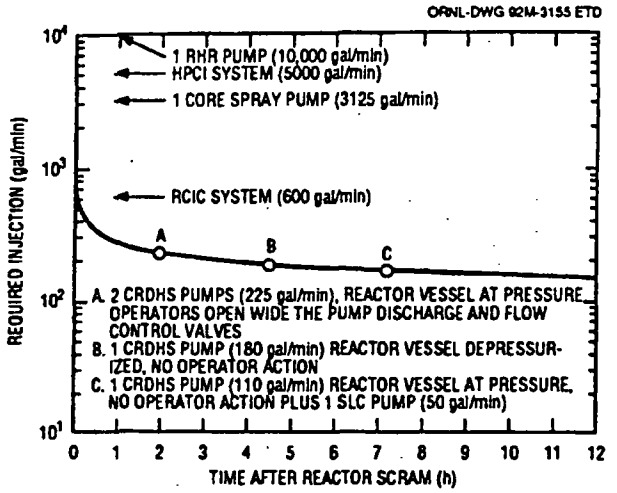

Figure 8.10 Typical BWR-4 injection system capabilities greatly exceed injected flow required to replace reactor vessel water inventory boiled away by decay heat

lower plenum, and there may be core debris within the lower plenum; the RHR system incorporates heat exchangers whereas core spray does not; and injection by core spray is subject to countercurrent flow limiting conditions.[33]

With respect to diagnostic and human factors concerns, the more knowledge that the operators have regarding the water level within the reactor vessel and the state of the core, the less the likelihood of losing control of the situation. For example, if the control blades have melted and relocated from the center of the core while the fuel rods in this region remain standing, then criticality may occur upon reflooding. Without proper training, the operators may be surprised by the conversion, for example, of a station blackout accident sequence into an unanticipated criticality. Even without criticality, steam generation during reflooding would cause rapid vessel pressurization, and the operators should be trained to expect this.

The general problem is that the plant ECCS systems are designed to deal with large-break LOCA, but would be automatically actuated for other accident sequences as well. Because the RHR system by itself has a total (four pump) capacity of ~40,000 gal/min (2.524 m^3/s), restoration of injection for the much more likely severe accident sequences not involving LOCA should be undertaken by operating one pump and throttling the pump discharge. Because the free volume of the vessel lower plenum is ~35,000 gal (133.089 m^3), rapid initial injection to raise the water level to the lower core region would be warranted if vessel bottom head dryout had occurred. The height within the reactor vessel at which an increasing water level would come on-scale with the available level instruments is plant-specific. The injection rate should be slowed as the water level rises above the core plate, if the time that this occurs can be determined. About 23,500 gal (89.198 m^3) of additional injection would be required to raise the level from the bottom to the top of the core region.

This strategy for operator control of ECCS pumps upon restoration of power is feasible using existing equipment; its effectiveness is related to the proposed strategy for injection of boron if control blade damage has occurred (Sect. 8.3), because the potential for criticality as water enters the core region is the major reason for limiting the injection rate during this period.

8.3 Inject Boron if Control Blade Damage Has Occurred

The goal of this candidate strategy is to prevent criticality upon restoration of reactor vessel injection or to return power to decay heat levels as quickly as possible if criticality cannot be prevented. The normal means of adding boron to the reactor vessel is by injection with the standby liquid control system (SLCS). Although this system is designed to inject sufficient neutron-absorbing sodium pentaborate solution into the reactor vessel to shut down the reactor from full power (independent of any control rod motion) and to maintain the reactor subcritical during cooldown to ambient conditions, the SLCS is not intended to provide a backup for the rapid shutdown normally achieved by scram. As indicated in Fig. 8.11, the basic system comprises a heated storage tank, two 100% capacity positive displacement pumps, and, as the only barrier to injection to the reactor vessel, two explosive squib valves. In most of the current BWR facilities, the sodium pentaborate solution enters the reactor vessel via a single vertical sparger located at one side of the lower plenum just below the core plate as indicated in Figs. 8.12 and 8.13. However, in an effort to improve the mixing and diffusion of the injected solution (which has a specific gravity of ~1.3) throughout the core region, some BWR facilities have been modified to provide a third positive displacement pump and to permit the injected solution to enter the reactor vessel via the core spray line and sparger.

For the purpose of reducing the time required for reactor shutdown for the ATWS accident sequence, the NRC has recently required that the SLCS injection be at a rate *equivalent* to 86 gal/min (0.0054 m^3/s) of 13 wt % sodium pentaborate solution, the boron being in its natural state with 19.8 at. % of the boron-10 isotope. Because the original SLCS standard design provided for single-pump operation at a rate of 43 gal/min (0.0027 m^3/s), the requirement

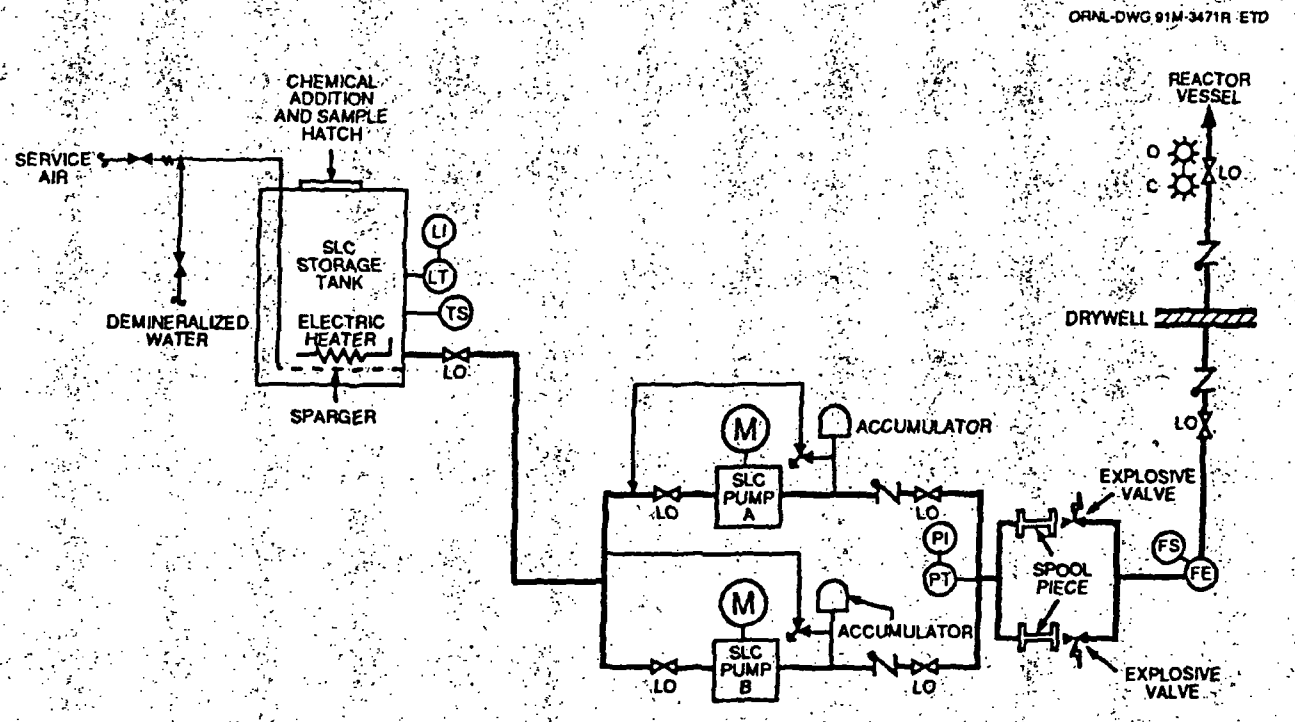

Figure 8.11 Abbreviated schematic of the typical BWR SLCS

for the increased equivalent control capacity can be satisfied by simultaneous operation of both of the installed pumps, by increasing the concentration of sodium pentaborate solution, or by enriching the boron within the sodium pentaborate solution in the isotope boron-10. Different BWR facilities have taken different approaches.

Under severe accident conditions, injection of a boron solution may be required for a situation very different than that normally associated with ATWS. If significant control blade melting and relocation from the core region were to occur during a period of temporary core uncovering, then criticality should be expected if reactor vessel injection capability is restored and the core is then covered with cold unborated water.[34] Obviously, a neutron poison should be introduced into the reactor vessel for reactivity control under these circumstances, but the question arises as to how best to do this. If the SLCS is used to inject sodium pentaborate solution at a relatively slow rate while the core is rapidly covered using the high-capacity low-pressure injection systems, then criticality would occur and the core would remain critical until sufficient boron for shutdown reached the core region. It would be preferable, if control blade relocation has occurred, to inject only the boron solution provided that this can be done at a rate sufficient to provide core cooling and terminate core damage.

The major diagnostic concern with respect to this strategy is that the operators would have no direct means of knowing whether significant control blade relocation has occurred. Therefore, either a boron solution would have to be injected after any nontrivial period of core uncovering or reliance would have to be made on precalculated values of time to control blade melting for the various accident situations to determine when injection of a neutron poison was required. (At the very least, operators should be trained to recognize that criticality might occur upon reflooding.)

With respect to human factors concerns, there is a strong potential for operator surprise and confusion should, for example, a station blackout accident sequence be suddenly converted into an uncontrolled criticality upon restoration of reactor vessel injection capability. Furthermore, should the SLCS be actuated only after the core had been covered (and become critical), the introduction of the sodium pentaborate solution into the core region would be inefficient. As mentioned previously, the SLCS flow is released into the vessel lower plenum in most BWRs, and a rapid upward flow into the core region is required to avoid stratification of the injected solution (specific gravity greater than one) within the lower plenum.

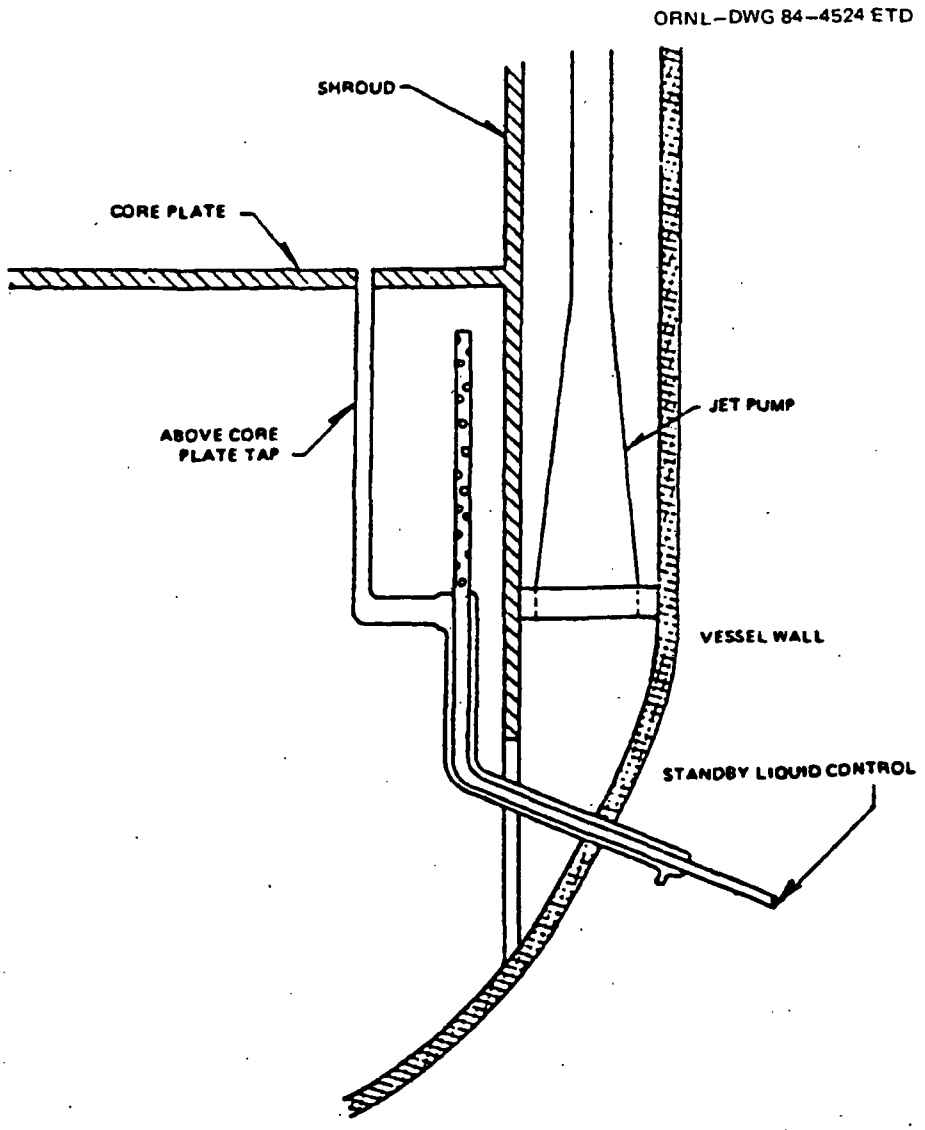

Figure 8.12 Location of SLCS injection sparger within BWR-4 reactor vessel

Therefore, to avoid the possibility of temporary criticality and to preclude thermal stratification of the injected solution, it would be desirable to inject effective quantities of boron together with the ECCS system flow being used to recover the core. One way to do this would be to prepare the boron solution directly within the condensate storage tank and to then take suction on the condensate storage tank with the low-pressure injection system pump to be used for controlled refilling of the reactor vessel. Here, in contrast to the concept employed when sodium pentaborate is injected from the SLC tank, the concept would be to prepare the desired concentration of boron within the total volume of water to be used to cover the core; it would not be intended that the concentration of the injected solution would be diluted within the vessel. A recent study[34] has indicated that a concentration of 700 to 1000 ppm of the boron-10 isotope would be required to ensure that criticality would not occur upon flooding a damaged core.

Although this is thought to be feasible and could be accomplished using only the existing plant equipment in most of the BWR facilities, it is not a simple matter to successfully invoke this strategy.

First, unless the condensate storage tank were drained down to approximately the water volume needed to refill the reactor vessel to above the top of the core, an untenable amount of borax and boric acid would have to be added to the tank. For example, a condensate storage tank at Peach Bottom has a capacity of 200,000 gal (757 m^3) whereas a Browns Ferry tank has a 375,000-gal (1420-m^3) capacity. Even with draining, the amount of borax and boric acid crystals to be manually added to the condensate storage tank under accident conditions would be large. Additional reduction could be achieved by increasing the effective enrichment of the boron. For example, enrichment to 60

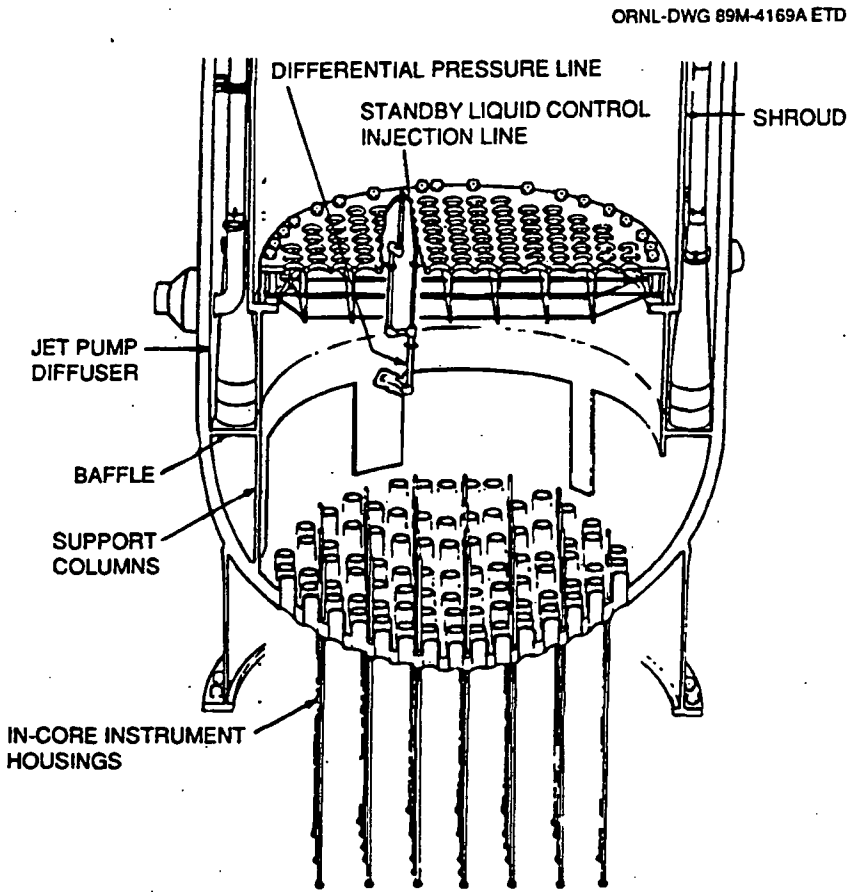

Figure 8.13 Differential pressure and standby liquid control injection line entering reactor vessel as two concentric pipes, which separate in lower plenum

at. % of the boron-10 isotope (instead of the naturally occurring 19.8 at. %) would reduce the required amounts of borax and boric acid by a factor of 3. The addition of these constituents to the condensate storage tank would have to be initiated before the restoration of electrical power because delay in injection of water to the reactor vessel while waiting for the mixture to be prepared would be intolerable after injection capability had been restored.

As a second major practical consideration for this candidate strategy, the condensate storage tank is located outside of the reactor building at most BWRs, and all provisions for heating of the associated piping would be inoperable during station blackout. Sodium pentaborate will precipitate from solution at temperatures dependent upon the concentration; for example, a solution of 9% sodium pentaborate (by weight) has a saturation temperature of –40°F (277.6 K). However, the concentration of sodium pentaborate within the condensate storage tank water would be <2% if natural boron were used and <1% with enriched boron. More of a problem would be presented in getting the borax and boric acid being added to the tank to go into solution at low temperature without mechanical stirring (provided by an air sparger during mixture preparation in the SLC tank). Therefore, employment of this strategy may be limited to accident conditions involving favorable atmospheric temperature if it is based upon a sodium pentaborate solution.

This candidate strategy was selected for detailed assessment, which is provided in Chaps. 9 through 13. Although it is expected that implementation could be effected using only the existing plant equipment at most BWR facilities, it is clear that a great deal of preplanning and training would be required to make the contemplated use of the condensate storage tank a viable maneuver. In this connection, it should be recognized that, in general, the earlier plants have provisions for both the RHR pumps and the core spray pumps to take suction on the condensate storage tank; the intermediate plants have provision for only the core spray pumps to take suction, and at the later (BWR 5/6) plants, only the motor-driven HPCS pump can inject to the reactor vessel from the condensate storage tank. The alternative for plants where injection of sodium pentaborate solution from the condensate storage tank is not feasible or practical due to weather or other limitations might be

to arrange for substitution of the fuel storage pool or for a combined pump suction from the SLC tank and the pressure suppression pool with the respective flows taken in the proper ratio. However, this would require significant modifications to existing equipment and therefore is beyond the scope of the present study. A more practical alternative would be to employ a different chemical form for the boron solution as will be described in Chap. 11.

8.4 Containment Flooding to Maintain Core and Structural Debris In-Vessel

The BWR Owners' Group EPGs currently provide (Contingency #6) for primary containment flooding where all other means of reactor vessel injection have failed; the concept is intended for application in LOCA situations where the water within the drywell could enter the reactor vessel through the break. In this section, consideration is given to flooding of the primary containment and the presence of water surrounding the lower portion of the reactor vessel as a means to provide sufficient cooling of the bottom head for severe accident sequences not involving LOCA. The purpose would be to maintain the core and structural debris within the vessel.

The proposed application of drywell flooding is illustrated for the BWR Mark I containment design in Fig. 8.14. Typically, the only means for water addition to the containment is provided by low-pressure pumping systems. Therefore, the drywell would have to be vented during filling to avoid exceeding the shutoff head of the low-pressure pumps taking suction on a large outside source such as river or reservoir. On the other hand, the wetwell should not be vented so as to trap the air volume in the upper portion of the wetwell and thereby reduce the amount of water required. For a plant the size of Peach Bottom or Browns Ferry, implementation of this concept would require the injection of ~1.5 million gallons (5680 m^3) of water.

As indicated in Fig. 8.15, the reactor vessel bottom head is surrounded by 3 in. (0.076 m) of mirror insulation, but the head and the insulation are nowhere in contact. Therefore, if the drywell were flooded with water to a level above the bottom head, the water would penetrate the mirror insulation, effectively removing the insulation from the heat transfer process by means of convection currents within the gap between the bottom head and the insulation. It follows that the effective thermal conductivity of the reactor vessel bottom head with the drywell flooded should be close to the thermal conductivity of carbon steel alone.

This strategy has been briefly considered previously, by a simple scoping analysis (Appendix D of Ref. 9), which indicated that water surrounding the reactor vessel bottom head would have the potential to prevent melting of the submerged vessel wall. Nevertheless, it was also concluded that the existing systems available for containment flooding would require too much time to fill the wetwell and then raise the water level within the drywell to surround the lower portion of the reactor vessel. To realistically

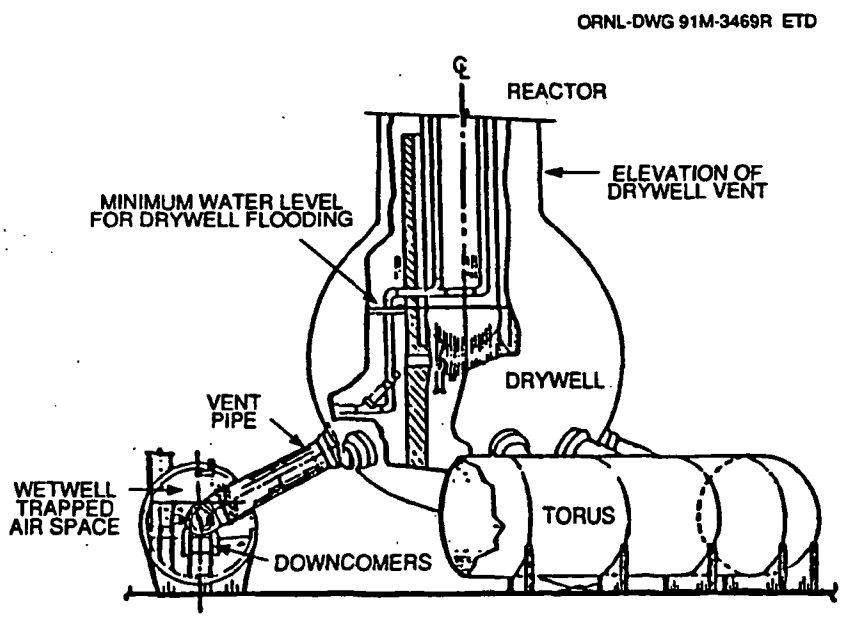

Figure 8.14 Flooding of BWR Mark I containment drywell to level sufficient to cover reactor vessel bottom head dependent on initial partial filling of wetwell torus

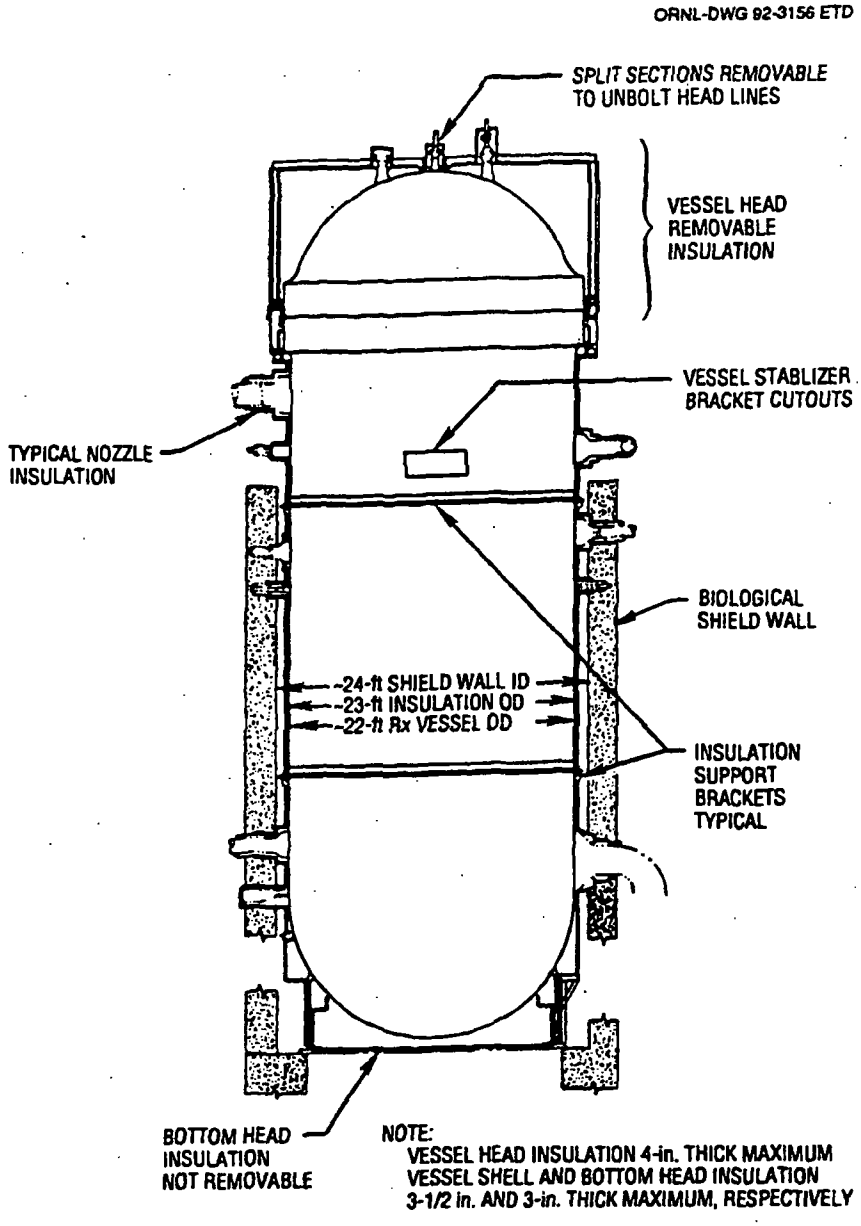

Figure 8.15 Reactor vessel insulation. *Source*: NRC, *Systems Manual—Boiling Water Reactors*

meet the requirements for retention of core and structural debris within the reactor vessel, there must be an ability to sufficiently flood the drywell within a very short time, because the operators would probably not resort to containment flooding until after core degradation had begun. If the water did not reach the vessel bottom head until after lower plenum dryout and heatup of the vessel wall, then the strategy might prove counterproductive, causing failure of the wall by thermal shock.

It seems worthwhile to consider this strategy again, especially in light of the current proposals[35] for preventing failure of the Mark I drywell shell by flooding the drywell floor with water, employing new or upgraded drywell spray systems for this purpose. The analyses associated with the current Mark I shell protection proposals are based upon a water level within the drywell extending only to the lower lip of the vent pipes [~2 ft (0.61 m) above the drywell floor], with the overflow entering the pressure suppression pool. However, equipment modifications to permit an increased drywell spray rate (or simultaneous use of the wetwell sprays) might be used to rapidly fill the wetwell, allowing the water within the drywell to flood the vent pipes and reach a level [~35 ft (10.7 m) above the drywell floor] sufficient to cover the reactor vessel bottom head. If drywell flooding to this level could be achieved quickly enough, then the water in the drywell might in

effect provide two lines of defense against containment failure: first, by serving to keep the debris within the reactor vessel and second, by protecting the drywell shell.

This candidate strategy has been selected for detailed assessment, the results of which are provided in Chaps. 14 through 22. From the standpoint of providing the necessary volume of water, implementation would require the availability of an independent drywell flooding system of sufficient capacity to deliver the required amount of water before reactor vessel lower plenum dryout and bottom head penetration failure. This, in general, would require equipment modifications but these provisions for effective and rapid drywell flooding might be required by separate considerations in support of resolution of the Mark I shell failure issue.[35]

9 Prevention of BWR Recriticality as a Late Accident Mitigation Strategy

S. A. Hodge M. Petek

A series of studies was undertaken at Oak Ridge National Laboratory (ORNL) during 1990 to consider candidate mitigative strategies for in-vessel events during the late phase (after core melting has occurred) of postulated boiling water reactor (BWR) severe accidents. The identification of new strategies was subject to the constraint that they should, to the maximum extent possible, make use of the existing equipment and water resources of the BWR facilities and not require major equipment modifications or additions. One of the recommendations developed as a result of these studies calls for further assessment of a candidate strategy for injecting borated water at the concentration necessary to preclude criticality upon reflooding of a damaged BWR core. This assessment is the subject of Chaps. 9 through 13 of this report.

9.1 Motivation for this Strategy

If significant control blade melting and relocation were to occur during a period of temporary core uncovering, then criticality would follow restoration of reactor vessel injection capability if the core were rapidly recovered with unborated water using the high-capacity low-pressure injection systems. If the relatively slow standby liquid control system (SLCS) were simultaneously initiated to inject sodium pentaborate solution, then the core would remain critical until sufficient boron for shutdown reached the core region. Furthermore, it is possible that injection of the SLCS tank contents may not produce a boron concentration sufficient for shutdown. For these reasons, it would be preferable, if control blade relocation has occurred, to inject only a boron solution at a rate sufficient to provide core cooling and terminate core damage.

The specific goal of the proposed strategy is to provide for the addition of the boron-10 isotope together with the injected flow being used to recover the core, in sufficient quantity to preclude criticality as the water level rises within the reactor vessel. It is expected that this could be accomplished using only existing plant equipment. One way to do this would be to mix the boron directly with the water in the condensate storage tank and then take suction on the condensate storage tank with the low-pressure system pump to be used for vessel injection. It is, however, not a simple matter to invoke this strategy, and preplanning and training would be required.

With respect to the rationale for incorporation of this strategy, a recent Pacific Northwest Laboratory (PNL) report[34] establishes that criticality upon reflooding a damaged core with unborated water is likely for either standing fuel rods or for a debris bed in the core region. It is conceivable that this alone might be a sufficient basis for incorporation of a boration strategy because there is a strong potential for operator surprise and confusion should, for example, a station blackout accident sequence be converted into an uncontrolled criticality with core damage upon restoration of reactor vessel injection capability. However, the PNL report concludes the following:

> — it appears that a super prompt-critical excursion (in which some fuel vaporization, dispersal of molten fuel debris, rapid molten fuel-coolant interaction, and the production of a large pressure pulse capable of directly failing the vessel and/or containment occurs) is not likely under conditions of reflooding a hot, degraded core; even under conditions of maximum reflood rate. Doppler feedback, in itself, appears to be adequate to limit the energetics of reflood recriticality to a level below which the vessel would be threatened by a pressure pulse. It is more likely that the reactor would either achieve a quasi-steady power level or enter an oscillatory mode in which water periodically enters and is expelled from the core debris. In either case, the average power level achieved is determined by the balance between reactivity added and the feedback mechanisms. Criticality in debris beds will probably produce power levels no larger than 10 to 20 percent of normal power. At these levels, the coolant makeup systems could provide adequate coolant to remove the heat generated within the debris bed.

Thus, one might conclude that the criticality attendant to reflooding could be controlled in the same manner as an anticipated transient without scram (ATWS), that it could be terminated by normal means [use of the SLCS], and that no dedicated strategy for preventing the criticality is required.

Criticality produced by reflooding after core damage has several important characteristics very different from those associated with ATWS, however, including not being addressed by current procedures, the lack of nuclear instrumentation, and the factor of operator surprise. The configuration of the critical masses in the core region might be standing fuel rods alone, a combination of standing fuel rods (outer core) and debris beds (central core), or a core-wide debris bed. Finally, the concentration of the boron-10 isotope produced by injection of the stored

contents of the SLCS tank may not be sufficient to terminate the criticality.

9.2 Assessment Outline

The most simple and straightforward strategy for injection of a boron solution into the reactor vessel under severe accident conditions would be based upon use of the SLCS. The capabilities of this system for such a purpose are discussed in Chap. 10.

The dominant set of BWR accident sequences leading to core damage involves station blackout, where simultaneous initiation of the SLCS would not be adequate to prevent criticality upon vessel reflooding if control blade damage has occurred. The only reliable strategy for prevention of criticality due to control blade damage for all recovered BWR severe accident sequences requires that the water used to recover the core contain a sufficient concentration of the boron-10 isotope to ensure that the reactor remains shutdown. Methods for accomplishing this are discussed in Chap. 11.

Chapter 12 provides the simplified cost-benefit analyses for the proposed strategy. As directed, this analysis is derived from the methodology described in NUREG-0933, *A Prioritization of Generic Safety Issues*.[18] It provides an evaluation of the estimated risk reduction associated with the proposed strategy and the estimated costs to the NRC and the industry in implementing such a strategy. Based upon these results, the priority ranking for the strategy is established in Chap. 13.

One of the conclusions of this assessment is that the use of Polybor® instead of a mixture of borax and boric acid to generate the boron solution would facilitate the implementation of the strategy. Appendix A provides information concerning the characteristics of this special sodium borate product.

Perhaps the most desirable characteristic of Polybor® from the standpoint of the proposed strategy is its ability to readily dissolve in cool water. Appendix B provides a discussion of several simple tabletop experiments performed at ORNL to investigate the limits of this high solubility.

10 Standby Liquid Control

S. A. Hodge M. Petek

The goal of the boiling water reactor (BWR) accident management strategy described in Chap. 9 is to prevent criticality upon restoration of reactor vessel injection following a core damage event in which the control blades have melted away. The normal means of adding boron to the reactor vessel is by dedicated injection by the standby liquid control system (SLCS). A brief description of the system arrangement is provided in the following section. Its function for mitigation of anticipated transient without scram (ATWS) is described in Sect. 10.2. The illustrative system dimensions and capacities given in this chapter are those associated with the 251-in. reactor vessel ID BWR-4 facility design installed at 1067-MW(e) plants such as Peach Bottom and Browns Ferry.

10.1 System Description

Although the SLCS is designed to inject sufficient neutron-absorbing sodium pentaborate solution into the reactor vessel to shut down the reactor from full power (independent of any control rod motion) and to maintain the reactor subcritical during cooldown to ambient conditions, the SLCS is not intended to provide a backup for the rapid shutdown normally achieved by scram. As indicated in Fig. 8.11, the basic system comprises a heated storage tank, two 100% capacity positive displacement pumps, and, as the only barrier to injection to the reactor vessel, two explosive squib valves. In most of the current BWR facilities, the sodium pentaborate solution enters the reactor vessel via a single vertical sparger located at one side of the lower plenum just below the core plate as indicated in Figs. 8.12 and 8.13. An effort to improve the mixing and diffusion of the injected solution (which has a specific gravity of about 1.3) throughout the core region has led some BWR facilities to provide a third positive displacement pump and to cause the injected solution to enter the reactor vessel via the core spray line and sparger.

10.2 Performance of Function

For the purpose of reducing the time required for reactor shutdown for the ATWS accident sequence, the Nuclear Regulatory Commission (NRC) has recently required that the SLCS injection be at a rate *equivalent* to 86 gal/min (0.0054 m^3/s) of 13 wt % sodium pentaborate solution, the boron being in its natural state with 19.8 at. % of the boron-10 isotope.* This requirement is established by the "ATWS rule," which states, in part:

> Each boiling water reactor must have a standby liquid control system (SLCS) with a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute of 13-weight percent sodium pentaborate solution.[36]

Because the original SLCS standard design provided for single-pump operation at a rate of 43 gal/min (0.0027 m^3/s), the ATWS rule permits the requirement for the increased equivalent control capacity to be satisfied by simultaneous operation of both of the installed pumps, by increasing the concentration of sodium pentaborate solution, or by enriching the boron within the sodium pentaborate solution in the isotope boron-10. Different BWR facilities have taken different approaches.

The sodium pentaborate solution is normally prepared by dissolving stoichiometric quantities of borax and boric acid within hot demineralized water according to the reaction

$$Na_2B_4O_7 \cdot 10H_2O + 6H_3BO_3 \rightarrow Na_2B_{10}O_{16} \cdot 10H_2O + 9H_2O.$$

As an illustrative example based upon the typical volume of the standby liquid control solution tank, 3458 lb (1569 kg) of borax and 3362 lb (1525 kg) of boric acid crystals dissolved within 3984 gal (15.08 m^3) of water will produce an aqueous solution containing 5350 lb (2427 kg) of sodium pentaborate. This is 13.4% sodium pentaborate by weight. The tank contains 980 lb (445 kg) of boron and, assuming that the boron is in its natural state (not enriched), 194 lb (88 kg) of the boron-10 isotope.

Continuing the example, the SLC tank contains 40,000 lb (18,100 kg) of solution so the concentration of natural boron within the tank would be 24,500 ppm. Because the mass of water within the reactor vessel (at normal water level and operating temperature) is 628,300 lb (285,000 kg),† the concentration of natural boron within the reactor vessel after the contents of the SLC tank had been added would be ~1560 ppm (the concentration of the boron-10 isotope would be ~308 ppm).

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*It is the $_5B^{10}$ isotope that has the large neutron absorption cross section (3840 barns). The reaction is $_5B^{10} + {}_0n^1 \rightarrow {}_3Li^7 + {}_2He^4$.

†Water mass for a 251-in.-ID BWR 3/4 reactor vessel, including the recirculation loops at the hot rated condition.

Standby

After the reactor had been brought subcritical, the next steps toward complete shutdown would involve cooldown and vessel filling. The reactor vessel water mass with normal water level at 70°F (294.3 K) would be 850,000 lb (385,000 kg) so that water addition during cooldown would reduce the concentration of natural boron to 1150 ppm. Finally, with the vessel completely filled after cooldown, the water mass would be 1,400,000 lb (635,000 kg), and the natural boron concentration would be 700 ppm. With the boron in its natural state, the concentration of the boron-10 isotope would be 138 ppm, which is sufficient to maintain the core shutdown in the cold, xenon-free condition.

Thus, the basic operational concept of the SLCS for ATWS control is that the very high concentration of boron in the relatively small SLC tank is diluted to the desired value when pumped into the much larger reactor vessel and mixed with the vessel water inventory.

Where BWR facilities have chosen to enrich the sodium pentaborate solution in the boron-10 isotope rather than to increase the pumping rate, it is the boric acid constituent that is enriched, typically to 92 at. %. This approach maintains the SLCS redundancy of having two pumps capable of independent operation. It also permits the sodium pentaborate concentration within the tank to be reduced from ~13.4% by weight to less than 9.2% by weight, which lowers the saturation temperature to ~40°F (277.6 K) and thereby eliminates the requirement for monitoring and maintaining the solution temperature.

10.3 Requirements for Liquid Poison in Severe Accidents

Under severe accident conditions, injection of neutron poison may be required for a situation very different than that normally associated with ATWS. If significant control blade melting and relocation from the core region were to occur during a period while the core was temporarily uncovered, then criticality should be expected if reactor vessel injection capability is restored and the core is then covered with cold unborated water.[34] This situation is most likely to occur with restoration of electrical power after a period of station blackout. If the SLCS were used to inject the sodium pentaborate solution at a relatively slow rate while the core was rapidly covered using the high-capacity low-pressure injection systems, then criticality would occur and the core would remain critical until sufficient boron for shutdown reached the core region.

In fact, it is possible that injection of the entire contents of the SLCS tank would not terminate the criticality. Reference 34 (Summary, page ix) reports:

> Analyses indicate that approximately 700 ppm ^{10}B are required to *ensure* subcriticality for all conditions, including standing fuel rods.

As noted in the previous section, injection of the SLCS tank contents would typically produce a concentration of ~308 ppm of the boron-10 isotope at normal reactor vessel water level and operating temperature, which would be diluted to ~225 ppm during cooldown to 70°F (294.3 K). This boron-10 concentration would be further reduced to ~138 ppm with the vessel filled and cold.

It would be preferable, if control blade melting and relocation have occurred, to reflood the vessel with a premixed solution of sufficient neutron poison concentration such that there would be no threat of criticality as the core was recovered. There must be a method for accomplishing this, however, at a rate sufficient to provide immediate core cooling and, thereby, terminate core damage. The major diagnostic concern with respect to this strategy is that the operators would have no direct means of knowing whether significant control blade melting and relocation had occurred. (In-core nuclear instrumentation would not be expected to survive control blade melting.) Therefore, either the injection source would have to be poisoned after any nontrivial period of core uncovering or reliance would have to be made on precalculated values of time to control blade melting for the various accident situations.

Methods for adequate poisoning of the injection source are described in Chap. 11.

11 Poisoning of the Injection Source

S. A. Hodge M. Petek

As described in the previous chapter, the standard means of adding boron to the reactor vessel involves the mixing of a highly concentrated injected boron stream with the normal vessel water inventory so that the resulting diluted solution attains a sufficient boron concentration to bring the core subcritical and to maintain subcriticality during vessel filling and cooldown. This method is intended for use in bringing the reactor to a gradual controlled shutdown in cases where cold shutdown cannot be obtained with control rods alone. It will not prevent criticality for cases where the core has been uncovered, the control blades have melted, and the core is then rapidly recovered with unborated water. Furthermore, a boron concentration sufficient for reactor shutdown may not be attained for these cases even after the entire contents of the standby liquid control system (SLCS) tank have been injected.

This chapter addresses means for accomplishing core covering and reactor vessel filling with a prepared solution of boron sufficiently concentrated to preclude criticality. Illustrative system dimensions and capacities are based upon the 251-in. reactor vessel ID BWR-4 facility design installed at 1067-MW(e) plants such as Peach Bottom and Browns Ferry; elevations within the reactor vessel are given as inches above vessel zero. The occasions for application of such a strategy and the associated frequencies for use in termination of core damage events will be discussed in Chap. 12.

11.1 Basic Requirements

The basic requirements to preclude criticality upon vessel reflooding with control blades melted from the core are, first, that the core be recovered with a poisoned solution and, second, that the solution contain a concentration of at least 700 ppm of the boron-10 isotope.[34]

A primary consideration involves the amount of water required to recover the core. Reactor vessel capacities at selected levels for Peach Bottom are provided in Table 11.1. It is important to recognize that the volumes listed in this table do not include allowance for filling of the recirculation or feedwater piping, because it would be expected that these loops would remain filled if the water level within the reactor vessel were lowered. Allowance has been made, however, for filling of the main steam lines [as the vessel water level rises above 647 in. (16.43 m)], which requires about 10,025 gal (37.95 m^3) of water.

The entry "Water height after ADS" in Table 11.1 pertains to the water volume remaining after operator actuation of the automatic depressurization system (ADS) under severe accident conditions. This operator action, which causes the opening of all safety/relief valves assigned to the ADS system (typically five or six), would be taken in accordance with the BWR Owners' Group Emergency Procedure Guidelines (EPGs) when the core became partially uncovered. Briefly, the purpose of this action is to induce flashing of a portion of the reactor vessel water volume, which provides temporary cooling of the previously uncovered region of the core. This rapid depressurization causes all of the water in the core region to be flashed and the reactor vessel water level to fall well beneath the core plate [at elevation 205 in. (5.21 m)] and into the vessel lower plenum. A detailed description of the background and purpose of the ADS maneuver is provided in Sect. 3.2.1.

Based upon the water volumes listed in Table 11.1, the mass of the boron-10 isotope that must be injected into the reactor vessel to achieve a 700-ppm concentration can be determined. The water temperature within the vessel is taken to be 70°F (294.3 K), so that the water density is 62.30 lb/ft^3 (998.0 kg/m^3). The results are shown in Table 11.2.

Table 11.1 Water volumes for the Peach Bottom reactor vessel

Location	Height above vessel zero (in.)	Water volume	
		ft^3	gal
Vessel filled	875.7	21,691	162,270
Normal operating level	561.0	12,722	95,172
Top of active fuel	366.3	7,566	56,600
Water height after ADS	137.0	2,191	16,391
Vessel zero	0.0	0	0

Table 11.2 Mass of the boron-10 isotope required to achieve a concentration of 700 ppm at 70°F

Location	Height above vessel zero (in.)	Water Volume (ft^3)	Water Mass (lb)	Boron-10 (lb)
Vessel filled	875.7	21,691	1,351,380	946
Normal operating level	561.0	12,722	792,600	555
Top of active fuel	366.3	7,566	471,370	330

With respect to the boron-10 concentration in the injected flow, two cases must be considered as appropriate to provide an upper and lower bound. The lower bound is provided by the case in which the vessel lower plenum is dry at the time when water injection is resumed. Under these conditions, the required boron-10 concentration for the injected flow simply equals the desired concentration in the vessel, which is 700 ppm. No allowance is made for the control blade B_4C powder originally in the vessel because there is no confidence that this powder would be available to mix with the injected flow.

The upper bound is provided by the case where some water remains in the lower plenum at the time injection is resumed. As indicated in Table 11.1, this might be as much as 16,400 gal (62.1 m^3). For this situation, the boron-10 concentration in the injected flow must exceed the desired concentration for the vessel, because the injected concentration will be diluted. The results are provided in Table 11.3. As indicated, the additional concentration required in the injected flow is inversely proportional to the total water mass to be injected. Stated another way, considerations regarding the excess concentration can be neglected if the reactor vessel is to be filled by the injected flow.

11.2 Alternative Boron Solutions

As described in Sect. 10.2, the SLCS injects to the reactor vessel from a tank containing a sodium pentaborate solution, prepared by dissolving stoichiometric quantities of borax and boric acid crystals within hot demineralized water. In the normal standby condition, the system tank contains about 190 lb (86 kg) of the boron-10 isotope. As indicated in Table 11.2, however, the mass of boron-10 isotope required to produce a concentration of 700 ppm within the reactor vessel is much greater than this.

In Sect. 11.1, it was shown that the boron-10 concentration of the injected solution must be between 740 and 1000 ppm (depending on the height to which the vessel is to be filled) to ensure a final concentration within the vessel of at least 700 ppm. It is now of interest to determine the corresponding concentrations of (natural) boron and of the sodium borate salt. As indicated in Appendix A, the boron-10 isotope constitutes 19.78% of natural boron and 3.62% of sodium pentaborate by weight. This leads to the results listed in Table 11.4.

It is easy to see that the formation of such high concentrations of sodium pentaborate in large volumes of water

Table 11.3 Concentration of the boron-10 isotope in the injected flow to achieve 700 ppm in the vessel

Location	Height above vessel zero (in.)	Water mass (lb)	Injected concentration Lower bound (ppm)	Injected concentration Upper bound (ppm)
Vessel filled	875.7	1,351,380	700	741
Normal operating level	561.0	792,600	700	846
Top of active fuel	366.3	471,370	700	985

Table 11.4 Concentrations of natural boron and sodium pentaborate corresponding to specified boron-10 concentrations

Boron-10	Natural boron (ppm)	Sodium pentaborate (ppm)
740	3741	20,421
1000	5056	27,596

would require the addition of large amounts of borax and boric acid. As an example, it is indicated in Table 11.2 that 555 lb (252 kg) of boron-10 would have to be injected to attain a concentration of 700 ppm at the normal reactor vessel water level. This corresponds to 2,806 lb (1,273 kg) of natural boron and 15,316 lb (6,947 kg) of sodium pentaborate. Furthermore, because each pound of sodium pentaborate formed requires the addition of 1.2749 lb (0.578 kg) of powder [0.6464 lb (0.293 kg) borax and 0.6285 lb (0.285 kg) boric acid], the total required mass addition would be 19,526 lb (8,857 kg). Even greater powder additions would have to be made if the reactor vessel were to be filled or if the injection tank where the boron solution is prepared has a larger volume.

A means of reducing the required quantity of powder to be added in carrying out the proposed accident management strategy would be to choose an alternative sodium borate solution for vessel injection. Polybor®, produced by the U. S. Borax Company, seems to be an ideal candidate. It is formed of exactly the same chemical constituents (sodium, boron, oxygen, and water) as is sodium pentaborate but has the advantages that, for the same boron concentration, it requires about one-third less mass of powder addition and has a significantly greater solubility in water. Detailed information concerning Polybor® and its comparison to sodium pentaborate as a means to form a boron solution are provided in Appendix A.

It is important to recognize that whereas sodium pentaborate solution is formed by adding borax and boric acid crystals to water, which then react to form the sodium pentaborate, a solution of Polybor® is formed simply by dissolving the Polybor® powder in water. This attribute, that there is no requirement for two separate powders to interact, is a major contributor to the greater solubility of Polybor®. The results of some tabletop experiments performed to investigate the solubility of Polybor® under adverse conditions (no mixing, cool water) are discussed in Appendix B. The advantage of Polybor® is obvious from these results.

To briefly illustrate the advantage of Polybor® with respect to weight of powder required for formation, the previous example will be repeated with Polybor® as the sodium borate solution. The 555 lb (252 kg) of boron-10 [corresponding to 2806 lb (1,273 kg) of natural boron] that must be injected to attain a concentration of 700 ppm at the normal reactor vessel water level would require the addition of 13,274 lb (6,021 kg) of Polybor® powder, which is about two-thirds of the borax/boric acid mass addition required for formation of the same boron-10 concentration.

The following section addresses the means by which a sufficient quantity of the prepared solution (at least 700 ppm of the boron-10 isotope) could be delivered to the reactor vessel as necessary to restore normal water level.

11.3 Practical Injection Methods

The condensate storage tank is an important source of water to the reactor vessel injection systems for each nuclear unit. As indicated in Fig. 11.1 (based upon the Browns Ferry arrangement), it is the normal suction source for the steam turbine-driven high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems and the alternate source for the electric motor-driven residual heat removal and core spray pumps. Other BWR facilities also have at least one motor-driven reactor vessel injection system [in addition to the control rod drive hydraulic system (CRDHS)] capable of taking suction upon the condensate storage tank.

A typical condensate storage tank structure, located outside the reactor building, is shown in Fig. 11.2. At the Browns Ferry Nuclear Plant, each unit's condensate storage tank has a cylindrical height of ~33 ft (10.1 m) and a total capacity of 375,000 gal (1420 m^3). Each condensate storage tank at Peach Bottom has a capacity of 200,000 gal (757 m^3). At least one BWR facility (Grand Gulf) currently has in place a procedure for adding borax and boric acid crystals directly to the (partially drained) tank, for use as backup to the SLCS if needed in the event of anticipated transient without scram (ATWS).

During normal reactor operation, the condensate storage tank provides makeup flow to the main condenser hotwells via an internal tank standpipe, as indicated on Fig. 11.3. The purpose of the standpipe is to guarantee a reserve supply of water for the reactor vessel injection systems that take suction from the bottom of the tank. These include the CRDHS, which provides cooling water to the CRD mechanism assemblies during normal reactor operation.

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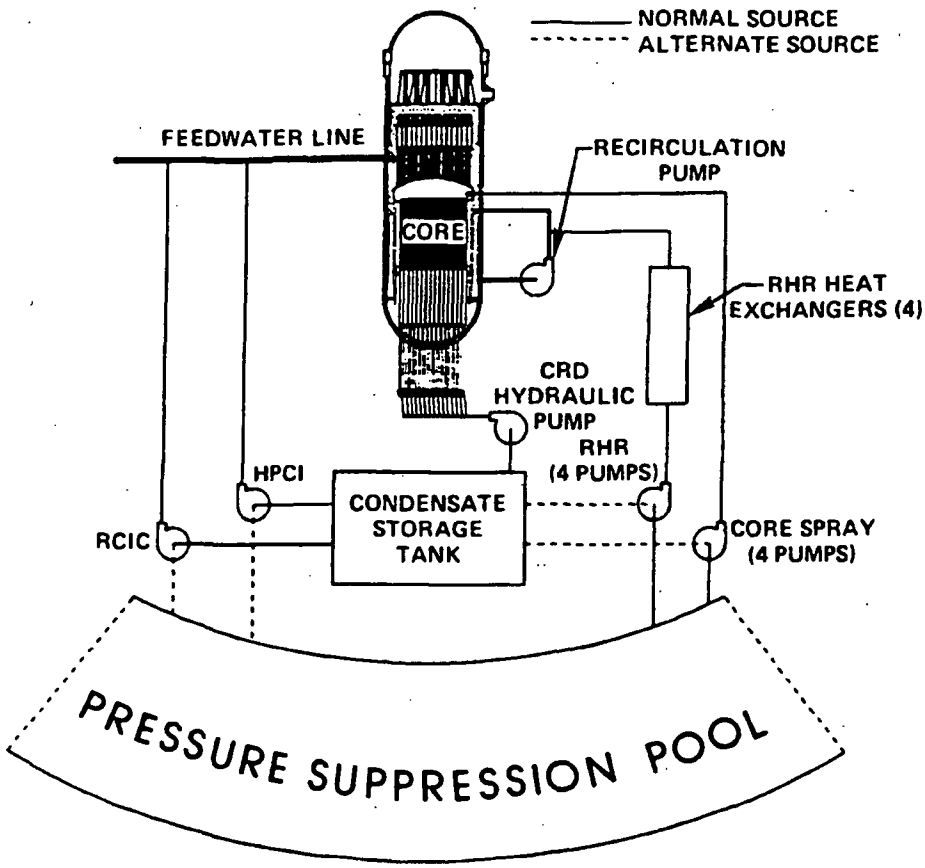

Figure 11.1 Condensate storage tank—an important source of water for use in accident sequences other than large-break LOCA

As discussed in Sect. 11.2, a much higher concentration of boron would be required for the prevention of criticality for the case of reflooding a degraded core than would be required for the termination of ATWS with the core intact. Indeed, the requirement specified in Ref. 34 for a concentration within the reactor vessel of 700 ppm of the boron-10 isotope is about three times greater than the vessel concentration (225 ppm) obtained [for normal vessel water level at 70°F (294.3 K)] by operation of the SLCS.

Any practical strategy for direct poisoning of the tank contents must provide for partial draining, particularly if boron-10 concentrations within the tank on the order of 740 to 985 ppm (Table 11.3) are to be achieved. The condensate storage tank could be gravity-drained through the standpipe under station blackout conditions. The remaining reserve water volume would be plant-specific, but a representative value for a 1060-MW(e) BWR-4 facility such as Browns Ferry or Peach Bottom is 135,000 gal (510 m^3). It is this reserve volume that would be poisoned.

Even with partial tank draining, however, the amount of powder required to obtain a boron-10 concentration of 740 to 985 ppm is large. Assuming the use of Polybor® to take advantage of its greater solubility, 20,400 to 27,200 lb (9,250 to 12,340 kg) would have to be added to a reserve volume of 135,000 gal (510 m^3). [If borax/boric acid were used, the requirement would be 30,050 to 40,350 lb (13,630 to 18,300 kg).] Clearly, this is too much to be manhandled [50-lb (23-kg) bags] to the top of the tank and poured in. The practical way to poison the tank contents would be to prepare a slurry of extremely high concentration in a smaller container at ground level, then to pump the contents of this small container into the upper opening of the condensate storage tank. (As indicated in Appendix B, Table B.4, extremely high concentrations can be achieved with Polybor®.)

For this concept, the arrangement of the condensate storage tank with its internal standpipe is almost ideal. As the majority of the added Polybor® mass settled toward the

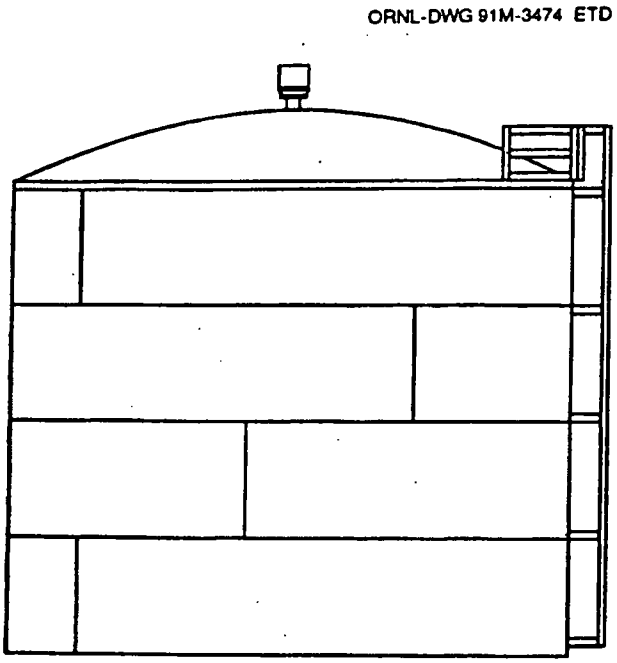

Figure 11.2 Condensate storage tank located external to reactor building and vented to atmosphere

bottom of the tank, the displaced solution would be removed at the location of the standpipe entrance, where the Polybor® concentration would be relatively low. On the other hand, when (and if) reactor vessel injection capability was restored, the injected solution would be taken from the bottom of the tank, where the Polybor® concentration would be the highest within the tank.

To avoid any requirement for procurement of additional plant equipment, a fire engine and independent portable (foldable) suction tank might be employed to perform the solution mixing and transfer function necessary for poisoning of the condensate storage tank. Foldable water tanks of 5000-gal (18.9-m^3) capacity that can be set up in seconds are commercially available. There would, however, be no need for such rush in implementation of this strategy.

The shortest time interval between the inception of a BWR accident sequence and the need for injection with a Polybor® solution if criticality induced by control blade melting is to be averted occurs in the short-term station blackout accident sequence. Even here, more than an hour would elapse before control blade melting began.

Upon total loss of station ac power, the foldable tank would be set up so that the high-concentration Polybor® solution could be prepared. The Fire Department would be notified to send engine pumpers. Although this is a plant-specific matter, these pumpers would in general be expected to have a capacity of 750 or 1000 gal/min (0.047 or 0.063 m^3/s) and to pump at that rate from a portable tank.[37] Obviously, plant procedures should be based upon the specific pumpers that would be available.

If the HPCI or RCIC systems were operational (long-term station blackout), then their use for reactor vessel injection during the period (~6 h) that battery power remained would

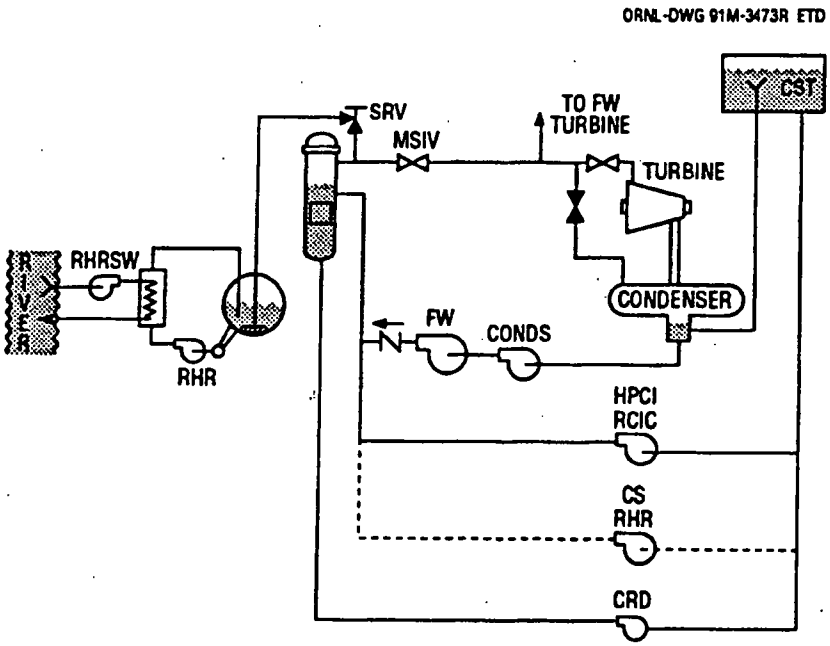

Figure 11.3 Condensate storage tank that can be drained to the main condenser hotwells via the internal standpipe, leaving sufficient water volume for reactor vessel injection

effectively reduce the condensate storage tank water inventory. If, on the other hand, these turbine-driven reactor vessel injection systems had failed upon demand (short-term station blackout), then no water would be taken from the condensate storage tank except by intentional draining. If there is no prospect for restoration of reactor vessel injection at 45 min after the inception of the loss of ac power, then tank draining (via the standpipe) should begin at that time. Typically, about 15 min would be required to reduce the tank contents to the level of the standpipe inlet by gravity draining at the maximum rate.

Should the accident sequence progress to the point where most of the core became uncovered, the EPGs require that the operators manually initiate the ADS; this action is intended to ensure that the core would become totally uncovered before its upper portion reached the runaway zirconium oxidation temperature. In addition, the upward rush of flashed steam would provide temporary cooling of the previously uncovered regions of the core. It is precisely at this point that actual introduction of Polybor® to the condensate storage tank should begin under the proposed strategy.

About 0.5 h after the ADS maneuver, the heatup of the upper core would recover from its temporary setback, and the control blades would begin to melt. Success of the proposed strategy would be counted if a sufficiently poisoned solution were available for vessel reflood from this point onward.

It is important to recognize that most of the difficulty associated with implementation of the proposed strategy derives from the large amount of powder that must be placed into solution.* For this reason, the strategy could be more easily implemented at smaller BWR facilities such as Monticello or Duane Arnold [~540 MW(e)], which have smaller condensate storage tanks and smaller reserve condensate volumes [75,000 gal (284 m^3)]. Because the required poison concentration is the same for all BWR facilities regardless of size, the required amount of powder addition is directly proportional to the water volume. Similarly, facilities such as Grand Gulf [1142 MW(e)] with larger condensate storage tank reserve volumes [170,000 gal (644 m^3)] would have to add additional powder.

Finally, nothing in the proposed strategy is intended to discourage the initiation of the SLCS upon restoration of ac power. The additional boron-10 concentration that would be delivered (slowly) by this system can be considered to compensate for some of the uncertainties associated with the injection from the condensate storage tank.

*This requirement derives from the results presented by NUREG/CR-5653. Because this strategy for prevention of criticality upon vessel reflood would be much simpler to implement if the necessary poison concentration were reduced, it seems that the results of NUREG/CR-5653 should be assessed by an independent body—particularly in view of the fact that the authors of that report describe their results as "conservative."

12 Cost-Benefit Analyses for the Boron Injection Strategy

S. A. Hodge M. Petek

This chapter describes the results of a cost-benefit analysis performed to assess the boron injection strategy proposed for mitigation of boiling water reactor (BWR) severe accidents in which control blade melting may have occurred. The analysis is based upon the standard methodology described in NUREG-0933, *A Prioritization of Generic Safety Issues.*[18] The detailed procedure and formats for listing the results are those explained in Refs. 38 and 38. The priority determination based upon these results is provided in Chap. 13.

12.1 Summary Work Sheet

TITLE: Boron Injection Strategy for Mitigation of BWR Severe Accidents

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

If significant control blade melting and relocation occurs during a period of temporary core uncovering, then criticality will occur if reactor vessel injection is restored and the core is flooded with unborated water. The goal of the proposed strategy is to prevent criticality by providing that borated water be used to recover the core. It is expected that the proposed strategy could be implemented using only the existing plant equipment.

AFFECTED PLANTS PWR: Operating = 0 Planned = 0
BWR: Operating = 37 Planned = 1

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION=	4856

OCCUPATIONAL DOSES:

Implementation =	0
Operation/Maintenance =	0
Total of Above =	0
Accident Avoidance =	19

COST RESULTS ($1E+06)

INDUSTRY COSTS:

Implementation =	2.7
Maintenance =	0.9
Total of Above =	3.6
Accident Avoidance =	1.6

NRC COSTS:

Strategy Development =	0
Implementation Support =	0.09
Maintenance Review =	0.18
Total of Above =	0.27

12.2 Proposed Accident Mitigation Strategy

The proposal involves a mitigative strategy for in-vessel events during the late phase (after core degradation has occurred) of postulated BWR severe accidents. The strategy addresses the prevention of undesired criticality.

If significant control blade melting and relocation were to occur during a period of temporary core uncovering, then criticality would follow restoration of reactor vessel injection capability if the core were rapidly recovered with unborated water using the high-capacity low-pressure injection systems. If the relatively slow standby liquid control system (SLCS) were simultaneously initiated to inject sodium pentaborate solution, then the core would remain critical until sufficient boron for shutdown reached the core region. It would be preferable, if control blade melting and relocation has occurred, to inject only a boron solution provided that this can be done at a rate sufficient to provide core cooling and terminate core damage.

The specific goal of the proposed strategy is to provide for the addition of the boron-10 isotope, together with the injected flow being used to recover the core, in sufficient quantity to preclude criticality as the water level rises within the reactor vessel. It is expected that this could be accomplished using only existing plant equipment. One way to do this would be to mix the boron directly with the water in the condensate storage tank and then take suction on the condensate storage tank with the low-pressure system pump to be used for vessel injection. It is, however, not a simple matter to invoke this strategy and preplanning and training would be required.

With respect to the rationale for incorporation of this strategy, a recent Pacific Northwest Laboratory (PNL) report[34] establishes that criticality upon reflooding with unborated water is likely for either standing fuel rods or for a debris bed subsequently formed in the core region. It is not unreasonable that this prediction alone should provide sufficient

motivation for incorporation of a boration strategy because there is a strong potential for operator surprise and confusion should, for example, a station blackout accident sequence be converted into an uncontrolled criticality event upon restoration of reactor vessel injection capability. However, the PNL report concludes that

> —it appears that a super prompt-critical excursion (in which some fuel vaporization, dispersal of molten fuel debris, rapid molten fuel-coolant interaction, and the production of a large pressure pulse capable of directly failing the vessel and/or containment occurs) is not likely under conditions of reflooding a hot, degraded core; even under conditions of maximum reflood rate. Doppler feedback, in itself, appears to be adequate to limit the energetics of reflood recriticality to a level below which the vessel would be threatened by a pressure pulse. It is more likely that the reactor would either achieve a quasi-steady power level or enter an oscillatory mode in which water periodically enters and is expelled from the core debris. In either case, the average power level achieved is determined by the balance between reactivity added and the feedback mechanisms. Criticality in debris beds will probably produce power levels no larger than 10 to 20 percent of normal power. At these levels, the coolant makeup systems could provide adequate coolant to remove the heat generated within the debris bed.[34]

Thus, one might conclude from the PNL analysis that the criticality attendant to reflooding could be controlled in the same manner as an anticipated transient without scram (ATWS), that it could be terminated by normal means [use of the SLCS], and that no dedicated strategy for preventing the criticality is required.

Nevertheless, criticality produced by reflooding after core damage has characteristics very different from those associated with ATWS, including not being addressed by current procedures, the probable lack of nuclear instrumentation, and the factor of operator surprise. The configuration of the critical masses in the core region might be standing fuel rods alone, a combination of standing fuel rods (outer core) and debris beds (central core), or a core-wide debris bed. The PNL report provides the estimate that a boron-10 concentration of between 700 and 1000 ppm would be required within the reactor vessel to preclude criticality once control blade melting had occurred. This is greater than the concentration attainable by injection of the entire contents of the SLCS tank (~170 ppm).

Thus, on two counts, operation of the SLCS would not prevent criticality upon vessel reflood following a period of temporary core uncovering with control blade melting. First, the injection of poison by this system would be too slow. Second, the amount of poison injected would be insufficient.

It would be preferable, if control blade melting and relocation has occurred, to reflood the vessel from an injection source such as the condensate storage tank containing a premixed solution of neutron poison so that there would be no threat of criticality as the core was recovered. This must be achievable, however, at a rate sufficient to provide immediate core cooling and, thereby, terminate core damage. The major diagnostic concern with respect to this strategy is that the operators would have no direct means of knowing whether significant control blade melting and relocation had occurred. Therefore, either the injection source would have to be poisoned after any non-trivial period of core uncovering or reliance would have to be made on precalculated values of time to control blade melting for the various accident situations.

In addition, formation of sodium pentaborate by the normal method of separately adding borax and boric acid crystals would not be feasible at low temperatures and without mechanical mixing. Information concerning an alternative boron form was obtained by contacting the U.S. Borax Company at Montvale, New Jersey. The company produces a disodium octaborate tetrahydrate ($Na_2B_8O_{13} \cdot 4H_2O$) in readily soluble powder form, under the tradename Polybor®. Boron constitutes 20.97% of the total weight of Polybor®, as opposed to 18.32% of the weight of sodium pentaborate. Using Polybor®, the total amount of material needed to form a given concentration of natural boron is significantly (about one-third) less than for borax and boric acid. Much of the difference lies in the excess water added with the borax ($Na_2B_4O_7 \cdot 10H_2O$).

Polybor® readily dissolves even in cool water to form supersaturated solutions. Simple tabletop experiments have demonstrated that Polybor® dissolves much more readily in water than does the normally used mixture of borax and boric acid crystals. (There is no need for two separate powders to interact in the case of Polybor®.) This is of interest because the primary application of the accident management strategy under consideration would be for use under station blackout conditions, when the water in the condensate storage tank may have cooled significantly at the time the borated solution was to be prepared and mechanical mixing of the tank contents would not be available.

The condensate storage tank is an important source of water to the reactor vessel injection systems. It is the normal suction source for the steam turbine-driven high-

pressure coolant injection and reactor core isolation cooling systems and the alternate source for at least one electric-motor-driven reactor vessel injection system, either the residual heat removal or the core spray system. At least one BWR facility currently has in place a procedure for adding borax and boric acid crystals directly to the (partially drained) condensate storage tank, for use as backup to the SLCS if needed in the event of ATWS.

During normal reactor operation, the condensate storage tank provides makeup flow to the main condenser hotwells via an internal tank standpipe. Any practical strategy for direct poisoning must provide for partial draining particularly if boron-10 concentrations as high as 700 to 1000 ppm are to be achieved. The condensate storage tank could be gravity-drained through the standpipe under station blackout conditions. The residual water volume would be plant-specific, but a representative value for a 1060-MW(e) BWR-4 facility such as Browns Ferry is 135,000 gal (511 m^3).

Even with partial condensate storage tank draining, however, the amount of powder required to obtain a boron-10 concentration of 1000 ppm is large. Assuming the use of Polybor® to take advantage of its greater solubility, 27,775 lb (12,600 kg) would have to be added to the partially drained tank. [If borax/boric acid were used, the requirement would be 41,000 lb (18,600 kg).]. Clearly, this is too much to be manhandled [50-lb (23-kg) bags] to the top of the tank and poured in. The practical way to poison the tank contents would be to prepare a slurry of extremely high concentration in a smaller container at ground level, then to pump the contents of this small container into the upper tank opening. (Extremely high concentrations can be achieved with Polybor®.). To avoid any requirement for procurement of additional plant equipment, a fire engine with its portable suction container might be employed to perform the slurry preparation and pumping function.

12.3 Risk and Dose Reduction

The calculations in this section are based upon the results of the recent PNL study *Recriticality in a BWR Following a Core Damage Event* published as NUREG/CR-5653 (PNL-7476) in December 1990.[34]

12.3.1 Public Risk Reduction

The reduction in public risk associated with implementation of the proposed strategy derives from the portion of the dominant station blackout severe accident sequences that have the potential to be terminated by restoration of electric power and reactor vessel injection capability before vessel breach and containment failure. If electric power is restored during the recriticality time window, then the associated restoration of reactor vessel injection capability does not terminate the station blackout severe accident sequence but rather converts it to an uncontrolled criticality event, which rapidly leads to vessel breach and containment failure. With successful implementation of the strategy, however, criticality is not a consequence of the restoration of reactor vessel injection and the potential for successful termination of the station blackout sequence is unchanged. The results for each step in the analysis for public risk reduction (following the NUREG-0933 methodology[18]) are provided in Table 12.1.

12.3.2 Occupational Dose

The results of the analysis for occupational dose are summarized in Table 12.2. An estimated occupational dose of 20,000 man-rem from postaccident cleanup, repair, and refurbishment is considered, per the guidance of Sect. 3.c of the Introduction to NUREG-0933.[18]

12.4 Costs

The industry and NRC costs associated with implementation of the proposed strategy are estimated in this section. In accordance with NUREG-0933 (Introduction, Sect. 3.d), costs are estimated in 1982 dollars.* Both NRC professional time and industry manpower costs are based upon $100,000/person-year. Results are summarized in Table 12.3.

It is important to recognize that this strategy for reactor vessel flooding with a sodium borate solution under severe accident conditions has been proposed as an effective yet inexpensive accident mitigation measure that might be implemented employing the existing plant equipment. This is in accordance with the general guidance for the Boiling Water Reactor In-Vessel Strategies Program:

> "Two criteria will be used in selecting the candidate strategies: (1) they shall require no major equipment modifications or additions by maximizing the use of the existing equipment and water resources in a plant, and (2) they are not currently available in the EPGs. The purpose ... is to identify candidate strategies that could enhance or extend the EPGs in handling severe accidents."

*Use of the 1982 dollar as a standard permits comparison of the results of the many cost-benefit analyses performed at various times during the last decade. Based upon the general inflation rate, 1992 cost = 1.448 times the 1982 cost.

Cost-Benefit

In this spirit, provision of a dedicated mixing tank and pumping system for preparation of the highly concentrated Polybor® solution and its transfer into the condensate storage tank has not been contemplated.

Rather, it is expected that the required mixing and transfer could be accomplished, with preplanning and training, by a fire department pumper and portable tank. In general, utilities have agreements in place for use of local municipality, county, or district fire department facilities in emergency situations. If a pumper and portable tank were to be purchased specifically for support of the proposed boration strategy, the associated cost (1982 dollars) would be about $125,000, which would have to be considered in the determination of the Per-Plant Industry Cost for Strategy Implementation (Item 7 of Table 12.3).

Table 12.1 Public Risk Reduction Work Sheet

1. Title of Proposed Strategy:

Boron Injection Strategy for Mitigation of BWR Severe Accidents.

2. Affected Plants (N) and Average Remaining Lives ($\overline{T}$):

		N	$\overline{T}$ (yr)
BWRs:	Operating	37	20.9
	Planned	1	30.0
	Total	38	21.1

3. Plants Selected for Analysis:

Peach Bottom 2—representative BWR.

4. Parameters Affected by Proposed Strategy:

The proposed strategy would serve to prevent criticality for accident sequences in which reactor vessel injection capability is restored after core damage has occurred. The probability that the strategy would be required has been calculated by Pacific Northwest Laboratories (PNL) based upon the NUREG-1150 risk study using the Peach Bottom plant as a technical basis. In their NUREG/CR-5653 analysis, PNL defines event tree parameters ACPOWXX to represent the estimated fraction (XX) of loss of offsite power events that will be terminated (power will be restored) during the recriticality time windows of the affected accident sequences.

5. Base Case Values for Affected Parameters:

The parameters ACPOWXX are accident-sequence dependent because the time after initiation of the loss of offsite power event that the recriticality time window begins and the length of this time window depend upon the accident sequence. The base-case parameters are established in NUREG/CR-5653 (Section 3.1.4) as:

Accident Sequence	ACPOWXX
PBTBO/PBTBUX	ACPOW12 = 0.12
PBEM2	ACPOW11 = 0.11
PBTBS	ACPOW01 = 0.01

These accident sequences are defined below.

Table 12.1 (continued)

6. Affected Accident Sequences and Base-Case Frequencies:

The affected accident sequences are

PBTBO/PBTBUX	short-term station blackout without ADS
PBEM2	short-term station blackout with ADS
PBTBS	long-term station blackout with ADS.

All are based upon Peach Bottom. PBTBO and PBTBUX are identical except that PBTBO was calculated by Battelle Memorial Institute as part of previous work upon NUREG-1150, whereas PBTBUX (as well as PBEM2 and PBTBS) were calculated specifically for the NUREG/CR-5653 study using a more recent version of the MARCH code. For additional information, see Section 2.2 of NUREG/CR-5653.

The base-case frequencies represent the probabilities of recriticality events based upon these accident sequences.

PBTB0/PBTBUX	x	ACPOW12	=	5.2E-07/py
PBEM2	x	ACPOW11	=	0.4E-07/py
PBTBS	x	ACPOW01	=	6.9E-07/py
				1.25E-06/py

These are the frequencies of recriticality events which, following NUREG/CR-5653 (Section 3.1.4), would lead to suppression pool saturation and containment over-pressurization in about half an hour.

7. Affected Release Categories and Base-Case Frequencies:

The release category for the recriticality event is BWR-3, defined in WASH-1400 (Appendix VI) as representing

> "a core meltdown caused by a transient event accompanied by a failure to scram or failure to remove decay heat. Containment failure would occur either before core melt or as result of gases generated during the interaction of the molten fuel with concrete after reactor-vessel meltthrough. Some fission-product retention would occur either in the suppression pool or the reactor building prior to release to the atmosphere. Most of the release would occur over a period of about 3 hours and would involve 10% of the iodines and 10% of the alkali metals. For those sequences in which the containment would fail due to overpressure after core melt, the rate of energy release to the atmosphere would be relatively high. For those sequences in which overpressure failure would occur before core melt, the energy release rate would be somewhat smaller, although still moderately high."

Thus the base-case release category and frequency is

BWR-3 = 1.25E-06/py.

8. Base-Case Affected Core-Melt Frequency ($\overline{F}$):

$\overline{F}$ = 1.25E-06/py

Table 12.1 (continued)

9. Base-Case, Affected Public Risk (W):

W = (1.25E-06/py) (5.1E+06 man-rem)

= 6.375 man-rem/py

The man-rem value associated with release category BWR-3 is taken from Exhibit B of Section III of the Introduction to NUREG-0933.

10. Adjusted-Case Values for Affected Parameters:

Following Section 3.1.4 of NUREG/CR-5653, it is assumed that implementation of the proposed strategy would reduce the frequency of recriticality caused by reactor vessel refill after control blade melting by 95%. Stated another way, it is assumed that if implemented, the boron injection strategy for mitigation of BWR severe accidents would fail to be properly applied 5% of the time.

11. Affected Accident Sequences and Adjusted-Case Frequencies:

PBTB0/PBTBUX	x	ACPOW12	x 0.05	=	2.6E-08/py
PBEM2	x	ACPOW11	x 0.05	=	2.0E-09/py
PBTBS	x	ACPOW01	x 0.05	=	3.5E-08/py
					6.25E-08/py

12. Affected Release Categories and Adjusted-Case Frequencies:

BWR-3 = 6.25E-08/py

13. Adjusted-Case, Affected Core-Melt Frequency ($\overline{F}^*$):

$\overline{F}^*$ = 6.25E-08/py

This is the adjusted frequency of recriticality events.

14. Adjusted-Case, Affected Public Risk (W*):

W* = (6.25E-08/py) (5.1E+06 man-rem)

= 0.319 man-rem/py

15. Reduction in Core-Melt Frequency ($\Delta\overline{F}$):

$\Delta\overline{F}$ = 1.25E-06/py – 6.25E-08/py

= 1.19E-06/py

Strictly speaking, this is the reduction in unmitigated core melt frequency. The proposed strategy affects the progression of events during the recriticality time window, which is initiated by the melting of some core structures (the control blades). Thus, some core damage is associated even with successful application of the strategy; vessel breach and containment failure would, however, be averted.

Table 12.1 (continued)

16. Per-Plant Reduction in Public Risk (ΔW):

$$\Delta W = 6.375 \text{ man-rem/py} - 0.319 \text{ man-rem/py}$$

$$= 6.056 \text{ man-rem/py}$$

17. Total Public Risk Reduction, (ΔW) Total:

$$\Delta W_{TOTAL} = (6.056 \text{ man-rem/py})\ (38 \text{ p})\ (21.1 \text{ y})$$

$$= 4856 \text{ man-rem}$$

Upper bound = 1.53E+05 man-rem

Lower bound = 0

Here the upper and lower bounds are established in accordance with the guidance of Section 3.5.1 of NUREG/CR-2800.

Table 12.2 Occupational Dose Work Sheet

1. Title of Proposed Strategy:

   Boron Injection Strategy for Mitigation of BWR Severe Accidents.

2. Affected Plants (N):

		N
BWR:	Operating	37
	Planned	1
	Total	38

3. Average Remaining Lives of Affected Plants ($\overline{T}$):

		$\overline{T}$ (yr)
BWR:	Operating	20.9
	Planned	30.0
	All BWRs	21.1

4. Per-Plant Occupational Dose Reduction Due to Accident Avoidance, $\Delta(FD_R)$:

$$\Delta \overline{F} D_R = (1.19E\text{-}06/py)\ (20{,}000\ \text{man-rem/py})$$

$$= 0.024\ \text{man-rem/py}$$

The reduction in unmitigated core melt frequency is obtained from Item 15 of Table 12.1. The estimated occupational dose of 20,000 man-rem incurred by post-accident cleanup, repair, and refurbishment is taken from Section III of the Introduction to NUREG-1150.

5. Total Occupational Dose Reduction Due to Accident Avoidance, (ΔU):

$$\Delta U = (0.024\ \text{man-rem/py})\ (38\ p)\ (21.1\ y)$$

$$= 19.24\ \text{man-rem}$$

Upper bound = 120

Lower bound = 0

Here the upper and lower bounds are established in accordance with Section 3.5.2 of NUREG/CR-2800.

Table 12.2 (continued)

6. Per-Plant Utility Labor in Radiation Zones for Strategy Implementation:

None.

7. Per-Plant Occupational Dose Increase for Strategy Implementation (D):

None.

8. Total Occupational Dose Increase for Strategy Implementation (ND):

None.

9. Per-Plant Utility Labor in Radiation Zones for Strategy Maintenance:

None.

10. Per-Plant Occupational Dose Increase for Strategy Maintenance (D_O):

None.

11. Total Occupational Dose Increase for Strategy Maintenance (NTD_O):

None.

12. Total Occupational Dose Increase (G):

Best Estimate (man-rem)	Error Bounds (man-rem) Upper	Lower
0	0	0

These upper and lower bounds are established in accordance with the guidance of NUREG/CR-2800, Section 3.5.3.

Table 12.3 Cost Work Sheet

1. Title of Proposed Strategy:

Boron Injection Strategy for Mitigation of BWR Severe Accidents.

2. Affected Plants (N):

		N
BWR:	Operating	37
	Planned	1
	Total	38

3. Average Remaining Lives of Affected Plants ($\overline{T}$):

		$\overline{T}$ (yr)
BWR:	Operating	20.9
	Planned	30.0
	All	21.1

Industry Costs (steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident Avoidance, $\Delta(\overline{F}A)$:

(1.19E-06/py) ($1.65E+09) = $1.96E+03/py

The estimated cost of $1.65E+09 for cleanup, repair, and refurbishment is taken from Table 3.5 of the *Handbook for Value-Impact Assessment, NUREG/CR-3568.*

5. Total Industry Cost Savings Due to Accident Avoidance, (ΔH):

(38 p) (21.1 y) ($1.96E+03/py) = $1.6E+06

Upper bound = $9.9E+06

Lower bound = 0

Here the upper and lower bounds are established in accordance with Section 4.3.1 of NUREG/CR-2800.

Table 12.3 (continued)

6. Per-Plant Industry Resources for Strategy Implementation:

Implementation of the proposed strategy will require acquisition of material (Polybor), engineering analysis, preparation of procedures, training, and management review.

7. Per-Plant Industry Cost for Strategy Implementation (I):

Resources	Cost ($/plant)
Engineering	10K
Procedures	20K
Management Review	5K
Training	20K
Material	15K
Total	70K

The estimated material cost is based on acquisition of 30,000 lb of Polybor at $0.50 per pound (1982 dollars). This invokes a conservative assumption that a special quality grade of powder would be purchased although it is by no means certain that this would be required for severe accident applications.

8. Total Industry Cost for Strategy Implementation (NI):

(38 plants) ($7.0E+4/plant) = $2.66E+06

9. Per-Plant Industry Labor for Strategy Maintenance:

It is estimated that an average of 20 man-hr/py will be required for periodic procedure review and training (including drills).

10. Per-Plant Industry Cost for Strategy Maintenance (I_O):

I_O = (20 man-hr/py) (1 man-wk/40 man-hr) ($2270/man-wk)
= $1135/py

11. Total Industry Cost for Strategy Maintenance ($N\overline{T}I_O$):

$N\overline{T}I_O$ = (38 plants) (21.1 yr) ($1135/py) = $9.1E+05

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
$3.6E+06	$5.0E+06	$2.2E+06

These upper and lower bounds are established in accordance with the guidance of NUREG/CR-2800, Section 4.3.2.

Table 12.3 (continued)

NRC Costs (Steps 13 through 21)

13. NRC Resources for Strategy Development:

The general strategy for boration of the water injection source has already been developed by the NRC Office of Research as a candidate accident management strategy. Implementation of the strategy would be carried out on a voluntary, plant-specific basis by the industry. Therefore, no additional NRC development costs would be incurred.

14. Total NRC Cost for Strategy Development (C_D):

None.

15. Per-Plant NRC Labor for Support of Strategy Implementation:

To support implementation by the industry, 1 man-week per plant is assumed.

16. Per-Plant NRC Cost for Support of Strategy Implementation (C):

C = (1 man-wk/plant) ($2270/man-wk) = $2270/plant

17. Total NRC Cost for Support of Strategy Implementation (NC):

NC = (38 Plants) ($2270/plant) = $8.6E+04

18. Per-Plant NRC Labor for Review of Strategy Maintenance:

Approximately 0.10 man-week per plant year is estimated for follow-up on maintenance of the proposed strategy.

19. Per-Plant NRC Cost for Review of Strategy Maintenance (C_O):

C_O = (0.10 man-wk/py) ($2270/man-wk) = $227/py

20. Total NRC Cost for Review of Strategy Maintenance ($N\overline{T}C_O$):

$N\overline{T}C_O$ = (38 plants) (21.1 yr) ($227/py) = $1.8E+05

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$2.7E+05	$3.7E+05	$1.7E+05

Here, the upper and lower bounds are established in accordance with Section 4.3.3 of NUREG/CR-2800.

13 Priority Ranking for Boron Injection Strategy and Recommendations

S. A. Hodge M. Petek

In this chapter, the accident frequency adjustment, consequence reduction, and cost estimates obtained by the steps described in Chap. 12 are applied to obtain the value/impact assessment and to establish a priority ranking for the proposed boron injection strategy for mitigation of boiling water reactor (BWR) severe accidents. These applications are made in accordance with the procedures specified in NUREG-0933, *A Prioritization of Generic Safety Issues*[18].

13.1 Frequency Estimate

Potential core-melt frequency reduction has been estimated in Table 12.1 by reference to the results of a Pacific Northwest Laboratories (PNL) analysis of *Recriticality in a BWR Following a Core Damage Event*, NUREG/CR-5653.[34] The PNL analysis is based upon the NUREG-1150 risk study[6] and uses the Peach Bottom Plant as a technical basis. Strictly speaking, the calculated reduction applies to the frequency of unmitigated core melting. The strategy proposed would, if implemented, affect the progression of severe accident events during the time window for recriticality, which is opened by the occasion of some core damage (the melting of the control blades). Thus, some core damage is associated even with successful implementation of the strategy. The goal of the strategy is to avert vessel breach and containment failure.

The proposed strategy would have the potential to affect the progression of the risk-dominant short-term and long-term station blackout accident sequences. It is assumed (following the PNL analysis) that implementation of the proposed strategy would reduce the frequency of criticality imposed by reactor vessel refill after control blade melting by 95%. The estimated change in frequency of unmitigated core melting is then 1.19E-06/RY.*

13.2 Consequence Estimate

The release category associated with conversion of a station blackout severe accident sequence, after core damage has occurred, into an uncontrolled criticality accident sequence is BWR-3. This release category (as defined in WASH-1400, Appendix VI) involves containment failure by overpressure. When the change in frequency of unmitigated core melting (Sect. 13.1) is multiplied by the public dose (5.1E+06) associated with release category BWR-3, the resulting estimated change in public risk is 6.06 man-rem/RY.

13.3 Cost Estimate

Implementation of the proposed strategy is estimated to involve expenditures (per plant) of $70,000 for engineering analysis, preparation of procedures, personnel training, management review, and acquisition of material (sodium borate powder in the form of Polybor®). In addition, it is estimated that 20 man-h/RY would be required for periodic procedure review and team training (including drills). With a cost of $56.75/man-h (1982 dollars) and an average remaining plant life of 21.1 years, the average industry cost per reactor is estimated to be about $93,950.

Nuclear Regulatory Commission (NRC) costs for implementation of the proposed strategy would be small because the general approach has already been developed by the Office of Research as a candidate accident management procedure. It is anticipated that the strategy would be implemented on a voluntary, plant-specific basis by the industry. Therefore, no additional NRC development costs would be incurred. Allowance is made, however, for the costs derived from oversight of the associated plant procedures and of the general readiness (status of personnel training) to successfully execute the plant-specific actions. These oversight activities are estimated as a cost per reactor of $2270 for support of strategy implementation and $227/year of remaining plant life for strategy maintenance. With an average remaining plant life of 21.1 years, the average NRC cost per reactor is estimated to be about $7100.

Based upon an average industry cost of $94,000/reactor and an NRC oversight cost of $7000/reactor, the total cost associated with implementation of this strategy for the 38 BWR facilities is estimated to be $3.84M.

13.4 Value/Impact Assessment and Priority Ranking

As indicated in Sect. 13.2, the estimated risk reduction associated with implementation of the proposed strategy is 6.06 man-rem/RY. Applying this estimate to the U.S.

*Here the nomenclature of NUREG-0933 is followed, using "RY" to represent reactor-year. In Chap. 12, the appellation "py" is used to represent plant-year, following NUREG/CR-2800. For the purposes of this report, the two terms are interchangeable.

inventory of 38 BWR facilities with an average remaining lifetime of 21.1 years, the total potential risk reduction is ~4860 man-rem.

With the total associated cost of about $3.84M derived in Sect. 13.3, the value/impact assessment consistent with the procedures of NUREG-0933 is

$$S = \frac{4860 \text{ man-rem}}{\$3.84M}$$

$$= 1266 \text{ man-rem}/\$M.$$

Section III.4.a of the Introduction to NUREG-0933 provides a priority ranking chart, reproduced here as Fig. 13.1 for the convenience of the reader. This chart shows how the tentative priority rankings are derived from the safety significance of an issue and its value /impact score. Entering this chart (Fig. 13.1) with a value of 1266 for S (the vertical axis) and value of 1.19E-06 core-melt/RY for the change in risk (horizontal axis—see Sect. 13.1), one obtains a priority ranking of MEDIUM for the proposed strategy.

13.5 Recommendations

Based upon the MEDIUM priority ranking for the proposed strategy, what further actions should be recommended? As pointed out in NUREG-0933, decisions should be tempered by the knowledge that the assessment uncertainties are generally large:

> The criteria and estimating process on which the priority rankings are based are neither rigorous nor precise. Considerable application of professional judgment, sometimes guided by good information but often tenuously based, occurs at a number of stages in the process when numerical values are selected for use in the formula calculations and when other considerations are taken into account in corroborating or changing a priority ranking. What is important in the process is that it is systematic, that it is guided by analyses that are as quantitative as the situation reasonably permits, and that the bases and rationale are explicitly stated, providing a "visible" information base for decision. The impact of imprecision is blunted by the fact that only approximate rankings (in only four broad priority categories) are necessary and sought.[18]

With these considerations in mind, it is recommended that each plant assess its need for the proposed strategy based upon the results of its individual plant examination (IPE). By far, the most important aspect of this recommended plant-specific assessment of the need for this strategy is the expected frequency of station blackout events that progress through the first stages of core damage (the melting of control blades). In the generic analysis of public risk reduction provided as item 6 of Table 12.1, the probability of a recriticality event was taken to be 1.25E-06/py, based upon the recent PNL study.[34]

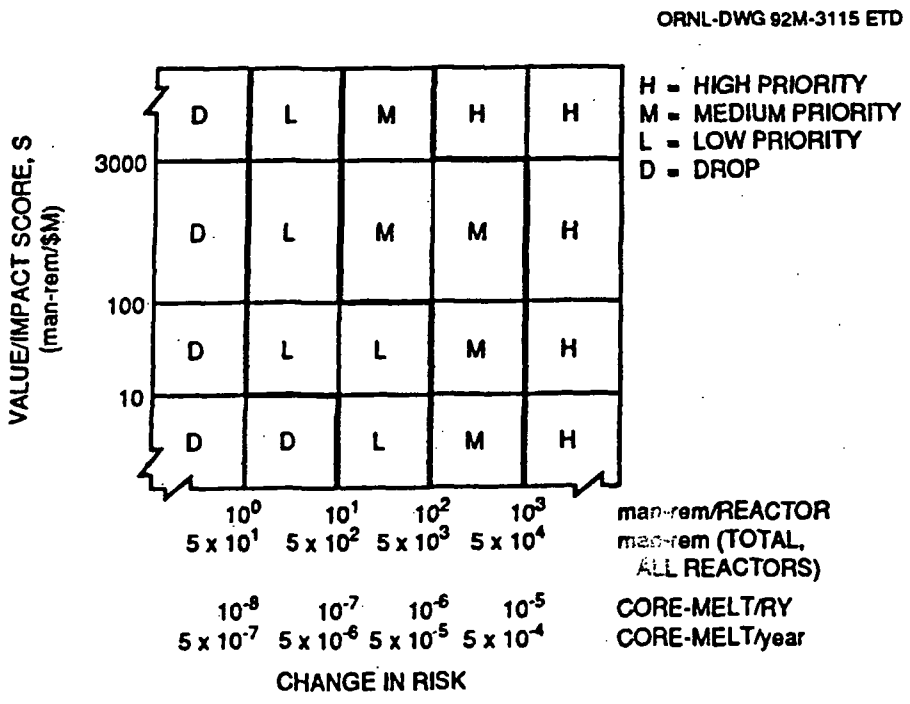

Figure 13.1 Priority ranking chart. ***Source:*** **Reproduced from Fig.1 of Introduction to NUREG-0933**

The PNL study is based upon the NUREG-1150 results for Peach Bottom, which include a core-melt frequency of ~4.5E-06/RY derived from station blackout events. If individual plants discover in their IPE process that a much lower station blackout core damage frequency applies, then correspondingly lower recriticality potential would also apply and implementation of the proposed strategy would probably not be practical for their facility.

As a final note, however, it is important to recognize that many of the BWR facilities are currently implementing accident management strategies, on a voluntary basis, to provide back-up capability for the standby liquid control system (SLCS). These back-up strategies invoke such methods as modification of the high-pressure coolant injection or reactor core isolation cooling system pump suction piping to permit connection to the SLCS tank or poisoning of the condensate storage tank. In all known cases, however, the effect of these plant-specific strategies is to provide a means to obtain a reactor vessel concentration of the boron-10 isotope similar to that attainable by use of the SLCS system itself. It seems highly desirable that these facilities should include within their training programs and procedural notes the information that, according to the analyses reported in NUREG/CR-5653, this concentration would be insufficient to preclude criticality associated with vessel reflood after control blade melting.

14 Containment Flooding as a Late Accident Mitigation Strategy

S. A. Hodge, J. C. Cleveland, T. S. Kress*

As discussed in Chap. 8, this effort for identification of new strategies has been subject to the constraint that candidate strategies should, to the maximum extent possible, make use of the existing equipment and water resources of the boiling water reactor (BWR) facilities and not require major equipment modifications or additions. One of the recommendations developed as a result of these studies calls for further assessment of a candidate strategy for containment flooding to maintain the core and structural debris within the reactor vessel if vessel injection cannot be restored as necessary to terminate a severe accident sequence.

14.1 Motivation for this Strategy

It is important to note that containment flooding to above the level of the core is currently incorporated within the *BWR Owners' Group Emergency Procedure Guidelines*[16] (EPGs) as an alternative method for providing a water source into the vessel in the event of design-basis loss-of-coolant-accident (LOCA) (the water would flow into the vessel from the containment through the break). The assessment undertaken here is to determine if containment flooding might also be effective in preventing the release of molten materials from the reactor vessel for the risk-dominant non-LOCA accident sequences such as station blackout.

The chief motivation for this assessment derives from the potential of the proposed strategy for application to the existing BWR facilities with the Mark I containment design. These are listed in Table 14.1 with their date of commercial operation, rated thermal power, and reactor vessel size. Because of its relatively small containment volume and drywell floor area, the Mark I containment structural boundary is particularly vulnerable to failure by overpressure or direct contact attack should core and structural debris leave the reactor vessel.

The proposed strategy for drywell flooding has the potential to serve not only as a first-line defense in preventing the release of molten material from the reactor vessel, but also as a second line of defense to prevent failure of the Mark I drywell shell should debris release from the reactor vessel occur. All current considerations of the Mark I shell melt-through issue[35] are based upon an assumption that the depth of water over the drywell floor would be limited to ~2 ft (0.610 m), the height at which overflow to the pressure suppression pool would occur. (This assumption derives from the limited pumping capacity available for containment flooding in the existing plants and the need to fill the pressure suppression pool before the water level could rise higher than this.)

While the study documented in Ref. 35 indicates that a water depth of 2 ft would protect the drywell shell (in the absence of steam explosions) from the *initial* debris release from the reactor vessel, the subsequent debris pours from the vessel would produce an island of debris leading to the shell. At some point, perhaps when about one-half of the core had left the vessel, the newly released debris would contact the shell above the water level and the shell would fail. However, drywell flooding to surround the lower portion of the reactor vessel with water would provide more than 30 ft (9.144 m) of water over the floor. This would preclude any possibility of direct failure of the drywell shell by late contact with debris and, therefore, has the potential to be an excellent late mitigation strategy, if the required pumping capacity can be provided.

14.2 Effects of Reactor Vessel Size

As indicated in Table 14.1, the reactor vessel sizes for the Mark I containment facilities range from 183-in. internal diameter (Duane Arnold) to 251-in. internal diameter (Browns Ferry, Peach Bottom). From the standpoint of the potential for removing decay heat by external cooling of the reactor vessel bottom head, an important measure of performance is the ratio of the plant rated thermal power to the internal surface area of the reactor vessel bottom head. This ratio varies from 37.4 MW/m^2 for Vermont Yankee to 51.6 MW/m^2 for the largest plants such as Peach Bottom. Because the heat transfer from a lower plenum debris bed would be by conduction through the vessel wall, the advantage of the smaller plants demonstrated here is magnified by consideration of the vessel wall thickness, which, as will be discussed in Chap. 16, is significantly less for the smaller reactor vessels.

The calculations discussed in this report have been performed, where appropriate, for the Browns Ferry, Hatch, Vermont Yankee, and Duane Arnold BWR facilities. This approach is intended to fully cover the spectrum of Mark I facility thermal capacities and reactor vessel dimensions.

*All work in connection with this project completed before becoming a member of the Advisory Committee on Reactor Safeguards.

Table 14.1 Mark I containment facilities in order of increasing reactor vessel size

Plant	Location	BWR	Commercial operation	Rated thermal power [MW(t)]	Reactor vessel internal diameter (in.)
Duane Arnold	Palo, IA	4	02/75	1593	183
Vermont Yankee	Vernon, VT	4	11/72	1593	205
Monticello	Monticello, NM	3	06/71	1670	206
Nine Mile Point 1	Scriba, NY	2	12/69	1850	213
Oyster Creek1	Forked River, NJ	2	12/69	1930	213
Cooper	Brownsville, NB	4	07/74	2381	218
Edwin I. Hatch 1	Baxley, GA	4	12/75	2436	218
Edwin I. Hatch 2	Baxley, GA	4	09/79	2436	218
James A. Fitzpatrick	Scriba, NY	4	07/75	2436	218
Brunswick 1	Southport, NC	4	03/77	2436	218
Brunswick 2	Southport, NC	4	11/75	2436	218
Pilgrim 1	Plymouth, MA	3	12/72	1998	224
Millstone 1	Waterford, CT	3	03/71	2011	224
Quad Cities 1	Cordova, IL	3	02/73	2511	251
Quad Cities 2	Cordova, IL	3	03/73	2511	251
Dresden 2	Morris, IL	3	06/70	2527	251
Dresden 3	Morris, IL	3	11/71	2527	251
Hope Creek 1	Salem, NJ	4	12/86	3293	251
Peach Bottom 2	Delta, PA	4	07/74	3293	251
Peach Bottom 3	Delta, PA	4	12/74	3293	251
Browns Ferry 1	Decatur, AL	4	08/74	3293	251
Browns Ferry 2	Decatur, AL	4	03/75	3293	251
Browns Ferry 3	Decatur, AL	4	03/77	3293	251
Fermi 2	Newport, MI	4	01/88	3293	251

14.3 Venting Requirement for Existing BWR Facilities

An undesirable aspect associated with the potential employment of this strategy for existing BWR facilities has an important effect upon the evaluation of the overall effectiveness and, therefore, should be mentioned in this introductory chapter. This disadvantage is the requirement for venting to the atmosphere while the containment is being filled by the low-pressure pumping systems (before the onset of core degradation) and the requirement that the drywell vents be kept open during the steaming from the water surrounding the reactor vessel lower head. As a direct consequence, employment of the drywell flooding strategy would require acceptance of an early noble gas release to the surrounding atmosphere as well as acceptance of the associated limited escape of fission product particulates. [These would enter the pressure suppression pool via the safety/relief valve (SRV) tailpipe T-quenchers and would be scrubbed by passage through the water in both the wetwell and the drywell.]

The requirement for venting associated with the proposed strategy is not unique among the currently considered mitigative measures for BWR severe accidents. For example, the containment drywell would also have to be vented for the strategy considered by Theofanous,[35] where attempts would be made to cool the debris on the drywell floor after it had left the reactor vessel. With only 2 ft (0.610 m) of water over the drywell floor, the released debris could be considered to be covered by water only for the early phase of the release from the vessel. Subsequently, the accumulating debris would rise above the water level with a direct path to the external atmosphere through the opened vents.

14.4 Computer Codes

The HEATING code[40] has been employed in this study for detailed analyses of reactor vessel wall response including conduction through the wall, the fin effect of the penetration housings, and the success of the water surrounding the housing tubes outside the vessel in preventing their failure by internal entry of molten material.

The response of the lower plenum debris bed after dryout and the effect of this bed upon the reactor vessel bottom head wall is calculated in this study by application of the coding developed at Oak Ridge within the BWRSAR code[2, 3] for this purpose. The lower plenum debris bed models, which have been made operational in the stand-alone mode, are currently being implemented into the MELCOR code framework at Oak Ridge. These models represent decay heating, conduction, and radiative heat transport within the debris bed as well as the effects of material melting, relocation, and freezing at a lower level. The formation of eutectic mixtures, defined by user input, is also represented. Additional information concerning these models is provided in Chap. 18.

14.5 Assessment Outline

If drywell flooding is to be effective in maintaining reactor vessel integrity, this strategy must be capable of quick implementation, because release of molten materials from the lower plenum debris bed to the drywell floor by means of failed penetration assemblies would otherwise be expected to occur soon after bottom head dryout. The general topic of drywell flooding, the means to accomplish it, and the effectiveness of this maneuver in cooling the reactor vessel exterior surface are addressed in Chap. 15.

While it can be shown that the submerged portions of the reactor vessel wall can be effectively cooled by the presence of water, there are physical limitations to the fraction of the overall decay power that can be removed downward through the lower portion of the debris bed boundary. These unfortunate realities are discussed in Chap. 16.

Cooling of the bottom head can greatly delay any failure of the reactor vessel structural boundary, but the first requirement to accomplish this goal is that failures of the penetration assemblies, induced by dryout and the entry of molten materials, be precluded. The success of water surrounding the vessel exterior in achieving this is described in Chap. 17.

The stand-alone models for the response of the lower plenum debris bed and the reactor vessel bottom head wall are described in Chap. 18. The results obtained by application of these models to the large BWR facilities such as Peach Bottom or Browns Ferry are discussed in Chaps. 19 and 20 for cases with and without venting of the atmosphere trapped within the reactor vessel support skirt.

Current provisions for reactor vessel depressurization as specified by the BWR EPGs are intended to lessen the severity of any BWR severe accident sequence. Therefore, it is important to recognize that drywell flooding, which would submerge the SRVs, might lead to failure of the dc power supply and thereby induce vessel repressurization. The potential for failure of SRV remote control due to drywell flooding and the effects of this eventuality are discussed in Chap. 21.

The summary and conclusions for the containment flooding strategy are summarized in Chap. 22. Application of the methodology to the smallest of the BWR facilities is demonstrated in Appendix C.

15 Method and Efficacy of Drywell Flooding

S. A. Hodge, J. C. Cleveland, T. S. Kress*

The objective of the proposed strategy for drywell flooding is to eliminate the threat of large releases of particulate fission products to the containment atmosphere that would be posed by core degradation and debris bed formation in the reactor vessel lower plenum and the associated challenge to the integrity of the vessel bottom head. This threat would be eliminated by providing a means to maintain the core and structural debris within the reactor vessel by cooling the vessel bottom head. To accomplish this goal, the water level within the drywell would have to be raised sufficiently quickly so that the lower portion of the reactor vessel would be submerged before any bottom head penetration assembly failures could occur. Furthermore, the direct availability of water to the reactor vessel exterior surface would have to be adequate to remove all of the fission product decay power that could be conducted through the wall. If these measures could be successfully implemented, then the drywell flooding strategy would be efficacious, although a penalty would be incurred in terms of early containment venting (to permit containment flooding) and the associated early release of fission product noble gases. Whether drywell flooding would prove effective in maintaining reactor vessel integrity over the long term will be the topic of subsequent chapters.

As will be explained in Chap. 16, the geometric effects of reactor vessel size are such that the effectiveness of external cooling of the vessel bottom head would be least for the largest vessels. Considering also that the motivation for maintaining any core and structural debris within the reactor vessel is greatest for the Mark I drywells, the primary focus of this assessment is upon the largest BWR Mark I containment facilities such as Peach Bottom or Browns Ferry.

15.1 Methods for Drywell Flooding

The concept of drywell flooding as a severe accident mitigation technique for boiling water reactor (BWR) applications has been briefly considered previously by a simple scoping analysis (Appendix D of Ref. 9) that indicated that water surrounding the reactor vessel bottom head would have the potential to prevent melting of the vessel wall. However, this analysis, which was based upon the Browns Ferry Nuclear Plant, also concluded that the existing pumping systems available for containment flooding would require too much time to fill the wetwell and then raise the water level within the drywell to surround the lower portion of the reactor vessel. Furthermore, these existing pumping systems would not be available for the dominant station blackout accident sequences.

To realistically provide a means for retention of core and structural debris within the reactor vessel, there would have to be a reliable ability to sufficiently flood the drywell within a short period of time, because emergency procedures cannot be expected to call for containment flooding (and the associated undesirable effects upon the installed drywell equipment) until after core degradation has begun. If the water did not reach the vessel bottom head until after lower plenum debris bed dryout and the beginning of heat-up of the vessel wall, it would be too late to prevent release of molten debris into the drywell by means of penetration assembly failures.

Despite the previously identified requirement for enhancement of the existing pumping systems, it seems worthwhile to again consider this strategy, especially in light of the current proposals[35] for preventing failure of the Mark I drywell shell (in case of debris release from the reactor vessel) by venting the containment and flooding the drywell floor with water, employing the existing drywell spray headers for this purpose. The analyses associated with these Mark I shell protection proposals are based upon attainment of a water level within the drywell extending only to the lower lip of the vent pipes [~2 ft (0.610 m) above the drywell floor], with the overflow entering the pressure suppression pool. Nevertheless, the necessary water would have to be capable of delivery to the drywell floor in case of station blackout; hence, there is an associated requirement for new or upgraded independently powered pumping systems.

Should it be decided to improve the conditional survival probability of the Mark I containment given core and structural release from the reactor vessel (a very improbable event[6]), then relatively minor modifications beyond the need for an independently powered dedicated pumping system might be employed to permit rapid filling of the wetwell, flooding of the vent pipes, and increase of the water level within the drywell to a height sufficient to cover the reactor vessel bottom head. If drywell flooding to this level could be achieved quickly enough, then the water in the drywell could provide two lines of defense against containment failure: first, by serving to keep the debris within

*All work in connection with this project completed before becoming a member of the Advisory Committee on Reactor Safeguards.

the reactor vessel and, second, by extending the protection of the drywell shell.

From the standpoint of providing the necessary volume of water, implementation of a strategy for maintaining the core and structural debris within the reactor vessel would require the availability of an independent containment flooding system of sufficient capacity to deliver the required amount of water before reactor vessel lower plenum debris bed dryout and the associated bottom head penetration failures could occur. As stated previously, this would in general require equipment modifications to existing plants, but similar modifications for rapid and effective drywell flooding would also be required by separate considerations in support of resolution of the Mark I shell failure issue.[35] In both cases, the drywell would have to be vented during the flooding process and beyond. The only additional requirement for the new or upgraded independently powered pumping systems necessary to deal with the (dominant) station blackout accident sequences would be with respect to increasing their capacity. Making allowance for the trapping of a portion of the containment atmosphere in the upper wetwell as indicated in Fig. 8.14, ~1,550,000 gal (5870 m^3) of water would have to be added to a Mark I containment of the size at Peach Bottom or Browns Ferry to submerge the reactor vessel hemispherical bottom head.

As will be explained in Chap. 19, the BWR severe accident sequence leading most rapidly to the formation of a reactor vessel lower plenum debris bed is short-term station blackout, for which the vessel bottom head would have to be submerged in no more than 150 min (2.5 h) after the onset of core degradation. For Browns Ferry or Peach Bottom, this is equivalent to a requirement for a pumping capacity of 10,000 gal/min (0.631 m^3/s).

Although the drywell would be vented during filling, the venting capacity may be limited so that a significant containment pressure would develop. (A procedure for venting during station blackout is provided in Ref. 41.) Allowing for a containment backpressure of 60 psig (0.515 MPa), an elevation head of 80 ft (24.38 m), and a pump efficiency of 70%, the required delivery of 10,000 gal/min (0.631 m^3/s) could be provided by an 800-bhp (0.60-MW) diesel. BWR facilities with containments smaller than those at Peach Bottom or Browns Ferry would, of course, require correspondingly smaller pumping capacities and driving horsepower.

15.2 Water Contact with the Reactor Vessel Wall

If a water level were established within the drywell sufficient to cover the lower portion of the reactor vessel, would the vessel insulation significantly impede the availability of water to the surface of the vessel wall? The reactor vessel insulation is an all-metal reflective insulation that does not tightly adhere to the vessel wall. Over the cylindrical shell of the vessel, it is 3 in. (0.076 m) thick and is held off the wall by support brackets (Fig. 15.1). Over the bottom head, the reflective insulation forms a cylindrical boxlike enclosure. The horizontal lower cap (disk) of this enclosure is 3 in. (0.076 m) thick with a 6-in. (0.152-m) separation* between its upper surface and the lowest point of the vessel wall. This insulation is "designed to permit complete submersion in water without loss of insulating material, contamination from the water, or adverse effect on the insulation efficiency of the insulation assembly after draining and drying."[42] More than 240 penetrations pass through this lower disk.

The results of simple experiments and supporting calculations to demonstrate that the standard reflective insulation would pose no significant impediment to leakage of water in sufficient quantities to cool the vessel wall have been provided in a recent paper by Henry.[43] Furthermore, this important paper [based upon pressurized water reactor (PWR) considerations] shows conclusively that the mode of heat transfer on the outer surface of the bottom head would be nucleate boiling and that the generated steam could escape from the confined space between the vessel wall and the inner surface of the insulation.

The central problem with attempting to remove the debris decay heat through the reactor vessel bottom head lies not with limitations to the heat removal rate at the outer surface, but rather with the limited conduction through the wall. This is particularly true for the BWR, as will be shown.

15.3 Atmosphere Trapping Beneath the Vessel Skirt

The weight of the BWR reactor vessel is carried by a vessel support skirt, as shown in Figs. 15.2 and 15.3. For the Browns Ferry reactor vessels, the skirt is 230 in. (5.842 m)

*This separation distance is plant-specific. The 6-in. (0.152 m) value applies to the Browns Ferry reactor vessels whereas the Peach Bottom separation is 2 in. (0.051 m). Because the greater separation distance (larger trapped air volume) tends to inhibit the effectiveness of drywell flooding, the calculations of this report have been based upon the Browns Ferry configuration.

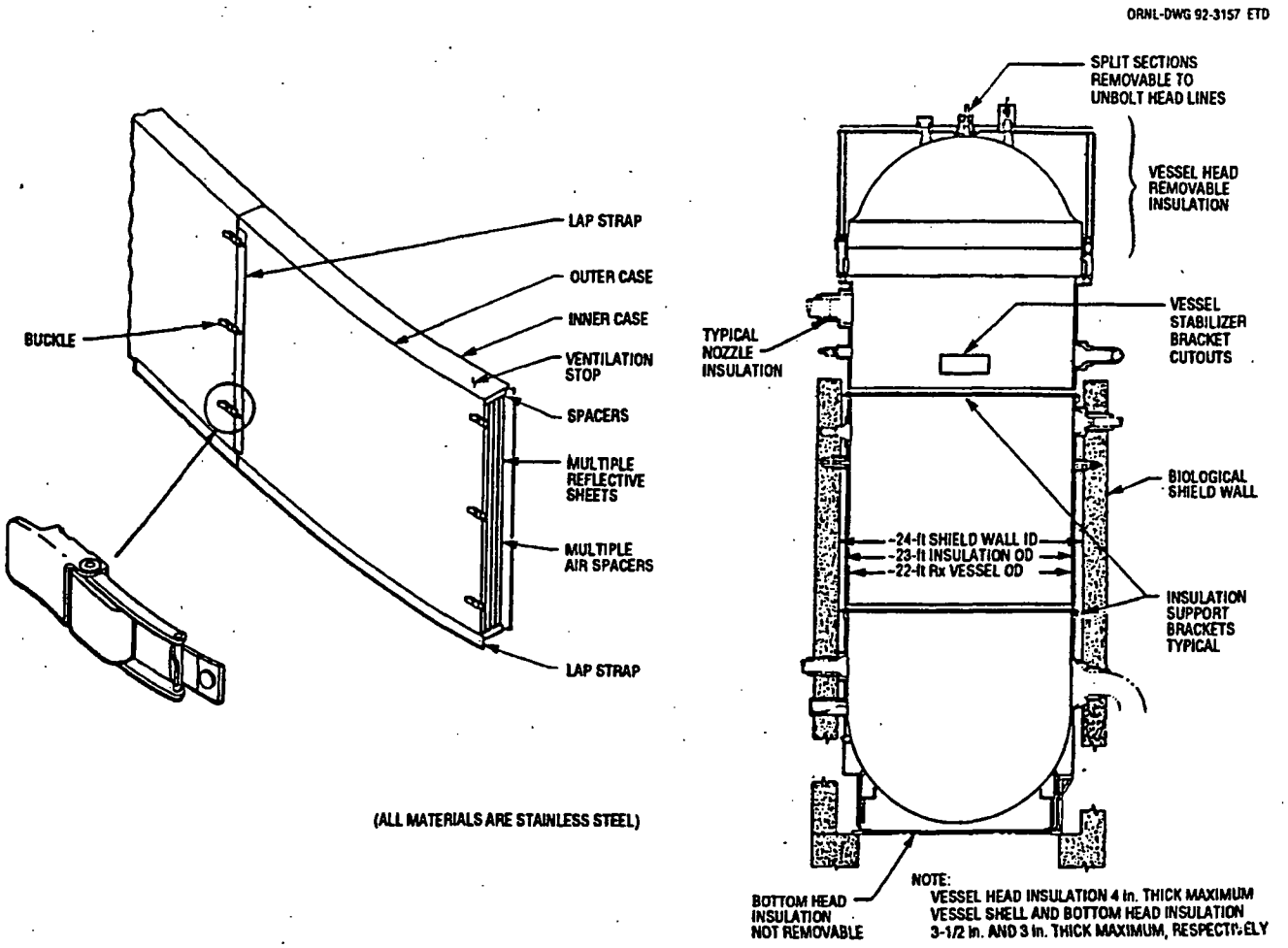

Figure 15.1 Reflective (mirror) insulation comprised of layered stainless steel panels held together by snap buckles but not watertight

in diameter, 1 in. (0.0254 m) thick, and has a vertical length of 77-3/8 in. (1.965 m) from the base of the lower flange to the vessel attachment weld. If the containment were flooded with water, a portion of the drywell atmosphere would be trapped within this skirt, as indicated in Fig. 15.4. Consequently, the effectiveness of the water in cooling the portion of the reactor vessel beneath the skirt attachment would be less than for the vessel wall above the skirt.

To assess the potential for water cooling beneath the skirt, it is necessary to estimate the height that the water would attain within the enclosed region. The first step in developing this estimate is to ascertain the uppermost point at which the atmosphere being compressed by an increasing drywell water volume could escape from the interior region of the reactor pedestal. As indicated in Fig. 15.5, there are four large openings in the upper pedestal to pass the numerous (370) individual hydraulic supply and return lines for the control rod drive mechanisms. If these openings constituted the highest atmosphere escape pathways, the trapped gas volume would prevent the water from reaching the bottom of the vessel.

The next higher possible escape path is at the juncture between the vessel support skirt and the vessel support ring girder (both are labeled on Fig. 15.2). The structure of this juncture is best illustrated within the shaded (underwater) area of Fig. 15.4. As shown, the lower flange of the vessel support skirt is bolted to the upper flange of the ring girder. The ring girder, in turn, is fastened to the concrete support pedestal by means of steel anchor bolts set in the concrete with the threads extending upward above the horizontal surface.[44] During construction, steel sole plates are set flat and level on the concrete. The lower flange of the ring girder is then set on top of the sole plate and shimmed as necessary to level the ring girder. (The anchor bolts extend through both the sole plates and the bottom flange of the ring girder.)

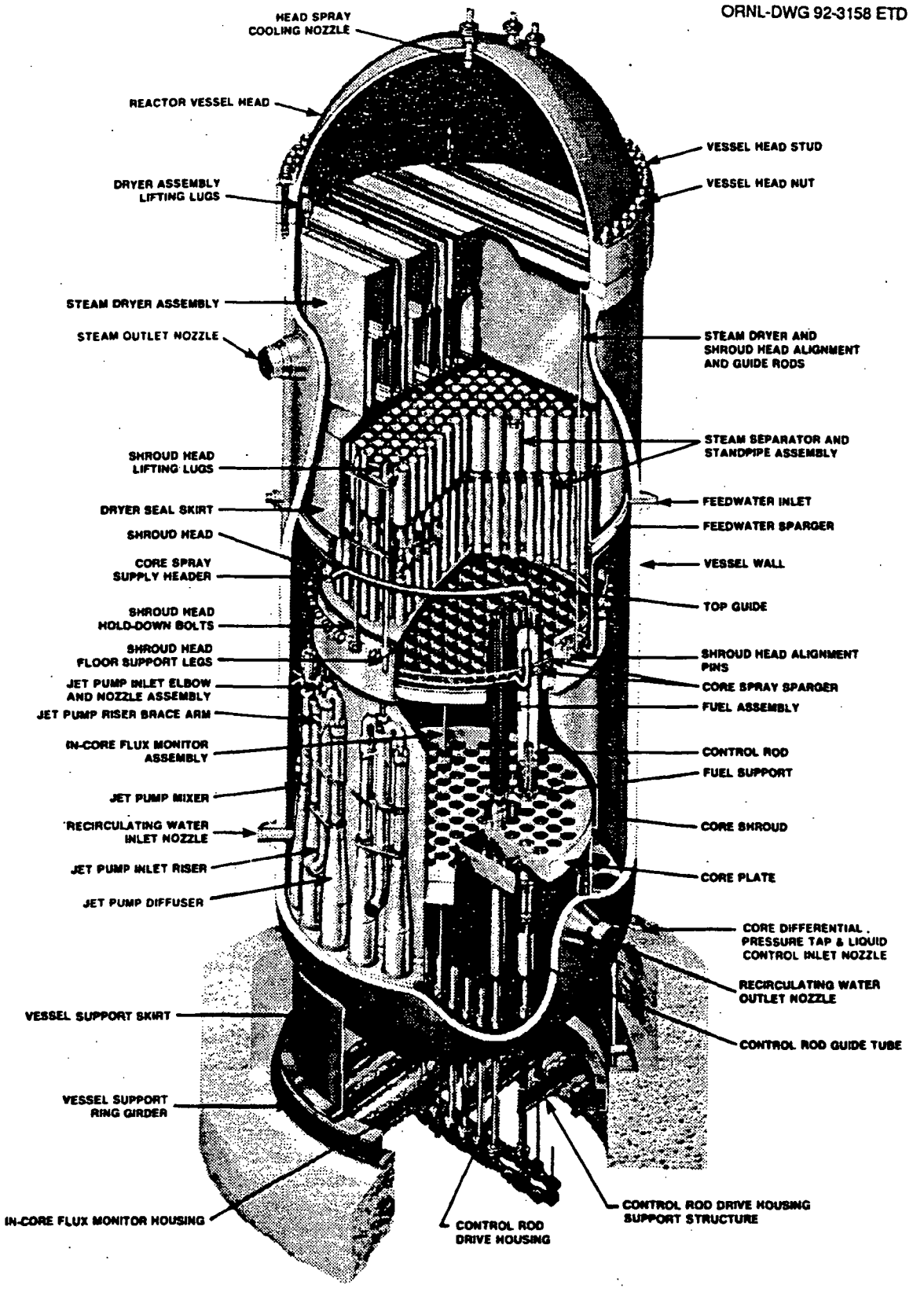

Figure 15.2 Cutaway drawing of BWR-4 reactor vessel

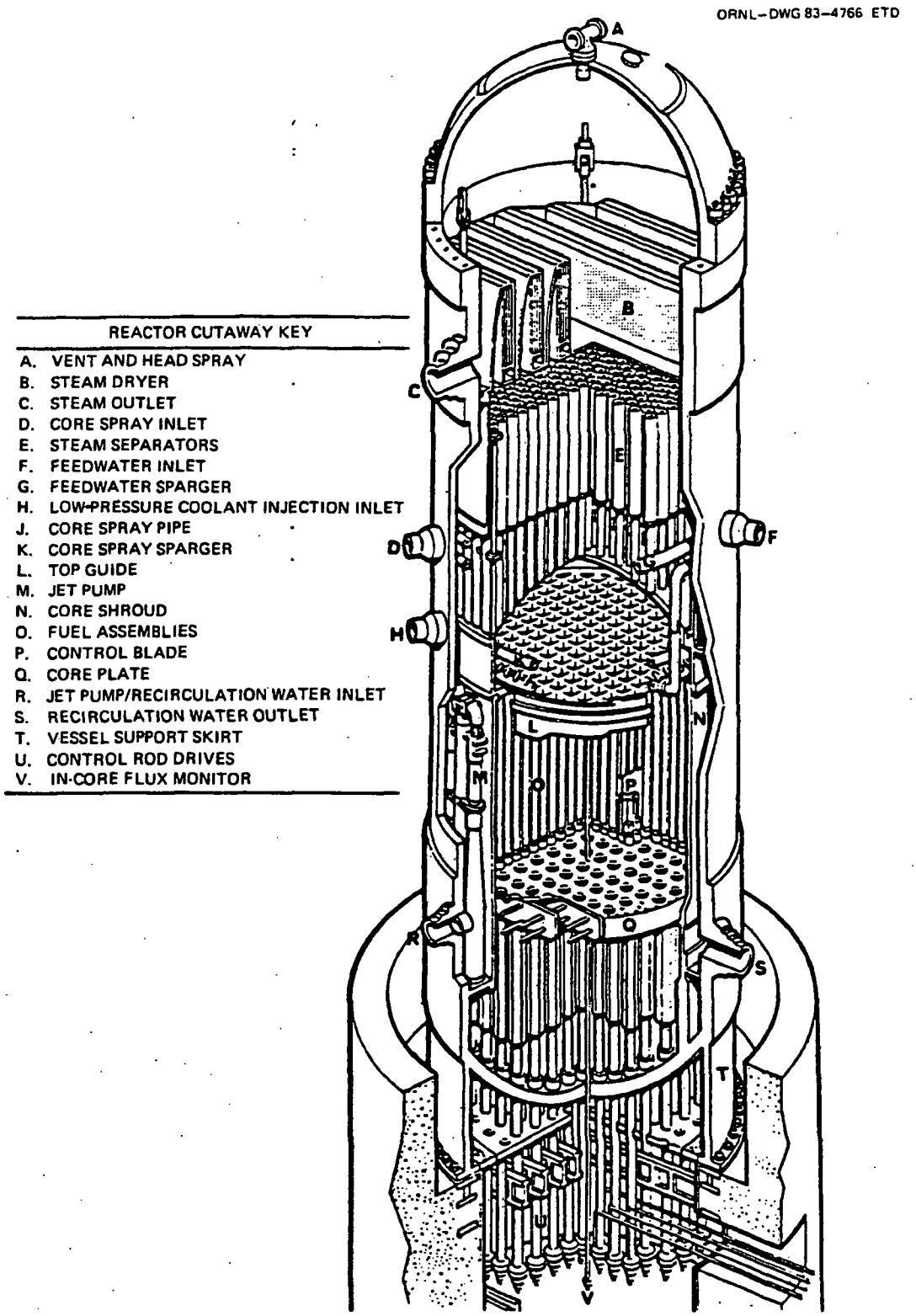

Figure 15.3 Reactor vessel support skirt (T) that transmits weight of reactor vessel to concrete reactor pedestal

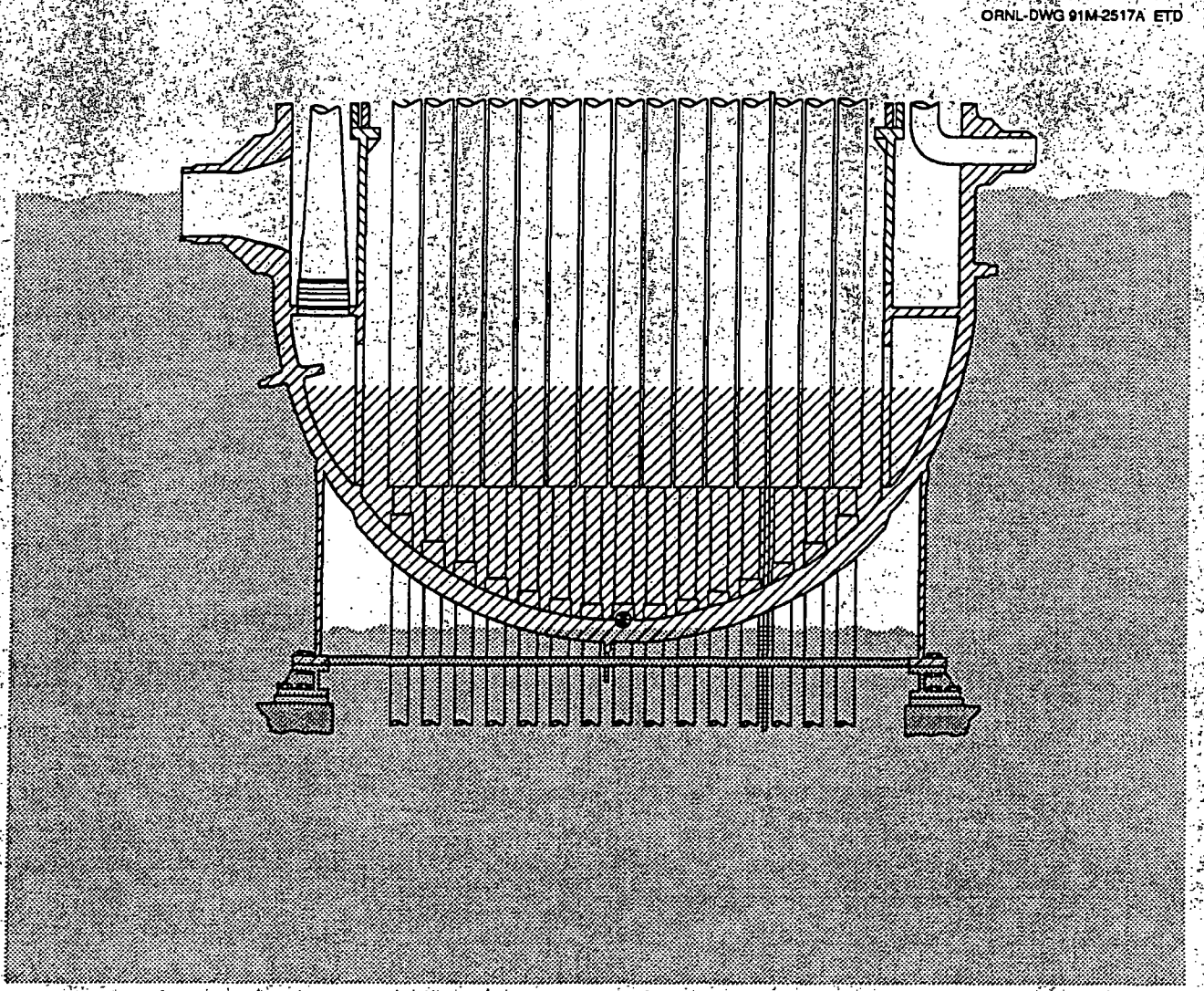

Figure 15.4 Atmosphere trapping within reactor vessel support skirt that would limit water contact with vessel wall in that region

Thus, no effort is made during construction to render air-tight either the juncture between the vessel support skirt lower flange and the support ring girder, or the juncture between the support ring girder and the concrete pedestal. In fact, the presence of shims at the latter juncture would seem to guarantee that gas leakage paths would be present. Accordingly, it is assumed for the purpose of this analysis that drywell flooding could raise the water level within the reactor pedestal to the bottom of the reactor vessel support flange unimpeded by local gas compression.

With the assumption of gas leakage at the lower surface of the reactor vessel support skirt flange, allowance for the volume occupied by the penetration assemblies descending from the bottom head, and application of the ideal gas law at constant temperature, the water height within the skirt can be calculated for *steady state conditions* as a function of the water height outside the skirt. The results, provided in Tables 15.1 and 15.2, are strongly dependent upon the assumed containment pressure, which in the actual case would be a time-dependent function of the balance between steam generation in the drywell (by heat transfer to the water surrounding the bottom head) and the capability of the drywell vents to release the steam.

As indicated in Table 15.1, a water level within the drywell at the height of the recirculation suction nozzle centerline would produce a steady-state water level within the skirt of 3.2 in. (0.081 m) above the low point of the outer surface of the reactor vessel bottom head. This is the situation depicted in Fig. 15.4, which corresponds to an assumed containment pressure of 20 psia (0.138 MPa). It is important to note from Table 15.2 that for a containment pressure of 40 psia (0.276 MPa), this same water level of 161.5 in.

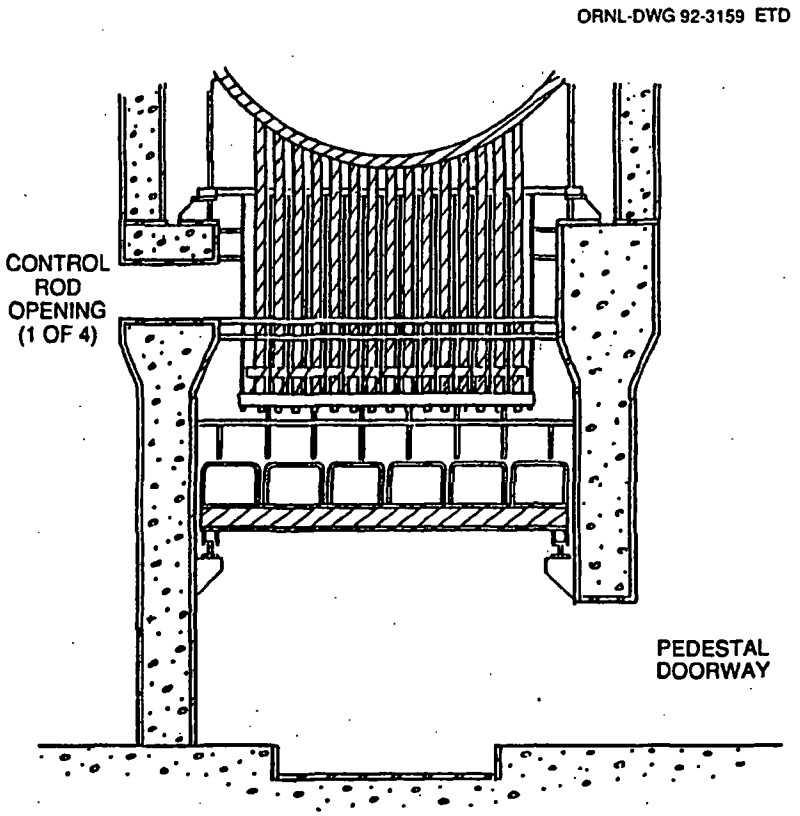

Figure 15.5 Uppermost openings between interior and exterior regions of the drywell pedestal for CRDHS supply and return lines

Table 15.1 Water level within the vessel skirt as a function of water level in the drywell for the Browns Ferry containment at 20 psia

Water level outside skirt		Water level inside skirt
Location of surface	Height relative to vessel zero[a] (in.)	Height relative to low point of vessel outer surface (in.)
Base of skirt	−14.375	−5.94
	90.0	0.0
Bottom head center of curvature	125.5	1.6
Recirculation nozzle centerline	161.5	3.2
Base of core	216.3	5.2
Core midplane	291.3	7.9
Top of core	366.3	10.2

[a]Vessel zero is the lowest point on the internal surface of the reactor vessel bottom head.

Table 15.2 Water level within the vessel skirt as a function of water level in the drywell for the Browns Ferry containment at 40 psia

Water level outside skirt		Water level inside skirt
Location of surface	Height relative to vessel zero[a] (in.)	Height relative to low point of vessel outer surface (in.)
Base of skirt	-14.375	-5.94
Bottom head center of curvature	125.5	-1.6
Recirculation nozzle centerline	161.5	-0.7
	188.5	0.0
Base of core	216.3	0.7
Core midplane	291.3	2.4
Top of core	366.3	3.9
Top of separators	607.5	8.3

[a]Vessel zero is the lowest point on the internal surface of the reactor vessel bottom head.

(4.102 m) above vessel zero (recirculation pump suction nozzle centerline) in the drywell would produce a steady-state water level within the skirt that did not quite reach the lower surface of the bottom head. Hence, the importance of maintaining the containment pressure as close to atmospheric pressure as possible during the period of drywell flooding.

In actuality, the situation within the vessel support skirt would be far from steady state. While a calculation of the time-dependent containment pressure and corresponding water level and gas temperature within the vessel skirt might be undertaken, such an endeavor does not seem worthwhile. Superimposed over the water level variation that would be predicted by this calculation would be the chugging cycle established by the generation of steam within the skirt; the temporary expulsion of the water, the condensation of steam on the water (and inner skirt) surface, and the reentry of the water to again contact the bottom head. Before proposing a detailed analysis of the cyclic variations of the conditions within the vessel skirt, it is prudent to first determine whether the integrity of the bottom head could tolerate the existence of any surface region without water contact. This determination will be made in Chap. 19.

15.4 Means for Venting the Vessel Skirt

The fraction of the bottom head surface area beneath the vessel skirt that is submerged in water could of course be increased by providing escape pathways for the trapped atmosphere (and generated steam) at elevations higher than the skirt lower flange. There are three conceivable means of doing this.

The simplest means of providing an elevated gas release pathway would be to take advantage of the existing access hole. The location of this 18-in.-(0.457-m) diameter hole for the Browns Ferry reactor vessel is indicated on Fig. 15.6. The top of the opening is ~26 in. (0.660 m) above the bottom of the skirt, or 12 in. (0.305 m) above vessel zero. The vertical distance along the skirt from the top of the opening to the attachment weld to the vessel is 48 in. (1.219 m).

The access hole is normally sealed by a bolted cover during power operation, and because no information to the contrary is available, it must be assumed to be gas-tight. However, if the upper bolts were loosened, then interpolation between the free volumes listed in Table 15.3 for water heights of 24 and 30 in. above the support skirt flange provides the result that the atmosphere volume initially trapped by the rising water would be 346 ft^3 (9.798 m^3). This may be compared with an initial trapped gas volume of 853 ft^3 [24.154 m^3 (first entry of Table 15.3)] for the case in which the uppermost gas escape pathway is at the lower flange of the vessel skirt. Clearly, much more of the bottom head would be covered with water if the upper bolts on the access hole cover were loosened. The results of calculations for the case with gas escape at the access hole will be considered in Chap. 20.

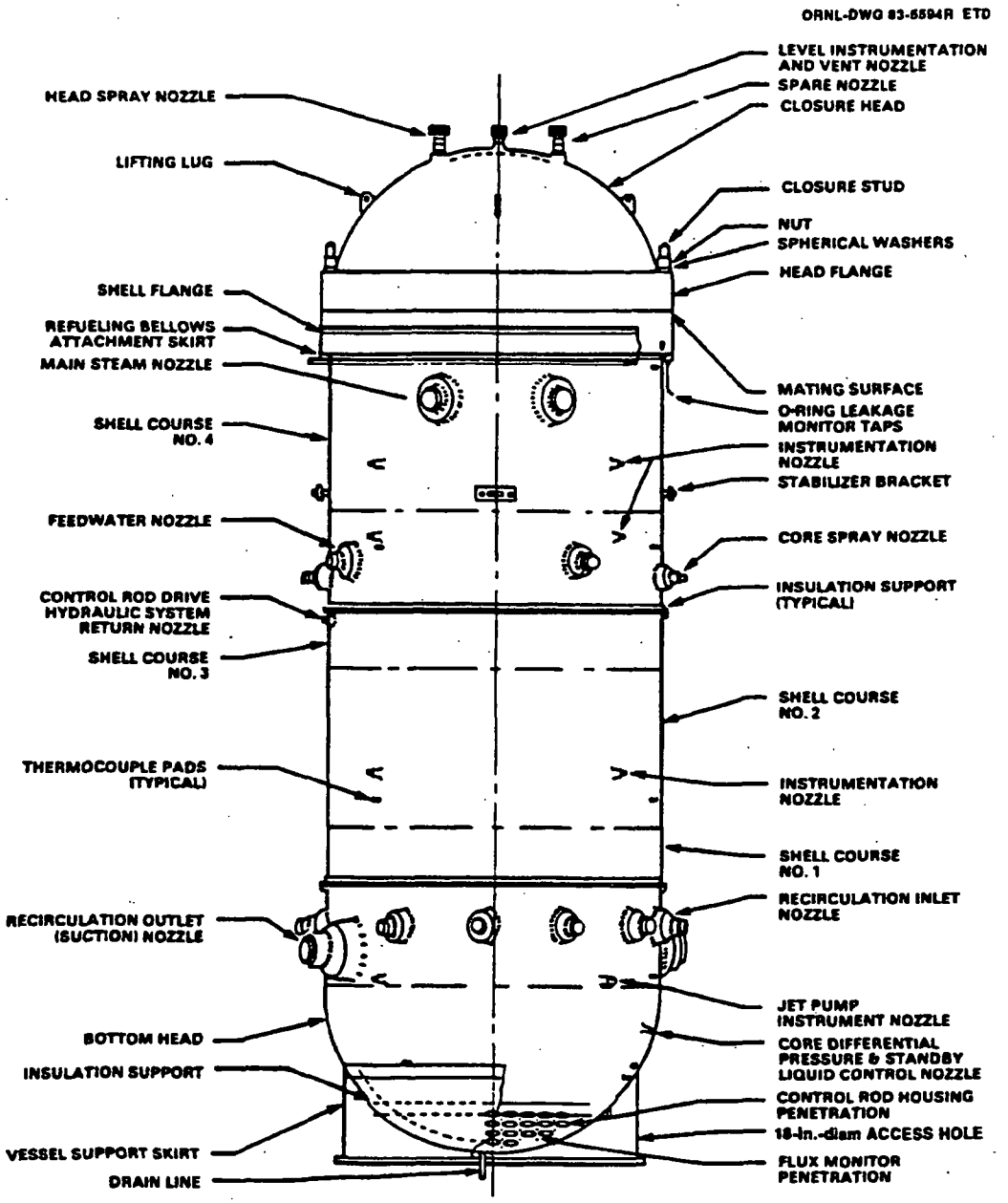

Figure 15.6 External appurtenances to Browns Ferry reactor vessel and indicating location of vessel support skirt access hole

Note that the previous discussion is based upon the vessel skirt access hole configuration at Browns Ferry. While it is believed that all BWR reactor vessel support skirts are fitted with access holes, the location, size, shape, and number of these holes and the type of cover used for sealing during plant operation are plant-specific. Therefore, the potential of each design for gas leakage and the practicality of providing for an elevated leakage path (such as by loosening cover bolts) are also plant-specific considerations.

A second means of venting the trapped atmosphere could be accomplished by methods only slightly more complicated than the first. This would be by providing a siphon tube (or tubes) leading from the upper armpit region of the skirt down through the highest available skirt access hole and upward through the outer drywell. While such a simple arrangement would accommodate the escape of noncondensible gases during the relatively slow initial drywell flooding, it would not be expected to survive the chugging

Table 15.3 Free volume within the vessel skirt above the waterline for the Browns Ferry reactor vessel

Height of waterline above support skirt flange (in.)	Total volume above waterline (ft^3)	Volume occupied by penetration assemblies (ft^3)	Free volume above waterline (ft^3)
0.0	899.3	46.3	853.0
5.9375[a]	752.0	27.7	724.3
6.0	750.5	27.5	723.0
12.0	610.4	15.1	595.3
18.0	487.4	8.3	479.1
24.0	379.7	4.0	375.7
30.0	287.8	1.4	286.4
36.0	209.6	0.0	209.6
42.0	145.7	0.0	145.7
48.0	94.0	0.0	94.0
54.0	54.8	0.0	54.8
60.0	26.4	0.0	26.4
66.0	8.9	0.0	8.9
72.0	0.7	0.0	0.7
74.375[b]	0.0	0.0	0.0

[a]Height at which water first comes into contact with the bottom head.
[b]Height of the skirt attachment weld to the vessel.

process of steam generation, bubble collapse, and water reentry within the skirt region unless a significant commitment were made to system design and reliability assurance. It is not reasonable to expect such a commitment for existing BWR facilities based upon their small numbers and the low core-melt probabilities predicted by the NUREG-1150 study[6].

The third (and most effective) way in which an elevated gas release pathway from the vessel skirt might be provided is by drilling several small holes through the skirt at points just below the attachment weld. From the standpoint of regulatory requirements, the NUREG-1150 results, cost-benefit analyses, radiation exposure to personnel, etc., this is clearly not a practical proposal for existing facilities; it might be considered, however, for advanced plant designs. Gas escape through the upper support skirt would permit a drywell flooding strategy to completely cover the BWR reactor vessel bottom head with water. The results of calculations for the case with water coverage of the entire vessel bottom head will be discussed in Chap. 20.

16 Physical Limitations to Bottom Head Heat Transfer

S. A. Hodge, J. C. Cleveland, T. S. Kress*

This chapter provides the results of scoping calculations performed to investigate the important physical limitations to lower plenum debris bed heat removal by means of cooling the reactor vessel bottom head and to demonstrate the associated effects of vessel size. Conduction through the vessel wall is addressed in Sect. 16.1, while the effect of natural circulation within the molten central portion of the debris bed is discussed in Sect. 16.2.

16.1 Heat Conduction Through the Vessel Wall

Simple hand calculations were performed to determine an upper bound for the ability to remove heat from a lower plenum debris bed through the hemispherical lower head of the reactor vessel of the Browns Ferry Nuclear Plant by flooding the containment. The following equation for heat flow is used for the hemisphere:

$$q = \frac{2\pi k r_0 r_i (T_i - T_0)}{(r_0 - r_i)} ,$$

with the following assumptions:

1. constant vessel wall conductivity k (independent of temperature),
2. the inner surface of the hemisphere at the melting temperature 2800°F (1811 K) of carbon steel, and
3. the outer surface of the hemisphere at the saturation temperature 267°F (404 K) of water at 40 psia (0.276 MPa).

This approach is based upon nucleate boiling heat transfer between the vessel wall outer surface and the surrounding water and therefore provides an upper limit of the capability for heat removal by conduction, without melting of the inner surface of the reactor vessel bottom head. Results show that ~17.2 MW would be removed by conduction through the vessel wall under these ideal circumstances. Because the decay heat generation rate of the core debris is in this same range, removal of the decay heat through the intact wall of the vessel bottom head is, from the standpoint of conduction, theoretically possible.

Additional conduction heat transport calculations were performed using a simple HEATING-7 model to compare the capabilities for removal of decay heat from a lower plenum debris bed through the bottom head for the Browns Ferry, Hatch, Vermont Yankee, and Duane Arnold reactor vessels. As were the hand calculations, these calculations were steady-state analyses with the inner surface of the lower head held at 2800°F (1811 K) and the outer surface held at 267°F (404 K). In contrast to the hand calculations, these machine calculations included consideration of temperature-dependent conductivity for the vessel wall. Results of these comparative calculations are shown in Table 16.1, where they are compared with the decay heat generation of the debris in the lower plenum at various times following accident initiation. Results show a larger

*All work in connection with this project completed before becoming a member of the Advisory Committee on Reactor Safeguards.

Table 16.1 Maximum possible conduction heat transport compared with debris bed heat generation

Parameter	Browns Ferry	Hatch	Vermont Yankee	Duane Arnold
Vessel wall thickness (in.)				
Shell	6.313	5.531	5.187	4.593
Region of penetrations	8.438	7.375	7.250	7.250
Bottom head radius r_i (in.)	125.5	109.0	102.5	91.5
Maximum possible heat removal rate through reactor vessel bottom head (MW)	17.6	14.7	14.2	12.1
Decay heat generation rate (MW) in debris				
600 min	16.4	12.1	7.9	7.9
800 min	15.1	11.2	7.3	7.3
1000 min	14.0	10.4	6.8	6.8

(favorable) margin between the maximum possible heat removal rate through the lower head and the debris bed decay heat generation rate for the smaller plants (e.g., Vermont Yankee and Duane Arnold).

As indicated, the calculations performed to produce the results listed in Table 16.1 did provide proper consideration of the increase in wall thickness in the region of the penetrations. The effect of the penetrations themselves, however, has been neglected. The justification for this is provided by the following argument.

Figure 16.1 is a photograph showing the region of penetrations at the lower portion of the reactor vessel bottom head for the Grand Gulf Unit 2 facility (never completed). The large [6-in. (0.152-m)] openings are for the control rod (CRD) drive housing penetrations, while the smaller [2-in. (0.051-m)] openings are for the instrument tube housings. While the metal removed to permit passage of the housings reduces the surface area for conduction heat transfer, the internal stub tubes and the housings themselves would tend to act as fins promoting heat transfer through the wall.

To determine whether the net effect of the penetrations and housings is to promote or detract from conduction heat transfer, heat transport calculations were performed with the HEATING-7 code. These unit cell calculations determine the heat transfer rates through the vessel bottom head with and without explicit representation of the stub tube and the CRD housing. Explicit representation of the CRD penetrations was found to reduce the calculated heat transport through the thickened section of the reactor vessel bottom head by 3.4%. Although this effect is sufficiently small to be neglected in view of other uncertainties, it can efficiently be represented, should one choose to do so, simply by reducing the input thermal conductivity for the thickened region of the wall by 3.4%. This adjustment has not been implemented for the calculations discussed in this report.

The primary purpose of the calculations reported in this section has been to demonstrate the effect of vessel size for the set of BWR facilities with the Mark I containment design. While the assumption of nucleate boiling at the submerged outer surface of the vessel wall is valid,[43] the assumption that the wall would remain intact with an inner surface temperature of 2800 °F (1810.9 K) is not. The portion of the reactor vessel bottom head beneath the skirt would be in tension* and would fail by creep rupture at temperatures significantly below this.†

16.2 Calculations for a Molten Hemisphere Contained Within its Own Crust

In this section, calculations recently performed by Henry[43] for the case of a pressurized water reactor (PWR) will be repeated with appropriate modification for application to the BWR, where the dimensions of the reactor vessel bottom head are much larger and the debris therein would include much more stainless steel with a significantly lower volumetric heat generation rate. The semi-empirical bases for the calculations reported by Henry are explained by Epstein,[45] who considered the (approximately) hemispherical volume of crusted molten material that formed within the Three Mile Island Unit 2 core.

Although the area Ad for downward heat transfer through the rounded lower surface of a hemisphere is twice the area Au of the upper surface disk, the analysis demonstrates the effects of natural circulation within the molten central region of the hemisphere in causing most of the heat transfer to occur through the upper surface.

Slightly modifying the approach of Henry,[43] the Rayleigh number for a heat generating hemispherical pool can be defined as

$$Ra = \frac{g \beta Q R^2}{2/3 \pi \alpha \upsilon K} .$$

The upward heat flux from the molten central region of the debris pool is

$$q_u = 0.36 \frac{K \Delta T}{R} Ra^{0.23} ,$$

while the downward heat flux is

$$q_d = 0.54 \frac{K \Delta T}{R} Ra^{0.18} .$$

Here ΔT is the superheat of the molten region relative to the inner surface temperature of the solidified crust.

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* The effectiveness of the water in cooling the portion of the bottom head beneath the vessel skirt attachment would be less than for the vessel wall above the skirt because of the trapping of drywell atmosphere within the skirt as the water level was raised within the drywell. With the reactor vessel depressurized, the portion of the vessel wall above the skirt attachment would be in compression.

† Creep rupture tests sponsored by the NRC to determine the susceptibility of carbon steel at very low stress levels are currently underway at Idaho National Engineering Laboratory. Available information concerning test results will be discussed in Sect. 18.2.3.

ORNL-PHOTO 6403-92

Figure 16.1 Lower portion of BWR reactor vessel bottom head showing control rod drive and instrument tube penetrations and the drain nozzle

Physical

The ratio of the downward to the upward heat transfer for the crusted molten hemisphere can then be expressed as

$$\frac{Qd}{Qu} = \frac{q_d A_d}{q_u A_u} = \frac{3.0}{Ra^{0.05}} .$$

The Rayleigh number depends upon the properties of the molten material, the decay power, and the hemisphere radius. As will be shown, the threat to the integrity of the BWR reactor vessel bottom head wall would begin about 10 h after scram (from 100% power). Accordingly, the property values listed in Table 16.2 are used to evaluate the Rayleigh number. Note that the values for debris thermal diffusivity and thermal conductivity are larger than those employed by Henry,[43] to reflect the much higher content of metals that would be included in BWR debris.

With the parameters listed in Table 16.2, the Rayleigh number can be expressed in terms of plant-specific values as

$$Ra = 2.2 \times 10^7 Q R^2 ,$$

where Q is the decay power in the bottom head debris (W) and R is the radius (m) of the bottom head.

Table 16.2 Parameters for evaluation of the Rayleigh Number for BWR core debris 10 h after scram

Parameter	Definition	Value
g	Acceleration of gravity	9.8 m/s^2
β	Thermal expansion coefficient	10^{-4} K^{-1}
α	Debris thermal diffusivity	2.9×10^{-6} m^2/s
ν	Debris kinematic viscosity	6×10^{-7} m^2/s
K	Debris thermal conductivity	12.4 W/m K

For a BWR reactor vessel of the size of Peach Bottom or Browns Ferry, the radius of the bottom head is 3.188 m, and the decay heat 10 h after scram would be 16.4 MW. Accordingly,

$$Ra = 3.7 \times 10^{15} ;$$

$$\frac{Qd}{Qu} = 0.50 ;$$

and one-third of the total decay heat is predicted to be transferred downward under steady-state conditions.

For the smallest BWR reactor vessel (Duane Arnold), the radius of the bottom head is 2.234 m, and the decay heat 10 h after scram would be 7.9 MW. Accordingly,

$$Ra = 8.7 \times 10^{14} ;$$

$$\frac{Qd}{Qu} = 0.54 ;$$

and the situation for downward heat transfer is only slightly more favorable, with 35% of the total decay heat predicted to be removed by this pathway.

The basic difficulty is that the internal natural convective motions of an internally heated molten pool cause the hottest liquid to rise toward the upper surface. The net effect of these internal flows in promoting heat transfer through the upper surface and limiting heat transfer through the lower surface cannot be significantly altered, no matter how efficiently the lower surface is cooled.

This question of the relative magnitudes of the upward and downward heat transfers from a molten pool in a reactor vessel lower plenum with external cooling of the bottom head has been again considered in two recent papers.[46,47] Both papers address the PWR vessel configuration and expected lower plenum debris characteristics. Both papers confirm that although the reactor vessel bottom head wall could be effectively cooled by nucleate boiling heat transfer to surrounding water, the preponderance of heat transfer from the debris pool would be projected upward within the reactor vessel. Neither of these papers addresses the consequences of melting of the upper reactor vessel internal structures and the addition of this molten material to the debris bed.

O'Brien and Hawkes[46] point out that turbulent conditions within the reactor vessel lower plenum debris pool might permit as much as 50% of the volumetric heat generation within the pool to be taken out by external cooling of the bottom head. They predict turbulent conditions within the lower plenum molten pool based upon a volumetric heat generation rate of 1.0 MW/m^3 and the expected characteristics of PWR debris. The much larger metallic content of BWR debris would, however, reduce its volumetric heat generation rate to no more than 0.40 MW/m^3; hence, turbulence in the BWR debris pool would not be expected.

Note that the theoretical approach (based upon the available experimental information) discussed in this section is not strictly applicable to the case at hand, because steady-

state conditions would never be attained in the BWR bottom head debris bed. This is because, without sufficient heat transfer (such as would be provided by overlying water), the upper crust would melt away so that heat transfer from the upper surface of the liquid pool by radiation to the upper reactor vessel internal stainless steel structures could occur. These structures would then melt with the resulting liquid metal being added to the debris pool, increasing both its depth and its metallic content.

Nevertheless, consideration of the predictions afforded by these simple hand calculations is useful in providing a basis for understanding the specific code results that will be discussed in Chap. 19. There it will again be seen that although the bottom head comprises two-thirds of the debris bed surface area (and is effectively cooled), less than two-thirds of the total heat transfer from the bed would follow this pathway.

17 Integrity of Bottom Head Penetrations

S. A. Hodge, J. C. Cleveland, T. S. Kress*

Because of the large number of penetrations, a boiling water reactor (BWR) vessel bottom head is highly perforated, as indicated in Fig. 17.1. In the unlikely event that an unmitigated severe accident should lead to formation of a lower plenum debris bed, failure of these penetrations after bed dryout might provide a release pathway for debris from the reactor vessel. This postulated release pathway would be initiated by molten material forming within the central portion of the debris bed, entering a locally failed (by melting) portion of a penetration guide tube, pouring through the inside of the tube, passing through the vessel bottom head, and causing failure of the tube wall outside the vessel. Failure of the lower plenum pressure boundary by this process is the subject of an ongoing Nuclear Regulatory Commission (NRC)-sponsored research project at Idaho National Engineering Laboratory.

If water surrounding the BWR reactor vessel bottom head is to be successful in maintaining the core and structural debris within the vessel, it must first be successful in preventing failure of the penetration assemblies below the vessel as they fill with relocating molten debris. Because the control rod drive penetration internal cross section is almost completely blocked by the presence of the movable index tube, the threat is to the in-core (instrument tube) penetrations, which have a relatively open cross section. These penetrations are shown in Fig. 17.2, and have the same dimensions for all BWR reactor vessel sizes.

An additional release mechanism associated with the penetrations is by failure of the welds holding the penetration assemblies in place. Here also, the threat of opening an escape pathway for the molten debris lies with the in-core instrument tube penetrations and not with the control rod guide tube penetrations. This is because BWRs are fitted with a structure beneath the vessel bottom head that would limit the downward movement of any control rod mechanism assembly to ~1 in. (0.0254 m) in the event of failure of its stub tube weld. (The purpose is to guard against the expulsion of a control blade from a critical core.) Because the bottom head thickness in the region of the penetrations ranges from 7-1/4 in.(0.1842 m) for Duane Arnold to 8-7/16 in. (0.2143 m) for Browns Ferry, this limited downward movement could not open a significant pathway through the vessel wall, and with wall cooling, any small

*All work in connection with this project completed before becoming a member of the Advisory Committee on Reactor Safeguards.

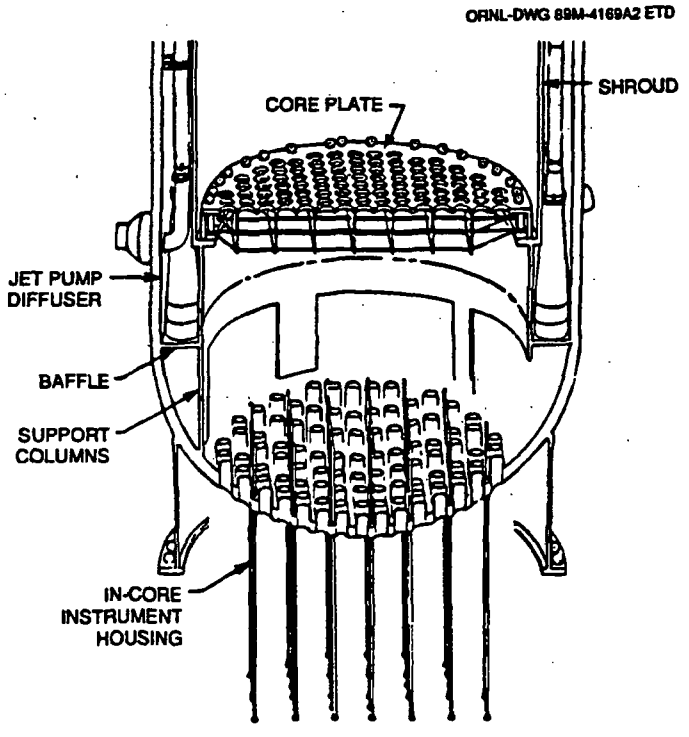

Figure 17.1 BWR bottom head that accommodates many control rod drive stub tube and in-core instrument housing penetrations

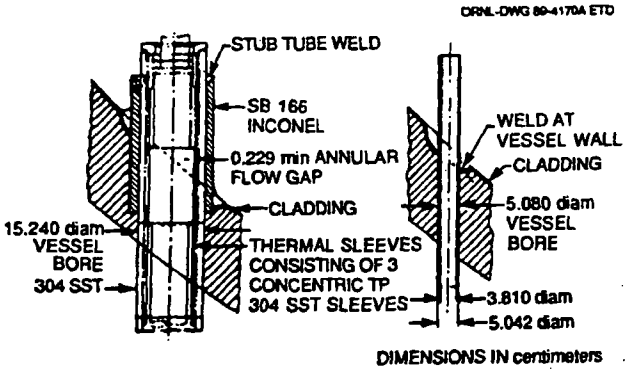

Figure 17.2 BWR control rod drive mechanism assemblies held in place by stainless steel-to-Inconel welds

pathway should become blocked with frozen material. The instrument tubes, however, have no external provision to limit their downward movement, and survival of their welds would depend upon the success of the surrounding water in cooling the vessel wall. The effectiveness of this cooling will be discussed in Chap. 19.

The remainder of the current chapter will address the response of the portion of the instrument guide tubes beneath the reactor vessel bottom head if subjected to the entry of molten debris. The HEATING code and the model employed for this assessment are briefly described in Sect. 17.1. The characteristics of the molten debris assumed to enter the tube are described in Sect. 17.2. Section 17.3 provides the results of the calculated tube response for the case without drywell flooding, while the results for the case with the instrument guide tube exterior surrounded by water are discussed in Sect. 17.4.

17.1 The Heating Model

The HEATING code[40,49,50] has been developed at Oak Ridge National Laboratory as a practical analytical tool to solve steady-state and transient heat conduction problems in one-, two-, or three-dimensional (1-, 2-, and 3-D) Cartesian, cylindrical, or spherical coordinates. The considered structure can be represented by variable mesh spacing along each axis, and may represent multiple materials. The thermal conductivity, density, and specific heat of each material may be both time- and temperature-dependent, and the thermal conductivity may be anisotropic. Materials may undergo change of phase.

The boundary conditions for the HEATING model may be specified temperatures or any combination of prescribed heat flux, forced convection, natural convection, and radiation. The boundary condition parameters may be time- and temperature-dependent.

For the current application, a simple 1-D HEATING-7 representation of a BWR instrument tube has been developed to investigate whether the portion of the tube outside the vessel would fail as a result of molten core and structural debris drainage into the tube. The model provides a cylindrical geometry representation of a uniform axial instrument tube section surrounded by air or water. For the case with water, the boiling mode (nucleate or film) is determined by HEATING in application of the boiling curve. The initial wall temperature profile for the calculations is either the entire wall at the temperature of the environment or a more conservative logarithmic radial temperature profile. The latter is established within the tube wall under the assumption that molten stainless steel has been pouring through the tube, thereby preheating the inner tube wall while the temperature of the outer wall surface remains at the temperature of the environment. The transient calculation then predicts the wall temperature response as the molten material (metallic or oxidic eutectic mixture) suddenly fills the tube section. Change of phase is calculated both for the tube wall (if it melts) and for the material filling the tube (as it freezes).

17.2 Characteristics of the Molten Debris

The first step in preparing to calculate the temperature response of the instrument guide tube wall to sudden filling of the tube with molten debris is to establish the thermal properties of the relocating debris. As the temperature of the central region of the lower plenum debris bed increased, the local metals would be expected to form a series of liquid eutectic mixtures followed by the formation of oxidic mixtures. Information as to the composition of these mixtures and the associated melting points has been provided by a recent small-scale experiment[51] employing prototypic BWR core constituent materials.

Based upon the experimental results described in Ref. 51, the two primarily metallic mixtures described in Tables 17.1 and 17.2 and the oxidic mixture described in Table 17.3 have been considered as the relocating molten materials for the HEATING calculations of tube wall response.

17.3 Results for the Dry Case

If the BWR drywell were not flooded with water, then the portion of the instrument guide tubes extending beneath the reactor vessel bottom head would be surrounded by air.

Table 17.1 Composition and mass-averaged properties for the first metallic mixture

Composition	Mole fraction	Molecular weight	Mass fraction
Zr	0.193	91.22	0.283
SS	0.807	55.34	0.717

Note:
Melting temperature: 2642°F (1723 K)
Density: 419.9 lb/ft^3 (6726 kg/m^3)
Specific heat: 0.1196 Btu/lb•°F [501 J/(kg•K)]
Thermal conductivity: 0.2362 Btu/min•ft•°F [80.5 W/(m^2•K)]
Heat of fusion: 116.9 Btu/lb (271,909 J/kg)

Table 17.2 Composition and mass-averaged properties for the second metallic mixture

Composition	Mole fraction	Molecular weight	Mass fraction
Zr	0.300	91.22	0.312
SS	0.600	55.34	0.380
UO_2	0.100	270.07	0.308

Note:
Melting temperature: 2912°F (1873 K)
Density: 438.9 lb/ft^3 (7031 kg/m^3)
Specific heat: 0.1126 Btu/lb•°F [471 J/(kg•K)]
Thermal conductivity: 0.2155 Btu/min•ft•°F [73.4 W/(m^2•K)]
Heat of fusion: 115.8 Btu/lb (269,351 J/kg)

Table 17.3 Composition and mass-averaged properties for the oxidic mixture

Composition	Mole fraction	Molecular weight	Mass fraction
ZrO_2	0.750	123.22	0.578
UO_2	0.250	270.07	0.422

Note:
Melting temperature: 4172°F (2573 K)
Density: 426.9 lb/ft^3 (6838 kg/m^3)
Specific heat: 0.1611 Btu/lb•°F [674 J/(kg•K)]
Thermal conductivity: 0.0546 Btu/min•ft•°F [18.6 W/(m^2•K)]
Heat of fusion: 225.5 Btu/lb (524,513 J/kg)

The boundary condition imposed for calculation of the tube wall temperature response is heat transfer from the tube surface to air by natural convection with a heat transfer coefficient of 0.625 Btu/h•ft^2°F [3.549 W/(m^2•K)]. Heat transfer by radiation to the surrounding structures, which are at the temperature of the atmosphere, is also modeled. (This includes the outer surface of the reactor vessel wall.) The dimensions of the stainless steel guide tubes are provided in Fig. 17.2.

The calculated results are presented in tabular form, considered to be the optimum means of displaying the time-dependent response of the temperature profile across the wall for the present discussion. The first case considered involves drainage of the oxidic mixture described in Table 17.3 into an instrument tube with an initial (time zero) uniform wall temperature of 400°F (477.6 K), the temperature of the surrounding atmosphere. Because this mixture includes a significant mass fraction of UO_2, a decay power of 0.400 MW/m^3 (appropriate for BWR debris with its high fraction of metallic materials) was associated with the mixture mass during the calculation. The predicted 1-D tube wall temperature response is provided in Table 17.4.

Although the melting temperature of stainless steel is ~2550°F (1672 K), creep rupture considerations dictate that stainless steel will not serve as a practical pressure boundary at temperatures above 2300°F (1533 K), because virtually all strength is lost at such elevated temperatures. Accordingly, the predicted wall temperatures provided in Table 17.4 lead to the conclusion that the integrity of the instrument guide tube would not survive introduction of the molten oxidic mixture at 4172°F (2573 K). Indeed, melting of this pressure boundary through almost 40% of the wall is predicted at time 60 s for this case.

It is logical, however, to argue that because the metallic eutectic mixtures melt at lower temperatures, they would pour into the instrument guide tubes and freeze long before the oxidic mixture could melt. Therefore, there would be no pathway for the molten oxidic mixture to enter the instrument guide tubes, and accordingly, this case should be excluded. Following this line of reasoning, the second case considered involves drainage of the metallic mixture described in Table 17.2.* Results are provided in Table 17.5.

The maximum predicted tube wall temperatures for this case of introduction of a metallic molten mixture at 2912°F (1873 K) are at time 30 s, as indicated in Table 17.5. The average wall temperature at this time is ~2070°F (1405 K), and the wall has everywhere cooled below 2000°F (1366 K) by time 1 min. With the reactor vessel depressurized, the tensile stress in the tube wall would be low [no more than 1.2 MPa (0.17 ksi)]; hence, the tube wall initially at ambient temperature would be expected to survive this transient.

*The decay heating associated with the very small UO_2 component of this mixture has been neglected in these calculations.

Table 17.4 Time-dependent response of an instrument tube wall initially at the ambient temperature following introduction of a molten oxidic mixture at 4172°F[a]

Time (s)	Inner surface (°F)	Fractional distance across tube wall				Outer surface (°F)
		0.20 (°F)	0.40 (°F)	0.60 (°F)	0.80 (°F)	
0	400	400	400	400	400	400
1	1844	1504	1186	915	728	660
5	2007	1891	1794	1720	1673	1654
10	2256	2196	2146	2109	2083	2068
20	2515	2483	2455	2432	2413	2398
30	2565	2541	2524	2508	2493	2478
60	2569	2550	2533	2517	2502	2488
120	2550	2533	2516	2501	2486	2472
300	1900	1894	1888	1881	1875	1869

[a]Debris initially molten at 4172°F with decay heating; air at 400°F.

Table 17.5 Time-dependent response of an instrument tube wall initially at the ambient temperature following introduction of a molten metallic mixture at 2912°F[a]

Time (s)	Inner surface (°F)	Fractional distance across tube wall				Outer surface (°F)
		0.20 (°F)	0.40 (°F)	0.60 (°F)	0.80 (°F)	
0	400	400	400	400	400	400
1	1740	1409	1092	825	643	579
5	1798	1702	1621	1559	1519	1503
10	1951	1908	1873	1846	1827	1817
20	2083	2064	2048	2035	2023	2014
30	2094	2085	2076	2067	2058	2049
60	1976	1971	1965	1959	1952	1945
120	1769	1765	1761	1756	1751	1746
300	1401	1399	1397	1394	1392	1388

[a]Debris initially molten at 2912°F; air at 400°F.

It now becomes necessary to consider the more conservative case with the tube wall initially heated by a passage of molten material immediately before local filling by the molten debris at 2912°F (1873 K). Here the initial tube wall temperatures are as indicated in the first line of Table 17.6. These initial temperatures follow the logarithmic radial profile produced by an inner surface temperature of 2550°F (1672 K), which is the melting temperature of the wall material, and an outer surface temperature of 400°F (477 K), which is equivalent to the temperature of the ambient surroundings.

With this very conservative assumption of wall preheating, it is now expected that the integrity of the instrument guide tube wall would not survive an introduction of the mixture of molten debris at 2912°F (1873 K). As indicated, the predicted average wall temperature exceeds 2300°F (1533 K) for ~1 min.

Finally, the case with filling of the instrument guide tube by the mixture of molten debris at 2642°F (1723 K) described in Table 17.1 will be considered. Based upon the previously discussed results for the higher-melting metallic mixture, it is obvious that only with the assumption of wall preheating might this second metallic mixture pose a threat to the integrity of the instrument guide tube wall. Accordingly, this is the only situation represented for this case; results are provided in Table 17.7.

Table 17.6 Time-dependent response of a preheated instrument tube wall following introduction of a molten metallic mixture at 2912°F[a]

Time (s)	Inner surface (°F)	Fractional distance across tube wall				Outer surface (°F)
		0.20 (°F)	0.40 (°F)	0.60 (°F)	0.80 (°F)	
0	2550	2069	1617	1189	785	400
1	2217	2001	1775	1567	1417	1361
5	2240	2176	2122	2080	2051	2036
10	2341	2309	2282	2259	2242	2229
20	2406	2388	2372	2357	2343	2330
30	2400	2386	2373	2360	2348	2336
60	2237	2230	2222	2213	2204	2194
120	1948	1943	1937	1931	1925	1918
300	1485	1483	1480	1477	1474	1470

[a]Debris initially molten at 2912°F; air at 400°F.

Table 17.7 Time-dependent response of a preheated instrument tube wall following introduction of a molten metallic mixture at 2642°F[a]

Time (s)	Inner surface (°F)	Fractional distance across tube wall				Outer surface (°F)
		0.20 (°F)	0.40 (°F)	0.60 (°F)	0.80 (°F)	
0	2550	2069	1617	1189	785	400
1	2128	1929	1713	1515	1373	1321
5	2129	2070	2021	1983	1957	1943
10	2216	2188	2163	2144	2128	2117
20	2276	2260	2246	2232	2220	2210
30	2276	2263	2252	2240	2229	2219
60	2149	2143	2135	2127	2219	2111
120	1891	1886	1881	1876	1870	1863
300	1462	1459	1457	1454	1450	1447

[a]Debris initially molten at 2642°F; air at 400°F.

While the predicted tube wall temperatures for this case are, as expected, lower than those listed in Table 17.6, failure of the instrument guide tube wall by creep rupture cannot be ruled out. The average tube wall temperature is predicted to exceed 2200°F (1478 K) for a period of ~ 30 s, which might be sufficient to induce failure of the tube wall pressure boundary.

In summary, HEATING calculations for the instrument guide tube surrounded by a dry atmosphere predict wall temperatures sufficiently high to induce certain loss of integrity if the tube is filled by the oxidic molten mixture at 4172°F (2573 K), probable loss of integrity if a *preheated* tube is filled by the metallic mixture at 2912°F (1873 K), and possible loss of integrity if a *preheated* tube is filled by a metallic mixture at 2642°F (1723 K). The temperature response of the tube wall if surrounded by water during the filling process is described in the following section.

17.4 The Case with Drywell Flooding

Heat transfer from the instrument guide tube outer surface would be greatly enhanced by the presence of water, because the mode of heat transfer would be shifted from natural convection of air to nucleate (or film) boiling of water. The results of the calculations discussed here demonstrate that the tube wall would be expected to survive even the introduction and freezing of the oxidic eutectic mixture if the tube were surrounded by water during the period while the mixture was being introduced.

Integrity

The water is assumed to be at the saturation temperature [267°F (404 K)] corresponding to a containment pressure of 40 psia (0.276 MPa).

The first case considered with the instrument guide tubes immersed in water involves drainage of the metallic mixture described in Table 17.1. The tube wall is assumed to be preheated by the prior passage of liquid stainless steel, as described in the preceding section. The predicted wall temperature response (one-dimension) is shown in Table 17.8. As indicated, the tube wall temperature is everywhere less than 2000°F (1367 K) within 5 s. These results may be compared with those presented in Table 17.7, to see the effect of water vs air cooling. Clearly, the presence of water prevents loss of tube wall integrity for this metallic mixture.

The next water case is similar to the first except that the second metallic mixture (Table 17.2) is assumed to fill the tube. As indicated in Table 17.9, the predicted wall temperatures are everywhere less than 2000°F (1367 K) within 10 s. These results may be compared with the results for an air environment shown in Table 17.6. Again, the effect of water is to quickly cool the instrument tube wall so that its integrity is not threatened.

The third case with water considers the oxide mixture described in Table 17.3 with decay heating. In Sect. 17.3, it was shown that HEATING predicts tube wall temperatures sufficient to cause loss of integrity for this mixture in an air environment even if the tube wall is not preheated (Table 17.4). The results for the water environment are shown in Table 17.10, where the maximum predicted wall

Table 17.8 Time-dependent response of a preheated instrument tube wall immersed in water following introduction of a molten metallic mixture at 2642°F[a]

Time (s)	Inner surface (°F)	Fractional distance across tube wall				Outer surface (°F)
		0.20 (°F)	0.40 (°F)	0.60 (°F)	0.80 (°F)	
0	2550	2039	1559	1105	676	267
1	2095	1881	1645	1415	1228	1119
5	1970	1878	1788	1703	1622	1548
10	1892	1824	1755	1685	1614	1544
20	1685	1628	1569	1507	1442	1375
30	1458	1409	1356	1300	1241	1178
60	823	785	745	707	654	604
120	318	310	302	294	285	277
300	267	267	267	267	267	267

[a]Debris initially molten at 2642°F; water at 267°F.

Table 17.9 Time-dependent response of a preheated instrument tube wall immersed in water following introduction of a molten metallic mixture at 2912°F[a]

Time (s)	Inner surface (°F)	Fractional distance across tube wall				Outer surface (°F)
		0.20 (°F)	0.40 (°F)	0.60 (°F)	0.80 (°F)	
0	2550	2039	1559	1105	676	267
1	2191	1958	1704	1456	1255	1140
5	2070	1973	1880	1790	1707	1630
10	1994	1924	1853	1781	1709	1636
20	1779	1720	1660	1596	1530	1461
30	1537	1487	1433	1376	1316	1251
60	872	834	729	748	699	647
120	325	316	307	297	288	279
300	267	267	267	267	267	267

[a]Debris initially molten at 2912°F; water at 267°F.

temperature does not exceed 2000°F (1367 K). Thus, the survival of the wall is not threatened for the case with water cooling.

Finally, calculated results are shown in Table 17.11 for the most challenging of the water cases, which involves the oxidic mixture (Table 17.3) with tube wall preheating. Here the predicted temperatures in the innermost 20% of the tube wall exceed 2200°F (1478 K) for ~ 20 s, but the remainder of the wall does not reach threatening temperatures. Therefore, the tube wall would not be expected to fail under this challenge.

In summary, the effectiveness of the water cooling is such that the submerged stainless steel instrument tubes would be expected to survive filling by any of the molten debris mixtures, even with the conservative assumption of wall preheating. Survival of the vessel drain, which is carbon steel and has a higher melting temperature, is addressed in Sect. 18.2.4.

Table 17.10 Time-dependent response of an instrument tube wall initially at the ambient temperature following introduction of a molten oxidic mixture at 4172°F[a]

Time (s)	Inner surface (°F)	Fractional distance across tube wall				Outer surface (°F)
		0.20 (°F)	0.40 (°F)	0.60 (°F)	0.80 (°F)	
0	267	267	267	267	267	267
1	1797	1442	1106	812	595	489
5	1882	1737	1603	1484	1385	1307
10	1949	1853	1761	1675	1595	1521
20	1891	1819	1747	1676	1604	1533
30	1761	1695	1628	1561	1494	1424
60	1364	1306	1248	1189	1128	1067
120	797	750	703	655	607	558
300	305	298	292	286	280	274

[a]Debris initially molten at 4172°F with decay heating; water at 267°F.

Table 17.11 Time-dependent response of a preheated instrument tube wall immersed in water following introduction of a molten oxidic mixture at 4172°F[a]

Time (s)	Inner surface (°F)	Fractional distance across tube wall				Outer surface (°F)
		0.20 (°F)	0.40 (°F)	0.60 (°F)	0.80 (°F)	
0	2550	2039	1559	1105	676	267
1	2369	2108	1834	1570	1354	1232
5	2318	2203	2095	1995	1905	1824
10	2292	2209	2127	2047	1969	1891
20	2123	2054	1984	1912	1840	1765
30	1927	1862	1796	1728	1659	1588
60	1446	1387	1328	1267	1206	1143
120	838	790	741	692	642	591
300	307	300	293	287	280	274

[a]Debris initially molten at 4172°F with decay heating; water at 267°F.

18 Lower Plenum Debris Bed and Bottom Head Models

S. A. Hodge, J. C. Cleveland, T. S. Kress*

The coding developed within the Boiling Water Reactor Severe Accident Response (BWRSAR) code framework[2] for calculating the behavior of a BWR lower plenum debris bed after dryout and the associated bottom head response is currently being made operational within the MELCOR code[52] at Oak Ridge. This Nuclear Regulatory Commission (NRC)-sponsored effort is to test the Oak Ridge lower plenum debris bed and bottom head models within the structure of a local version of MELCOR and, when successful, to make recommendations for formal adoption of these models to the NRC and to the MELCOR code development staff at Sandia National Laboratories.

As an interim step toward their implementation within MELCOR, the lower plenum debris bed and bottom head response models have been made operational completely independently of their parent code. This has been done by means of a special driver that provides the input and timestep-dependent information once provided by BWRSAR. These models in their independent form have been exercised extensively in support of the current analysis of the effectiveness of water surrounding the lower portion of the reactor vessel in cooling the bottom head.

The Oak Ridge BWR lower plenum debris bed and bottom head response models intended for MELCOR are described in detail in the "BWR Lower Plenum Debris Bed Package Reference Manual" to be incorporated into the MELCOR Computer Code Manual. Accordingly, the discussion of these models in this chapter is limited to a description of their application as utilized in the present analysis, where penetration assembly failures are precluded and all materials, both solids and liquids, are retained within the debris bed.

18.1 Lower Plenum Debris Bed Models

The control volumes used to represent the initial structure (after dryout) of the lower plenum debris bed for calculations based upon the short-term station blackout severe accident sequence for Browns Ferry or Peach Bottom are shown in Fig. 18.1. The drawing is to scale, correctly indicating the relative sizes of the calculational control volumes employed by the lower plenum model. These initial volumes (surfaces of revolution) are listed in Table 18.1.

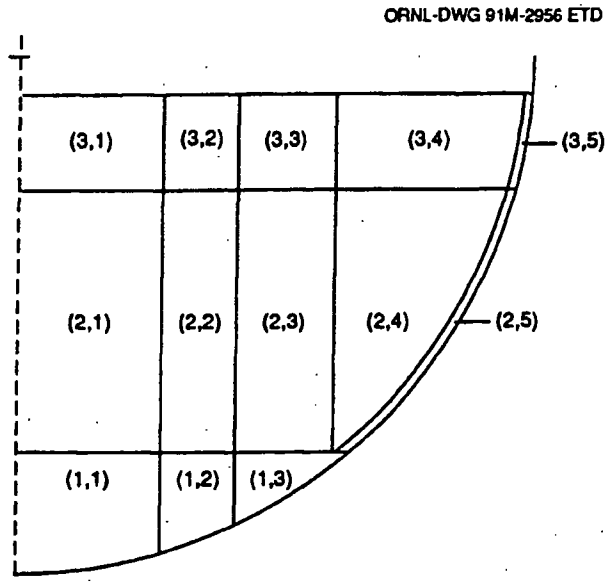

Figure 18.1 Representation (to-scale) of nodalization of lower plenum debris bed based upon Peach Bottom Plant

Table 18.1 Reactor vessel control volumes considered in the lower plenum debris bed calculation

Nodal designation	Volume (ft^3)
(1,1)	63.0
(1,2)	63.0
(1,3)	63.0
(2,1)	145.4
(2,2)	186.2
(2,3)	357.0
(2,4)	561.1
(2,5)	57.4
(3,1)	53.5
(3,2)	68.4
(3,3)	131.2
(3,4)	373.2
(3,5)	21.1
Total	2143.5

Note that the entire debris bed is contained below the center of curvature of the bottom head hemisphere. The total volume occupied by the debris bed is, of course, determined by the assumed bed porosity, which is user input.

*All work in connection with this project completed before becoming a member of the Advisory Committee on Reactor Safeguards.

Lower

For the calculations discussed in this report, a porosity of 0.40 was assigned for the bed oxides and a porosity of 0.20 was utilized for the bed metals. These are considered to be reasonable values, based upon the available information.[53]

Debris bed control volumes (2,5) and (3,5) are intended to represent the narrow solidified crust of oxidic debris that would be maintained at the inner surface of the bottom head wall after the central portion of the bed had melted. Such a provision is not considered necessary for the relatively small bottom debris layer, because it is comprised almost entirely of metals, as indicated in Table 18.2.

18.1.1 Eutectic Formation and Melting

As decay heating increases the temperature of the debris after bed dryout, the lower plenum models calculate the melting, migration, freezing, and remelting of the various material species making up the bed. Within each control volume, the eutectic mixtures defined by user input are formed according to the local availability of their constituents. Whenever the solid phase of one of the constituents of a particular eutectic mixture is exhausted within a control volume, the temperature of that control volume is permitted to increase under the impetus of decay heating to the melting temperature of the eutectic mixture or independent material species having the next higher melting point. The eutectic mixtures considered for the calculations discussed in this report are described in Table 18.3. The composition and melting points of these mixtures are based upon the results of the small-scale BWR materials melting experiment[51] performed at Oak Ridge in 1987.

The independent material species represented in these calculations are listed with their assigned melting points and other information in Table 18.4. The actual material melting temperatures are used for all metals except iron and nickel, where the assigned melting temperatures have been increased to slightly exceed the melting temperature [2912°F (1873 K)] of the Zr-SS-UO_2 eutectic mixture (Table 18.3) of which they are constituents. The assigned independent UO_2 species melting temperature has been lowered from its actual value [5066°F (3070 K)] by ~100 F° (56 K) in recognition of the effect of internal fission products.

18.1.2 Relocation and Settling

Within the debris bed, molten materials move downward from one control volume to another as long as void space (free volume) remains within the lower control volume. Once the interstitial spaces in the lower control volumes are filled, the liquid can move horizontally within the bed as necessary to keep the liquid level approximately equal within a layer. In all cases, the rate of movement of molten material through the debris bed is controlled by a user-input time constant, set at 1 min for the calculations discussed in this report. Thus, with a calculational timestep of 0.20 min, 20% of the liquid within a control volume is permitted to move downward each timestep. The rate of horizontal movement (for a control volume for which no void space remains in the underlying control volume) is set to one-half the rate for downward movement or 10% of the liquid mass each timestep for these calculations.

Table 18.2 Material masses (lb) included in the initial setup of the debris bed layers for Peach Bottom short-term station blackout

Material	Layer 1	Layer 2	Layer 3	Total
Zr	26,780.	71,318.	11,901.	109,999.
Fe	28,052.	84,684.	92,147.	204,883.
Cr	6,823.	20,599.	22,414.	49,836.
Ni	3,039.	9,181.	9,962.	22,182.
B_4C	592.	1,660.	186.	2,438.
ZrO_2	1,846.	26,124.	9,561	37,531.
FeO	52.	186.	0.	238.
Fe_3O_4	90.	435.	51.	576.
Cr_2O_3	37.	163.	13.	213.
NiO	6.	31.	5.	42.
B_2O_3	13.	32.	0.	45.
UO_2	1,966.	266,223.	88,842.	357,031.
Totals	69,296.	480,636.	235,082.	785,014.

Table 18.3 Eutectic mixture compositions considered for the lower plenum debris bed

Eutectic mixture	Mole fractions	Melting temperature °F	Melting temperature K
Zr – SS[a]	0.193 – 0.807	2642.	1723.
Fe – Cr – Ni[b]	0.731 – 0.190 – 0.079	2660.	1733.
Zr – SS – UO_2	0.300 – 0.600 – 0.100	2912.	1873.
ZrO_2 – UO_2	0.750 – 0.250	4172.	2573.

[a]SS represents stainless steel.
[b]This is the stainless steel eutectic mixture.

Table 18.4 Independent material species considered for the lower plenum debris bed

Material species	Molecular weight	Melting temperature (°F)	Heat of fusion (Btu/lb_m)
Fe	55.85	2960.[a]	117.
Ni	58.70	2960.[b]	129.
Cr	52.00	3400.	136.
Zr	91.22	3365.	108.
B_4C	55.26	4450.	814.
FeO	71.85	2510.	190.
Fe_3O_4	231.54	2850.	256.
NiO	74.71	3580.	292.
Cr_2O_3	152.02	4170.	296.
B_2O_3	69.62	4450.	148.
ZrO_2	123.22	4900.	304.
UO_2	270.07	4960.[c]	118.

[a]Actual melting temperature is 2800°F.
[b]Actual melting temperature is 2650°F.
[c]Actual melting temperature is 5066°F.

The adoption of a 1-min time constant for the movement of material liquids within the debris bed is the result of testing and experience. Use of too large a time constant will result in unrealistic predictions of free-standing liquid columns within the central control volumes. On the other hand, a time constant that is too small will result in the prediction of unrealistic sloshing of liquids between adjacent control volumes. Experience has shown that the use of a 1-min constant for lower plenum debris bed applications will result in a prediction of smooth and realistic spreading of liquids from their source control volumes.

18.1.3 Material Properties

The lower plenum debris bed model calculates composition-dependent properties of density, porosity, specific heat, and thermal conductivity for the debris mixture within each bed control volume each timestep. Specifically, the local porosity is based upon the relative mass fractions of solid metals and oxides within the control volume, while the representative local density, specific heat, and thermal conductivity are mass-averaged values based upon the relative amounts of each debris constituent present. (The relative masses of the solid and liquid phase of each constituent are also considered in the calculation of density and thermal conductivity.) The variation of material properties with temperature is considered where appropriate. A detailed discussion of the method by which these properties are calculated is provided in the "BWR Lower Plenum Debris Bed Package Reference Manual."

It is important to note here that almost all of the previous lower plenum debris bed response calculations have been performed for applications in which drywell flooding was not considered and bottom head penetration failures were predicted to occur soon after lower plenum dryout. Accordingly, the liquid fraction within any calculational control volume remained small in these cases, because the liquid would drain from the reactor vessel as it was formed. For the calculations discussed in this report, however, the lower plenum debris bed model is exercised without the provision of penetration failures so that the upper central bed control volumes eventually become primarily or even totally liquid. Within the upper liquid regions of the debris bed, heat transport would be greatly enhanced by the buoyancy-driven circulation of molten liquids. While the model has no representation of this liquid circulation, the associated increase in heat transport is crudely (but adequately) represented by increasing the effective mass-averaged and phase-averaged local thermal conductivity by a factor of 10 whenever the liquid mass within a control volume exceeds two-thirds of the total control volume mass.

As in the case of the relocation time constant, the use of a factor of 10 for enhancement of conduction to represent the effect of liquid circulation is the result of testing and experience. Use of too large an enhancement factor will result in a series of rapid phase changes within a control volume as excessive heat removal causes the liquid to freeze, the concomitant reduction in conduction heat transfer causes the solid to melt, and a new cycle begins. On the other hand, an enhancement factor that is too small will result in

the prediction of unrealistic temperature differences between adjacent liquid-dominated control volumes. Experience has demonstrated that enhancement of conduction by a factor of 10 will maintain the affected control volumes in a realistic condition of increasing liquid proportion together with a realistic radial temperature profile between adjacent liquid-dominated control volumes.

18.1.4 Effects of Water Trapped in the Downcomer Region

BWRs are fitted with an automatic depressurization system (ADS) that, upon actuation, causes rapid opening of several (five at Peach Bottom) of the reactor vessel safety/relief valves (SRVs). The BWR Emergency Procedure Guidelines[16] direct the operators, under severe accident conditions, to manually actuate the ADS when the core has become partially uncovered (but before any significant core damage has occurred). The flashing attendant to the resulting rapid depressurization of the reactor vessel causes the rapid loss of all water from the core region and core plate dryout. However, much of the cooler water in the downcomer region between the lower core shroud and the vessel wall is not flashed during this maneuver.

After lower plenum debris bed dryout, the water surrounding the jet pump assemblies in the downcomer region is the only water remaining in the reactor vessel. The proximity of the baffle plate and lower core shroud boundaries of this water-filled region to the bottom head hemisphere is illustrated in Fig. 18.2.

In the lower plenum debris bed energy balances, heat transfer by conduction is calculated between the bed control volumes and from the outer control volumes to the vessel wall. Additionally, radiation and convection from the bed upper surface to the vessel atmosphere and to intact structures above the bed are considered. Radiation to the lower core shroud from the bed surface, radiation and convection to the lower core shroud from the vessel atmosphere, and axial conduction along the vessel wall all contribute to heating and evaporation of the water trapped in the downcomer region.

While water remains in the downcomer region, the lower core shroud would be maintained at a temperature close to the saturation temperature of the water, which, with the reactor vessel depressurized, would be in the neighborhood of 300°F (422 K). Because this is much lower than the temperature of the upper surface of the debris bed, it is obvious that the core shroud would constitute a major heat sink for radiation from the upper bed.

After the water in the downcomer region had boiled away, the shroud temperature would increase to its melting temperature [2550°F (1672 K)], and the shroud would melt. The resulting liquid stainless steel would then enter the debris bed, providing a cooling effect while increasing the volume of the molten pool. These events have been considered in the calculations discussed in this report, and their effects will be described in detail in Chap. 19.

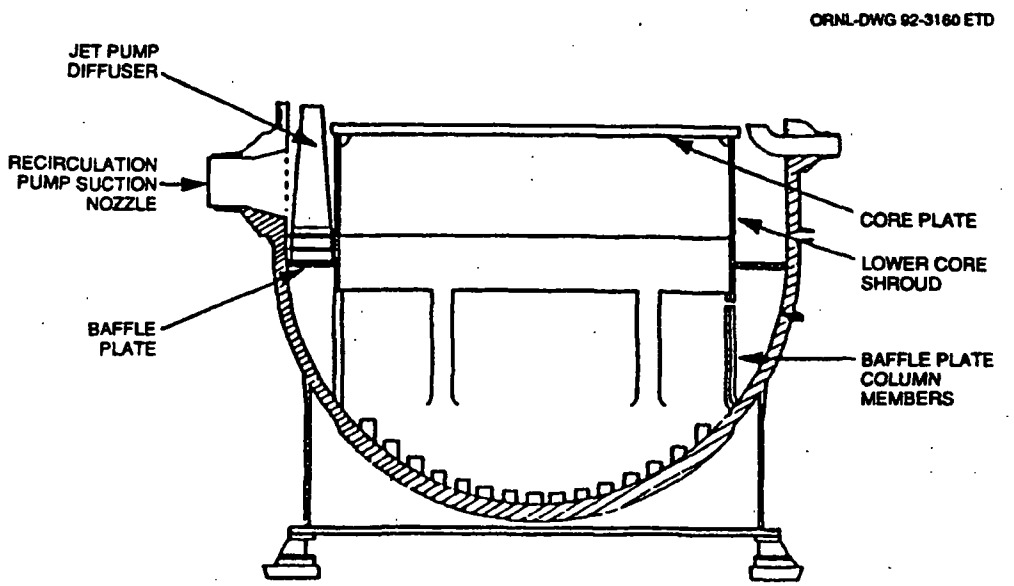

Figure 18.2 Portion of BWR reactor vessel beneath core plate divided into cylindrical region and bottom head hemisphere

18.1.5 Release of Fission Products from Fuel

The reduction of the decay heat source within the fuel associated with the loss of volatile fission products from the fuel-clad gap and during subsequent fuel heatup and melting is represented in accordance with the current recommendations of R.A. Lorenz of Oak Ridge National Laboratory. For the calculations discussed in this report, 80.3% of the total decay power at the time of lower plenum dryout is predicted to remain with the fuel in the debris bed. [The remainder is divided between the pressure suppression pool (16.3%) and the containment atmosphere (3.4%).] An additional release (equivalent to ~3 1/2% of the total decay power) from the fuel within the debris bed is predicted during the course of the calculation as a result of fuel melting within the bed.

The reduction of the total decay power with time is determined in accordance with the recommendations of Ostmeyer.[54] The decay heat curve is basically in conformance with the rigorous ANS standard, which provides values varying from 10 to 30% lower than those obtained by application of the ANS "simplified method." The decay heat values and fission product release methodology employed for the current calculations are identical to those employed for recent calculations in support of the Containment Performance Improvement Program.[21]

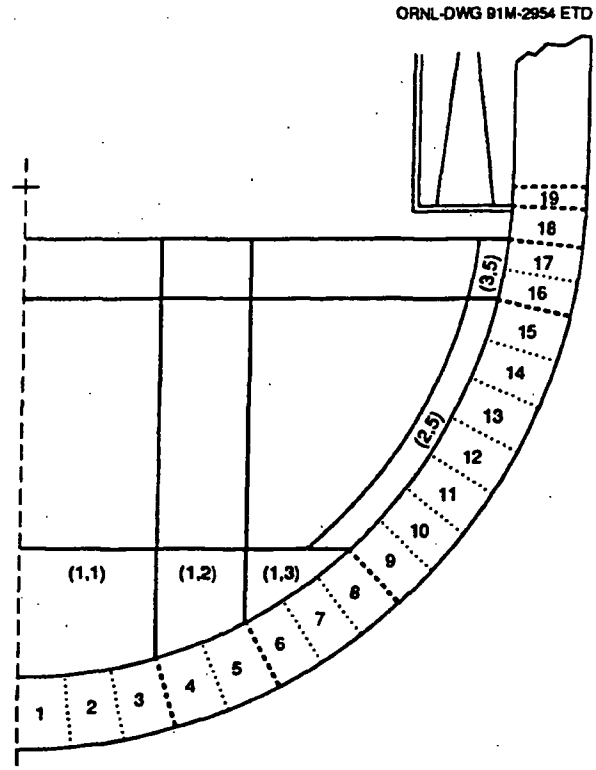

Figure 18.3 BWRSAR nodalization of reactor vessel bottom head wall

18.2 Bottom Head Models

18.2.1 The Vessel Wall

The nodalization employed for the hemispherical portion of the reactor vessel bottom head wall is shown in Fig. 18.3. As indicated, eight wall nodes are placed adjacent to the lowest debris layer, seven nodes abut debris control volume (2,5), and two wall nodes are adjacent to control volume (3,5).

The portion of the vessel wall between the upper surface of debris layer three and the bottom of the shroud baffle is represented by a single node. One wall node (node 19 in Fig. 18.3) represents the wall adjacent to the water trapped above the shroud baffle in the downcomer region. The upper surface of this last node is at the elevation of the center of curvature of the hemispherical bottom head.

For the purpose of calculating the bottom head wall temperatures, each node is divided into three equal-volume segments as shown in Fig. 18.4. Heat is transferred from the adjacent debris bed control volumes into the wall nodes by conduction. Heat transport along and across the wall by conduction from segment-to-segment is also calculated.

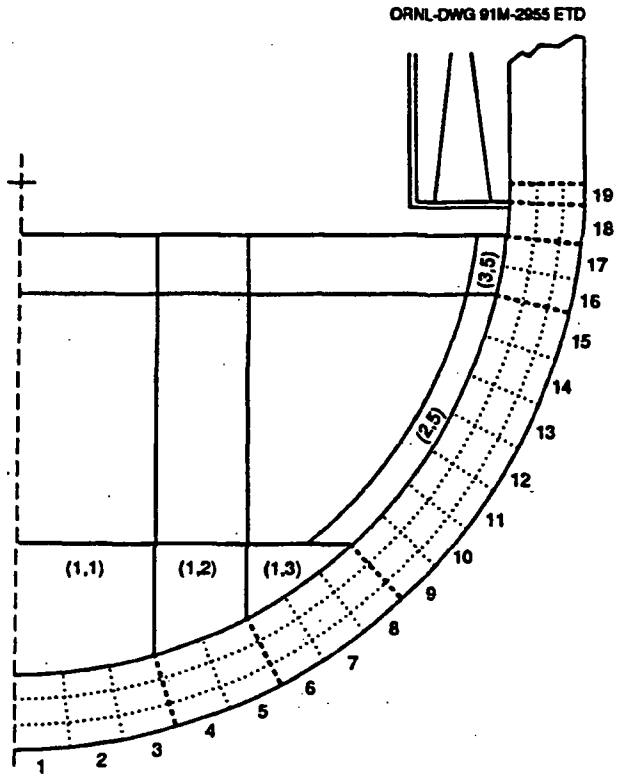

Figure 18.4 Each vessel bottom head wall node is divided into three radial segments for wall temperature calculation

Lower

Wall nodes above the elevation of the upper debris bed surface receive heat transfer by radiation from the bed and by radiation and convection from the vessel atmosphere.

Although not indicated in Fig. 18.4, the thickness of the BWR reactor vessel wall increases at some point (plant-specific) between the cylindrical section of the vessel and the lower portion of the bottom head where the many penetrations are located. (This transition in wall thickness is illustrated in Fig. 15.4.) The vessel wall nodalization established for the calculations discussed in this report recognizes the user-input location of this transition point and adjusts the thickness of the wall nodes above and below this location accordingly. Furthermore, the lengths of the two adjacent wall nodes are adjusted (one shortened, one lengthened) so that the transition point falls exactly on their nodal boundary. The resulting arrangement of the wall nodes is provided in Table 18.5, where the listed heights are relative to the low point of the vessel bottom head outer surface. [The heights relative to vessel zero can be obtained simply by subtracting the thickness of the lower bottom head, 8.4375 in. (0.2143 m) for this case.]

Table 18.5 Height of right-hand outer boundary of vessel wall nodes relative to low point of vessel outer surface for Peach Bottom

Debris control volume	Wall node index	Vertical height of outer right boundary (in.)
(1,1)	1	0.6125
	2	2.4445
	3	5.4791
(1,2)	4	8.7924
	5	12.8538
	6	18.3615
(1,3)	7	24.7657
	8	32.0170
	9	39.9115
	10[a]	44.8237
	11	58.9573
(2,5)	12[b]	68.6107
	13	78.7534
	14	89.3093
	15	100.1995
(3,5)	16	112.3382
	17	124.6686
	18	129.7362
	19	133.9375

[a]Wall thickness: nodes 1-10, 8.4375 in. nodes 11-19, 6.313 in.

[b]The vessel support skirt attaches at wall node 12, at height 68.4375 in. above the low point of the bottom head outer surface.

18.2.2 Heat Transfer from the Wall

The rates of heat transfer from the inner segment of the uppermost wall node (No. 19 in Figs. 18.3 and 18.4) to the water in the downcomer region are governed by nucleate boiling and conduction through the wall.

As indicated in Fig. 15.4, the effect of the atmosphere trapped in the "armpit" region between the reactor vessel support skirt and the bottom head is to divide the outer surface of the vessel wall into a lower wetted section, a dry section facing the armpit region, and an upper wetted section above the vessel skirt attachment. Within each section, heat transfer from the outer segment of each wall node to the drywell is calculated by means of user-input overall heat transfer coefficients of 0.625 Btu/h•ft^2•°F [3.549 W/(m^2•K)] where the wall is exposed to the atmosphere and 600,000 Btu/h•ft^2•°F [3,400,000 W/(m^2•K)—nucleate boiling] where the wall is covered by water. (Independent modeling of radiative heat transfer from the outer vessel surface is not employed.) The trapped atmosphere transfers heat to the vessel skirt by convection and through the vessel skirt by conduction.

18.2.3 Wall Stress and Creep Rupture

In considering the effects of tensile stress in the reactor vessel wall under severe accident conditions, it is important to recall that the BWR reactor vessel is supported from below. Accordingly, with the reactor vessel depressurized, only the portion of the vessel bottom head beneath the support skirt attachment weld would be in tension. The pertinent dimensions for a reactor vessel of the size at Peach Bottom or Browns Ferry are shown in Fig. 18.5. The bottom head loadings during normal operation and with the reactor vessel depressurized and dry are provided in Table 18.6.

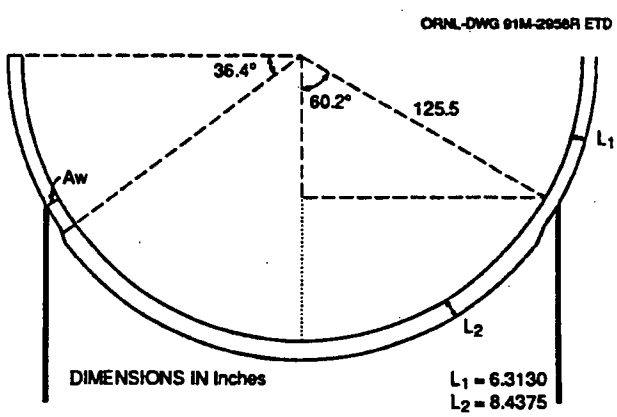

Figure 18.5 Physical dimensions of Browns Ferry reactor vessel bottom head

Table 18.6 Loading of the Browns Ferry reactor vessel bottom head wall underneath the skirt attachment

Load	Tensile stress psi	Tensile stress MPa
Normal operation		
Weight of water	152	1.05
Weight of core and structures	192	1.32
Weight of shroud and separators	36	0.25
Weight of bottom head beneath support skirt and pendants	36	0.25
Vessel pressure[a]	9,555	65.88
Total tensile stress at support skirt attachment	9,971	68.75
Vessel depressurized and dry		
Weight of lower plenum debris bed	204	1.41
Weight of shroud and separators	36	0.25
Weight of bottom head beneath support skirt and pendants	36	0.25
Vessel pressure[b]	469	3.23
Total tensile stress at support skirt attachment	745	5.14

[a]Based upon a normal operating pressure of 1000 psia.
[b]Based upon a vessel-to-drywell differential pressure of 50 psi.

The weights of the control rod drive mechanism assemblies, the instrument tube housings, the core shroud, and the core itself are transmitted via the stub tube, instrument housing welds (Fig. 17.2), and the shroud support columns and are ultimately borne by the portion of the bottom head beneath the skirt attachment.* It is reasonable, therefore, to determine the bottom head loading by adding these weights to the force-equivalent of the vessel internal pressure (relative to the drywell pressure). Based upon vessel-to-drywell differential pressures of 985 psi (6.791 MPa) for normal operation and 50 psi (0.345 MPa) for the depressurized case,† and with the vessel dimensions indicated in Fig. 18.5, the calculated wall tensile stress at the vessel support skirt attachment weld is ~10,000 psi (68.9 MPa) under normal operating conditions, but only 745 psi (5.1 MPa) under severe accident conditions with the reactor vessel depressurized.

*The reader may wish to review Figs. 15.3, 15.4, and 17.1 to fully recognize the manner in which these reactor vessel internal weights are supported.

†The two-stage Target Rock reactor vessel SRVs used at Browns Ferry and several other BWR facilities would shut when the vessel pressure dropped to within 20 psi (0.138 MPa) of the drywell pressure and would reopen when the vessel pressure increased to 50 psi (0.345 MPa) above the drywell pressure. Other types of valves behave differently when signalled to remain open while the vessel pressure approaches the drywell pressure; thus, the vessel wall stress under severe accident conditions is another plant-specific consideration. However, assumption of a 50-psi (0.345-MPa) pressure differential between vessel and drywell provides a reasonable upper bound.

The creep-rupture curves derived from recent tests[48] performed at Idaho National Engineering Laboratory (INEL) for the SA533B1 carbon steel of the BWR reactor vessel are shown in Fig. 18.6. The normal operating wall tensile stress of 68.75 MPa is indicated on this figure, from which a creep rupture time of ~2 h at 1340°F (1000 K) can be ascertained.

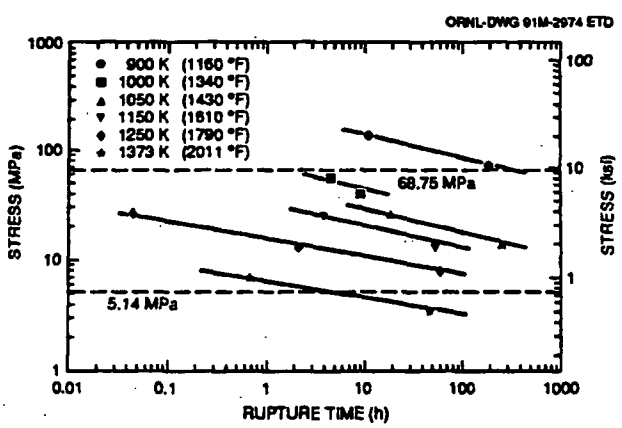

Figure 18.6 Creep rupture curves for SA533B1 carbon steel from recent tests at INEL

Also indicated on Fig. 18.6 is the wall tensile stress of 5.14 MPa corresponding to the depressurized and dry reactor vessel with a debris bed in the lower plenum. For this

relatively low value of stress, the figure indicates that a wall temperature of 2011°F (1373 K) would be expected to cause failure by creep rupture after ~4 h.

Extrapolation of the available creep rupture data shown in Fig. 18.6 to higher wall temperatures can best be performed by use of the Lawson-Miller parameter as described in Appendix B of Ref. 48. The results of such an extrapolation are provided in Table 18.7, which indicates the calculated failure times (min) over the stress range of interest for a depressurized reactor vessel for temperatures as high as 2500°F (1644 K). While it is undesirable to have to resort to extrapolation over such a large range, this table does provide some guidance concerning the wall failure times for the case of a depressurized reactor vessel with wall temperatures approaching the carbon steel melting point. The reactor vessel wall temperatures calculated in this study for the depressurized vessel after bottom head dryout will be discussed in the next chapter.

18.2.4 The Vessel Drain

The effectiveness of water cooling of the stainless steel instrument guide tubes beneath the reactor vessel bottom head has been discussed in Chap. 17. There it is shown that relocation of molten metallic or oxidic debris mixtures into the interior of an instrument guide tube would not induce wall temperatures sufficiently high to threaten the integrity of the tube, provided the tube wall is surrounded by water. In this section, results of a similar analysis for the carbon steel vessel drain will be discussed.

It is important to recognize that the pathway by which molten debris would enter the vessel drain is different than for the instrument guide tube. The vessel drain, shown in Figs. 15.4, 15.6, and 16.1, is located at the bottom of the lower plenum, offset 6 in. (0.152 m) from the point of vessel zero. Therefore, molten material forming within the debris bed and relocating downward would enter the vessel drain. On the other hand, molten material would move laterally to enter the failure location of the instrument tube within the bed only after the lower portions of the bed had been filled with liquid, to the level of the failure location. The upshot is that the vessel drain would probably be filled earlier by a lower-melting-temperature metallic mixture; the instrument guide tube would probably be filled later by a higher-melting-temperature mixture.

Nevertheless, any argument as to just which molten debris mixture might fill the vessel drain is rendered moot for the case in which the drain is submerged in water. This is true because it can be demonstrated that the drain pipe wall would not reach threatening temperatures even if filled with the oxidic mixture (with decay heating) described in Table 17.3. The configuration of the drain wall is shown to scale in Fig. 18.7.

If the vessel drain were surrounded by the dry containment atmosphere when filled by the molten oxidic mixture, the HEATING 7.0 code predicts the drain pipe wall temperature response shown in Table 18.8. Reference to the extrapolated creep rupture failure times listed in Table 18.7 clearly indicates that failure of the drain pipe wall would be expected shortly after 30 s, as the average wall temperature approached 2500°F (1644 K). [As noted by J. L. Rempe of INEL, we "are applying SA533 data to the drain pipe, which is composed of SA105/SA106. The drain pipe material is not a high temperature material. Although there is no high temperature data for this material, it's performance will undoubtedly be worse than SA533."[48]]

Table 18.7 Time (min) to creep rupture by extrapolation of the data for SA533B1 carbon steel using the Larson-Miller parameter

Temperature (F°)	Stress (MPa)							
	2.0	2.5	3.0	3.5	4.0	4.5	5.0	5.5
2100	4499.4	1672.8	622.20	433.80	135.00	82.20	50.40	31.98
2150	1714.3	649.6	246.14	172.95	54.93	33.81	20.82	13.39
2200	677.2	261.4	100.86	71.34	23.16	14.40	8.94	5.82
2250	276.9	108.8	42.72	30.42	10.08	6.30	3.96	2.58
2300	116.9	46.7	18.66	13.38	4.50	2.88	1.80	1.20
2350	50.9	20.6	8.40	6.06	2.10	1.32	0.84	0.56
2400	22.9	9.4	3.84	2.82	0.96	0.66	0.42	0.27
2450	10.5	4.4	1.85	1.34	0.48	0.31	0.20	0.14
2500	5.0	2.1	0.90	0.66	0.24	0.16	0.10	0.07

If, however, the vessel drain were surrounded by water as it was filled by the molten oxidic mixture, then the HEATING prediction of drain pipe wall temperature is as shown in Table 18.9. The wall temperatures do not reach threatening values.

These results indicate that drywell flooding, to the point of submersion of the reactor vessel bottom head, would prevent failure of the pressure boundary at the vessel drain. This is an important result, because it is a conclusion of the study reported in Ref. 48 that the drain is the most probable failure location given relocation of molten debris into the vessel lower plenum with a dry containment.

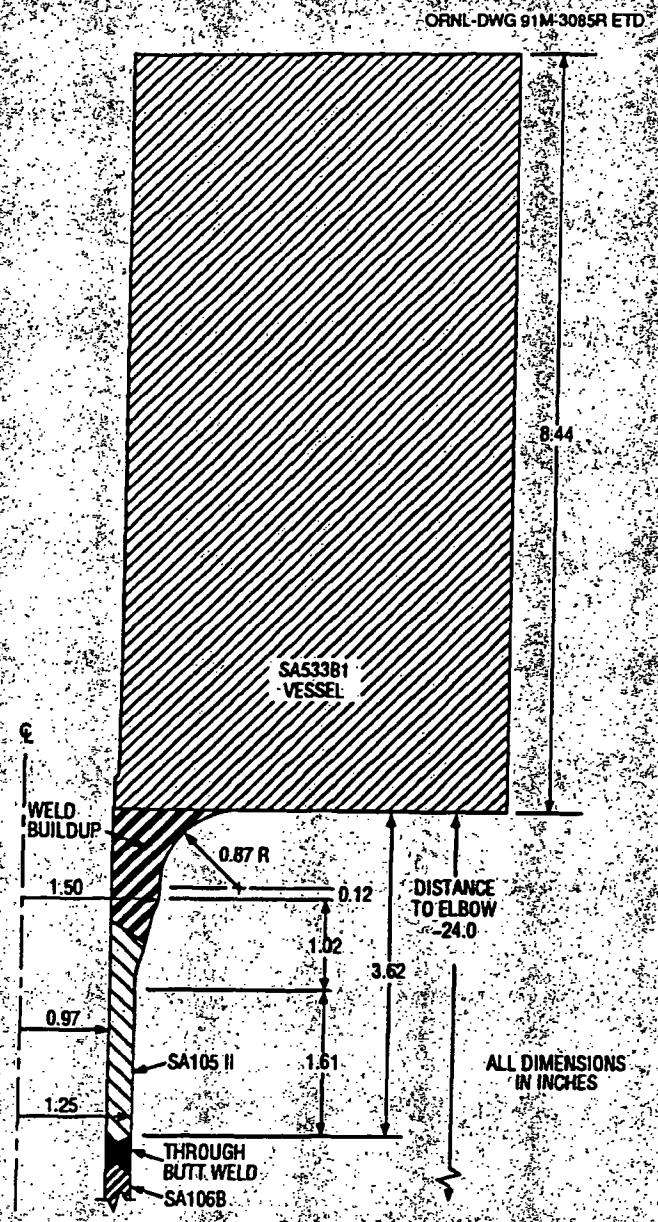

Figure 18.7 BWR drain line configuration and dimensions ***Source:*** **Adapted from Ref. 48**

Table 18.8 Time-dependent response of the vessel drain wall initially at the ambient temperature following introduction of a molten oxidic mixture at 4172°F[a]

Time (s)	Inner surface (°F)	Fractional distance across tube wall				Outer surface (°F)
		0.20 (°F)	0.40 (°F)	0.60 (°F)	0.80 (°F)	
0	400	400	400	400	400	400
1	1637	1242	963	776	668	632
5	1744	1556	1399	1301	1246	1228
10	1892	1773	1670	1590	1537	1515
20	2148	2086	2035	1996	1968	1953
30	2315	2271	2233	2203	2180	2163
60	2514	2486	2461	2438	2417	2398
120	2514	2494	2474	2455	2436	2417
300	2157	2146	2135	2123	2112	2100

[a]Debris initially molten at 4172°F with decay heating; air at 400°F.

Table 18.9 Time-dependent response of the vessel drain wall initially at the water temperature following introduction of a molten oxidic mixture at 4172°F[a]

Time (s)	Inner surface (°F)	Fractional distance across tube wall				Outer surface (°F)
		0.20 (°F)	0.40 (°F)	0.60 (°F)	0.80 (°F)	
0	267	267	267	267	267	267
1	1557	1158	873	676	555	502
5	1631	1426	1268	1152	1069	1011
10	1711	1568	1437	1331	1249	1183
20	1731	1631	1533	1441	1360	1288
30	1692	1607	1523	1441	1365	1295
60	1477	1411	1346	1284	1223	1165
120	980	941	903	866	829	792
300	333	325	317	309	301	294

[a]Debris initially molten at 4172°F with decay heating; water at 267°F.

19 Calculated Results for the Base Case

S. A. Hodge, J. C. Cleveland, T. S. Kress*

This chapter provides the results of calculations performed for the case of drywell flooding with the reactor vessel support skirt in its normal operating status, without a gas leakage pathway through the access hole. The calculations are based upon the short-term station blackout accident sequence, which is described in Sect. 19.1. The reference plant is a BWR Mark I containment facility of the size [3293 MW(t)] at Peach Bottom or Browns Ferry.†

19.1 Accident Sequence Description

Information concerning the definition of station blackout and other risk-dominant boiling water reactor (BWR) severe accident sequences is provided in Sect. 2.2. The short-term station blackout accident sequence has been selected for this study of the efficacy of drywell flooding as a severe accident mitigation technique because it involves formation of a reactor vessel lower plenum debris bed in the shortest time of all the dominant BWR accident sequences. A brief summary of the sequence of events is provided in the following paragraphs; detailed information concerning the events associated with BWR station blackout is available in Sect. 2.2.1 and in Ref. 7.

If a Peach Bottom unit were operating at 100% power when station blackout occurred and if both high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) were to fail upon demand, then the swollen reactor vessel water level would fall below the top of the core in ~40 min. After the core is uncovered, events would progress rapidly because the decay heat level is relatively high for the short-term station blackout accident sequence.

The dc power from the unit battery remains available during this accident sequence so that the operators can take the actions regarding manual safety relief valve (SRV) operation that are directed by the *BWR Owners Group Emergency Procedure Guidelines* (EPGs).[16] Specifically, the RPV Control Guideline directs the operators to open one SRV if the reactor vessel water level cannot be determined and to open all (five) ADS valves when the reactor vessel pressure falls below 700 psig. (Water level indication would be lost when the level drops below the indicating range at about one-third core height.) These actions provide temporary cooling of the partially uncovered core and are predicted to be taken at times 77 and 80 min after scram, respectively.

The high rate of flow through the open SRVs would cause rapid loss of reactor vessel water inventory, and core plate dryout would occur at about time 81 min. Heatup of the totally uncovered core would then lead to significant structural relocation (molten control blade and canister material), beginning at time 131 min. Because the core plate would be dry at this time, heatup and local core plate failure would occur immediately after debris relocation began. Subsequent core damage would proceed rapidly, with the fuel rods in the central regions of the core predicted to be relocated into the reactor vessel lower plenum at time 216 min.

Following the BWRSAR[2, 3] code prediction, the continually accumulating core debris in the reactor vessel lower plenum would transfer heat to the surrounding water over a period of ~30 min until initial bottom head dryout and, in the process, the lower layer of core debris would be cooled to an average temperature of 1310°F (983 K). After vessel dryout occurred at time 246 min, the temperature in the middle debris layer would be sufficient [2375°F (1575 K)] to cause immediate failure of the control rod guide tubes in the lower plenum. Failure of this supporting structure would then cause the remaining intact portions of the core to collapse into the lower plenum.

The calculated sequence of events through the point of bottom head dryout and control rod guide tube failure is summarized in Table 19.1. The lower plenum debris bed would be fully established, dry, and with its temperature increasing under the impetus of decay heat. The stainless steel masses of the control rod guide tubes, instrument guide tubes, and other lower plenum structures would, as they melt, be subsumed into the surrounding volumetrically heated debris. This is a major source of the large proportion of metallic content characteristic of the expected composition of a BWR lower plenum debris bed.

As discussed in Chap. 17, failure of the instrument guide tubes within the central portion of the lower plenum debris bed is expected to open an escape pathway for molten debris to flow through the reactor vessel bottom head wall.

*All work in connection with this project completed before becoming a member of the Advisory Committee on Reactor Safeguards.

†Small reactor calculations are provided in Appendix C. As explained in Chap. 16, the efficiency of water cooling of the reactor vessel bottom head would be least for the largest vessels, and these are considered in the main body of this report.

Table 19.1 Calculated sequence of events for Peach Bottom short-term station blackout with ADS actuation

Event	Time (min)
Station blackout-initiated scram from 100% power. Independent loss of the steam turbine-driven HPCI and RCIC injection systems	0.0
Swollen water level falls below top of core	40.3
Open one SRV	77.0
ADS system actuation	80.0
Core plate dryout	80.7
Relocation of core debris begins	130.8
First local core plate failure	132.1
Collapse of fuel pellet stacks in central core	215.9
Reactor vessel bottom head dryout; structural support by control rod guide tubes fails; remainder of core falls into reactor vessel bottom head	245.8

However, the presence of water surrounding the portion of the instrument guide tubes external to the vessel wall would prevent local failure of these tubes as they filled with relocating molten debris. Failure of the vessel drain and other penetrations would also be precluded by the water surrounding the bottom head. Accordingly, the lower plenum debris bed response calculations performed for this study of the effect of drywell flooding have been carried out without the modeling of penetration failures; all liquid debris remains within the lower plenum.

Before proceeding to a detailed discussion of the results calculated by the lower plenum debris bed model, it is important to recognize that many uncertainties attend the process of relocation of core and structural debris from the core region into the reactor vessel lower plenum. The question of core plate survival in the BWR severe accident sequence is pivotal. For example, if much of the relocating molten core debris were to not reach the core plate, but instead were to form a frozen crust above the plate, subsequent debris bed formation and melting above the core plate would lead to an accident event sequence more like the Three Mile Island experience (PWR) than the sequence calculated by BWRSAR.

Nevertheless, the purpose here is to analyze the effect of water cooling of the lower portion of the reactor vessel outer surface under the assumption that a whole-core lower plenum debris bed exists within the vessel. While some of the details such as the initial temperatures of the individual calculational control volumes within the bed and the relative local distributions of the bed constituents would vary, the basic characteristics of the whole-core lower plenum debris bed *after bottom head dryout* should not differ significantly for different assumptions concerning the relocation pathways from the core region into the lower plenum. In other words, the total bed mass and the fundamental mix of debris constituent mixtures would not change. Furthermore, the effects of differences in the initial control volume temperatures become small as the influence of decay heating after bed dryout becomes dominant and the bed materials begin to melt. (While the validity of this hypothesis that the response of the whole-core lower plenum debris bed after dryout would not be significantly altered by varying the material relocation pathways by which the bed is initially established cannot be demonstrated at this time, the lower plenum debris bed models are being made operational within the MELCOR code architecture; once this is accomplished, other debris relocation pathways can be invoked to produce the initial lower plenum debris bed configuration.)

19.2 Containment Pressure and Water Level

The containment pressure is important to the reactor vessel bottom head response calculations because it determines the pressure within the reactor vessel and the saturation temperature of the water surrounding the bottom head. For these calculations, the drywell is assumed to be vented as necessary to maintain the containment pressure at 40 psia (0.276 MPa). The reactor vessel pressure then cycles between 60 and 90 psia (0.414 and 0.621 MPa) due to the action of the SRVs, which open when the vessel-to-drywell pressure differential reaches 50 psi (0.345 MPa) and close when the vessel pressure falls to within 20 psi (0.138 MPa) of drywell pressure.[5]

Note that the available venting capacity is more than sufficient to maintain the drywell pressure below 40 psia (0.276 MPa). If the 18-in. (0.457-m) drywell exhaust butterfly valves at Peach Bottom were opened fully when the containment pressure reached 40 psia (0.276 MPa), the steam flow would be ~60 lb/s (27.2 kg/s). However, the maximum steam generation rate from the water surrounding the reactor vessel bottom head would be ~6 lb/s (2.72 kg/s). Accordingly, the containment pressure would rapidly decrease to ~15 psia (0.103 MPa) and remain in this vicinity as long as the vents remained fully open.

A method for opening the containment vents* under station blackout conditions is discussed in Ref. 41. While this discussion as written is specific to the torus exhaust vent valves, the method, involving the connection of copper tubing and a bottled supply of compressed gas, is also applicable to the drywell exhaust valves. There is no guarantee, however, that the valves would be fully opened under station blackout conditions; the proposed method includes consideration of an alternate vent pathway in which the valve boot seals would be deflated and only the inboard butterfly valve would be opened, providing a much smaller effective venting area. For this reason, it was decided to base the current calculations upon the conservative assumption of a containment pressure of 40 psia (0.276 MPa).

With the drywell pressure held near 40 psia (0.276 MPa), the saturation temperature of the containment water would be 267°F (408 K). For the base case calculations, the water is assumed to cover the outer surfaces of the four lowest bottom head wall calculational nodes. As indicated in Table 18.5, this requires a water height within the skirt of 8.79 in. (0.223 m) above the lower point of the reactor vessel outer surface. This water level is depicted in the to-scale representation of Fig. 19.1.

At this point, it is well to review the discussion provided in Sect. 15.3 concerning the effect that the drywell pressure *at the time the rising water reaches the skirt lower flange* has upon the subsequent relative water levels outside and

*Full opening of these valves implies opening to the maximum extent against the mechanical stops. For the drywell exhaust valves, this is equivalent to ~58% of the total flow area. See Ref. 41 for additional information.

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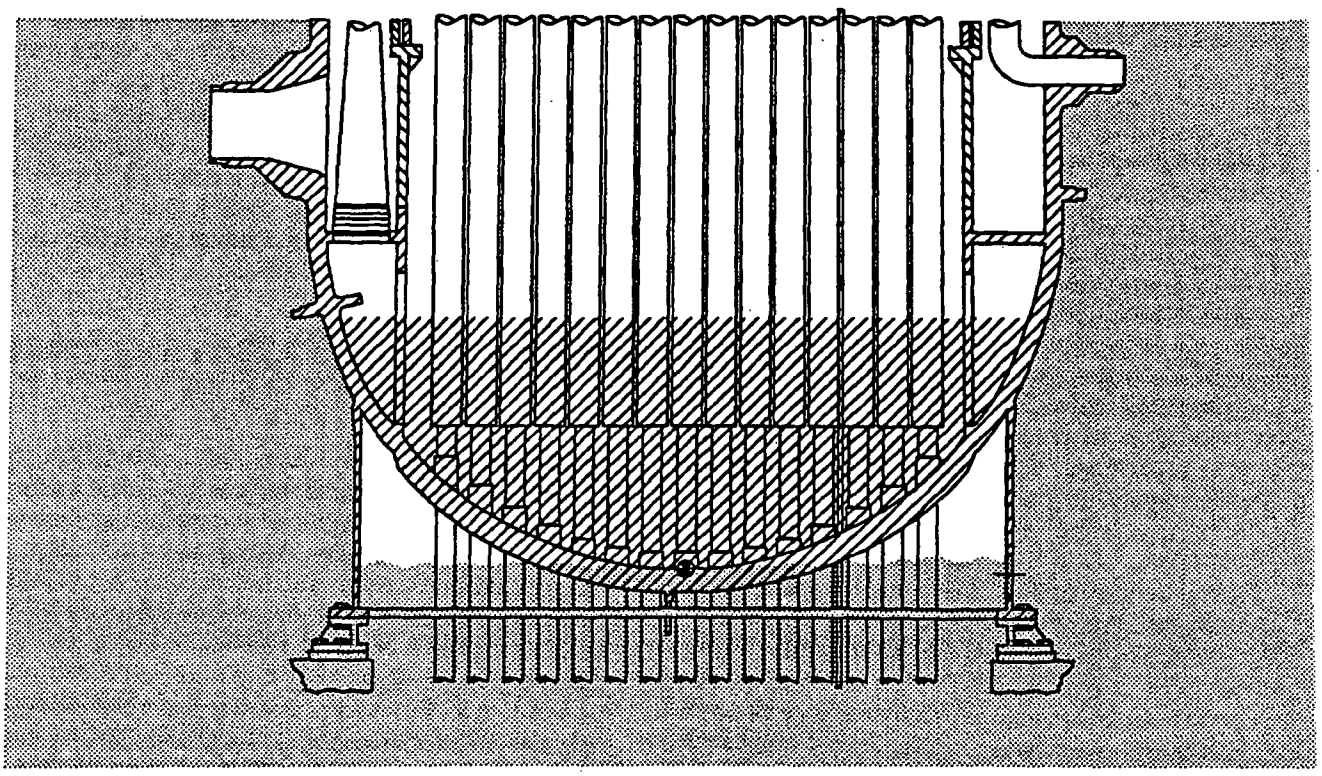

Figure 19.1 Water level within the vessel skirt for the base-case calculations of 8.79 in. above the low point of bottom head outer surface

Calculated

inside the skirt. As indicated in Table 15.2, a water level outside the skirt in excess of 600 in. (15.24 m) relative to vessel zero would be required to attain the water level within the skirt depicted in Fig. 19.1, if the containment were at 40 psia (0.276 MPa) during the flooding process. However, the inlet to the drywell vent for the Browns Ferry containment is located at a height equivalent to 528 in. (13.41 m) above vessel zero; therefore, this is the maximum height to which the water level could be raised in the containment and it would not be possible to wet the bottom head to the extent depicted in Fig. 19.1 with a containment pressure of 40 psia (0.276 MPa).

As indicated in Table 15.1, however, a water level within the skirt of 8.79 in. (0.223 m) above the lowest point of the reactor vessel outer surface could be attained with a water level outside the skirt of about 320 in. (8.13 m) if the containment pressure were 20 psia (0.138 MPa) during flooding. The rationale for assumption of the water level shown in Fig. 19.1 combined with a containment pressure of 40 psia (0.276 MPa) during the period of the calculations is that it is considered that drywell venting would maintain the pressure significantly below 40 psia (0.276 MPa) during the period of containment flooding; the pressure would subsequently increase to 40 psia (0.276 MPa) as a result of the steam generation initiated when the water came into contact with the vessel bottom head.

The observant reader will note from Fig. 19.1 that some of the outer bottom head penetration assemblies are shown with a dry length between the exterior of the bottom head and the water surface. Might bottom head penetration failures occur in these dry sections?

As discussed in Chap. 17, the threat of bottom head penetration failure is confined to the instrument guide tubes (not the control rod drive mechanism assembly housings). With the water surface shown in Fig. 19.1, 17 of the 55 instrument guide tubes would have exposed (dry) axial sections of varying lengths. It may be of interest to see just how these 17 partially exposed instrument guide tubes are distributed and what their dry lengths would be.

Counting from the left of Fig. 19.1, 15 control rod drive mechanism assembly housings are shown beneath the reactor vessel bottom head; one instrument guide tube housing is visible between the 12th and 13th members of this row. The three-dimensional configuration represented by this cross-sectional view actually comprises seven radial rings of control rod drive mechanism assembly housings surrounding one placed centrally, for a total of 185 control rod drive penetrations. The 55 instrument guide tube penetrations are interposed between these radial rings.

Counting outward from the centrally placed control rod drive mechanism assembly housing and considering the water level beneath the skirt shown in Fig. 19.1, eight instrument guide tube housings between the fourth and fifth radial rings would be exposed (dry) for a length of 4 in. (0.102 m), five instrument guide tube housings between radial rings five and six would be dry for a length of 10 in. (0.254 m), and four instrument guide tubes between radial rings six and seven would be exposed for a length of 17 in. (0.432 m).

Nevertheless, no attempt has been made to introduce a separate calculation of the response of the exposed portions of the outer instrument guide tubes, given the water level shown in Fig. 19.1. The reader is reminded that the water level would not in actuality be quiescent, but rather would fluctuate violently within the skirt in response to steam generation, expulsion, and condensation. It is reasonable to assume that the enhanced heat transfer induced by splattering would protect the upper sections of these outer instrument guide tube penetrations. Furthermore, as discussed in Sect. 15.3, it is prudent to determine whether or not the integrity of the reactor vessel bottom head could be maintained with any pocket of trapped atmosphere beneath the vessel skirt before undertaking a calculation of the in-skirt water level and its transient behavior more sophisticated than the simple approach outlined above. The effect of the trapped air pocket for the base-case calculation (Fig. 19.1) will be described in Sect. 19.4.

19.3 Initial Debris Bed Configuration

The initial configuration of the lower plenum debris bed immediately after dryout and control rod guide tube collapse is indicated in Fig. 19.2. The left-hand portion of this figure represents the 13 debris bed control volumes, which are numbered and shown to scale in Fig. 18.1. Because the outermost control volumes (2,5) and (3,5) are relatively small, however, they are shown on an expanded scale at the center of Fig. 19.2.

For each debris bed control volume, the current temperature, total mass of debris, and nodal free volume are listed on Fig. 19.2. (The total volumes of all bed control volumes in the initial configuration are provided in Table 18.1.) Because all of the initial control volume temperatures are below the melting temperature of the first eutectic mixture (Table 17.1), all of the bed constituents are entirely solid at this time.*

*Note that this initial condition of a quenched lower plenum debris bed comprised of solid particles is totally different from the initial condition of a molten pool assumed in other studies.[46, 47] In reviewing these other studies, one should consider the fate of the water initially in the lower plenum.

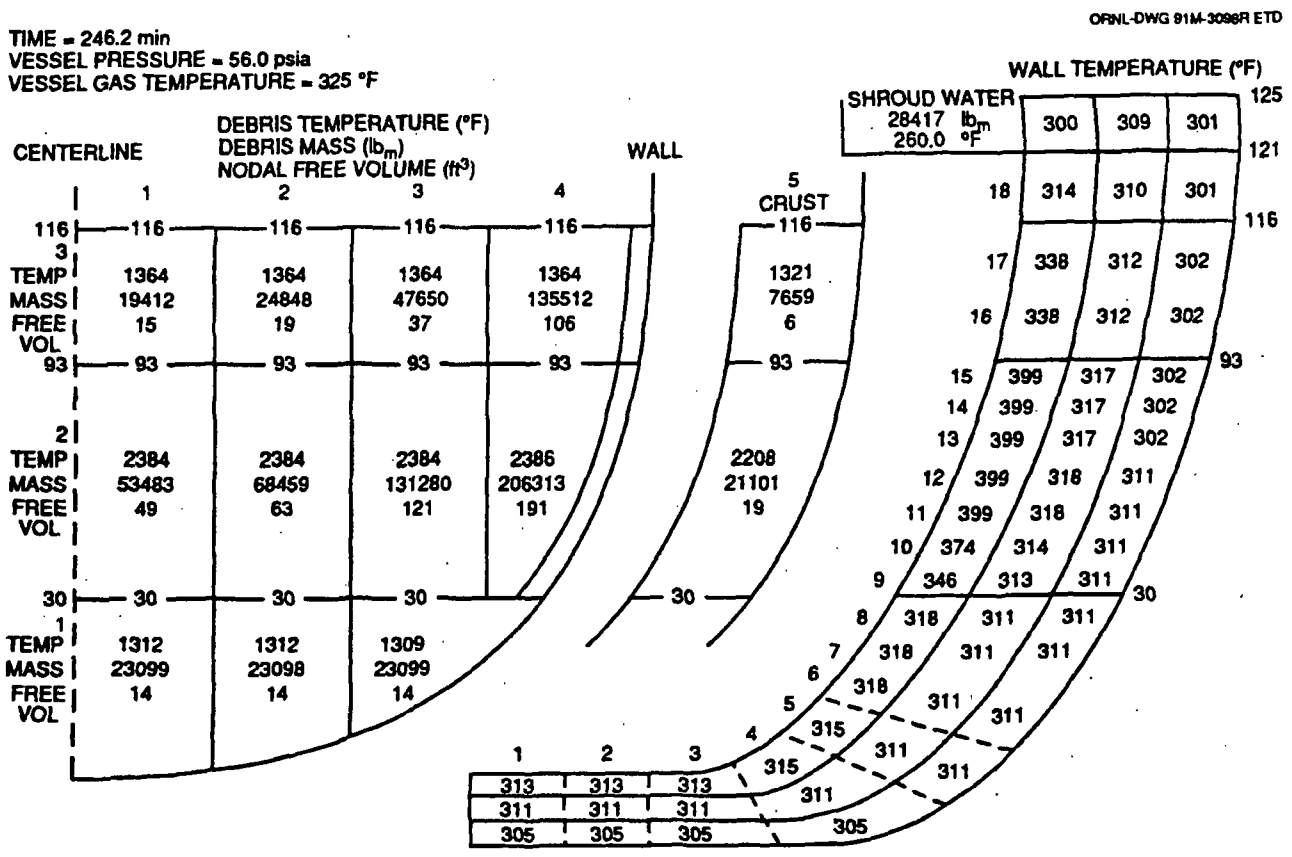

Figure 19.2 Initial configuration of lower plenum debris bed and initial bottom head wall temperatures

Immediately above each control volume is indicated the local height above vessel zero of the bed upper surface, in inches. In this initial bed configuration, the upper surface heights of all control volumes within a layer are equal [93 in. (2.36 m) for layer two, for example]. As the calculation proceeds, however, the debris constituents will melt and relocate downward, causing the bed to settle. The basic pictorial representation of the bed structure will remain the same as shown in Fig. 19.2 while the current local bed heights will be indicated numerically. The heights indicated to the left of the vertical dashed center line will, however, always represent the initial bed layer heights.

The right-hand portion of Fig. 19.2 represents the reactor vessel bottom head wall. (The wall nodalization and the division of each node into three radial segments has been described in Sect. 18.2.1 and illustrated in Figs. 18.3 and 18.4). Wall nodes 1 through 18 are numbered along the inner wall surface in Fig. 19.2. Within the wall, temperatures are indicated for each wall segment. Along the outer wall surface are indicated the heights above vessel zero of the inner wall termini of the dividing lines between the eight wall nodes originally adjacent to debris layer one, the seven nodes adjacent to debris layer two, the two nodes adjacent to debris layer three, the single node (18) between the upper bed surface and the bottom of the shroud baffle, and the single node (19) adjacent to the lower portion of the downcomer region. While the wall node locations will never change, the debris control volumes adjacent to the wall can settle downward. For example, bed control volume (3,5) might be adjacent to wall node 15 late in the calculation instead of control volume (2,5) as in the original configuration shown in Fig. 19.2.

Finally, Fig. 19.2 also provides information as to the mass and temperature of the water remaining in the downcomer region between the core shroud and the vessel wall. This information is listed under the heading "SHROUD WATER" adjacent to the inner surface of the uppermost wall node. As indicated, the initial mass [28,417 lb (12,890 kg)] is significant. The source of this water and the effect that the water has upon the debris bed response calculation have been discussed in Sect. 18.1.4.

19.4 Lower Plenum and Bottom Head Response

In this section, the calculated lower plenum debris bed and reactor vessel bottom head response will be described at

time 300 min and at 60-min intervals thereafter. The initial debris bed configuration at time 246.2 min after the inception of the accident has been provided in Fig. 19.2 and described in Sect. 19.3.

The reader is reminded that bottom head wall nodes 1 through 4 transfer heat in this calculation by nucleate boiling of water on their outer surfaces. Wall nodes 5 through 12 transfer heat by natural convection to the atmosphere trapped in the armpit region between the vessel skirt and the vessel wall (Fig. 19.1), while wall nodes 13 through 19 transfer heat to the water surrounding the vessel wall above the skirt attachment (located at the upper end of node 12). Additional information concerning the heat transfer pathways and the heat transfer coefficients employed is provided in Sect. 18.2.2.

The calculated situation at time 300 min is illustrated in Fig. 19.3. No free volume remains in the layer one control volumes because molten materials relocating from layer two have filled the previously existing interstitial regions. The increased masses and temperatures of the layer one control volumes also reflect the effect of these material relocations. Because the layer one temperatures are well below the melting temperature [2642°F (1723 K)] of the first eutectic mixture, however, all of the relocating liquids have solidified within layer one.

Molten debris materials do, however, exist at this time within layer two. As indicated, the temperatures of the central four control volumes of this layer exceed the melting temperatures of the first three eutectic mixtures (Table 18.3). The inventories of the solid and liquid phases of each debris component within layer two at this time are listed in Table 19.2. The interested reader may wish to compare the total constituent masses listed in this table with the original constituent masses for layer two given in Table 18.2. The differences are the masses predicted to be melted and relocated downward into layer one.

The height of the central region of layer two has been reduced, and most of the previously existing free volume within this layer has disappeared as the interstitial pores filled with liquid. The temperature of the thin crust control volume (2,5) is close to the temperature of the wall. The mass within this control volume has increased since debris

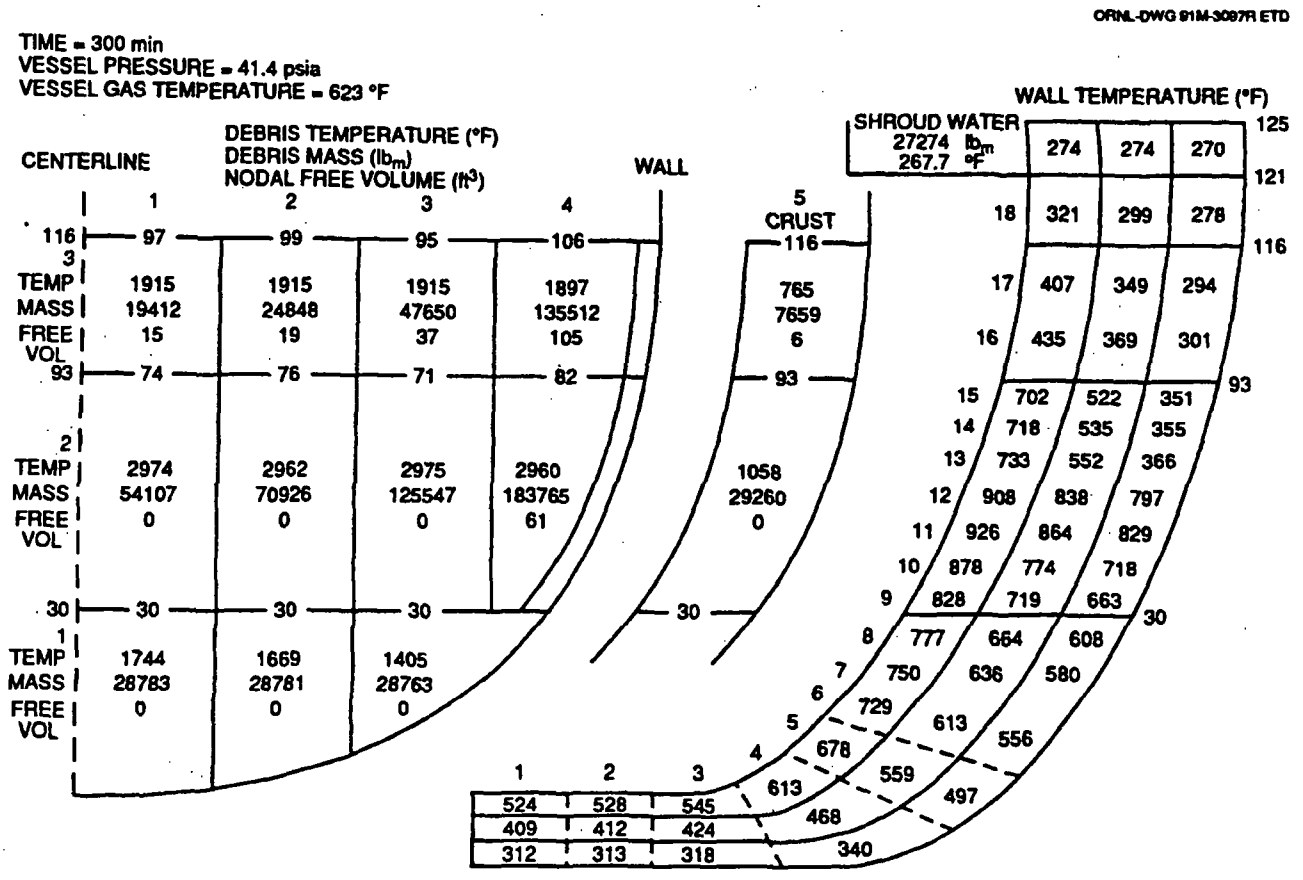

Figure 19.3 Lower plenum debris bed and vessel wall response at time 300 min after scram for base case

Table 19.2 Solid and liquid masses within debris layer two at time 300 min

Constituent	Solid mass (lb)	Liquid mass (lb)	Total (lb)
Zr	28,840	37,696	66,536
Fe	9,806	65,296	75,102
Cr	2,109	16,002	18,111
Ni	403	8,776	9,179
B_4C	1,660	0	1,660
ZrO_2	26,125	0	26,125
FeO	8	0	8
Fe_3O_4	19	416	435
Cr_2O_3	163	0	163
NiO	31	0	31
B_2O_3	32	0	32
UO_2	259,403	6,820	266,223
Totals	328,599	135,006	463,605

bed dryout as liquid materials from the adjacent control volume (2,4) relocated laterally and solidified.

All of layer three remains solidified at this time. The masses and free volumes of these control volumes remain unchanged, but the elevations of the upper and lower boundaries within the central region have been reduced as the layer two control volumes collapsed beneath them.

While the vessel wall temperatures have increased, they have not reached threatening values. The effect of the reduced heat transfer in the armpit region is reflected in the higher outer surface temperatures of wall nodes 5 through 12. About 1100 lb (499 kg) of water has evaporated from the downcomer region.

Moving ahead 1 h to time 360 min, the calculated situation is illustrated in Fig. 19.4. Within the bed, the major dimensional change is in layer two control volume (2,4), where additional melting has eliminated all of the previous free volume and the local bed height has decreased. Although the temperatures of all bed control volumes have increased, constituent liquids continue to be found only in layer two. The relative solid and liquid masses in this layer at this time are provided in Table 19.3. The metallic solids

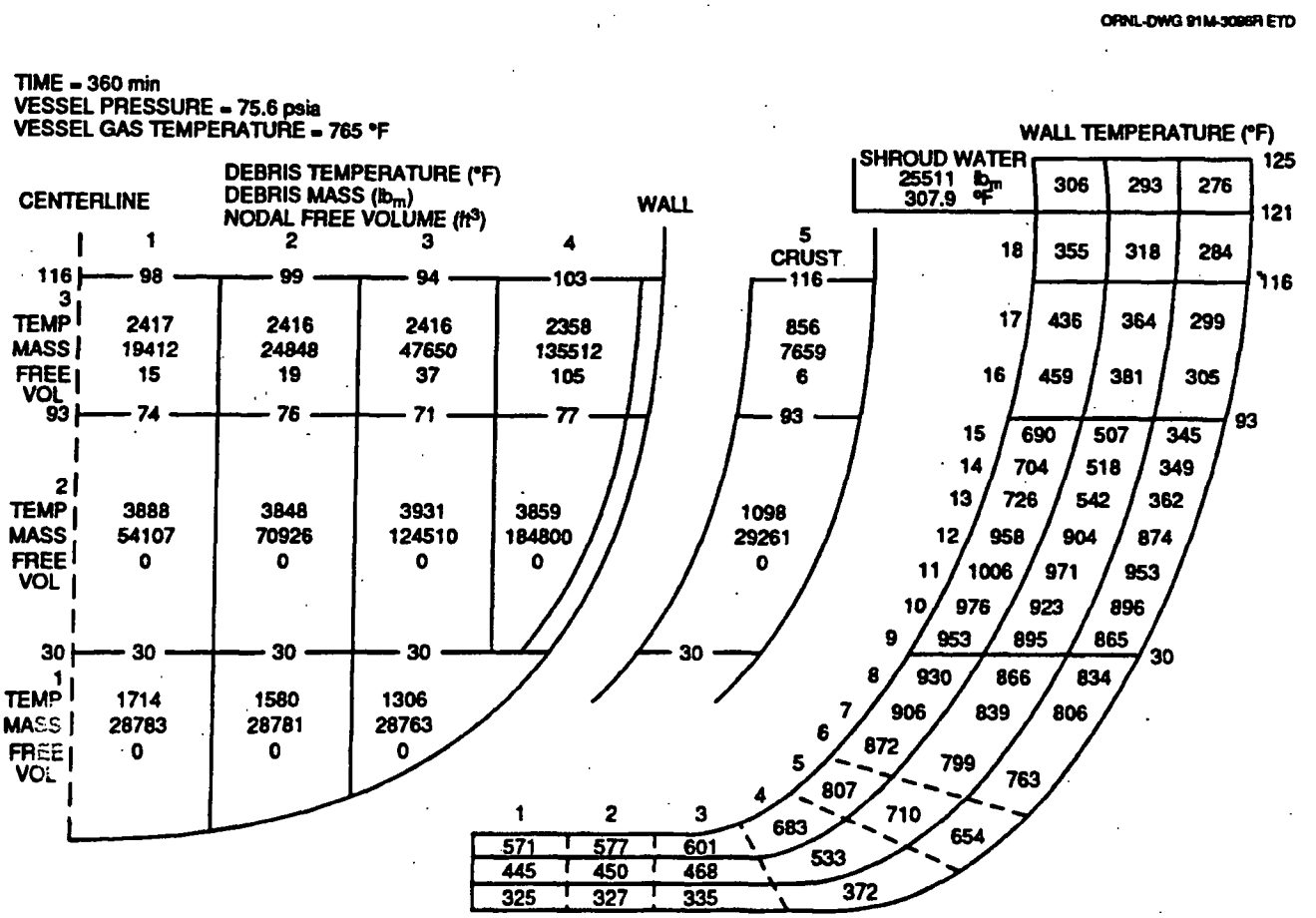

Figure 19.4 Lower plenum debris bed and vessel wall response at time 360 min after scram for base case

remaining within layer two are, of course, confined to control volume (2,5).

Table 19.3 Solid and liquid masses within debris layer two at time 360 min

Constituent	Solid mass (lb)	Liquid mass (lb)	Total (lb)
Zr	5,448	61,088	66,536
Fe	8,356	66,746	75,102
Cr	2,109	16,002	18,111
Ni	403	8,776	9,179
B_4C	1,660	0	1,660
ZrO_2	26,125	0	26,125
FeO	8	0	8
Fe_3O_4	19	416	435
Cr_2O_3	163	0	163
NiO	1	30	31
B_2O_3	32	0	32
UO_2	259,403	6,820	266,223
Totals	303,727	159,878	463,605

The calculated debris bed configuration at time 420 min is shown in Fig. 19.5. Control volumes (3,1) and (3,2) no longer exist as separate entities. Their masses and energies have been subsumed into the underlying control volumes (2,1) and 2,2), which had become primarily liquid and could no longer support the overlying solid debris. Melting of the first two eutectic mixtures is under way within control volumes (3,3) and (3,4).

By time 480 min, the central region of layer three no longer exists; it has been entirely subsumed into the control volumes of layer two. The calculated bed configuration at this time is shown in Fig. 19.6. The debris temperatures in the central three control volumes of layer two are sufficiently high to indicate the melting of some of the oxides (see Tables 18.3 and 18.4 for melting temperatures). The relative solid and liquid masses within layer two at this time are provided in Table 19.4.

The melting process within such a large debris bed proceeds slowly under the impetus of the decay power 9 h or more after scram. The calculated situations within the lower plenum debris bed and the bottom head wall at times

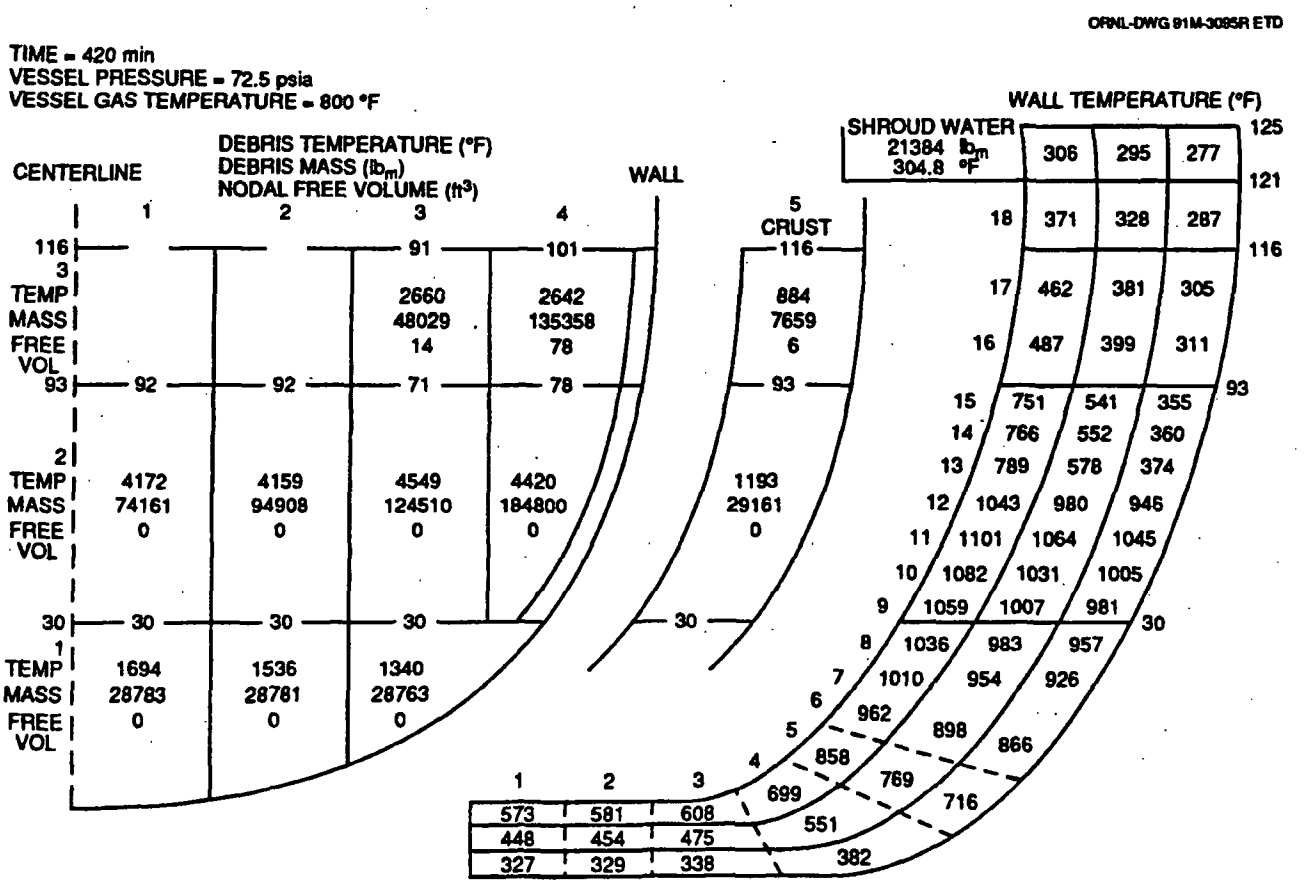

Figure 19.5 Lower plenum debris bed and vessel wall response at time 420 min after scram for base case

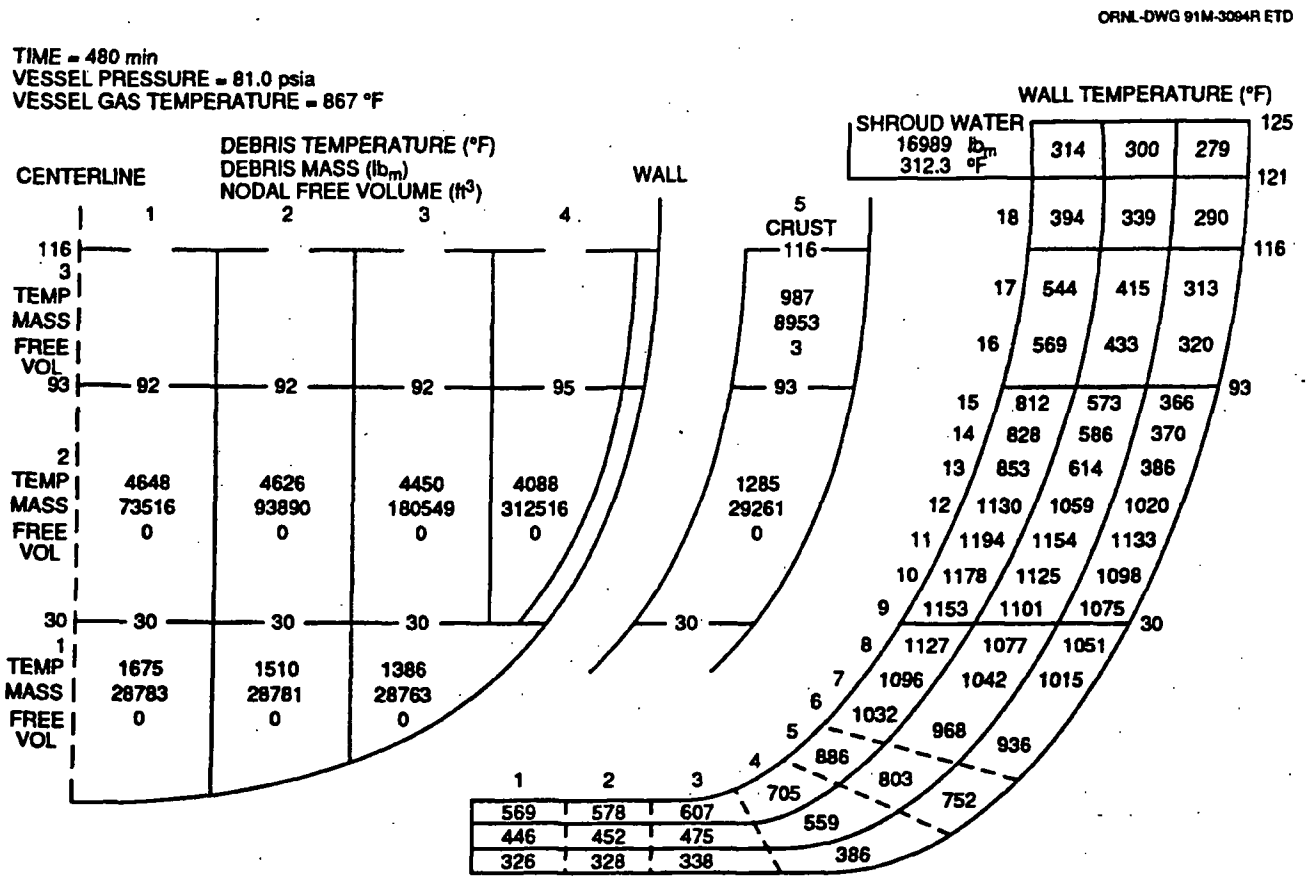

Figure 19.6 Lower plenum debris bed and vessel wall response at time 480 min after scram for base case

Table 19.4 Solid and liquid masses within debris layer two at time 480 min

Constituent	Solid mass (lb)	Liquid mass (lb)	Total (lb)
Zr	5,448	72,411	77,859
Fe	8,356	155,049	163,405
Cr	2,109	37,475	39,584
Ni	403	18,361	18,764
B_4C	1,020	820	1,840
ZrO_2	17,880	17,494	35,374
FeO	8	0	8
Fe_3O_4	19	465	484
Cr_2O_3	85	91	176
NiO	1	34	35
B_2O_3	15	17	32
UO_2	335,472	16,699	352,171
Totals	370,816	318,916	689,732

540 min and 600 min are shown in Figs. 19.7 and 19.8. Increasing temperatures and decreasing densities during the melting process have caused a slight increase in the thickness of the central region of layer two. By time 600 min, the three central control volumes have reached the melting temperature of UO_2. The relative amounts of solids and liquids within layer two at this time are listed in Table 19.5. The only solid materials other than UO_2 remaining within layer two are located within the small crust control volume (2,5).

It is worthwhile to pause and take note of two points with respect to Fig. 19.8. First, it should be noted that the highest wall temperatures occur within node 11, the second node beneath the location of the skirt attachment weld. These wall temperatures at time 600 min have reached levels at which creep rupture would be anticipated (see Fig. 18.6) if the vessel were not depressurized.

Second, it should be noted that the water in the downcomer region is almost exhausted at this time. This water has played an important role as a vessel heat sink in removing heat from the upper vessel wall and, more importantly, in cooling the vessel shroud. After it is exhausted (at about time 610 min), the upward radiation from the top of the debris bed will cause the shroud temperature to increase;

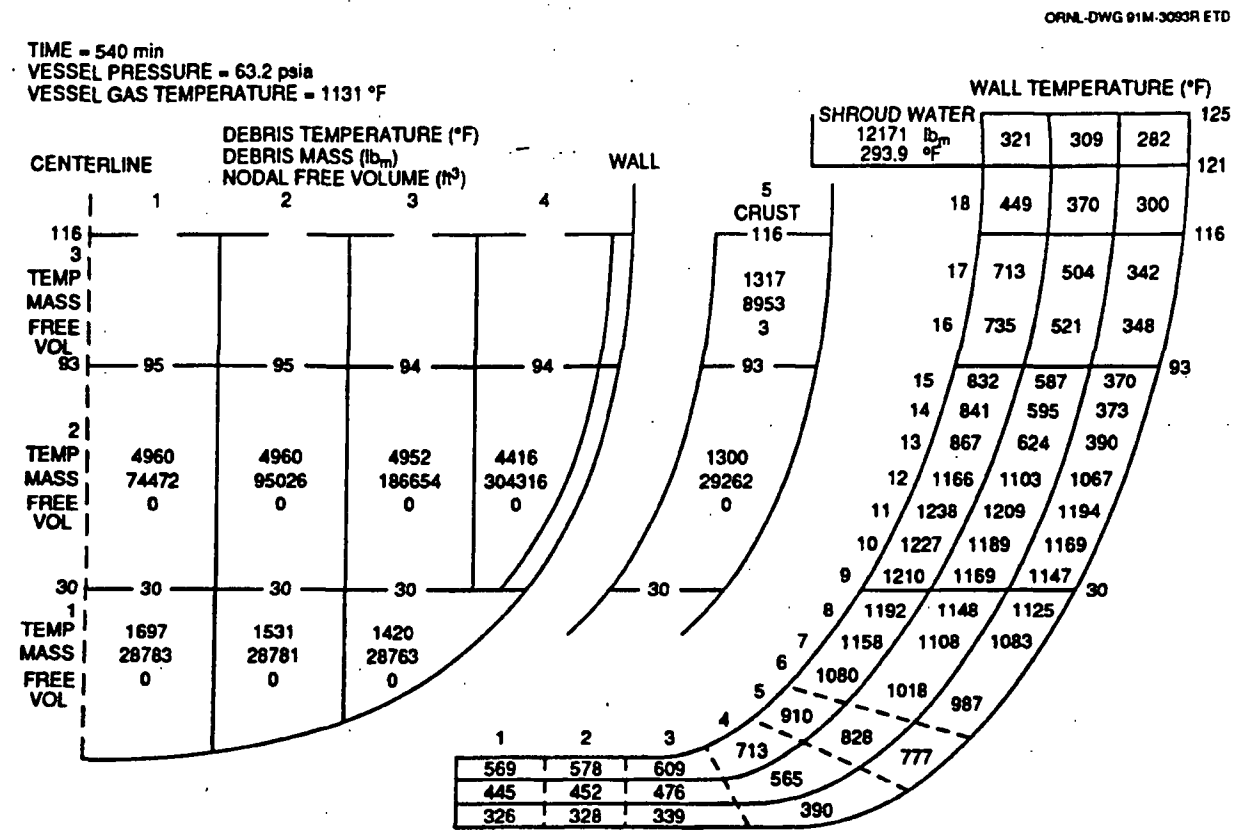

Figure 19.7 Lower plenum debris bed and vessel wall response at time 540 min after scram for base case

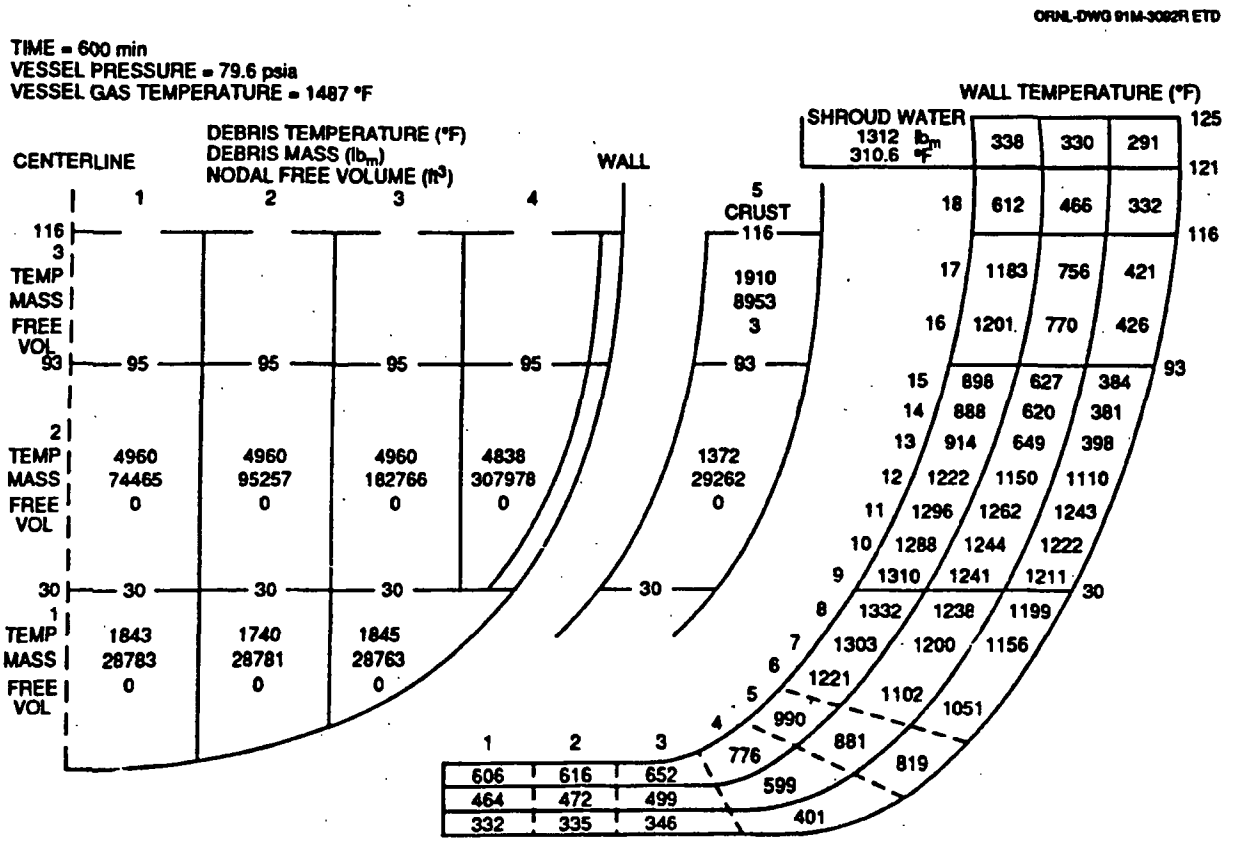

Figure 19.8 Lower plenum debris bed and vessel wall response at time 600 min after scram for base case

Table 19.5. Solid and liquid masses within debris layer two at time 600 min

Constituent	Solid mass (lb)	Liquid mass (lb)	Total (lb)
Zr	5,448	72,411	77,859
Fe	8,356	155,049	163,405
Cr	2,109	37,475	39,584
Ni	403	18,361	18,764
B_4C	73	1,767	1,840
ZrO_2	1,147	34,227	35,374
FeO	8	0	8
Fe_3O_4	19	465	484
Cr_2O_3	7	169	176
NiO	1	34	35
B_2O_3	1	31	32
UO_2	281,699	70,472	352,171
Totals	299,271	390,461	689,732

previously, this radiative energy transfer has primarily been consumed in evaporating water from the downcomer region.

The calculated situation at time 660 min is shown in Fig. 19.9. The temperature of the now dry shroud has increased to 1105°F (869 K). The reader may wish to refer to Figs. 15.2, 15.3, 17.1, and 18.2 to review the location of the core shroud [a stainless steel mass of ~160,000 lb (72,500 kg) modeled as a single heat sink*] with respect to the debris bed. In these calculations, the reverse side of the shroud is modeled to radiate heat to an additional single heat sink representing the hidden vessel internal structures such as the jet pumps, steam separators, and dryers—in all, an additional vessel internal stainless steel heat sink of ~225,000 lb (102,000 kg).

As the temperature of the shroud increases, so does the temperature of the vessel atmosphere. By time 720 min, the predicted temperature of the vessel atmosphere has reached 1800°F (1255 K) as indicated on Fig. 19.10. The corresponding temperature of the vessel shroud at this time is 1531°F (1106 K), and the maximum average wall node

*Although the lumped parameter representation of the core shroud is considered adequate for the present calculations, it is intended that a multicomponent shroud model will be utilized when the lower plenum debris bed models are operational within the MELCOR code architecture.

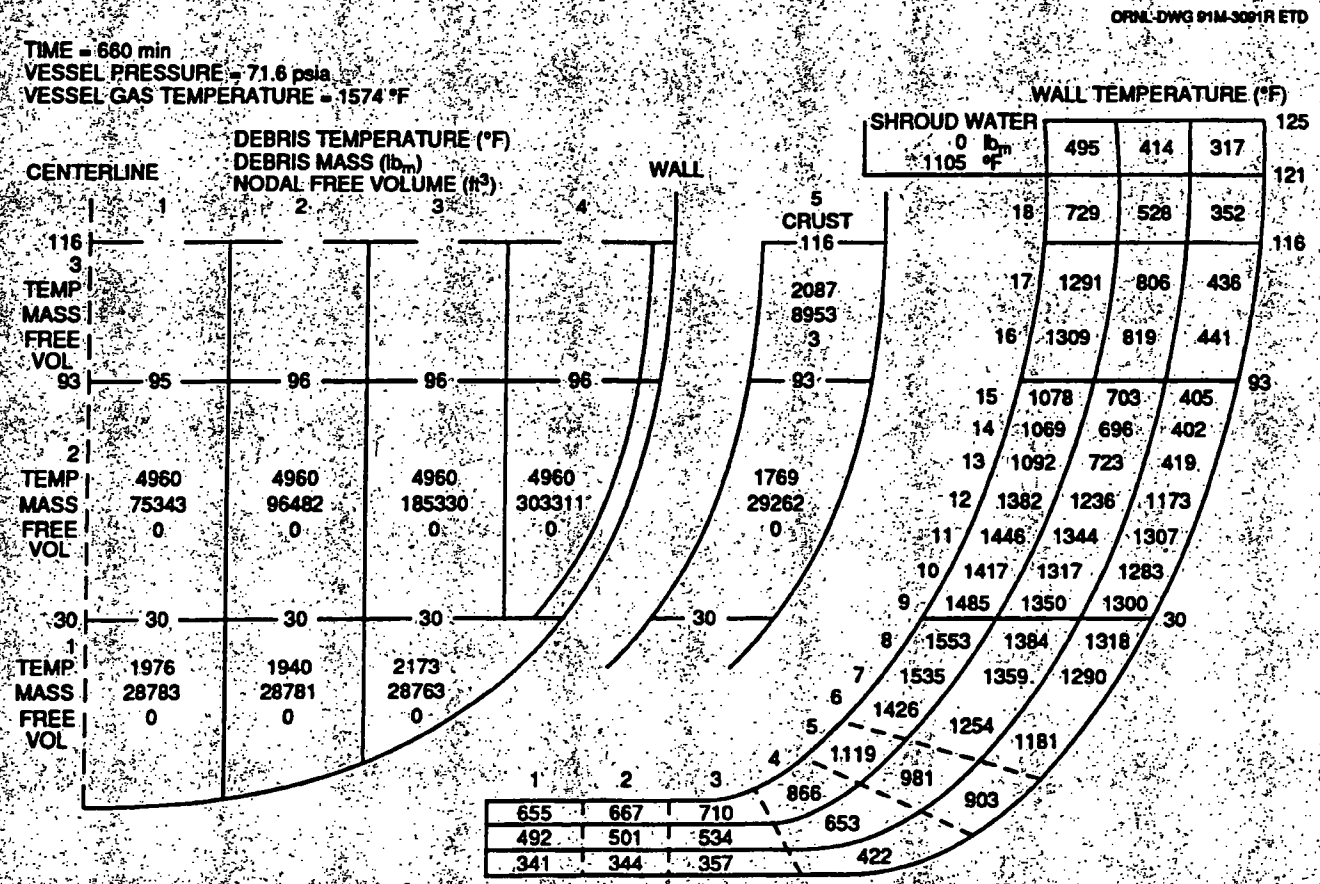

Figure 19.9 Lower plenum debris bed and vessel wall response at time 660 min after scram for base case

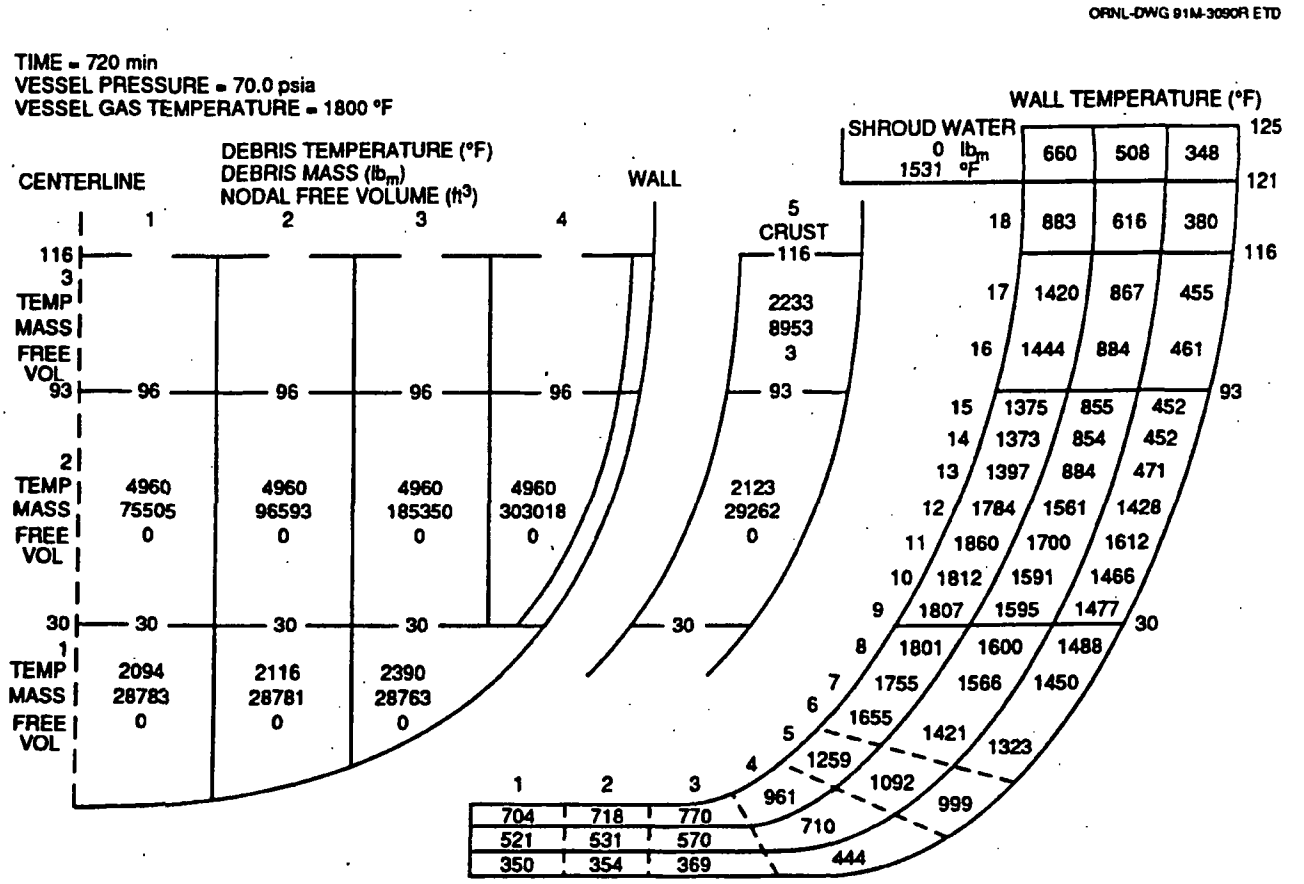

Figure 19.10 Lower plenum debris bed and vessel wall response at time 720 min after scram for base case

temperature (node 11) is 1724°F (1213 K). This is not quite sufficient to raise concerns with respect to creep rupture of the bottom head with the vessel in its depressurized state.*

By time 780 min, however, the average wall temperature at node 11 is predicted to have reached 2029°F (1383 K). The calculated general situation within the lower plenum and vessel bottom head at this time is represented in Fig. 19.11. Reference to Fig. 18.6 and Table 18.7 indicates that creep rupture of the vessel wall in the vicinity of node 11 should be expected within the next hour, even at the low (5.14-MPa) tensile stress considered with the reactor vessel depressurized. The calculated amounts of solid and liquid debris within layer two at time 780 min are listed in Table 19.6.

With the assumption that the wall does not fail, the calculated situation at time 840 min is shown in Fig. 19.12. The average wall temperature at node 11 is now 2221°F (1489 K), confirming that the time and temperature combination necessary for creep rupture would, according to the

Table 19.6 Solid and liquid masses within debris layer two at time 780 min

Constituent	Solid mass (lb)	Liquid mass (lb)	Total (lb)
Zr	5,448	72,411	77,859
Fe	8,356	155,049	163,405
Cr	2,109	37,475	39,584
Ni	403	18,361	18,764
B_4C	73	1,767	1,840
ZrO_2	1,147	34,227	35,374
FeO	8	0	8
Fe_3O_4	19	465	484
Cr_2O_3	7	169	176
NiO	1	34	35
B_2O_3	1	31	32
UO_2	229,621	122,550	352,171
Totals	247,193	442,539	689,732

*The average wall node temperature is the average temperature of the three radial segments that make up the node. In other words, it is the average temperature across the vessel wall at the nodal location.

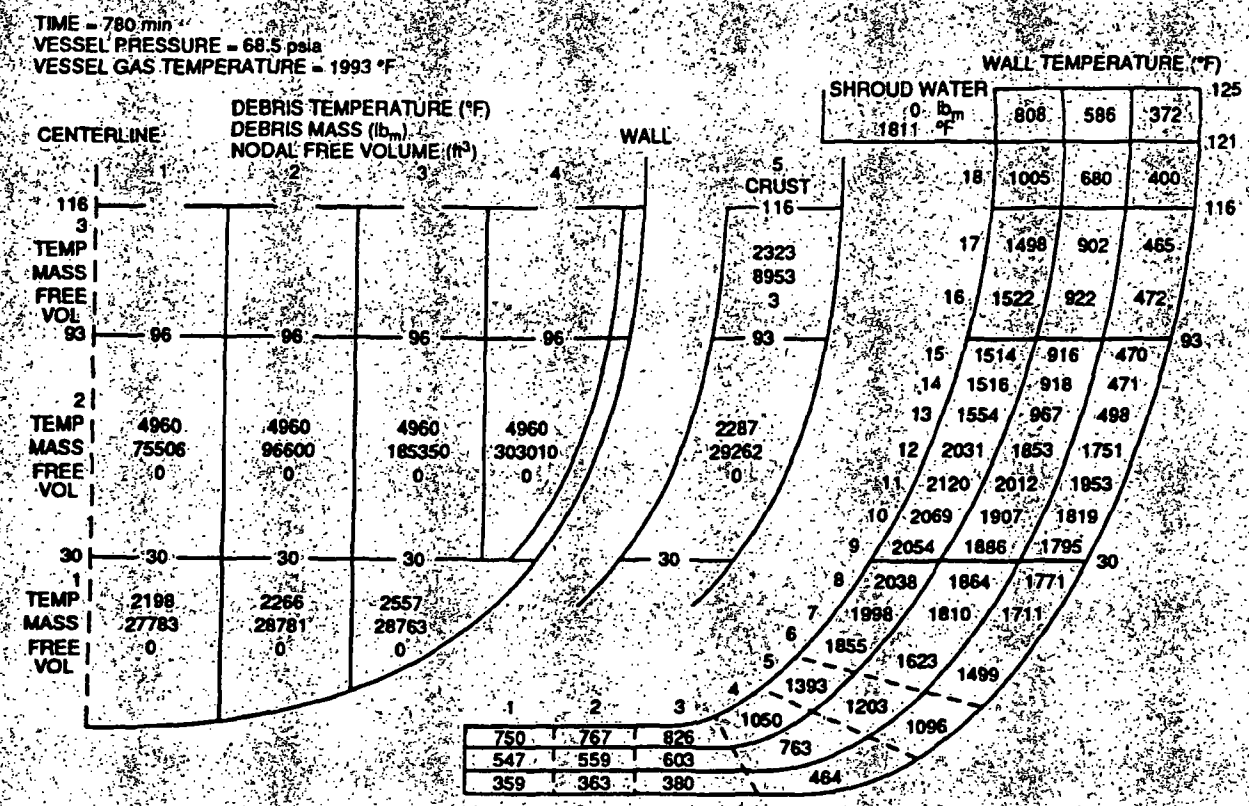

Figure 19.11 *Lower plenum debris bed and vessel wall response at time 780 min after scram for base case*

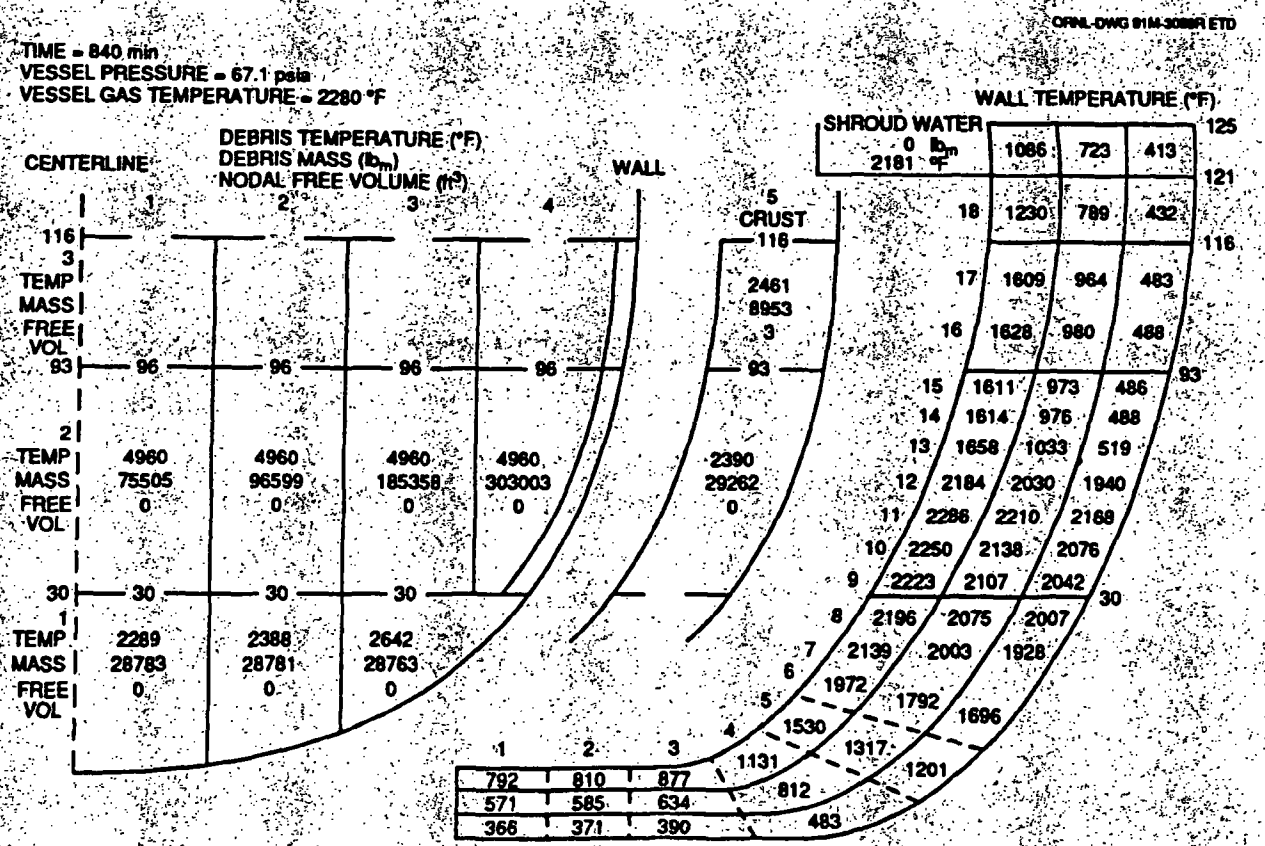

Figure 19.12 *Lower plenum debris bed and vessel wall response at time 840 min after scram for base case*

present prediction, occur at some point between 780 and 840 min. (A precise determination is beyond the reliability of the available information.)

These results indicate that the presence of the pocket of gas trapped in the armpit region beneath the skirt is ultimately fatal to the survival of the adjacent wall. The improved temperature response of the bottom head under conditions of a reduced or eliminated trapped gas pocket will be discussed in Chap. 20. First, however, the effectiveness of the water for the base case just presented will be evaluated. Although containment flooding was not predicted to prevent bottom head wall failure, it does provide a significant delay, as described in the following section.

19.5 The Effect of Drywell Flooding

The first and most important effect of drywell flooding with respect to delaying the release of core and structural debris from the reactor vessel is the prevention of bottom head penetration failures, as described in Chap. 17. (Without drywell flooding, bottom head penetration failures would be predicted to occur as early as 250 min after scram, or ~9 h before the creep rupture failure described in Sect. 19.4.) For completeness, however, the effect of drywell flooding in delaying gross failure of the bottom head will now be described. In other words, the effect of drywell flooding in delaying creep rupture of the bottom head wall under the assumption that bottom head penetration failures did not occur *in the dry case* will now be considered.

The calculated situation within the lower plenum debris bed and bottom head wall at time 600 min is shown for the dry case in Fig. 19.13. Without water to cool the lower portion of the reactor vessel, more energy is radiated upward within the vessel and downcomer dryout occurs earlier, at about time 585 min. The wall temperatures are also much higher, as may be seen by comparison of Figs. 19.13 and 19.8. The maximum average wall temperature of 2042°F (1390 K) occurs at node 12, but average nodal temperatures in excess of 2000°F (1366 K) extend from node 11 through node 15.

Average bottom head wall temperatures in excess of 2200°F (1478 K) are predicted by time 640 min for the dry

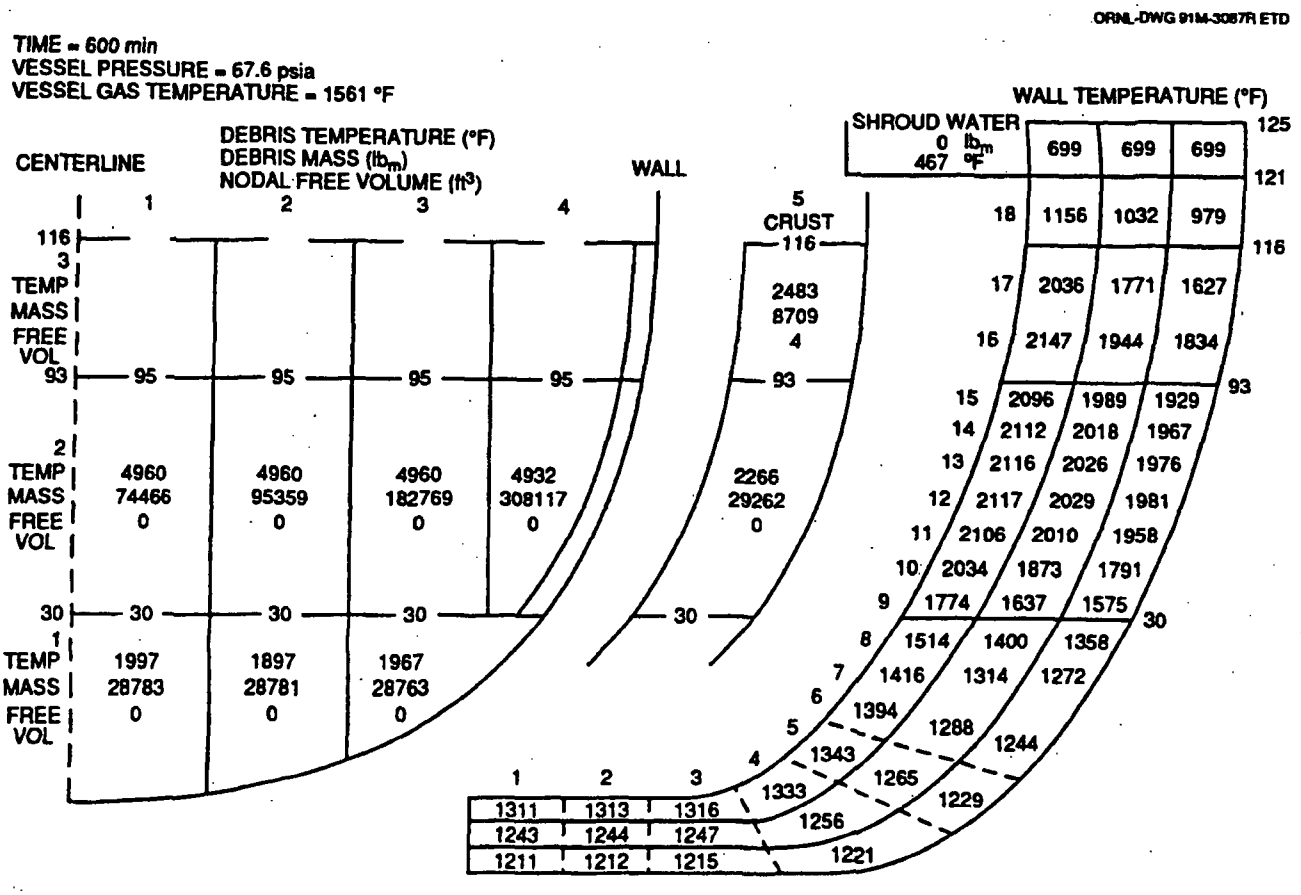

Figure 19.13 Lower plenum debris bed and vessel wall response at time 600 min after scram for the case without drywell flooding and without penetration failures

case as indicated in Fig. 19.14. Here, predicted average wall temperatures exceeding 2200°F (1478 K) are found in nodes 11 through 16, with the maximum average temperature [2248°F (1504 K)] occurring at node 12. Based upon these results, it is estimated that creep rupture failure would occur in the vicinity of wall node 12 for the dry case at some time between 600 and 640 min. This is ~3 h earlier than the creep rupture failure time estimated for the base case with drywell flooding.

A comparison of the calculated energy releases through the outer surface of the reactor vessel bottom head wall for the cases with and without drywell flooding is provided in Table 19.7. As indicated, this energy transfer increases during the period of the calculation as the temperature of the wall increases. The effectiveness of the drywell flooding is demonstrated by the much greater heat transfer from the wall for the base case, equivalent to approximately one-third of the decay heat release at the time of predicted wall failure. The vessel wall outer surface heat transfer for the dry case, on the other hand, never exceeds 1% of the decay heat release.

Because the conditions at the time the lower plenum debris bed is initially established are the same for the base case and the dry case and the amount of energy release by decay heating is very nearly the same,* it is of interest to consider where the energy retained in the dry case is stored. For example, it can be determined from the results listed in Table 19.7 that the heat transferred from the vessel wall during the period 246.2 to 600 min is 45.992×10^6 Btu (4.852×10^{10} J) for the base case and 1.704×10^6 Btu (0.180×10^{10} J) for the dry case. Where is the difference of 44.288×10^6 Btu (4.673×10^{10} J) stored in the dry case?

Table 19.8 provides the relative apportionment of the additional stored energy for the dry case at time 600 min. This apportionment is of interest because it is the *removal* of

*An insignificant difference arises because of a slightly earlier escape of some fission products from the debris for the dry case, where melting begins earlier.

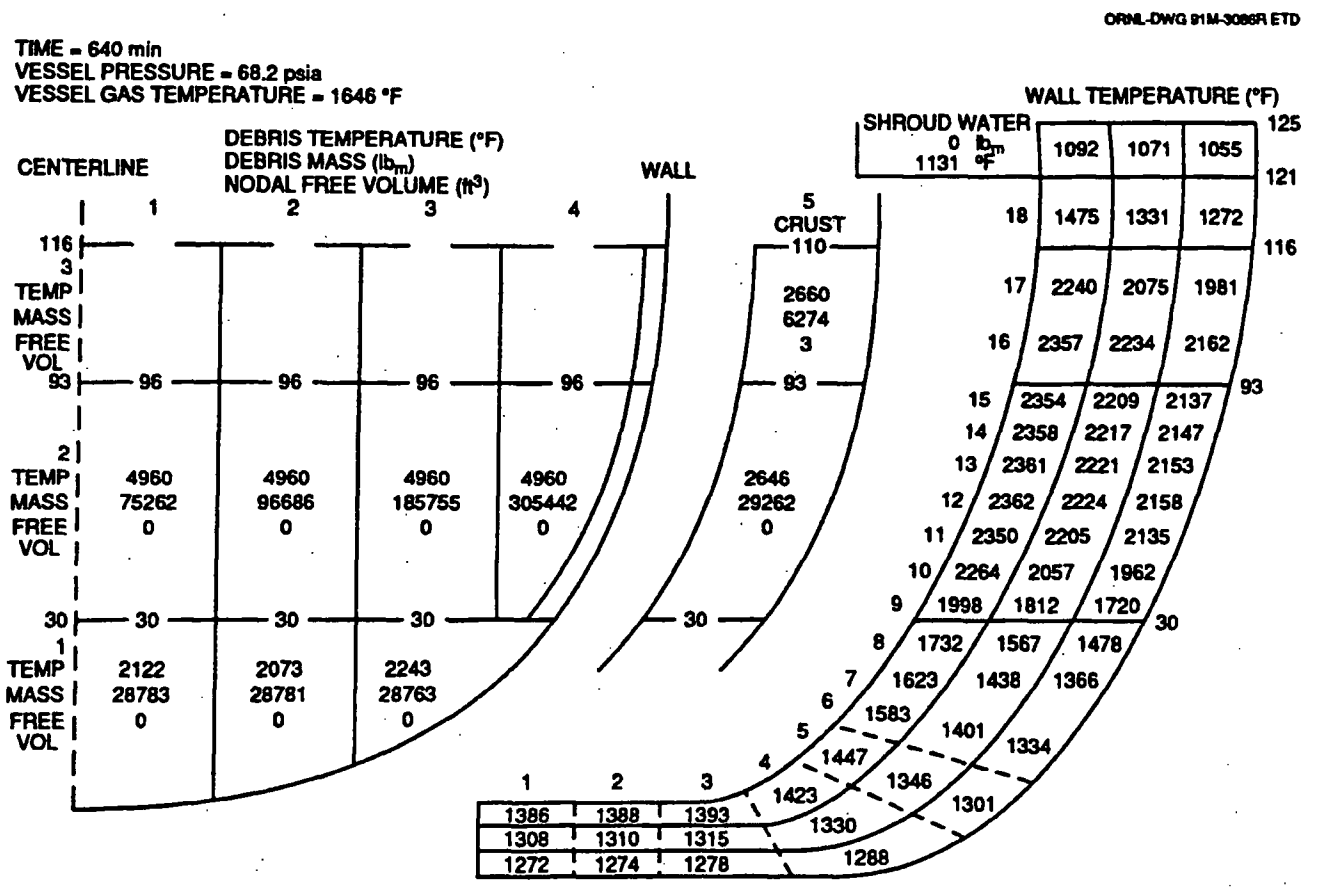

Figure 19.14 Lower plenum debris bed and vessel wall response at time 640 min after scram for the case without drywell flooding and without penetration failures

Table 19.7 Comparison of integrated heat transfers from the outer surface of the reactor vessel bottom head with and without drywell flooding

Period (min)	Decay heat released (Btu × 10^6)	Heat transferred from the vessel wall			
		Base case		Dry case	
		Btu × 10^6	%	Btu × 10^6	%
246.2–300	64.874	4.883	7.2	0.084	0.1
300–360	68.258	6.113	9.0	0.199	0.3
360–420	65.147	6.912	10.6	0.266	0.4
420–480	62.355	7.536	12.1	0.321	0.5
480–540	58.531	8.950	15.3	0.381	0.7
540–600	56.595	11.598	20.5	0.453	0.8
600–660	54.992	13.985	25.4	0.560	1.0
660–720	53.581	17.310	32.3		
720–780	52.462	19.579	37.3		

Table 19.8 Relative locations of the additional energy storage within the vessel and the debris for the dry case

Location	Percent of additional stored energy
Vessel atmosphere	0.1
Upper vessel structure	10.4
Central molten region of debris bed	12.0
Debris bed crust nodes adjacent to wall	7.2
Bottom head wall	70.3
Total	100.0

this additional energy that is the effect of the external vessel wall cooling provided by drywell flooding. It should be noted that the majority (77.5%) of this energy is stored in the vessel wall and the thin adjacent debris crust control volumes. One might think that downward heat transfer from the molten central region of the debris bed would be greatly reduced by a lack of wall cooling. However, the lesson of Table 19.8 is that downward heat transfer from the molten central region continues for the dry case, but the transferred energy is held up (stored) in the debris crust and in the wall rather than passed through to the drywell. That the effect of the increased wall and crust temperatures in reducing the downward heat transfer is small is due to the very large temperature difference between the molten central region of the debris and the wall; this temperature difference remains large in the dry case.

20 Results with Venting of the Vessel Support Skirt

S. A. Hodge, J. C. Cleveland, T. S. Kress*

It was shown in the previous chapter that although surrounding the reactor vessel bottom head with water would certainly delay the onset of failure by creep rupture, such a failure would ultimately occur in the region of the bottom head adjacent to the trapped gas pocket underneath the vessel support skirt. In Sect. 15.4, means were suggested for reducing or eliminating this gas pocket. In this chapter, the results of calculations based upon a reduced gas pocket are discussed.

20.1 Leakage from the Manhole Access Cover

The relation between the water level within the drywell outside of the reactor vessel support skirt and the water level within the skirt for the case with venting at the manhole access cover is provided in Table 20.1. It is assumed that the containment pressure at the time the water levels are established is 20 psia (0.138 MPa). The advantage of venting at the access cover can be recognized by comparison of the information in Table 20.1 with that in Table 15.1. For example, with a drywell water level equivalent to the top of the core [366.3 in. (9.304 m) above vessel zero], the height of water within the skirt relative to the low point of the bottom head outer surface is 29.7 in. (0.754 m) if there is gas escape from the access cover, but only 10.2 in. (0.259 m) if the access cover is tight.

*All work in connection with this project completed before becoming a member of the Advisory Committee on Reactor Safeguards.

In this section, the results of calculations performed to assess the improvement in bottom head cooling obtained by venting from the manhole access cover are discussed. For these calculations, it was assumed that wall nodes one through eight are covered with water and transfer heat by nucleate boiling. This coverage is depicted on Fig. 20.1, which may be compared with Fig. 19.1 showing the coverage for the base case. For the base case, wall nodes 5 through 12 underneath the skirt were assumed to be uncovered, equivalent to 44.8% of the total bottom head outer surface. For the calculations with venting at the access hole, the uncovered surface beneath the skirt attachment is limited to wall nodes 9 through 12. This reduces the uncovered portion of the bottom head from 44.8% to 27.3% of the total outer surface area.

Because the top of the outer surface of wall node eight is located 32.02 in. (0.813 m) above the low point of the bottom head outer surface (Table 18.5), it can be determined by interpolation between the last two entries of Table 20.1 that the corresponding height of water within the drywell is 506 in. (12.852 m). This is less than the elevation of the drywell vent [528 in. (13.411 m) above vessel zero] and is, therefore, attainable.

In Sect. 19.4, it was shown that the results for the base case indicate that the calculated wall temperatures are sufficiently high by time 780 min that the potential for creep rupture failure of the wall becomes of concern. For

Table 20.1 Water level within the vessel skirt with gas leakage at the access hole for the Browns Ferry containment at 20 psia

Water level outside skirt		Water level inside skirt
Location of surface	Height relative to vessel zero[a] (in).	Height relative to low point of vessel outer surface (in.)
Top of access hole	12.0	20.4
Bottom head center of curvature	125.5	23.9
Recirculation nozzle centerline	161.5	24.9
Base of core	216.3	26.4
Core midplane	291.3	28.2
Top of core	366.3	29.7
Top of separators	607.5	33.7

[a]Vessel zero is the lowest point on the internal surface of the reactor vessel bottom head.

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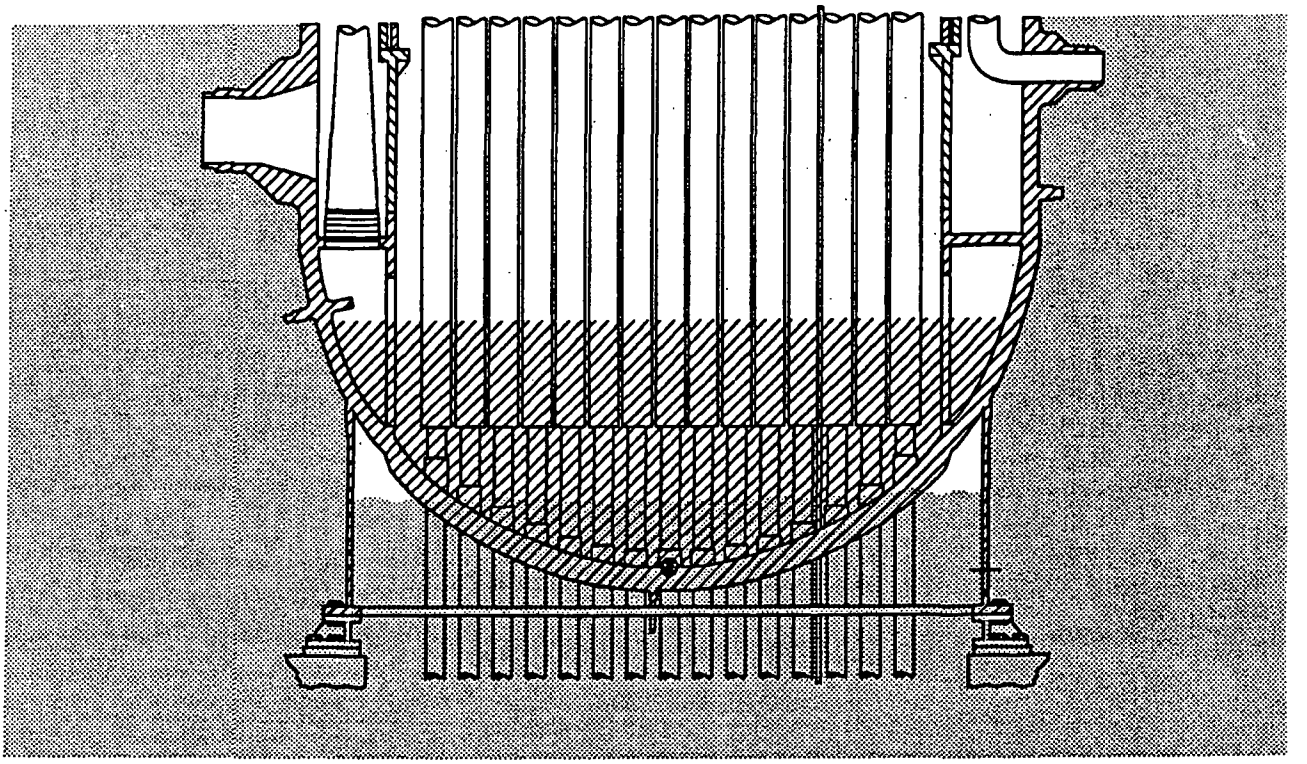

Figure 20.1 Volume of gas trapped beneath the reactor vessel support skirt reduced by providing vent path from manhole access cover

comparison, the calculated wall temperatures at time 780 min for the case with venting at the vessel skirt access hole are shown in Fig. 20.2. Whereas the average wall temperature at this time for wall node 11 is 2029°F (1383 K) for the base case (Fig. 19.11), it is 1937°F (1331 K) for the vented case (Fig. 20.2).

For the base case, creep rupture failure of the wall was judged certain by time 840 min (Fig. 19.12), for which the average temperature of wall node 11 was predicted to be 2221°F (1489 K). For the case with venting of the skirt, the calculated situation at time 840 min is shown in Fig. 20.3. The average wall temperature at node 11 is now 2124°F (1435 K). In general, the effect of the reduced gas pocket is to decrease the calculated maximum average wall node temperature by about 100°F (56 K). This delays the wall heating, but will not prevent ultimate failure by creep rupture.

By time 900 min, the calculated wall temperatures for the case with venting of the skirt have reached the levels shown in Fig. 20.4. The average temperature of wall node 11 is now 2233°F (1496 K), ensuring that creep rupture of the wall would have occurred before this time.

Because the creep rupture failure was estimated to occur at some time during the period 780 to 840 min after scram for the base case, it can be concluded that the advantage obtained by venting the skirt during containment flooding is limited to delaying the predicted failure of the wall by ~1 h.

20.2 The Case with Complete Venting

It was suggested in Sect. 15.4 that a drywell flooding strategy to completely cover the BWR reactor vessel bottom head with water might be achieved if several small holes were drilled through the skirt at points just below the attachment weld. This would provide an elevated gas release pathway such that the pocket of trapped atmosphere that would be produced by drywell flooding strategies for existing plants could be completely eliminated. It is recognized that drilling of the vessel support skirt for an existing facility is not practical, but the provision of gas escape holes in advanced plant designs might be feasible. It has been shown previously that partial wetting of the bottom head can only delay the failure of the wall by creep rupture. In this section, the effect of complete venting of the vessel support skirt will be investigated.

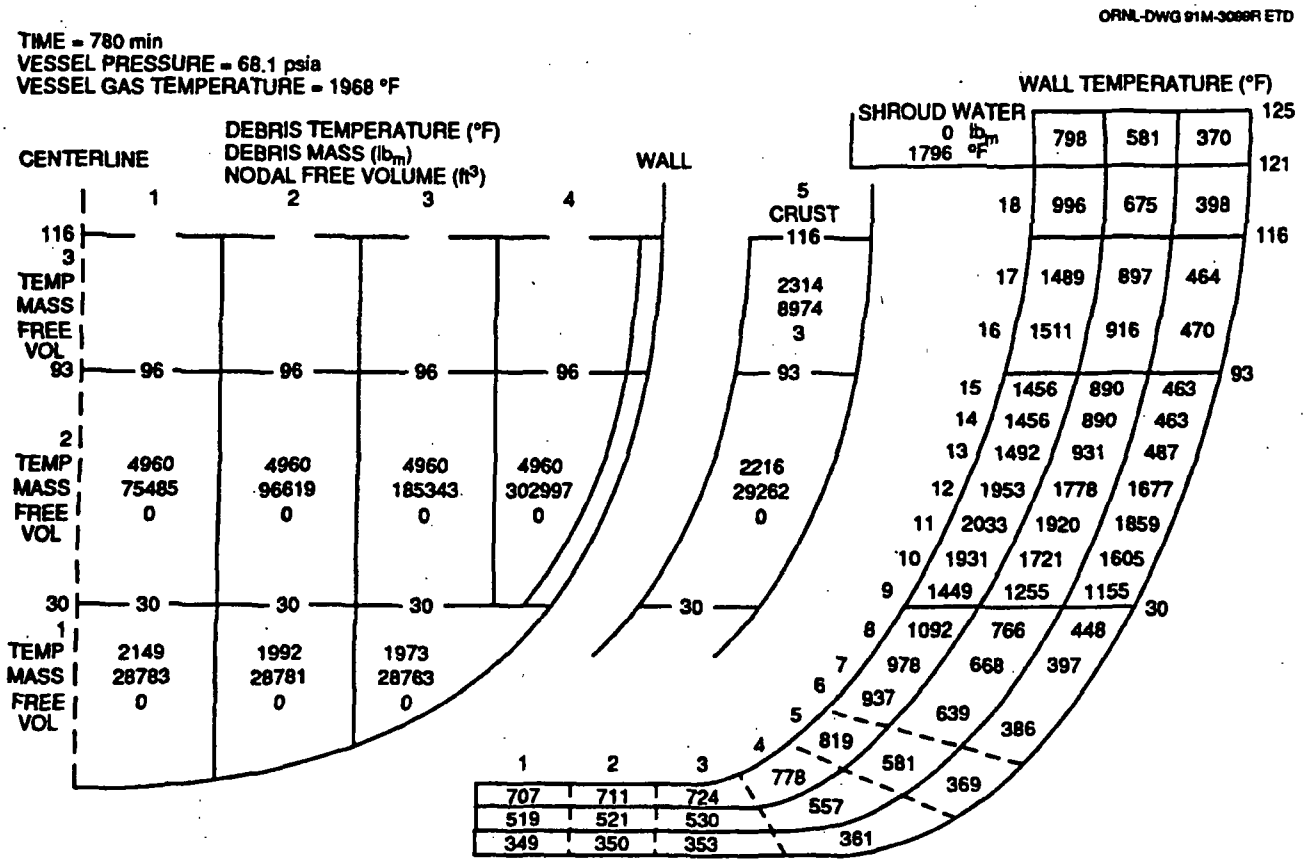

Figure 20.2 Lower plenum debris bed and vessel wall response at time 780 min after scram for case with venting from vessel skirt access hole cover

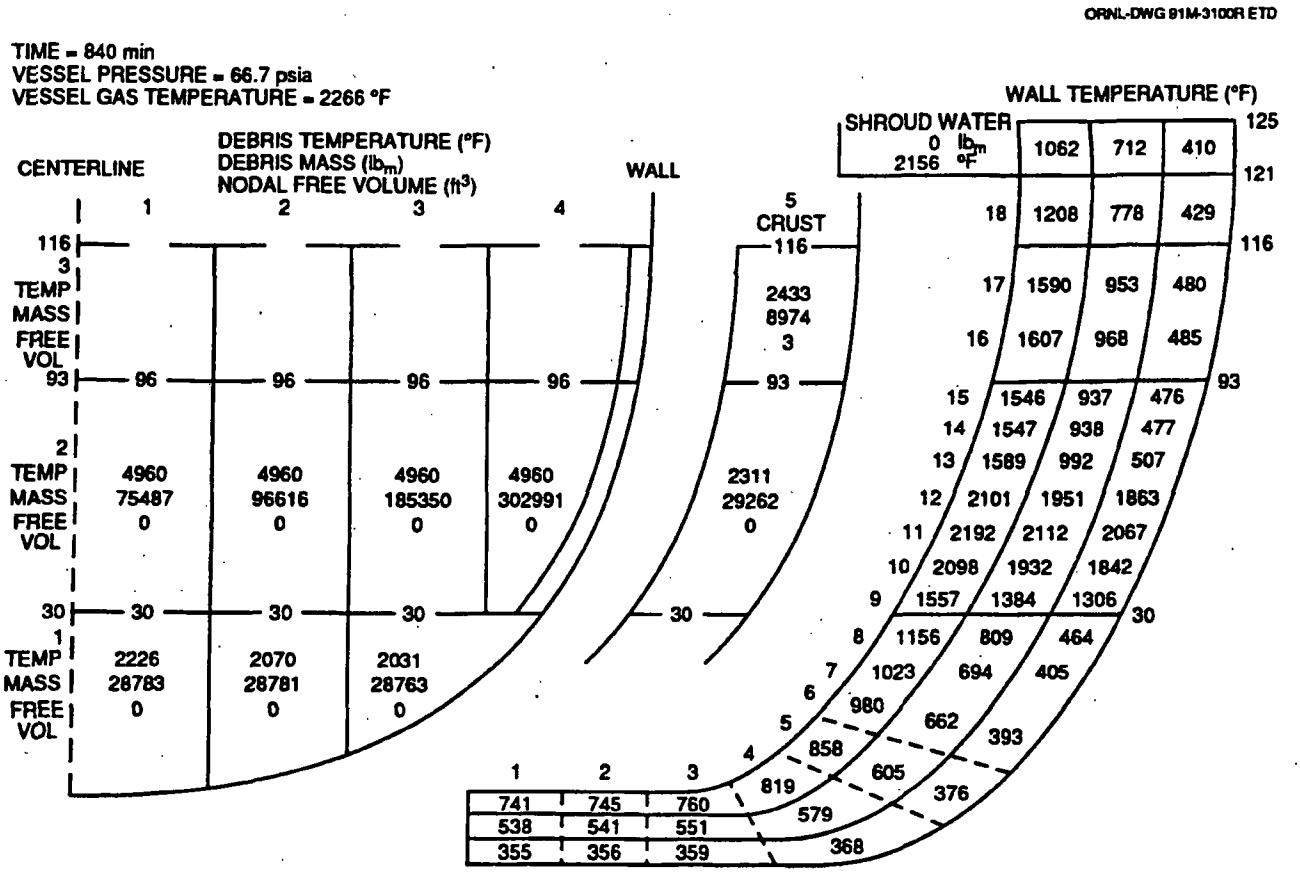

Figure 20.3 Lower plenum debris bed and vessel wall response at time 840 min after scram for case with venting from vessel skirt access hole cover

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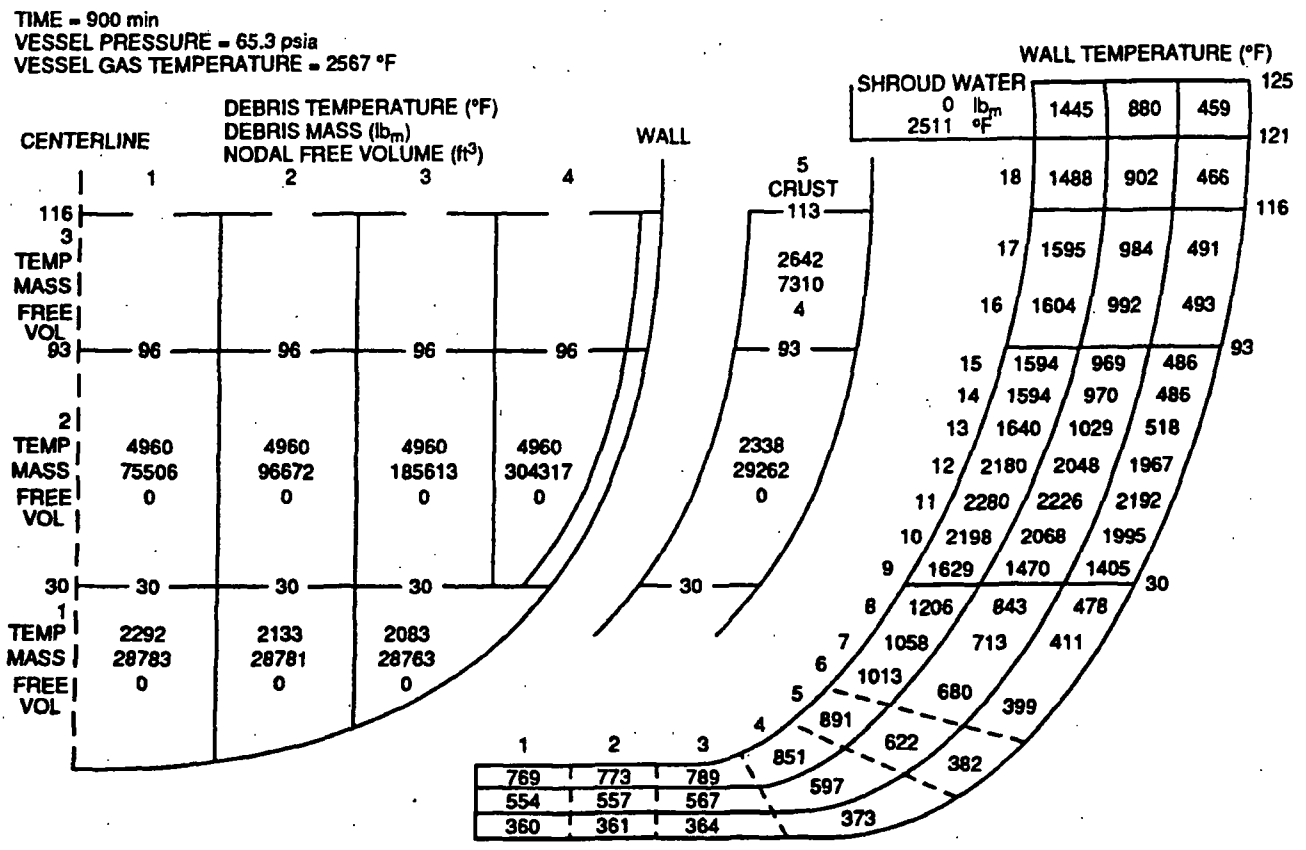

Figure 20.4 Lower plenum debris bed and vessel wall response at time 900 min after scram for case with venting from vessel skirt access hole cover

In Sect. 20.1, results presented for the case with partial venting at the skirt access manhole cover indicated that wall failure would occur in the vicinity of wall node 11 before time 900 min. For the case with complete venting, however, the predicted vessel wall temperatures at this time are nowhere near the values required to induce creep rupture (with the vessel depressurized). The dramatic advantage gained by complete venting vs partial venting can be appreciated by comparing the wall temperatures shown in Fig. 20.5 with those displayed in Fig. 20.4.

A comparison of the integrated heat transfers from the outer surface of the reactor vessel bottom head for each 1-h period of the calculations for the cases of complete and partial venting is provided in Table 20.2. As indicated, the heat transferred from the vessel wall represents a larger percentage of the decay heat release during each period for the case of complete skirt venting. (Similar information for the base case and the case without drywell flooding is available in Table 19.7, should the reader desire to compare the relative wall heat transfers for all four cases.)

Moving ahead 1 h to time 960 min, the calculated situation in the lower plenum debris bed and bottom head for the case with complete skirt venting is as shown in Fig. 20.6. As indicated, the shroud temperature at this time is 2550°F (1672 K), the melting temperature of stainless steel. Shroud melting is predicted to begin at about time 919 min in this calculation, and ~42,000 lb (19,000 kg) of liquid stainless steel has entered the debris bed by time 960 min. This is the reason for the increased elevation of the upper surface of the debris, from 96 in. (2.438 m) at time 900 min (Fig. 20.5) to the 99 in. (2.515 m) indicated in Fig. 20.6.

Also note that the temperatures of the debris crust control volumes and of wall nodes 6 through 17 indicated in Fig. 20.6 are lower than the temperatures calculated for these regions 1 h earlier (Fig. 20.5). This reduction occurs because much of the decay heating is now consumed in increasing the temperature of the liquid stainless steel entering the central region of the bed from the melting point to the local bed temperature. The calculated

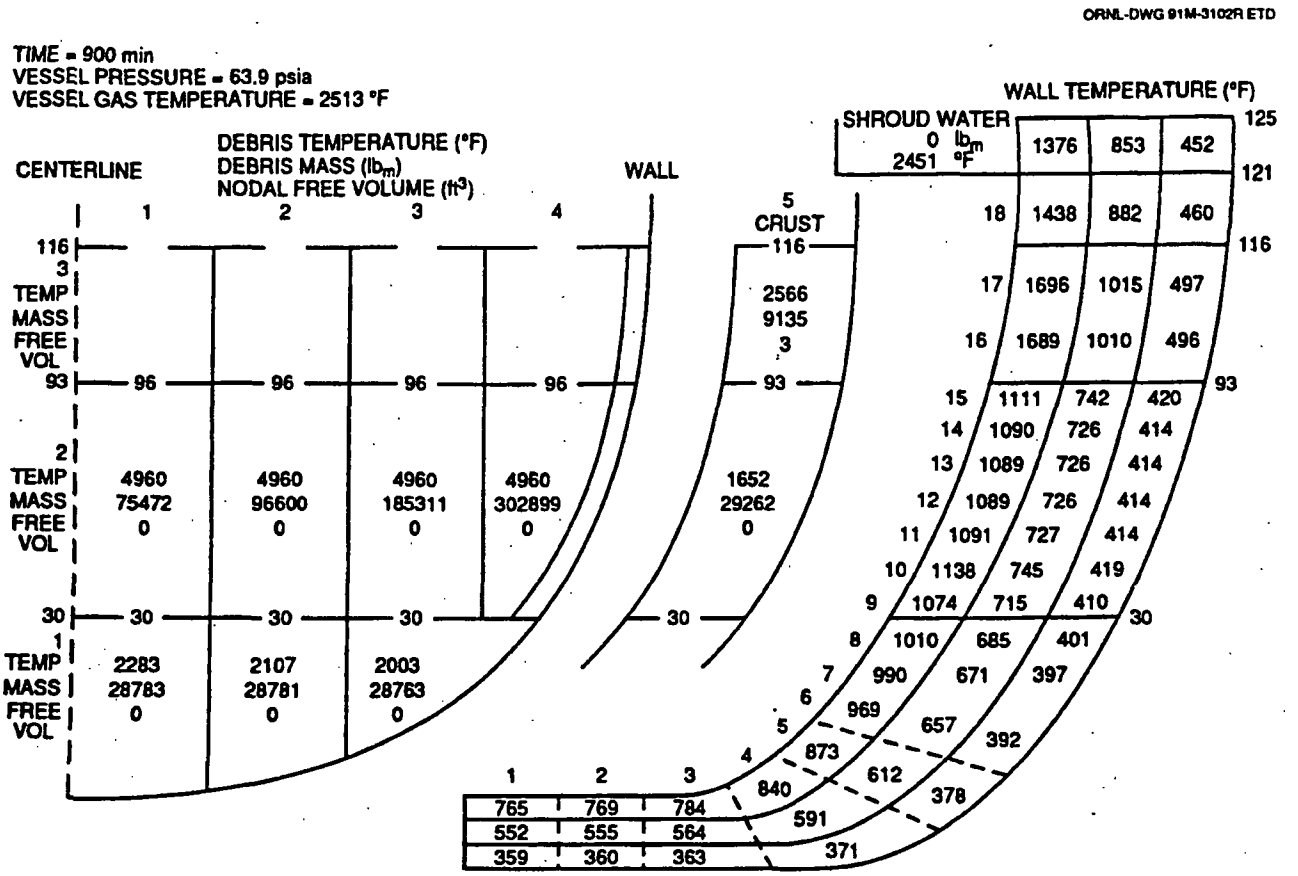

Figure 20.5 Lower plenum debris bed and vessel wall response at time 900 min after scram for case with interior of vessel skirt completely vented

Table 20.2 Comparison of integrated heat transfers from the outer surface of the reactor vessel bottom head for complete and partial skirt venting

Period (min)	Decay heat released (Btu × 10^6)	Heat transferred from the vessel wall			
		Complete venting		Partial venting	
		Btu × 10^6	%	Btu × 10^6	%
246.2–300	64.874	8.489	13.1	5.693	8.8
300–360	68.258	10.138	14.9	7.958	11.7
360–420	65.147	9.716	14.9	8.485	13.0
420–480	62.355	9.999	16.0	8.909	14.3
480–540	58.531	11.173	19.1	10.185	17.4
540–600	56.595	13.644	24.1	12.782	22.6
600–660	54.992	16.628	30.2	15.629	28.4
660–720	53.581	22.083	41.2	19.491	36.4
720–780	52.462	26.030	49.6	22.128	42.2
780–840	51.344	27.398	53.4	24.037	46.8
840–900	50.223	28.682	57.1	26.086	51.9

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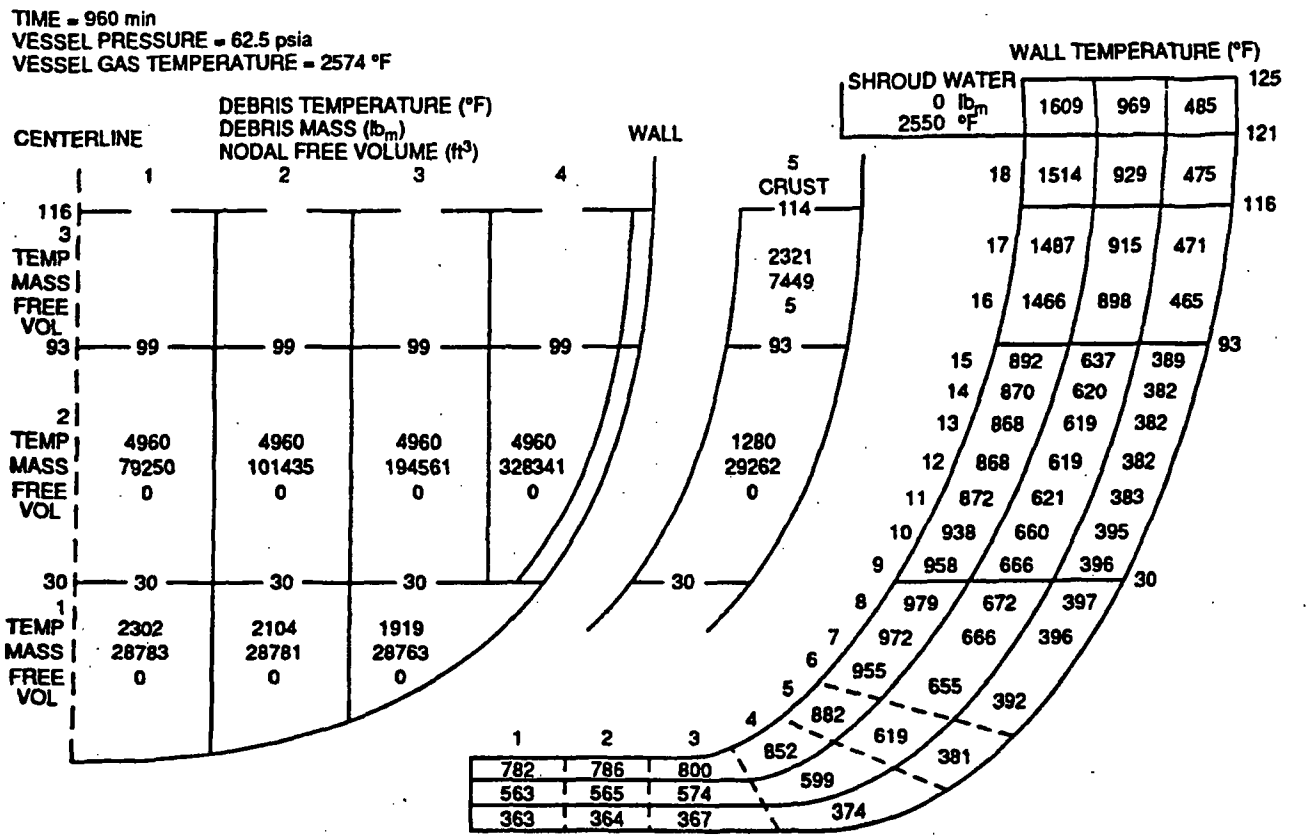

Figure 20.6 Lower plenum debris bed and vessel wall response at time 960 min after scram for case with interior of vessel skirt completely vented

compositions of the layer two debris at the beginning and end of this period are indicated in Tables 20.3 and 20.4. Comparison of these two compositions reveals that some of the previously molten UO_2 has reverted to the solid phase to release additional energy as necessary to superheat the entering stainless steel. [Some of the metallic constituents of debris control volume (3,5) have also melted and relocated during this period; this is the reason for the additional zirconium found in layer two at time 960 min.]

The BWR reactor vessel internal mass of stainless steel components is very large. For these calculations based upon Peach Bottom or Browns Ferry, ~375,000 lb (170,000 kg) would remain intact at the time melting induced by radiation from the upper debris bed surface began. Melting of such a large mass would, of course, occur over a long period of time. Moving ahead 3 h in the calculated results, the predicted situation in the lower plenum debris bed and the bottom head wall at time 1140 min after scram is depicted in Fig. 20.7. The corresponding debris bed layer two composition at this time is described in Table 20.5.

Table 20.3 Solid and liquid masses within debris layer two at time 900 min

Constituent	Solid mass (lb)	Liquid mass (lb)	Total (lb)
Zr	5,448	72,387	77,835
Fe	8,356	154,929	163,285
Cr	2,109	37,445	39,554
Ni	403	18,353	18,756
B_4C	73	1,766	1,839
ZrO_2	1,147	34,227	35,374
FeO	8	0	8
Fe_3O_4	19	463	482
Cr_2O_3	7	167	174
NiO	1	33	34
B_2O_3	1	30	31
UO_2	229,695	122,476	352,171
Totals	247,267	442,276	689,543

Table 20.4 Solid and liquid masses within debris layer two at time 960 min

Constituent	Solid mass (lb)	Liquid mass (lb)	Total (lb)
Zr	5,448	72,849	78,297
Fe	8,356	186,694	195,050
Cr	2,109	45,187	47,296
Ni	403	21,688	22,091
B_4C	73	1,765	1,838
ZrO_2	1,147	34,227	35,374
FeO	8	0	8
Fe_3O_4	19	463	482
Cr_2O_3	7	167	174
NiO	1	33	34
B_2O_3	1	30	31
UO_2	244,796	107,375	352,171
Totals	262,368	470,478	732,846

About 258,000 lb (117,000 kg) of liquid stainless steel is predicted to have relocated from the upper vessel internal structures into the debris bed. As indicated in Fig. 20.7, this has increased the calculated elevation of the bed upper surface to 116 in. (2.946 m), the same as when the bed was initially formed (Fig. 19.2). However, the bed was initially comprised of solid particles with free volume within the interstitial pores, while now the upper central portion is primarily comprised of liquid metals.

The calculated temperature of debris bed control volume (2,4) at time 1140 min has fallen below the melting temperature [4960°F (3011 K)] of UO_2 and the information provided in Table 20.5 confirms that most of the UO_2 in layer two is predicted to have solidified by this time. As stated previously, decay heating within the bed is inadequate to superheat all of the entering stainless steel to the local bed temperatures; some of the liquid UO_2 must revert to the solid phase to release the additional required energy.

Subsequent to the onset of melting of the reactor vessel upper internal stainless steel structures, the predicted vessel wall temperatures decrease slowly and remain far below threatening values. However, all remaining intact stainless steel of the vessel internals is predicted to be exhausted at about time 1189 min in these calculations. After this, the

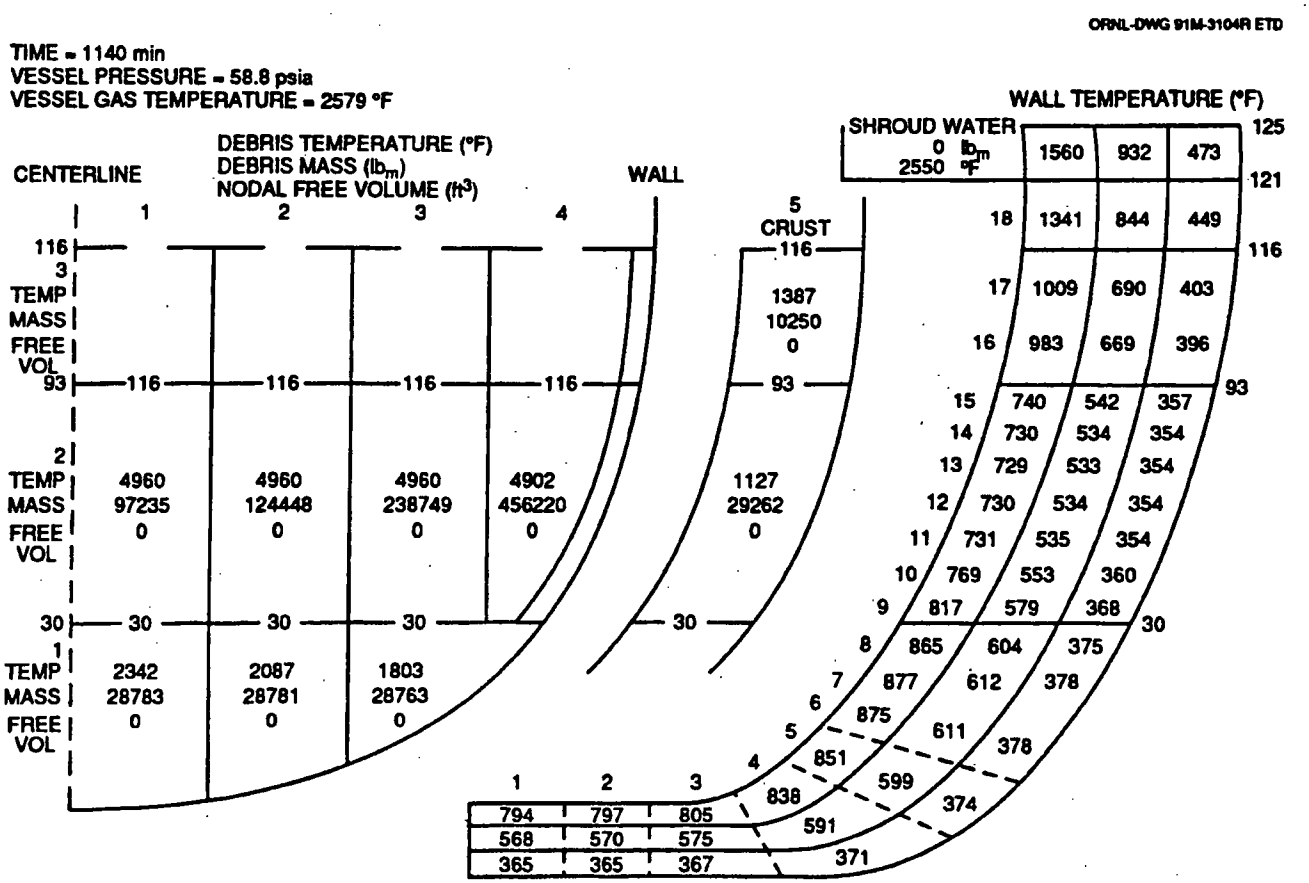

Figure 20.7 Lower plenum debris bed and vessel wall response at time 1140 min after scram for case with interior of vessel skirt completely vented

Table 20.5 Solid and liquid masses within debris layer two at time 1140 min

Constituent	Solid mass (lb)	Liquid mass (lb)	Total (lb)
Zr	5,448	72,849	78,297
Fe	8,356	344,364	352,720
Cr	2,109	83,539	85,648
Ni	403	38,733	39,136
B_4C	73	1,765	1,838
ZrO_2	1,147	34,227	35,374
FeO	8	0	8
Fe_3O_4	19	463	482
Cr_2O_3	7	167	174
NiO	1	33	34
B_2O_3	1	30	31
UO_2	316,292	35,879	352,171
Totals	333,864	612,049	945,913

receiver material for radiation from the surface of the debris bed is the carbon steel of the upper reactor vessel wall. Because carbon steel has a higher melting temperature than stainless steel [2800°F vs 2550°F (1811 vs 1672 K)], the introduction of liquid steel into the debris bed would be temporarily interrupted while the temperature of the upper reactor vessel wall increased to its melting point.

The predicted situation at 20 h after scram is shown in Fig. 20.8. The temperature of the upper vessel is 2772°F (1795 K), so melting of the carbon steel has not yet begun. The debris bed volume has increased to the point that it occupies the entire bottom head hemisphere. The calculated bottom head wall temperatures are not threatening, as long as the reactor vessel remains depressurized.

All debris bed control volume temperatures at this time are below the melting temperature of UO_2 as an independent species (Table 18.4). The layer two material compositions are described in Table 20.6. The only remaining liquid UO_2 is that associated with the eutectic mixtures described in Tables 17.2 and 17.3. The central region of debris bed layer two now consists primarily of solid UO_2 surrounded by liquid metals.

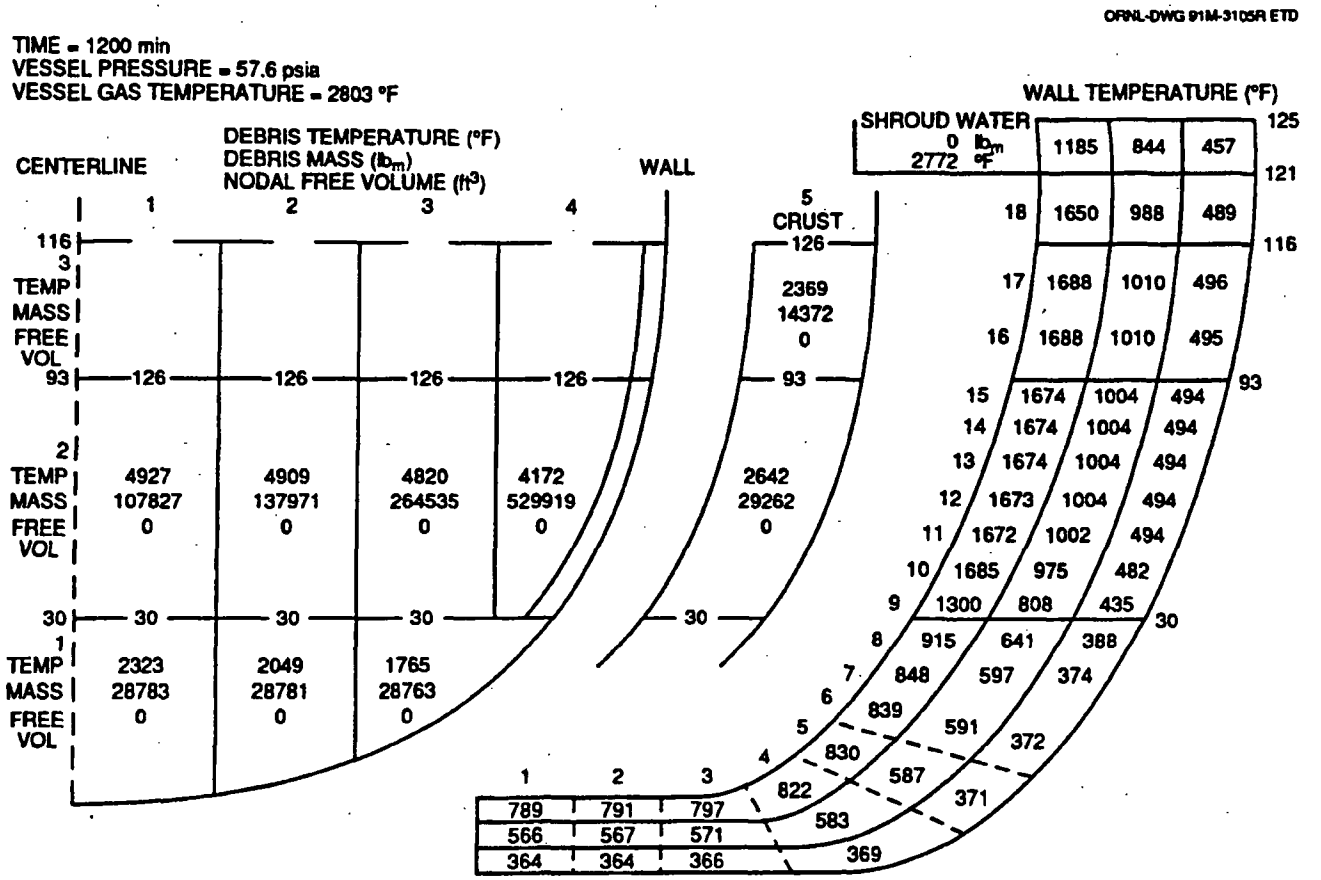

Figure 20.8 Lower plenum debris bed and vessel wall response at time 1200 min after scram for case with interior of vessel skirt completely vented

Table 20.6 Solid and liquid masses within debris layer two at time 1200 min

Constituent	Solid mass (lb)	Liquid mass (lb)	Total (lb)
Zr	4,612	73,685	78,297
Fe	6,683	437,501	444,184
Cr	1,674	106,222	107,896
Ni	403	48,621	49,024
B_4C	894	944	1,838
ZrO_2	1,147	34,227	35,374
FeO	0	8	8
Fe_3O_4	19	463	482
Cr_2O_3	7	167	174
NiO	1	33	34
B_2O_3	15	16	31
UO_2	327,165	25,006	352,171
Totals	342,620	726,893	1,069,513

Beyond 20 h, radiation from the upper surface of the debris bed would continue and the carbon steel of the upper reactor vessel wall (above the elevation of the water in the drywell) might melt. To determine the capacity for heat conduction through the wall, a simple HEATING model was developed to represent an axial section of the reactor vessel cylindrical shell with consideration of temperature-dependent thermal conductivity. With the inner surface of the vessel wall at its [2800°F (1811 K)] melting temperature, the calculated heat conduction rates through the wall are as listed in Table 20.7. As indicated, the thermal loading of the interior surface of the submerged portion of the wall would have to exceed 1557.8 Btu/min/ft^2 (295,000 W/m^2) for melting to proceed.

With the drywell flooded to a height equivalent to 505 in. (12.827 m) above vessel zero, the wetted and dry regions of the reactor vessel are as shown in Fig. 20.9. It is easy to show that the average thermal loading of the interior wall surface at time 1200 min after scram would be much less than 1550 Btu/min/ft^2 (295,000 W/m^2). The total debris bed decay heating at this time is 13.8 MW or 760,000 Btu/min. We make the very conservative assumptions that (1) *all* of the decay heat release is radiated upward, and (2) the radiated energy falls *only* on the submerged portion of the wall. The thermal loading of the interior wall surface of the submerged portion of the wall is then

$$\frac{760,000}{2,078} = 365.7 \text{ Btu/min/ft}^2$$

or 69,000 W/m^2. Because this is much less than the heat removal capacity with the interior of the wall held at 2800°F (1811 K), the interior wall surface temperature cannot reach its melting point, and general melting of the submerged portion of the wall cannot occur.

Local surface melting might occur, however, over the portion of the vessel wall just above the surface of the debris pool. To calculate the local effects of radiative heating, it is necessary to consider the respective view factors for the capped cylindrical structure above the debris pool. Based upon the upper pool temperatures shown in Fig. 20.8 and the drywell water level shown in Fig. 20.9, a HEATING calculation predicts local surface melting over the first 3 ft (0.914 m) of cylindrical wall above the pool surface. Nevertheless, this would not threaten the integrity of the wall because there is a steep temperature gradient across the wall in this vicinity and slight thinning would increase the local conductive capacity, terminating interior melting.

Of more interest is the fate of the reactor vessel wall above the waterline in the drywell. This portion of the upper wall would have a much more uniform transverse temperature profile, and melting, once started, would proceed across the entire wall, destroying its integrity. With the drywell water level shown in Fig. 20.9, however, the view factors are such that <10% of the radiant energy from the debris

Table 20.7 Heat conduction rates for the reactor vessel cylindrical shell with the inner surface held at 2800°F

Outer surface boundary condition	Receiver temperature (°F)	Heat transfer per square foot of wall (Btu/min/ft^2)
Nucleate boiling to water	267	1557.8
Convection to air heat transfer coefficient 0.625 Btu/h/ft^2/°F	400	25.7
	600	23.6
	800	21.5
	1000	19.5

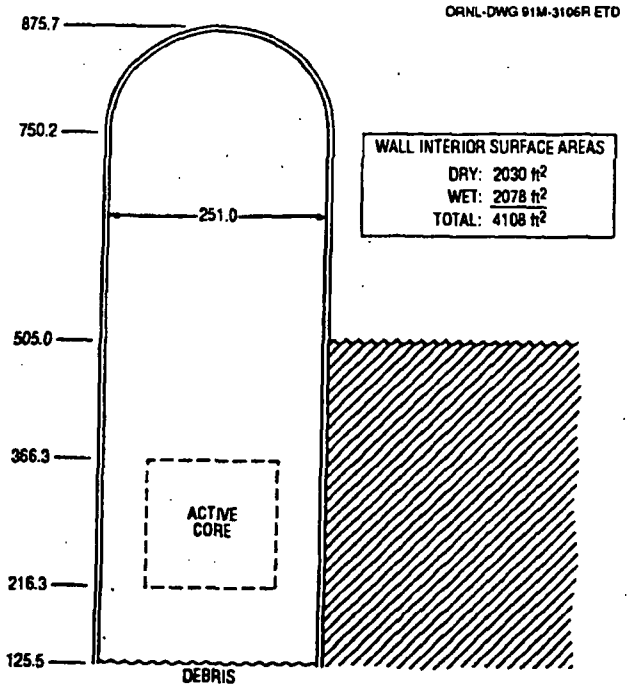

Figure 20.9 To-scale representation of portion of reactor vessel above surface of lower plenum debris bed

pool surface would reach the dry portion of the wall. The HEATING calculation predicts that none of this upper surface would reach threatening temperatures.

Obviously, there is a minimum drywell water level for which a HEATING calculation would predict survival of the dry portion of the upper reactor vessel wall. A trial-and-error approach has indicated that the water level in the drywell must extend at least 9 ft (2.743 m) above the surface of the debris pool within the vessel to preclude loss of wall integrity. (With this configuration, the radiant energy from the pool surface would be distributed approximately equally between the wetted and dry portions of the upper wall.)

To summarize the findings for the case with complete venting of the reactor vessel skirt, the calculated results demonstrate that the submerged portion of the reactor vessel wall cannot undergo significant melting because the cooling provided by the water is sufficient to maintain all portions of the wall beyond the inner surface well below the (carbon steel) melting temperature. On the other hand, the heat transport from the debris through the wall to the water is conduction-limited, and only a fraction of the decay power is removed by this pathway (Table 20.2). Consequently, a liquid pool forms in the central and upper portions of the bed, and the energy not transferred through the wall or consumed in debris melting and superheating is radiated upward within the vessel. This ultimately leads to melting of the large masses of stainless steel vessel internals (shroud, separators, dryers) with the resulting liquid metal entering the debris pool and raising the level of its surface. Although the lower plenum debris bed models have only a crude representation of the heatup and melting of the upper vessel internals, sufficient energy is radiated upward that there can be no doubt that melting of this stainless steel [at 2550°F (1672 K)] would occur. Once the upper vessel internals have melted, the radiated energy would fall upon the upper vessel carbon steel wall [melting temperature 2800°F (1811 K)].

Thus, it seems that the drywell flooding strategy with complete venting of the reactor vessel support skirt would greatly delay failure of the reactor vessel wall and would, in the strict sense, meet its goal of maintaining the core and structural debris within the vessel; the ultimate failure of the wall, should it occur, would be above the waterline in the containment and above the surface of the debris pool in the vessel. (Survival of the upper vessel wall could be guaranteed, of course, if the entire vessel were submerged in water; this cannot be done, however, because it would require raising the containment water level above the drywell vents.) The worst aspect of this ultimate wall failure is that it would open a direct pathway from the superheated debris pool within the reactor vessel to the drywell, which would be vented to the atmosphere. Even though the volatile fission products would have long before escaped from the debris to the pressure suppression pool (via the SRVs), the opening of this pathway is very undesirable. If the drywell vents were shut at this time, the containment would soon fail on overpressure. Although the drywell flooding strategy can only delay failure of the vessel wall in the existing plants, the current analyses demonstrate the potential for assured prevention of vessel failure in future plants, by provision of means for total vessel submergence.

21 Results with the Vessel Pressurized

S. A. Hodge, J. C. Cleveland, T. S. Kress*

It is an important and well-established feature of boiling water reactor (BWR) accident management that the reactor vessel should be depressurized if the core becomes uncovered. If the low-pressure emergency core cooling systems (ECCSs) are available, then the accident would be terminated by this maneuver, which would initiate vessel flooding. If the low-pressure ECCSs are not available, then vessel depressurization (manually initiated) is still beneficial because it would provide temporary steam cooling of the uncovered region of the core and would eliminate water from the core region before the fuel cladding reached runaway zirconium oxidation temperatures. (Additional information concerning the reactor vessel depressurization strategy is found in Ref. 16 and in Sect. 3.2.1.)

All of the calculations whose results are discussed in Chaps. 19 and 20 were carried out under the assumption that the reactor vessel remains depressurized during the formation of the lower plenum debris bed and during the subsequent period of bed heatup and melting. In this chapter, the potential for failure of safety/relief valve (SRV) remote control due to drywell flooding and the effects of this eventuality are discussed.

*All work in connection with this project completed before becoming a member of the Advisory Committee on Reactor Safeguards.

21.1 Motivation for the Analysis

As indicated in Fig. 21.1, each main steam line emerges from the upper reactor vessel, drops vertically [~45 ft (13.716 m)], runs horizontally about one-eighth of the way around the vessel, then drops vertically again before making a final turn and short horizontal run through the drywell shell. As shown, the SRVs are mounted on the first horizontal run of the main steam line. The location of this horizontal piping run relative to the minimum water level for drywell flooding (the center of curvature of the vessel bottom head) is shown in Fig. 8.14.

As discussed in Sect. 15.3, a portion of the drywell atmosphere would be trapped within the reactor vessel support skirt as the water level rose within the drywell. Because there is no provision in existing BWR facilities for venting from the skirt region, the drywell water level would have to be much higher than the minimum shown in Fig. 15.1 for a significant portion of the exterior surface of the vessel bottom head to be covered by water. In fact, the water level would be in the vicinity of the vessel midplane, and the SRVs would be submerged.

At this point, it is well to provide a brief review of the expected operation of the SRVs during the late phase of the

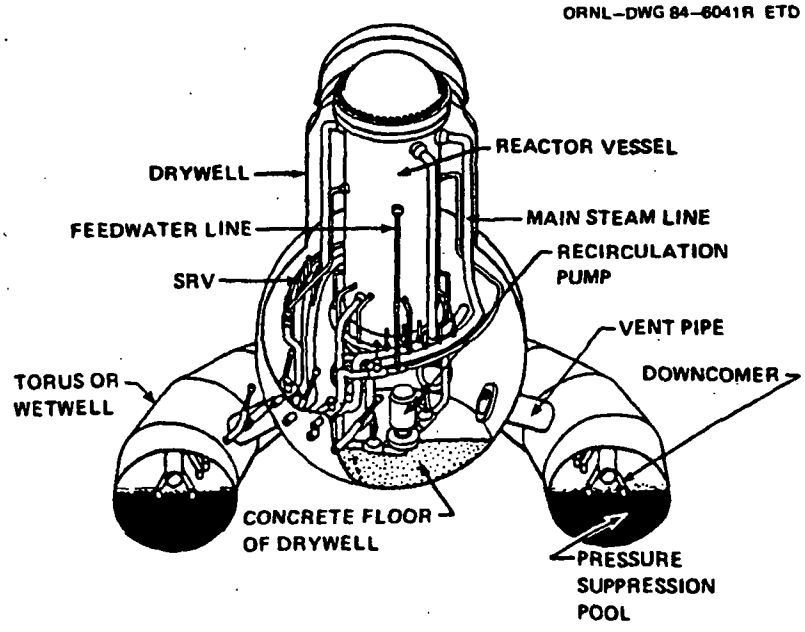

Figure 21.1 Reactor vessel SRVs are located on horizontal runs of main steam lines, near bottom of vessel

short-term station blackout accident sequence. The automatic depressurization system (ADS) would have been manually initiated at the time the core had become partially uncovered with no reactor vessel injection systems available. This provides a continuous open signal to the SRVs associated with the ADS (six at Browns Ferry). The open signal, however, merely positions a pilot valve permitting the main valve to open (or remain open) *if the differential pressure between the reactor vessel and the drywell is sufficient to lift the main valve piston.* Whenever the reactor vessel pressure falls to within 20 psi (0.138 MPa) of the drywell pressure, the valves close even though the open signal remains in effect. When the reactor vessel pressure subsequently increases to 50 psi (0.345 MPa) above the drywell pressure, the valves reopen. Additional information concerning the operation of the two-stage Target Rock SRVs installed at Browns Ferry and several other BWR facilities is provided in Chap. 4 of Ref. 5.

Control air for the pilot valve positioning is provided by the drywell control air system backed up by the ADS accumulators. The electrical and mechanical components of the ADS accumulator system, associated equipment, and control circuitry are seismic category 1 and are environmentally qualified for conditions associated with normal operation, maintenance, testing, and postulated accidents as analyzed in the plant final safety analysis report. Therefore, it seems probable that the SRVs would continue to function in the ADS-actuated mode even if submerged in water. This has not been demonstrated, however, and it seems prudent to consider what would happen if the open signal were lost upon drywell flooding.

Before proceeding to a discussion of the analysis, two additional points with respect to the potential for loss of reactor vessel pressure control must be addressed. Figure 21.2 shows the location of a typical SRV within the drywell and the arrangement of its tailpipe piping, which terminates in a quencher device near the bottom of the pressure suppression pool. It is important to note the two check valves located on the vertical run of tailpipe piping within the drywell. Under normal conditions, these check valves prevent pressure suppression pool water from being drawn up into the tailpipe as the steam within the tailpipe condenses after SRV actuation and reclosure. With the drywell flooded, however, these check valves would admit water rather than drywell atmosphere into the tailpipe. Each subsequent SRV opening would then initiate clearing of water from the entire tailpipe. With reactor vessel pressure only 50 psi (0.345 MPa) above drywell pressure, the threat of failures induced by water hammer would probably be small. Nevertheless, this has not been demonstrated.

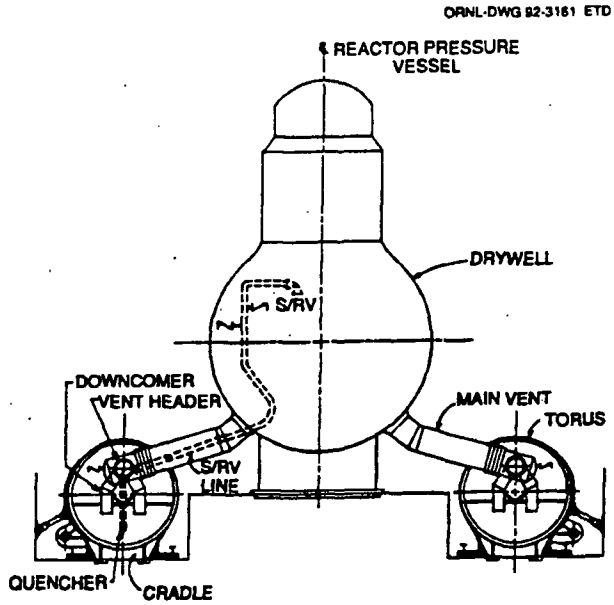

Figure 21.2 Location of typical SRV and its tailpipe within BWR Mark I containment

The second point to be addressed involves the long-term station blackout accident sequence, for which loss of reactor vessel pressure control is a characteristic feature. In this accident sequence, the core remains covered until battery power is exhausted; loss of dc power then causes both loss of reactor vessel injection and loss of pressure control. The reactor vessel would be pressurized during the period of lower plenum debris bed formation and melting.

Because long-term station blackout has been identified to be among the dominant BWR accident sequences leading to core melt,[6] it would seem that this alone would justify the discussion of the effects of failure of reactor vessel pressure control provided in the following sections of this chapter. The reader is reminded, however, that an increased reliability of the battery supply to the SRVs, implemented as a result of the ongoing individual plant examination (IPE) process,[29] might eliminate loss of reactor vessel pressure control as a credible accident sequence event. In the meantime, additional information with respect to this important consideration of long-term reactor vessel pressure control is provided in Sect. 8.1.

21.2 Loss of Pressure Control After Lower Plenum Dryout

To achieve its purpose of preventing the release of core and structural debris from the reactor vessel bottom head, the first requirement for the containment flooding strategy is that the water level within the vessel skirt be raised sufficiently quickly to surround the penetration assemblies

and the vessel drain before lower plenum dryout. Subsequently, the drywell water level might be increased more slowly as necessary to cool the upper portions of the reactor vessel, because some time will be required for evaporation of the water in the downcomer region. Therefore, if loss of reactor vessel pressure control should be a consequence of submergence of the SRVs, this might occur before or after lower plenum dryout. In this section, the consequences of the latter possibility are examined. The effects of failure of reactor vessel pressure control *before* lower plenum dryout will be discussed in Sect. 21.3.

The only water within the reactor vessel after lower plenum dryout would be the water surrounding the jet pumps in the downcomer region. With the SRVs closed, the rate of pressure increase within the reactor vessel would depend upon the rate of heat transfer to this water, which would be primarily by radiation from the upper surface of the debris bed.

Calculations have been carried out with the lower plenum debris bed model that differ from those described in Chap. 19 only in that pressure control is assumed to be lost at time 250 min. This is shortly after lower plenum dryout, which is predicted to occur at time 246.2 min. It is important to note that the calculated results for these two cases are exactly the same through time 360 min. For the case *with* pressure control, the SRVs are predicted to be closed at the time of lower plenum dryout, and the reactor vessel-to-drywell pressure differential is not predicted to reach 50 psi (0.345 MPa) before time 360 min. Subsequently, the SRVs reopen, and the vessel pressure decreases. For the case without pressure control, however, the SRVs remain closed, and the calculated reactor vessel internal pressure continues to increase after time 360 min.

The calculated situation within the lower plenum debris bed and reactor vessel bottom head wall at time 600 min is shown in Fig. 21.3 for the case without pressure control. Comparison with Fig. 19.8, which represents the situation at the same point in time for the case with pressure control, reveals the effects of the hypothetical loss of SRV operability due to drywell flooding. While there is very little difference in the predicted maximum wall temperatures (vicinity of wall node 11), the vessel pressure is significantly higher in Fig. 21.3, 519 psia (3.578 MPa) vs 79.6 psia (0.549 MPa) in Fig. 19.8.

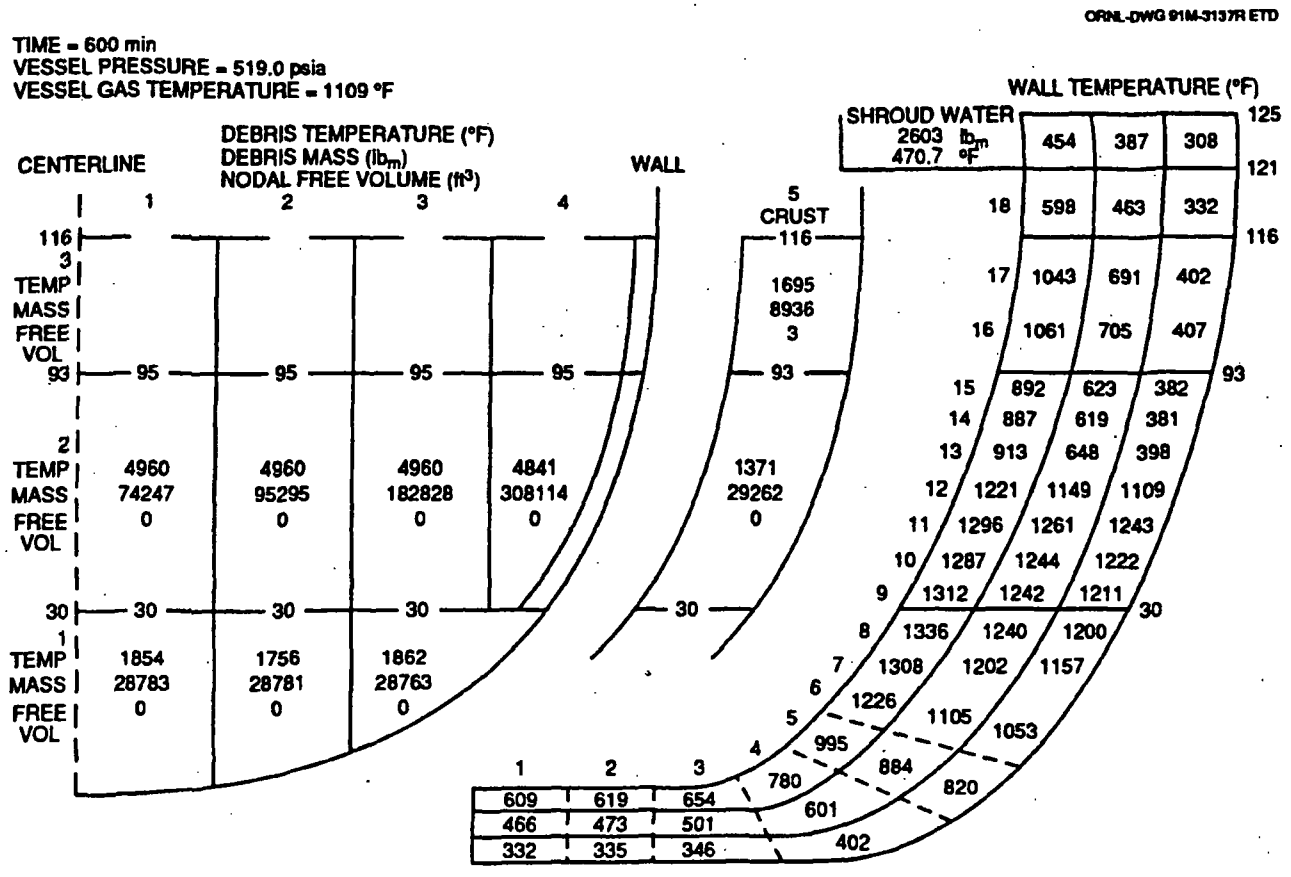

Figure 21.3 Lower plenum debris bed and vessel wall response at time 600 min after scram for case with loss of reactor vessel pressure control at time 250 min

Results

The higher vessel pressure is associated with a higher wall tensile stress at the skirt attachment weld, about 4920 psia (33.9 MPa). As indicated in Table 21.1, however, a tensile stress of 33.9 MPa would not be expected to induce creep rupture at wall temperatures as low as those shown in Fig. 21.3.

Moving ahead 1 h to time 660 min, the calculated situation is as shown in Fig. 21.4. (The corresponding debris bed conditions and wall temperatures for the base case are shown in Fig. 19.9.) Dryout of the downcomer region has occurred during this hour; consequently, the vessel pressure increase has been only ~38 psi (0.262 MPa), to a pressure at time 660 min of 556.8 psi (3.839 MPa). The corresponding wall tensile stress at the skirt attachment weld is about 5280 psi (36.40 MPa), and reference to Table 21.1 indicates that failure due to creep rupture would not be expected at time 660 min for the average wall temperatures shown in Fig. 21.4.

The calculated situation at time 700 min is shown in Fig. 21.5, where the maximum average temperature across the wall (at node 11) has reached 1598°F (1143 K). Although the predicted vessel pressure has decreased slightly [no more water is being evaporated within the lower plenum and a small amount of (normal) leakage continues through the closed main steam isolation valves], the tensile stress at the skirt attachment weld is ~5200 psi (35.85 MPa). From the information provided in Table 21.1, it can be concluded that failure of the wall by creep rupture in the vicinity of node 11 would be expected to occur at some point during the 40-min interval between 660 and 700 min after scram.

For the base case with the reactor vessel remaining depressurized, the calculated results (discussed in Sect. 19.4) indicate a wall failure by creep rupture sometime during the period 780 to 840 min after scram. Therefore, closure of the SRVs shortly after bottom head dryout is predicted to advance the time of wall failure by ~2 h. It is important to recognize that the wall failure time would have been advanced significantly more than this had the reactor vessel-to-drywell differential pressure reached and subsequently been maintained at the setpoint [about 1100 psia (7.584 MPa)] for actuation of the SRVs in the automatic (spring-loaded) mode. The reactor vessel pressure increase was limited, however, by the amount of water in the downcomer region. Boiling of all of the available water increased the vessel pressure only to about 560 psia (3.861 MPa).

21.3 Pressure Restored Before Lower Plenum Dryout

In this section, the case is considered in which containment flooding submerges the SRVs while ample water for vessel repressurization remains in the lower plenum. In addition, it is assumed that this submergence causes failure of vessel pressure control. This occurs at time 210 min (3-1/2 h) after scram for the calculations discussed here.

With the reactor vessel pressurized during the period of about one-half hour immediately preceding lower plenum dryout, the predicted time of dryout is advanced slightly, and the initial conditions within the debris bed are shown in Fig. 21.6. These may be compared with the initial conditions for the base case, shown in Fig. 19.2. The basic configuration of the debris bed is about the same in both cases, with no molten materials within the bed.

Table 21.1 Time (min) to creep rupture by interpolation of the data for SA533B1 carbon steel using the Larson-Miller parameter

Temperature	Stress (MPa)							
(°F)	15	20	25	30	35	40	45	50
1350							764.05	379.54
1400						630.46	206.91	104.73
1450					324.44	177.56	60.00	30.92
1500				559.16	95.99	53.35	18.53	9.71
1550			420.65	168.23	30.17	17.02	6.07	3.23
1600		448.68	131.21	53.65	10.03	5.74	2.10	1.13
1650	1273.99	143.65	43.25	18.06	3.51	2.04	0.76	0.42
1700	408.78	48.48	15.01	6.40	1.29	0.76	0.29	0.16
1750	138.08	17.19	5.46	2.37	0.50	0.30	0.12	0.06
1800	48.94	6.38	2.08	0.92	0.20	0.12	0.05	0.03

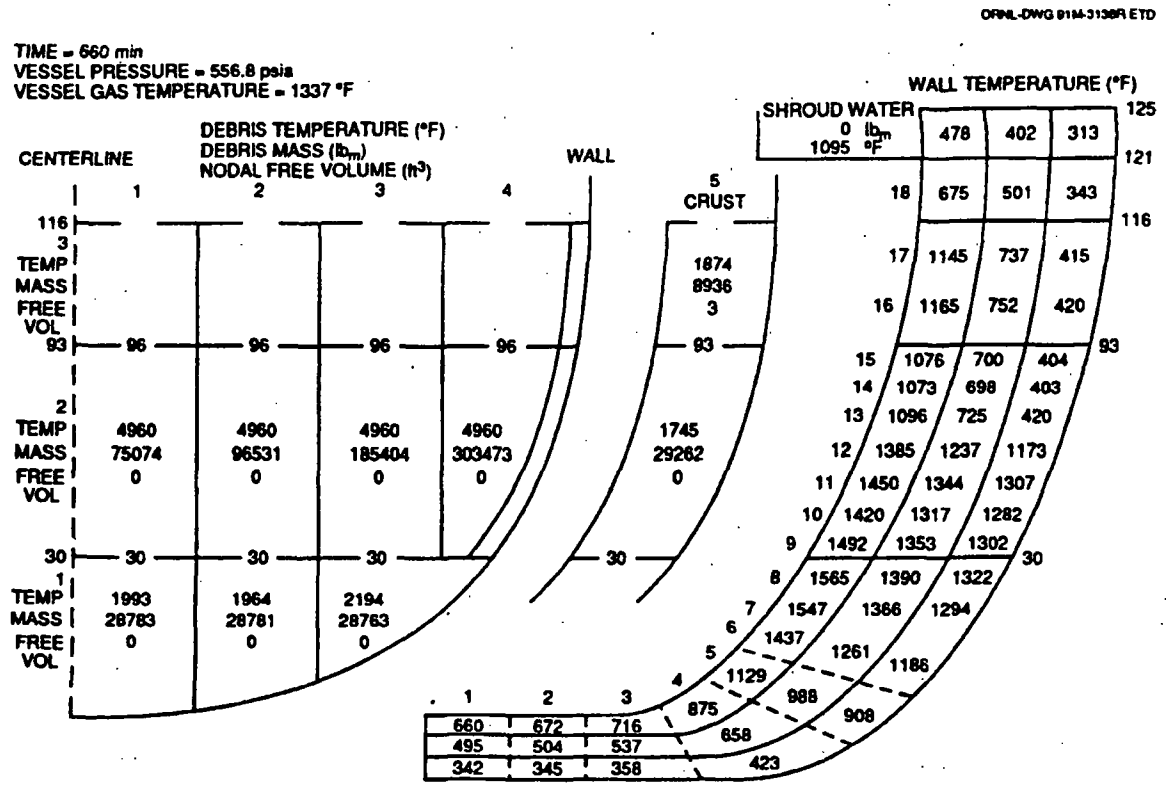

Figure 21.4 Lower plenum debris bed and vessel wall response at time 660 min after scram for case with loss of reactor vessel pressure control at time 250 min

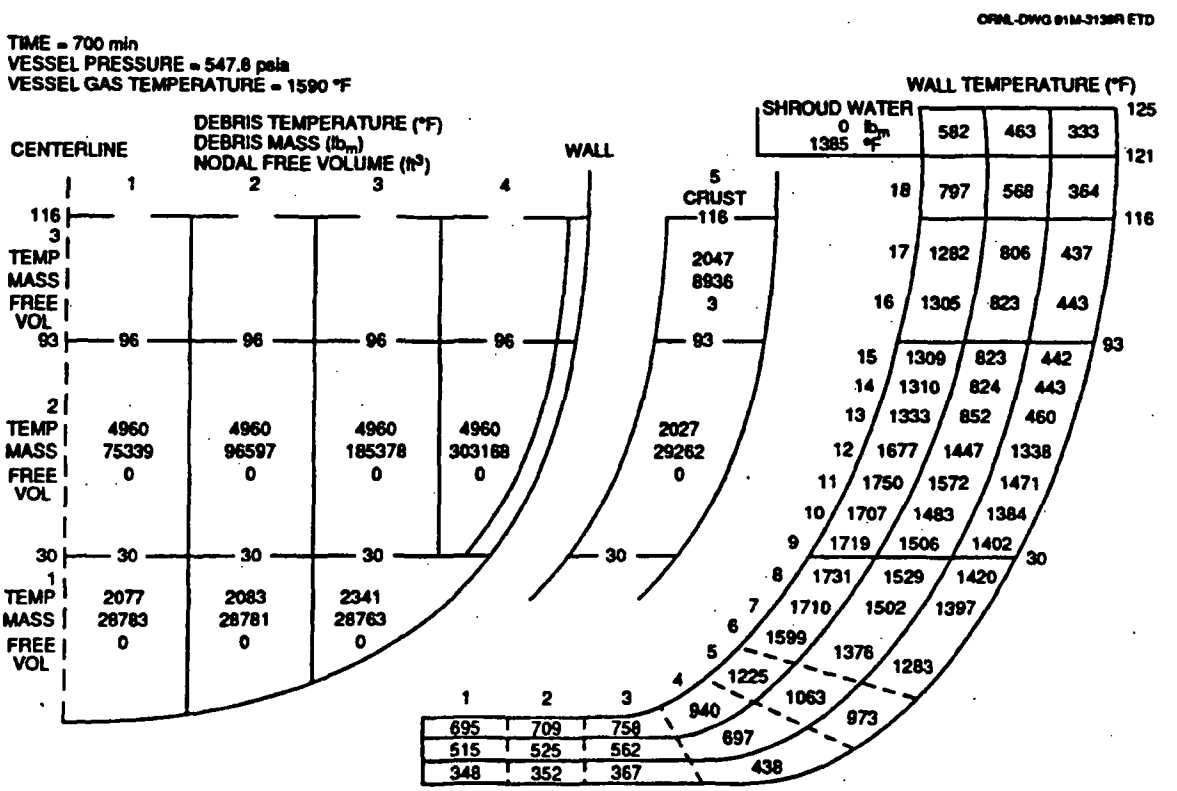

Figure 21.5 Lower plenum debris bed and vessel wall response at time 700 min after scram for case with loss of reactor vessel pressure control at time 250 min

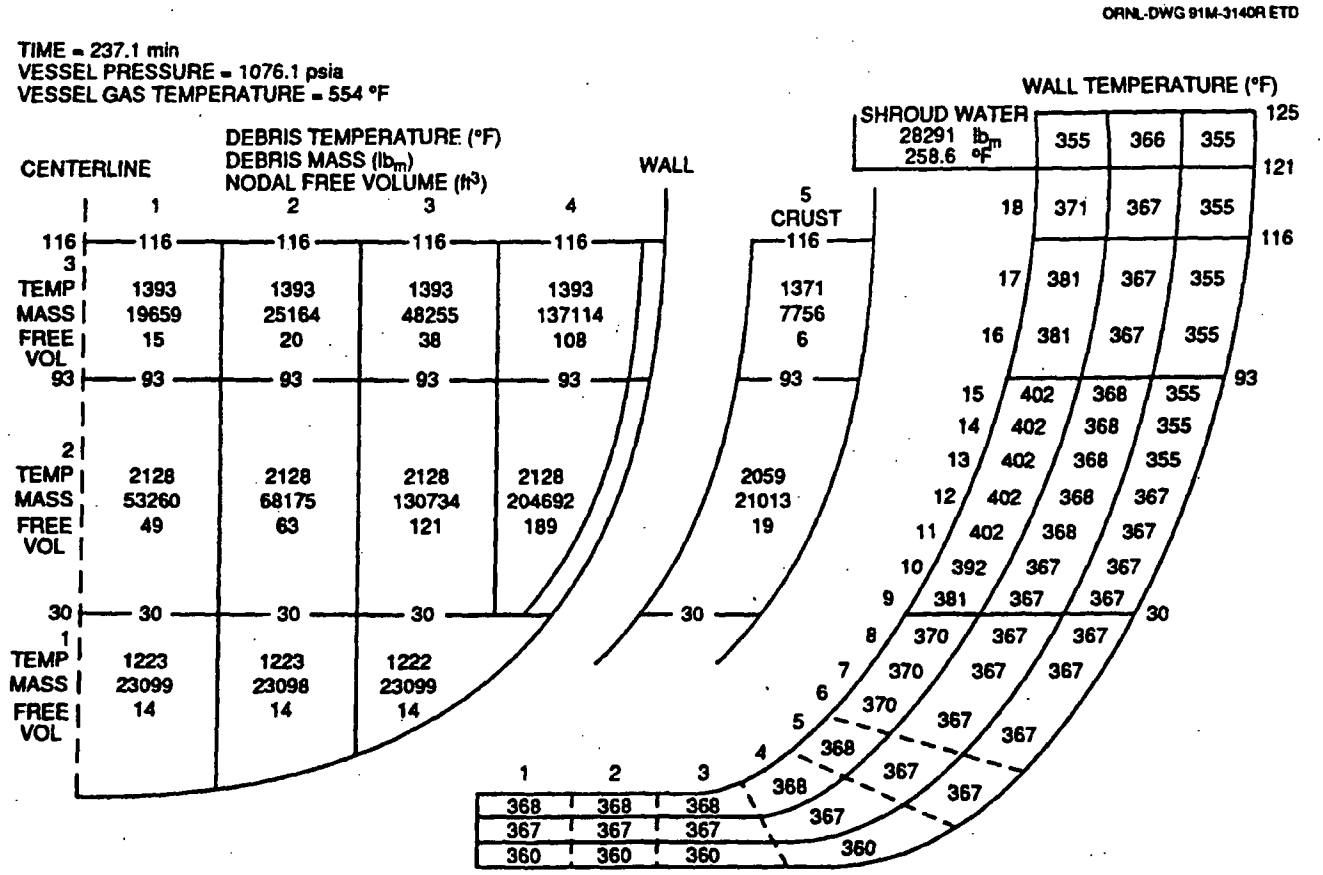

Figure 21.6 Initial configuration of lower plenum debris bed and initial wall temperatures for case with reactor vessel pressure restored before lower plenum dryout

As indicated in Fig. 21.6, the temperature of the water in the downcomer region is ~260°F (400 K), far below the saturation temperature [554°F (563 K)] corresponding to a reactor vessel pressure of 1076 psia (7.419 MPa). Therefore, boiling of the downcomer water cannot occur until a combination of reduced vessel pressure and increased water temperature brings the water to saturation conditions. This is just what the calculation predicts. The vessel pressure slowly decreases due to normal leakage and the temperature of the water in the downcomer region slowly increases due to radiative heating from the surface of the lower plenum debris bed. The downcomer water becomes saturated at time 380 min after scram at a vessel pressure of (coincidentally) 380 psi (2.620 MPa). Subsequently, the continued heating of the water in the downcomer region causes the predicted reactor vessel pressure to reverse its previous trend and begin to increase.

The calculated situation in the lower plenum debris bed and bottom head wall at time 600 min is shown in Fig. 21.7. The predicted reactor vessel pressure at this time [784 psia (5.405 MPa)] is significantly higher than for the case with loss of vessel pressure control after lower plenum dryout, shown in Fig. 21.3. The calculated maximum average wall temperatures (wall nodes 7 through 11) are, however, about the same for these two cases. At the vessel support skirt attachment weld, the wall tensile stress corresponding to a vessel pressure of 784 psia (5.405 MPa) is ~7490 psi (51.64 MPa). Reference to Table 21.1 indicates that creep rupture of the wall would not be expected before time 600 min for the wall temperatures shown in Fig. 21.7.

By time 660 min, however, the maximum average wall temperature has increased to 1413°F (1040 K) at node 8, and the vessel pressure has increased to 813 psia (5.605 MPa), as indicated in Fig. 21.8. Twenty minutes later, the predicted average wall temperature at node 8 is 1475°F (1075 K), while the calculated vessel pressure remains above 800 psia (5.516 MPa), equivalent to a wall tensile stress of about 7640 psi (52.68 MPa). From the information provided in Table 21.1, failure of the wall by creep rupture in the vicinity of the wall node 8 would be expected to occur at about time 680 min after scram. This is about the same failure time as estimated for the case with loss of vessel pressure control after lower plenum

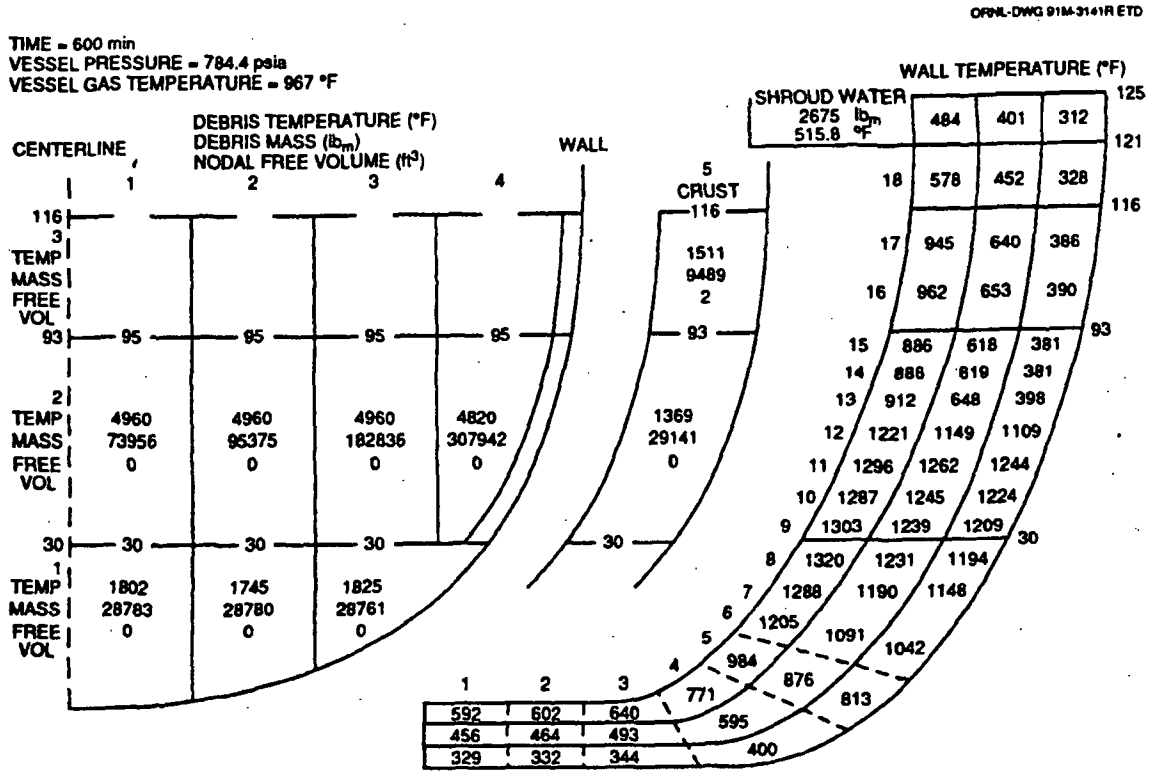

Figure 21.7 Lower plenum debris bed and vessel wall response at time 600 min after scram for case with reactor vessel pressure restored before lower plenum dryout

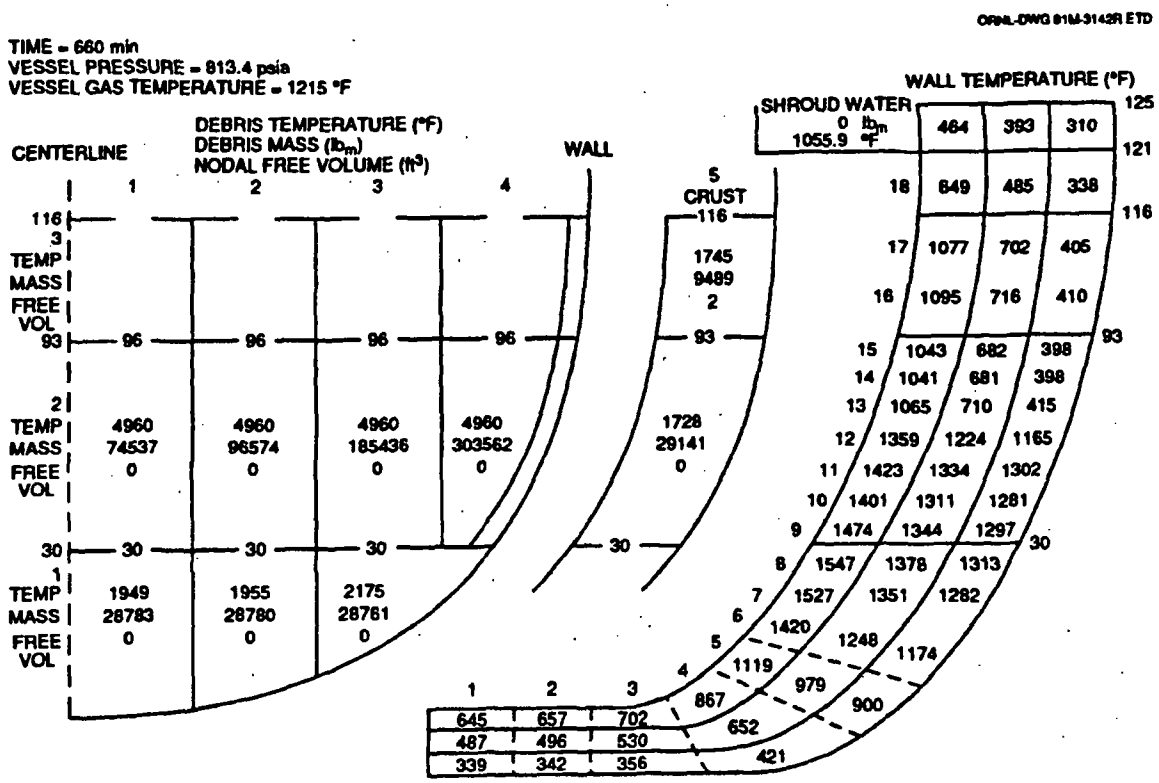

Figure 21.8 Lower plenum debris bed and vessel wall response at time 660 min after scram for case with reactor vessel pressure restored before lower plenum dryout

dryout and is some 2 h earlier than the creep rupture failure time expected for the base case (with the vessel remaining depressurized) discussed in Sect. 19.4.

In assessing the effectiveness of drywell flooding as an accident mitigation technique, it is important to remember that the most important contribution of this maneuver with respect to delaying release of debris from the reactor vessel lies in preventing the establishment of release pathways through the instrument guide tubes or vessel drain. As described in Chap. 16, such pathways would be expected to be established at ~250 min after scram if the drywell is not flooded. While some of the delay in bottom head failure by creep rupture that might be obtained by drywell flooding would not be achieved if reactor vessel pressure control were lost, the estimated time of failure (680 min after scram) is still some 7 h later than the instrument guide tube or drain failures associated with the dry case.

22 Summary and Conclusions

This chapter provides a concise summary of the major findings and conclusions of this report, which comprises the information previously documented in five letter reports addressing the general subject of boiling water reactor (BWR) severe accident management. This summary is divided into three sections following the major divisions of the report.

22.1 Status of BWR Severe Accident Management

The conclusions and recommendations of Chaps. 2 through 8 are addressed in this section. Each of the following items is listed in the order in which it is developed within the report, with reference to the appropriate section of the report where a detailed discussion can be found.

1. Station blackout and anticipated transient without scram (ATWS) are the dominant contributors to the overall risk of BWR core melt (Sect. 2.1).
2. Short-term station blackout is characterized by immediate loss of reactor vessel injection capability. For long-term station blackout, reactor vessel water makeup is lost following exhaustion of the unit battery (Sect. 2.2.1).
3. Without battery power, the safety/relief valves (SRVs) could not be actuated, and the reactor vessel could not be maintained depressurized (Sect. 2.2.1).
4. The BWR Owners' Group Emergency Procedure Guidelines (EPGs) require unequivocally that the operators act to manually depressurize the reactor vessel should the core become partially uncovered under station blackout conditions (Sect. 2.2.1.1).
5. It is beneficial for the operators to depressurize the reactor vessel early in the initial phase of a long-term station blackout, while dc power for SRV operation remains available (Sect. 2.2.1.2).
6. The most severe form of ATWS is initiated by main steam isolation valve (MSIV) closure with a complete failure of the scram function (Sect. 2.2.2).
7. Rapid shutdown by scram from power operation is required for ATWS with failure of recirculation pump trip (Sect. 2.2.2).
8. It is the compounded case of ATWS with failure of the standby liquid control system (SLCS) that requires special accident management strategies (Sect. 2.2.2).
9. Experience has shown that plant-specific differences preclude any simple extension of the results obtained by a detailed analysis of one BWR plant to other plants of the same classification (Sect. 2.3).
10. The report *Assessment of Candidate Accident Management Strategies* (NUREG/CR-5474)[17] provides a set of accident management strategies derived from a review of various NRC and industry reports on the subject of prevention or mitigation of core damage (Sect. 3.1).
11. The *BWR Owners' Group Emergency Procedure Guidelines* (EPGs) are generic to the BWR plant designs and are intended to be adapted for application to individual plants (Sect. 3.2).
12. The EPGs provide effective guidance for dealing with station blackout, provided the accident sequence can be brought under control and terminated (Sect. 3.2.1).
13. The guidance of the EPGs for dealing with ATWS should terminate the accident sequence without core damage. The principal challenge to this desired conclusion is that the operator actions undertaken while attempting to achieve the pressure control directed by the EPGs might create an unstable situation (Sect. 3.2.2).
14. With the exception of seven items, the candidate BWR accident management strategies identified by the NUREG/CR-5474 report are represented within the EPGs (Sect. 4.1).
15. Of the seven items not addressed, six are highly dependent upon plant-specific arrangements and, therefore, should be implemented within the plant Emergency Operating Procedures rather than the generic EPGs. The seventh item pertains to control blade melting under severe accident conditions and the potential for criticality upon recovery of reactor vessel injection; this has generic applicability and is appropriate for inclusion in the generic EPGs (Sect. 4.2).
16. Although several questions remain with respect to the existing guidance for the ATWS accident sequence, most of the potential benefit that could derive from enhancement of the existing strategies lies in the realm of severe accident management (Sect. 4.3).
17. Station blackout is the leading contributor to BWR core damage frequency because the majority of the reactor vessel injection systems are dependent upon the availability of ac power, and BWRs are vulnerable to loss of injection (Sect. 5.1).
18. The relative probabilities of the long-term and short-term versions of the station blackout accident sequence depend upon the plant-specific configuration of the battery systems (some plants have independent starting batteries for the diesels) and whether the plant has one or two steam turbine-driven injection systems (Sect. 5.2).
19. For the ATWS accident sequences, as for all other BWR severe accident sequences, core degradation can occur only after failure of adequate reactor vessel injection. Injection would be lost in an unmitigated ATWS accident sequence because of events occurring in the overheated and pressurized primary containment (Sect. 5.3).

20. It is recommended that consideration be given to the separation of the ATWS guidelines from the symptom-oriented guidelines of the EPGs for all other accident sequences. It is recommended that care be taken to avoid leading the operators to attempt manual depressurization of a critical reactor. It is recommended that consideration be given to control of the reactor vessel injection rate as a means for reduction of reactor power, and it is recommended that the guidance to the operators regarding manual insertion of control blades be expanded (Sect. 6.2).
21. The availability of the various plant instruments under accident conditions is a plant-specific consideration, and this should be an important part of each individual plant examination (IPE) for severe accident vulnerabilities (Sect. 7.1).
22. The availability of information concerning plant status is much greater for accident sequences such as short-term station blackout [with mechanical failure of high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC)], ATWS, loss-of-coolant accident (LOCA), or loss of decay heat removal for which electrical power is maintained after loss of reactor vessel injection capability (Sect. 7.2).
23. Four candidate accident management strategies for control of in-vessel events during the late phase (after core melting has occurred) of postulated BWR severe accidents have been proposed and considered (Chap. 8).
24. The severe accident management strategy to keep the reactor vessel depressurized is not recommended for further consideration because it is expected that the IPE process will produce practical means for accomplishing this (Sect. 8.1).
25. The severe accident management strategy to restore injection in a controlled manner is recommended for consideration in conjunction with the strategy to prevent criticality upon vessel reflood (Sect. 8.2).
26. The severe accident management strategy for injection of boron if control blade damage has occurred is selected for detailed assessment (Sect. 8.3).
27. The severe accident management strategy for containment flooding to maintain core and structural debris in-vessel is selected for detailed assessment (Sect. 8.4).

22.2 Strategy for Prevention of Criticality Upon Reflood

The conclusions and recommendations of Chaps. 9 through 13 for injection of a boron solution if control blade damage has occurred are summarized in this section.

1. Criticality upon reflooding a damaged core with unborated water is likely for either standing fuel rods or for a debris bed in the core region. The concentration of the boron-10 isotope produced by injection of the stored contents of the SLCS tank may not be sufficient to terminate the criticality (Sect. 9.1).
2. Although the SLCS is designed to inject sufficient sodium pentaborate to shut down the reactor from full power and to maintain the reactor subcritical during cooldown, the SLCS is not intended to provide a backup for the rapid shutdown normally achieved by scram (Sect. 10.1).
3. To reduce the time required for shutdown, the NRC has recently required that the SLCS injection be at a rate *equivalent* to 86 gal/min (0.0054 m^3/s) of 13-wt % sodium pentaborate solution in its natural state with 19.8 at. % of the boron-10 isotope (Sect. 10.2).
4. The ATWS rule permits the requirement for the increased equivalent control capacity to be satisfied by simultaneous operation of both of the installed SLCS pumps, by increasing the concentration of sodium pentaborate solution, or by enriching the boron within the solution in the isotope boron-10 (Sect. 10.2).
5. If the SLCS were used to inject sodium pentaborate at a relatively slow rate while the damaged core was rapidly covered using the high-capacity, low-pressure injection systems, then criticality would occur. It would be preferable, if control blade melting and relocation have occurred, to reflood the vessel with a premixed solution of sufficient neutron poison concentration such that there would be no threat of criticality as the core was covered (Sect. 10.3).
6. The basic requirements to preclude criticality upon vessel reflooding with control blades melted from the core are, first, that the core be recovered with a poisoned solution and, second, that the solution contain a concentration of at least 700 ppm of the boron-10 isotope. This required concentration derives from the PNL report *Recriticality in a BWR Following a Core Damage Event*, NUREG/CR-5653[34] (Sect. 11.1).
7. Polybor® is formed of exactly the same chemical constituents as is sodium pentaborate but, for the same boron concentration, requires one-third less mass of powder addition and has a significantly greater solubility in water (Sect. 11.2).
8. Even with partial tank draining, the amount of powder required to obtain the required boron-10 concentration is large. Assuming the use of Polybor®, 20,400 to 27,200 lb (9,250 to 12,340 kg) would have to be added to a condensate storage tank reserve volume of 135,000 gal (510 m^3). The practical way to poison the tank contents would be to prepare a slurry of extremely high concentration in a smaller container at ground level, then to pump the contents of this small container into the condensate storage tank. A fire engine and independent portable suction tank might be used to perform this solution mixing and transfer function (Sect. 11.3).

9. A cost-benefit analysis has been performed to assess a "Boron Injection Strategy for Mitigation of Severe Accidents." The analysis is based upon the standard methodology described in NUREG-0933, *A Prioritization of Generic Safety Issues*[18] (Sect. 12.1).
10. Criticality produced by reflooding after core damage has characteristics very different from those associated with ATWS, including not being addressed by current procedures, the probable lack of nuclear instrumentation, and the factor of operator surprise. The configuration of the critical masses in the core region might be standing fuel rods alone, a combination of standing fuel rods (outer core) and debris beds (central core), or a core-wide debris bed (Sect. 12.2).
11. The reduction in public risk associated with implementation of the proposed strategy derives from the portion of the dominant station blackout sequences that have the potential to be terminated by restoration of electric power and reactor vessel injection capability before vessel breach. With successful implementation of the strategy, criticality is not a consequence of the restoration of reactor vessel injection (Sect. 12.3).
12. In accordance with the standard NUREG-0933 methodology,[18] costs are estimated in 1982 dollars (Sect. 12.4).
13. The estimated reduction in frequency of unmitigated core melting due to implementation of the boron injection strategy is estimated to be 1.19E–06/RY (Sect. 13.1).
14. The estimated reduction in public risk is 6.06 man-rem/RY (Sect. 13.2).
15. The average industry cost per reactor associated with implementation of the proposed strategy is estimated to be $94,000 (1982 dollars). The average NRC cost per reactor is $7,000 (Sect. 13.3).
16. Considering a U.S. inventory of 38 BWR facilities with an average remaining lifetime of 21.1 years, the value/impact assessment consistent with the procedures of NUREG-0933 is

$$\frac{4860 \text{ man-rem}}{\$3.84\text{M}} = 1266 \text{ man-rem/ \$M} ,$$

which leads to an assignment of MEDIUM priority (Sect. 13.4).
17. It is recommended that each plant assess its need for the proposed strategy based upon the results of its IPE. By far, the most important aspect of this assessment is the expected frequency of station blackout events that progress through the first stages of core damage including the melting of control blades (Sect. 13.5).

22.3 Strategy for Drywell Flooding

This section provides a concise summary of the major findings and conclusions of Chap. 14 through 21, which describe an assessment of the efficacy of drywell flooding as a BWR severe accident mitigation technique. Because geometric effects of reactor vessel size dictate that external cooling of the vessel bottom head would be least effective for the largest vessels and the motivation for maintaining core and structure debris within the reactor vessel is greatest for the Mark I drywells, the primary focus of this assessment is upon the largest BWR Mark I containment facilities such as Peach Bottom or Browns Ferry. Each of the following items is listed in the order in which it is developed within the report, with reference to the appropriate section of the report where a detailed discussion may be found.

1. If water did not reach the reactor vessel bottom head external surface until after lower plenum debris bed dryout and the beginning of heatup of the vessel wall, it would be too late to prevent release of molten debris into the drywell by means of penetration assembly failures (Sect. 15.1).
2. Provision of an independently powered containment flooding system of sufficient capacity would in general require equipment modifications to existing plants, but similar modifications would also be required to support the proposed resolution of the Mark I shell failure issue (NUREG/CR-5423).[35] In both cases, the drywell would have to be vented during the flooding process and beyond (Sect. 15.1).
3. The BWR severe accident sequence leading most rapidly to the formation of a reactor vessel lower plenum debris bed is short-term station blackout, for which the vessel bottom head would have to be submerged within 150 min (2-1/2 h) after the onset of core degradation. Making conservative allowance for a containment back pressure of 60 psig (0.515 MPa), an elevation head of 80 ft (24.38 m), and a pump efficiency of 70%, the required delivery of 10,000 gal/min (0.631 m^3/s) could be provided by an 800-bhp (0.60-MW) diesel. BWR facilities with containments smaller than those at Peach Bottom or Browns Ferry would require correspondingly smaller pumping capacities and driving horsepower (Sect. 15.1).
4. Recently published information indicates that the all-metal reflective reactor vessel insulation would not significantly impede the availability of water to the vessel bottom head. Furthermore, the mode of heat transfer on the vessel outer surface would be nucleate boiling, and the generated steam could escape from the space between the vessel wall and the inner surface of the insulation (Sect. 15.2).
5. If the containment were flooded with water, a portion of the drywell atmosphere would be trapped within the reactor vessel support skirt. The height of water within

the skirt at any given time would depend upon the height of water outside the skirt, the drywell pressure at the time the water level was raised, and the current drywell pressure. In addition, the water level within the skirt would vary in accordance with the chugging cycle established by the generation of steam within the skirt, the temporary expulsion of the water, the condensation of steam on the water (and inner skirt) surface, and the reentry of the water to again contact the vessel bottom head. However, analyses to determine whether the integrity of the bottom head could tolerate the existence of any surface region without water contact have been pursued in lieu of a detailed analysis of the cyclic variation of the water level within the vessel skirt (Sect. 15.3).

6. The fraction of the bottom head surface area beneath the skirt that is submerged in water could be increased by venting from the manhole access cover or by drilling several small holes in the vessel support skirt just below the vessel attachment weld. While the latter option would provide total water coverage for the bottom head, it is not considered to be a practical proposal for existing facilities (Sect. 15.4).
7. The ratio of the maximum possible rate of heat transfer by conduction through the reactor vessel bottom head to the rate of energy release by decay heating within a whole-core lower plenum debris bed is highest (most favorable) for the smaller BWR facilities. This indicates that calculations based upon the large Browns Ferry or Peach Bottom reactor vessel designs provide the most challenging test of the proposed drywell flooding strategy (Sect. 16.1).
8. The available experimental evidence provides sufficient information to support theoretical calculations for the case of a molten hemisphere contained within its own crust. Based upon the characteristics of (high metallic content) BWR debris, these calculations indicate that about one-third of the total decay heat would be removed downward under steady-state conditions. These results are not strictly applicable to the case at hand, because melting of the reactor vessel upper internal structures would prevent steady-state conditions from ever being attained within the BWR lower plenum. They do, however, identify important energy transport mechanisms affecting the distribution of decay heat removal pathways (Sect. 16.2).
9. As the temperature of the central region of the lower plenum debris increased after bed dryout, the available experimental evidence indicates that two metallic mixtures [melting at 2142°F and 2912°F (1445 K and 1873 K)] and one oxidic mixture [melting at 4172°F (2573 K)] would form within the bed (Sect. 17.2).
10. HEATING code calculations for the portion of an instrument guide tube surrounded by a dry atmosphere beneath the vessel bottom head wall predict tube wall temperatures sufficiently high to induce certain loss of integrity if the tube were suddenly filled by the oxidic mixture, probable loss of integrity if a *preheated* tube were filled by the higher-melting metallic mixture, and possible failure if a *preheated* tube were filled by the lower-melting metallic mixture (Sect. 17.3).
11. Heat transfer from the instrument guide tube outer surface would be greatly enhanced by the presence of water because the heat transfer mode would be shifted from natural convection of air to nucleate or film boiling of water. HEATING calculations demonstrate that the effectiveness of the water cooling is such that the submerged stainless steel instrument tubes would be expected to survive filling by any of the molten debris mixtures, even with the conservative assumption of tube wall preheating (Sect. 17.4).
12. After lower plenum debris bed dryout, the water surrounding the jet pumps in the downcomer region would be the only water remaining in the reactor vessel. As long as this water remains, the core shroud would constitute a major heat sink for radiation from the upper surface of the debris bed. Once this water is exhausted, the shroud temperature would increase to the melting point of stainless steel and the shroud would melt. The liquid steel would enter the debris, providing a cooling effect while increasing the volume of the central molten pool (Sect. 18.1.4).
13. For the depressurized reactor vessel, only the portion of the bottom head wall beneath the vessel support skirt would be in tension. Failure of the wall by creep rupture would become of concern as the local average wall temperature exceeded 2100°F (1422 K) (Sect. 18.2.3).
14. Water surrounding the reactor vessel bottom head would preserve the integrity of the carbon steel vessel drain as it filled with relocating molten material from the overlying debris bed (Sect. 18.2.4).
15. The short-term station blackout accident sequence involves formation of a reactor vessel lower plenum debris bed in the shortest time of all the dominant BWR severe accident sequences. For this reason, it is the accident sequence considered in this study of the efficacy of drywell flooding as a severe accident mitigation technique (Sect. 19.1).
16. Without special measures to reduce the volume of drywell atmosphere trapped within the reactor vessel support skirt, creep rupture of the reactor vessel bottom head would be expected to occur between 780 and 840 min after scram if drywell flooding were employed as a mitigating strategy for the short-term station blackout severe accident sequence (Sect. 19.4).
17. Without drywell flooding, bottom head penetration failures would be expected for short-term station blackout as early as 250 min after scram. If bottom head penetration failures did not occur, then global creep rupture of the bottom head would be expected for the dry case at some point between 600 and 640 min after scram; this is still some 3 h earlier than

the failure time for the base case with drywell flooding (Sect. 19.5).

18. Venting from the upper portion of the vessel support skirt manhole access cover during drywell flooding would reduce the unwetted portion of the bottom head from 44.8% to 27.3% of the total outer surface area. In general, the effect of the reduced gas pocket is to decrease the calculated maximum average wall node temperatures by about 100°F (56 K). This delays the wall heatup but does not prevent ultimate failure by creep rupture, which is estimated to occur ~1 h later (time 880 min) than for the base case (~820 min after scram) (Sect. 20.1).
19. For the case with complete venting of the vessel support skirt so that the pocket of trapped gas is entirely eliminated, all of the bottom head is adequately cooled, and creep rupture of the wall is not predicted to occur. However, the upward radiation from the surface of the molten debris pool would induce melting of the reactor vessel upper internal stainless steel structures, predicted to begin at about time 920 min after scram (Sect. 20.2).
20. Melting of the large mass [~375,000 lb (170,000 kg)] of reactor vessel internal structures would occur over a long period of time, during which the debris decay heat not removed through the bottom head wall or radiated upward would be consumed in increasing the temperature of the liquid stainless steel entering the molten debris pool. However, the vessel internal stainless steel is predicted to be entirely melted by about time 1200 min. After this, the receiver material for radiation from the upper debris bed surface would be the carbon steel of the upper reactor vessel wall. Because carbon steel has a higher melting temperature [2800°F (1811 K)] than stainless steel [2550°F (1672 K)], the introduction of liquid steel into the debris pool would be temporarily interrupted while the temperature of the upper vessel wall increased to its melting point (Sect. 20.2).
21. The thermal loading applied by radiation from the upper debris bed surface is insufficient to induce significant melting of the lower portion of the reactor vessel wall that is cooled by water on its outer surface. (A steep temperature gradient would exist over this wetted potion of the wall.) Failure of the upper reactor vessel wall above the water level within the drywell may occur, however, as a result of this radiative heating. The temperature profile across this portion of the wall is quite uniform, and temperatures sufficient for loss of all strength are possible even if melting does not occur (Sect. 20.2).
22. Drywell flooding with complete venting of the reactor vessel support skirt would serve to maintain the core and structural debris within the vessel. Any ultimate failure of the wall would occur above the waterline in the drywell and above the debris pool within the vessel and would be delayed until more than 20 h after scram. [Calculations based upon the Peach Bottom reactor vessel indicate that the water level within the drywell would have to extend at least 9 ft (2.743 m) above the surface of the debris pool to preclude loss of wall integrity.] The worst aspect of an ultimate wall failure is that it would open a direct pathway from the superheated debris pool within the reactor vessel to the drywell, which would be vented to the atmosphere. Even though the volatile fission products would have long before escaped from the debris to the pressure suppression pool (via the SRVs), the opening of this pathway is very undesirable. If the drywell vents were shut at this time, the containment would soon fail on overpressure (Sect. 20.2).
23. Drywell flooding as necessary to raise the water level within the reactor vessel support skirt would submerge the SRVs. While these valves are qualified for the harsh environmental conditions associated with design basis accidents, they have not been demonstrated to operate in the pressure control mode while under water and with their tailpipes flooded. If reactor vessel pressure control were lost at the time the drywell was flooded, then the vessel pressure and wall tensile stress would increase, leading to creep rupture of the wall at temperatures lower than for the depressurized case. Although loss of pressure control is considered unlikely, it cannot be ruled out, and its consequences have been calculated for the base case (without venting from the vessel support skirt) (Sect. 21.1).
24. Loss of reactor vessel pressure control shortly after lower plenum debris bed dryout would advance the time of wall creep rupture to about time 680 min, which is 2 h earlier than for the base case with the vessel remaining depressurized. The wall failure time would be advanced much more than this if the reactor vessel-to-drywell differential pressure were predicted to reach and be maintained at the set point [~1100 psi (7.584 MPa)] for actuation of the SRVs in the automatic (spring-loaded) mode. The reactor vessel pressure increase is limited, however, by the amount of water available in the downcomer region. Boiling of all of the available water serves to increase the vessel pressure only to ~560 psi (3.861 MPa) (Sect. 21.2).
25. If drywell flooding were to cause loss of reactor vessel pressure control before dryout of the vessel lower plenum, then the continued boiloff of water in the lower plenum would quickly restore the vessel to full pressure. The water in the downcomer region would remain subcooled, however. After lower plenum dryout, the vessel pressure would decrease due to normal vessel leakage while the temperature of the water in the downcomer region would increase as a result of radiation from the upper surface of the debris bed. The reactor vessel pressure would continue to decrease until the downcomer water became saturated after which the vessel pressure would increase. Creep rupture failure of the wall would occur at about the

same time as for the case with loss of pressure control after lower plenum dryout, which is 2 h earlier than for the base case with the vessel remaining depressurized (Sect. 21.3).

26. In assessing the effectiveness of drywell flooding as an accident mitigation technique, it is important to remember that the most important contribution of this maneuver with respect to delaying release of debris from the reactor vessel lies in preventing the establishment of release pathways through the instrument guide tubes or vessel drain. While some of the potential delay in bottom head creep rupture that might be obtained by drywell flooding could not be achieved if reactor vessel pressure control were lost, the estimated time of failure is still some 7 h later than the instrument guide tube or drain failures associated with the dry case.
27. Although the effect of gas trapping within the vessel skirt is such that implementation of the drywell flooding strategy for existing plants could only delay failure of the bottom head, the current analyses demonstrate the potential for preventing vessel failure entirely in future plants. It is recommended that consideration be given to providing means for complete coverage of the bottom head and submergence of most or all of the upper vessel wall as a severe accident mitigation strategy for future plant designs.

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*Available for purchase from National Technical Information Service, Springfield, VA 22161.

†Available in public technical libraries.

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**Available from American National Standards Institute, 1430 Broadway, New York, NY 10018, Copyrighted.

§Available from Radiation Shielding Information Center as CCC-545.

Appendix A

Characteristics of Polybor®

The object of the strategy described in Chaps. 9 through 13 is to provide the boron-10 isotope, in sufficient quantities to preclude criticality, together with the injected flow being used to recover a boiling water reactor (BWR) core that has temporarily been uncovered.

As discussed in Chap. 11, formation of sodium pentaborate by the normal method of separately adding borax and boric acid crystals would not be feasible at low temperatures and without mechanical mixing. Information concerning an alternative boron form was obtained by contacting U.S. Borax Company at Montvale, New Jersey. The company produces a spray-dried disodium octaborate tetrahydrate ($Na_2B_8O_{13} \cdot 4H_2O$) with a tradename Polybor®, which readily dissolves in water forming even supersaturated solutions. Using Polybor®, the total amount of material needed to form a 3540-ppm concentration of natural boron (700 ppm of the boron-10 isotope) is 32.0% less than for borax and boric acid. For example, preparation of this concentration within 135,000 gal (511 m^3) of condensate storage tank water would require the addition of 28,400 lb (12,880 kg) of borax and boric acid crystals, but only 19,300 lb (8,750 kg) of Polybor®.

Much of the difference lies in the excess water added with the borax ($Na_2B_4O_7 \cdot 10H_2O$). About 735 gal (2.782 m^3) of free water is formed during the reaction of the borax and boric acid in this example, whereas no free water is formed when Polybor® is dissolved in water.

Because the only reason for discussing Polybor® in this report is in regard to its advantages as a substitute for sodium pentaborate, its characteristics will be described in comparison to those of sodium pentaborate. The compositions of these two salts are compared below:

Table A.1 Compositions of Polybor® and sodium pentaborate by weight

Constituent	Polybor® $Na_2B_8O_{13}.4H_2O$	Sodium pentaborate $Na_2B_{10}O_{16}\cdot 10H_2O$
Na	45.98	45.98
B	86.49	108.11
O	207.99	255.99
H_2O	72.06	180.15
Formula weight	412.52	590.23

It is important to note from the information presented in this table that boron constitutes 20.97% of Polybor® by weight but only 18.32% of the sodium pentaborate by weight.

Another important point of comparison is the weight fraction within each salt of the boron-10 isotope. Because this isotope contributes 19.78% of the weight of natural boron, we are lead directly to the following result:

Table A.2 Weight fraction of the boron-10 isotope in Polybor® and sodium pentaborate

Weight fraction of boron-10	
Polybor®	Sodium pentaborate
0.0415	0.0362

This comparison indicates only part of the overall reason why about one-third less weight of powder need be added to a given amount of water to obtain the same concentration of the boron-10 isotope when Polybor® is used. There are two other contributory factors.

First, each pound (kg) of Polybor® added yields 1 lb (kg) of Polybor® in solution. However, each pound (kg) of sodium pentaborate in solution requires that 0.6464 lb (kg) of borax and 0.6285 lb (kg) of boric acid crystals be reacted, a total of 1.2749 lb (kg) of powder added per pound (kg) of sodium pentaborate formed.

Second, the free water formed [0.2749 lb (kg) per pound (kg) of sodium pentaborate created] acts in a self-defeating manner to dilute the resulting solution. This effect is very small at low concentrations but increases as larger boron concentrations are sought.

The following expressions lead toward a clear understanding of the relation between the weights of powder to be added when a given boron concentration (parts per million) is obtained by the use of Polybor® or sodium pentaborate. Let the desired concentration of boron be represented by PPMB. Then if Polybor® is added to water, the mass to be added is

$$WPLY = \frac{8.3285 \times PPMB \times VWI}{209,700 - PPMB} \text{ (lb)}, \quad (A.1)$$

where VWI is the initial volume of water [at 70°F (294.3 K)] in gallons.

If, on the other hand, borax and boric acid crystals are added to water, then the required mass of sodium pentaborate to be formed is

$$WSPB = \frac{8.3285 \times PPMB \times VWI}{183,200 - 1.2749 \times PPMB} \text{ (lb)}. \quad (A.2)$$

The mass of borax and boric acid crystals to be added is

$$WTSPB = 1.2749 \times WSPB \text{ (lb)}, \quad (A.3)$$

and the mass and volume of free water formed are

$$WH_2O = 0.2749 \times WSPB \text{ (lb)}, \quad (A.4)$$

and

$$VH_2O = 0.0330 \times WSPB \text{ (gal)}, \quad (A.5)$$

respectively. These relations were used in calculating the results for the example discussed at the beginning of this appendix so the interested reader may check their application.

Division of Eq. (A.2) into Eq. (A.1) provides the ratio of Polybor® to sodium pentaborate in solution required to produce a given boron concentration:

$$\frac{WPLY}{WSPB} = \frac{183,200 - 1.2749[PPMB]}{209,700 - PPMB}. \quad (A.6)$$

Division of Eq. (A.6) by Eq. (A.3) then provides the ratio of Polybor® added to borax plus boric acid crystals added as necessary to produce a given boron concentration.

$$\frac{WPLY}{WTSPB} = \frac{143,698 - PPMB}{209,700 - PPMB}, \quad (A.7)$$

which clearly indicates that this ratio decreases as the desired boron concentration increases.

The ratios predicted by Eqs. (A.6) and (A.7) are listed in Table A.3 for a spectrum of boron concentrations.

Table A.3 Relative weights of Polybor®, sodium pentaborate, borax, and boric acid required to achieve a given boron concentration (ppm) in water

PPMB	Polybor®	Polybor®
	Sodium pentaborate	Borax + boric acid
10	0.8736	0.6852
100	0.8734	0.6851
1,000	0.8717	0.6837
3,540	0.8667	0.6798
10,000	0.8535	0.6695
20,000	0.8313	0.6521
30,000	0.8066	0.6327
40,000	0.7790	0.6111

The special concentration 3540 ppm has been included in this table because, with the boron in its natural form, this concentration corresponds to 700 ppm of the boron-10 isotope.

To this point, the discussion of this appendix has focused upon the characteristic property of Polybor® that less of it is required to form the same boron concentration in water. However, with respect to the boration strategy that is the primary consideration of this report, it is also important to note the high solubility of Polybor®, particularly at lower temperatures. Exhibits A.1 and A.2 display the two sides of Technical Data Sheet IC-13 (prepared by the U.S. Borax Company), which provides information concerning Polybor®; its solubility is described at the center of Exhibit A.1.

The high solubility of Polybor® relative to that of borax and boric acid is explored further in Appendix B. However, the fundamental reason for this advantage can easily be recognized. When borax and boric acid crystals are introduced into water, they must find each other and react to form sodium pentaborate. When Polybor® is added to water, it merely dissolves.

The Material Safety Data Sheet (pages 1–5) provided for Polybor® by the U.S. Borax Company is reproduced as Exhibit A.3. There are no special requirements for the handling of Polybor® beyond those necessary for the handling of borax and boric acid.

Polybor® is currently sold in 50-lb (23-kg) bags at a price of $0.50 /lb. Special quality grade (as is employed for the borax and boric acid currently supplied to nuclear power

plants) is not currently available. However, a U.S. Borax Company representative has indicated that arrangements for a special quality grade of Polybor® could be made, at approximately double the current price.

POLYBOR (Disodium Octaborate Tetrahydrate) is a special sodium borate product, in readily soluble powder form having the approximate composition $Na_2B_8O_{13} \cdot 4H_2O$, and the formula weight of 412.52.

HYDROGEN ION CONCENTRATION: Aqueous solutions of POLYBOR range from mildly alkaline at low concentrations to practically neutral as concentration increases at ordinary temperatures.

Percent POLYBOR by weight of solution	pH at 23°C (73.4°F)
1%	8.5
2	8.4
5	8.0
10%	7.6
15	7.3

CAS NUMBER 12008-41-2

SOLUBILITY IN WATER, and corresponding concentrations of B_2O_3 compared with borax at same temperatures:

Temperature °C.	°F.	weight % POLYBOR	% Concentration of B_2O_3 in saturated solutions of: POLYBOR	Borax
0	32	2.4	1.6	0.73
10	50	4.5	3.0	1.13
20	68	9.5	6.3	1.72
30	86	21.9	14.5	2.63
40	104	27.8	18.4	4.10
50	122	32.0	21.2	6.54
60	140	35.0	23.2	11.07
75	167	39.3	26.0	14.67
94	201	45.3	30.0	21.0

Solubilities in the above table are for equilibrium conditions. POLYBOR readily dissolves even in cool water to give supersaturated solutions of considerably higher concentration than indicated in the table. At temperatures above 60°C (140°F), concentrated POLYBOR solutions become very viscous, and may have application where heavy coatings rather than impregnations are desired.

INDUSTRIAL USES: Fire retardant treatment of lumber by heavy spray application or by immersion treatment of decorative and other cellulosic materials.

U.S. Patent No. 2,998,310

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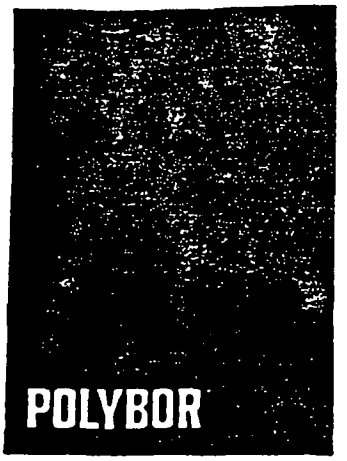
Exhibit A.1 Front of Technical Data Sheet IC-13



TYPICAL ANALYSIS

CHEMICAL

Sodium Oxide (Na_2O)	14.7%
Boric Oxide (B_2O_3)	67.1
Water (H_2O by difference)	18.2
Equiv. Disodium Octaborate Tetrahydrate	99.4

TECHNICAL GRADE

SCREEN DESCRIPTION	AVG. BULK DENSITY Pounds per Cubic Foot Loose Pack	Tight Pack
A fine, rapidly soluble spray-dried product	25	35

CONTAINERS

Multiwall paper bags with a polyethylene free film moisture-resistant barrier, 50 pounds net.

NOTICE: THERE ARE NO WARRANTIES, EXPRESS OR IMPLIED INCLUDING ANY WARRANTY OF FITNESS FOR A PARTICULAR PURPOSE, WHICH EXTEND BEYOND THE DESCRIBED USES IN THIS BULLETIN.

WARNING: ON ALL PRODUCTS, AVOID PROLONGED INHALATION OR PROLONGED SKIN CONTACT. NOT FOR FOOD OR DRUG USE. READ ALL INSTRUCTIONS RELATING TO THE PRODUCTS BEFORE USE.

If possible uses of these products have been mentioned herein, it is not intended that the above products be used to practice any applicable patent, whether mentioned in this Bulletin or not, without procurement of a license, if necessary, from the owner, following investigation by the user. Nor is it intended or recommended that the products be used for any such described purposes without verification of their safety and efficacy for such purposes.

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UNITED STATES BORAX & CHEMICAL CORPORATION
[illegible]
REGIONAL OFFICES [illegible]

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Exhibit A.2 Reverse of Technical Data Sheet IC-13

U.S. BORAX

MATERIAL SAFETY DATA SHEET

Meeting OSHA Standard 29CFR § 1910.1200 (g)
CAL OSHA Standard Title 26 § 8—5194 (g)

EFFECTIVE DATE: August 31, 1990

SECTION I — PRODUCT IDENTIFICATION

PRODUCT TRADE NAME: POLYBOR®

TSCA NO.: 12008-41-2

CHEMICAL NAME AND SYNONYMS:
Disodium octaborate tetrahydrate

CAS NO.: 12280-03-4

CHEMICAL FAMILY: Sodium Borate

FORMULA: $Na_2B_8O_{13} \cdot 4H_2O$

PHYSICAL HAZARD RATING: National Fire Protection Association

Health	0
Flammability	0
Reactivity	0

SECTION II — HAZARDOUS INGREDIENTS

MATERIAL OR COMPONENT % :

Sodium octaborate tetrahydrate >98% CAS No. 12008-41-2

WARNING: This product contains trace amounts of arsenic, a chemical known to the State of California to cause cancer.

SECTION III — PHYSICAL DATA

APPEARANCE: White, odorless powder

VAPOR PRESSURE: Negligible

SOLUBILITY IN WATER: 9.7% @ 20°C (68°F); 40.7% @ 60°C (140°F);

FORMULA WEIGHT: 412.52

pH 3% SOLUTION: @ 23°C 1%-8.5; 10%-7.6; 15%-7.3

24 HOUR EMERGENCY TELEPHONE NUMBER: (714) 774-2673

CONTACT: P.L. Strong; Manager, Product Safety

The information and recommendations contained herein are based upon data believed to be correct. However, no guarantee or warranty of any kind expressed or implied is made with respect to the information contained herein.

UNITED STATES BORAX & CHEMICAL CORPORATION • 3075 WILSHIRE BLVD., LOS ANGELES, CA 90010-1294 USBORAX

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Exhibit A.3 Page 1 of Material Safety Data Sheet for Polybor®

SECTION IV - HEALTH HAZARD INFORMATION

EFFECTS OF ACUTE EXPOSURE

INGESTION:

ACUTE ORAL LD_{50} : 2.55 gram/kg of body weight (Sprague-Dawley rats). Reversible.

HUMAN ACCIDENTAL EXPOSURE: Nausea, vomiting

EYE: Irritant (rabbits - per 16 CFR §1500.42) May be slightly irritating to humans. Reversible

DERMAL:

ACUTE DERMAL LD_{50}: Greater than 2.0 gram/kg of body weight (rabbits - per 16 CFR §1500.40)

PRIMARY SKIN IRRITATION INDEX: 0.5 (rabbits - per 16 CFR §1500.41)

SKIN: Not known to be an irritant

CORROSIVE: Not corrosive per 49 CFR §240.

INHALATION: May be irritating to nose and throat.

EFFECTS OF CHRONIC OVEREXPOSURE

INGESTION: Animal testing for carcinogenicity of boric acid has been negative.

Animal studies show that ingestion of large amounts of borates over prolonged periods of time causes a decrease in sperm production and testicle size in male laboratory animals and developmental effects in fetuses of pregnant female laboratory animals. No evidence of such effects in humans.

EYE: May be irritating

DERMAL: May be irritating on wet skin.

INHALATION: May be irritating to nose and throat.

Exhibit A.3 Page 2 of Material Safety Data Sheet for Polybor®

HEALTH HAZARD INFORMATION (cont. from page 2)

REGULATORY INFORMATION

OSHA PERMISSIBLE EXPOSURE LIMIT (PEL): Not listed 29CFR§1910 SUBPART Z

ACGIH RECOMMENDED THRESHOLD LIMIT VALUE: Not listed

NOT LISTED IN THE NATIONAL TOXICOLOGY PROGRAM ANNUAL REPORT ON CARCINOGENS (1989)

NOT LISTED IN THE INTERNATIONAL AGENCY FOR RESEARCH ON CANCER (IARC) MONOGRAPH

NOT LISTED ON THE OSHA CARCINOGENS LIST

EMERGENCY AND FIRST AID PROCEDURES:

EYES: Flush with tepid water for 15 minutes. If irritation persists, consult a physician.

SKIN: Wash with mild soap and water.

INHALATION: Remove to fresh air.

INGESTION: Drink plenty of milk or water. Induce vomiting.

NOTE TO PHYSICIAN:

Gastric lavage with 5% sodium bicarbonate is suggested. This should be followed by saline catharsis. Assure adequate hydration. POLYBOR® is not considered an acute poison. After ingestion or absorption into the bloodstream of large amounts (15 grams or more), symptoms may appear after 24-72 hours. Borates are readily dissipated through the urine (70% in the first 24 hours). Complimentary blood analysis is available for physicians and emergency rooms. Medical consultation is also available. Call (714) 774-2673.

Exhibit A.3 Page 3 of Material Safety Data Sheet for Polybor®

SECTION V - FIRE AND EXPLOSION HAZARD DATA

FLASH POINT (METHOD USED): N/A

FLAMMABLE LIMITS: N/A

EXTINGUISHING MEDIA: None required. Product is an inherent fire retardant.

SPECIAL FIREFIGHTING PROCEDURES: None are required. No potential for fire or explosion hazard. Product is an inherent fire retardant.

UNUSUAL FIRE AND EXPLOSION HAZARDS: None

SECTION VI - REACTIVITY DATA

STABILITY: POLYBOR® is a stable product.

INCOMPATIBILITY (MATERIALS TO AVOID): None

HAZARDOUS DECOMPOSITION PRODUCTS: None

HAZARDOUS POLYMERIZATION WILL NOT OCCUR:

CONDITIONS TO AVOID: None

SECTION VII - SPILL OR LEAK PROCEDURES

STEPS TO BE TAKEN IN CASE MATERIAL IS RELEASED OR SPILLED: Sweep or vacuum followed by water rinse.

WASTE DISPOSAL METHOD: Refer to local disposal requirements and regulations for waste disposal methods. Not regulated under §313 of SARA Title III or RCRA (40 CFR 261.33)

Exhibit A.3 Page 4 of Material Safety Data Sheet for Polybor®

SECTION VIII - SPECIAL PROTECTION INFORMATION

RESPIRATON PROTECTION (SPECIFY TYPE): Recommend use of light duty dust mask (such as 3M Model 5800) in areas of airborne concentrations greater than 10mg/m^3.

VENTILATION: Local exhaust is sufficient.

PROTECTIVE GLOVES: Leather, cloth or rubber.

EYE PROTECTION: Dust goggles with side shields if dust level is high.

OTHER PROTECTIVE EQUIPMENT: None

SECTION IX - SPECIAL PRECAUTIONS

PRECAUTIONS TO BE TAKEN IN HANDLING AND STORING: Dry indoor storage.

OTHER PRECAUTIONS: None

Revised Dates: November, 1985; October, 1986; February, 1990

DATE: Aug 31, 1990 SIGNATURE: Philip L. Strong

P.L. Strong, Manager, Product Safety

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Page 5 POLYBOR®

Exhibit A.3 Page 5 of Material Safety Data Sheet for Polybor®

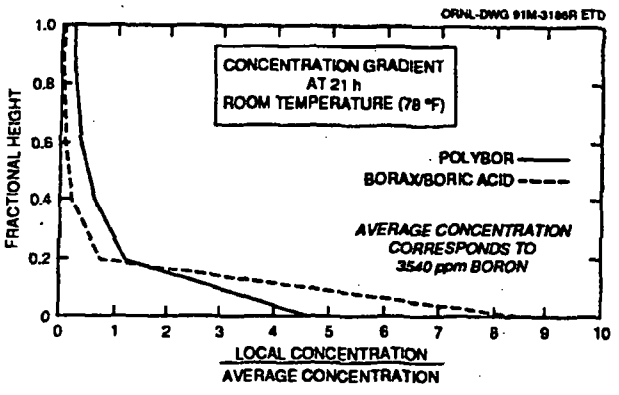
Appendix B

Tabletop Experiments

Simple testing demonstrates that Polybor® [spray-dried disodium octaborate tetrahydrate ($Na_2B_8O_{13} \cdot 4H_2O$)] dissolves much more readily in water than does the normally used mixture of borax and boric acid crystals. This is of interest when considering an accident management strategy for use under station blackout conditions, where the water in the condensate storage tank may have cooled significantly at the time the borated solution was to be prepared and mechanical mixing of the tank contents would not be available.

A series of stationary experiments was carried out to investigate the greater solubility of Polybor® with respect to borax/boric acid by determining the concentration distributions of the two boration agents in a cylindrical container as functions of the time elapsed after dumping and of the water temperature. The concentrations of Polybor® and of sodium pentaborate were determined by titration with hydrochloric acid. (The validity of the titration method was checked by calibration using five known boron concentrations in the range of interest.)

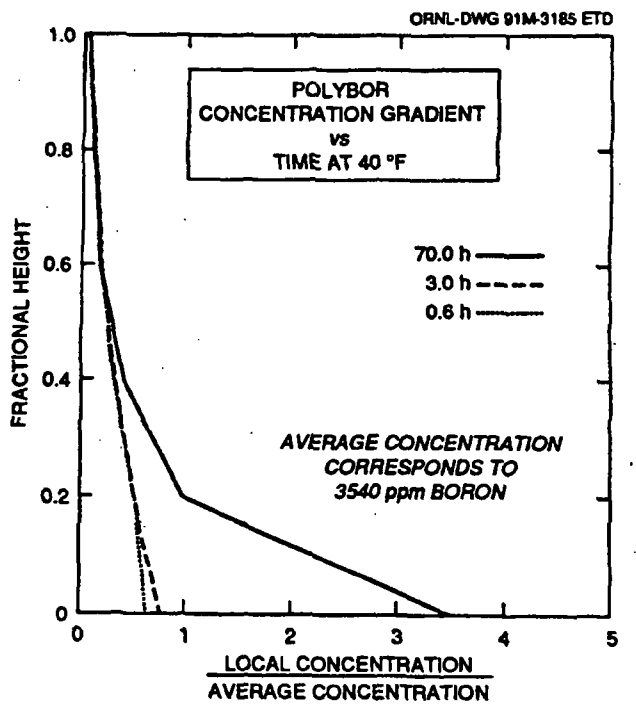
At room temperature [78°F (298.7 K)], the relative concentrations of boron after 21 h in solutions produced by addition of Polybor® and of borax/boric acid are as shown:



In this figure, the vertical axis represents the normalized height of the location within the cylindrical container where the test samples were taken, while the horizontal axis represents the normalized boron concentration. The average boron concentration (3540 ppm) is that which would be uniformly present within the beaker if it were "well-mixed" and corresponds to 700 ppm of the boron-10 isotope. It is evident that the Polybor® produces a more even distribution of boron. The very high concentration of boron at the bottom of the container in the case of borax/boric acid is caused by the presence of some undissolved solids.

At intermediate temperature [60°F (288.7 K)], the solubility rate for Polybor® was 2 to 3 times faster than for borax/boric acid, which was manifested by higher boron concentrations throughout the container. After 24 h, large undissolved crystals were observed at the bottom of the tank for the case of borax/boric acid whereas a fine suspension was observed for the case of Polybor®.

At low temperature [40°F (277.6 K)], only Polybor® was tested with the results shown below.



The powder dissolved very slowly, as evidenced by the low local concentrations throughout the container as late as 3 h after dumping (nowhere is the concentration corresponding to a well-mixed solution average attained at this time). Finally, however, the Polybor® was completely dissolved, as indicated by the results for time 70 h.

These experiments indicate that without any mixing, boration would proceed prohibitively slowly at low temperatures in stagnant water. However, in the real situation, appreciable mixing may be induced by the water surge through the tank exit nozzle, which is located near the bottom of the condensate storage tank.

The simple stationary tests described above demonstrate that Polybor® dissolves much more readily in water than does the normally used mixture of borax and boric acid crystals. Subsequently, two tabletop experiments were performed to investigate the effects of convection and mixing induced by the draining of a boron solution from the bottom of a tank. The basic procedure employed for these experiments follows:

1. Prepare a cylindrical container of demineralized water at the desired test temperature. [The water height-to-diameter ratio is the same as that for a typical BWR condensate storage tank drained to its standpipe reserve (0.42)].
2. Slowly sprinkle a predetermined amount of powder (either Polybor® or borax/boric acid crystals) onto the water surface. The total amount of powder added would achieve a boron concentration of C_1 ppm if the solution were well-mixed. However, no mixing is applied to the container.
3. After a brief pause, drain the container through the horizontal exit pipe at the base. [The exit pipe is positioned such that approximately the same portion of the original water volume remains in the container (~3%) as would remain in a drained condensate storage tank (~4%)].
4. Determine the boron concentration C_2 in the drained solution and compare this to the concentration C_1 originally prepared (perfect mixing) for the container.

The first experiment provided a comparison of the C_2/C_1 ratios for Polybor® and for borax/boric acid crystals at room temperature. Sufficient powder for a boron-10 concentration of 700 ppm (uniformly dissolved) was added in each case at comparable rates. It was observed that practically all of the Polybor® was dissolved, whereas some of the borax/boric acid crystals settled to the bottom of the container. Immediately after adding the salts, the solution [~3 gal (0.011 m^3)] was drained from the container, requiring ~15 min to empty.

The amounts of Polybor® and of sodium pentaborate (formed from dissolved borax/boric acid) were determined for the drained solution and for the residual liquid remaining in the container. The corresponding boron-10 concentrations are shown in Table B.1.

The results of this simple experiment again demonstrate the clear superiority of Polybor® for the purpose intended, to poison a storage tank in preparation for its contents to be injected into a reactor vessel.

The second experiment investigated the effectiveness of Polybor® for this purpose at low temperature. Sufficient Polybor® was added to the container for a (well-mixed) concentration of the boron-10 isotope of 1000 ppm; the temperature of the container solution increased from 36°F to 40°F (275.4 K to 277.6 K) during the 15-min period that the powder was being added. It was observed that the impact of small clumps of Polybor® falling onto the water surface causes breakup and spreading in the radial directions. Furthermore, a vivid interaction of the fragments with water as they sink toward the bottom of the container enhances the solubility. After draining, the results are shown in Table B.2.

This low-temperature test did not include consideration of a sodium pentaborate solution because it was thought that the case for the clear superiority of the solubility of Polybor® had been sufficiently demonstrated by the previous test (Table B.1).

Table B.1 Boron-10 concentrations (ppm) in released and residual container liquids for a room-temperature (78°F) test

Salt	Initial (well-mixed) concentration	Concentration of release	Concentration of residual
Borax/boric acid	700	225	2660
Polybor®	700	456	1323

Table B.2 Boron-10 concentrations (ppm) in released and residual container liquids for a test at low temperature (38°F)

Salt	Initial (well-mixed) concentration	Concentration of release	Concentration of residual
Polybor®	1000	527	2580

Assuming that Polybor® would be used, it now becomes of interest to consider the amount of excess powder that would have to be added to the tank to ensure that the desired concentration would be observed in the drainage (see Table B.3).

From these results, it seems that enough Polybor® must be added to the condensate storage tank under accident conditions to produce a solution of approximately double the desired reactor vessel concentration (if well mixed). This excess within the tank is necessary if the desired reactor vessel concentration is to be achieved without mixing in all weather. Accordingly, a final set of four simple tabletop experiments was performed to investigate the loading capacity of Polybor® in water.

Weighed amounts of Polybor® were poured into individual beakers containing 50 mL of distilled water cooled to 50°F (283.2 K). The target concentrations of the boron-10 isotope were 4000, 6000, 8000, and 10,000 ppm. Mixing was applied by bubbling a gentle stream of compressed air for 10 min, after which the mixtures were left unstirred. The density of the four solutions thus obtained was determined by weighing 5.0 mL of each solution. For the two cases where the Polybor® powder was incompletely dissolved, the samples were taken from the clear solutions above the sediment. The results are summarized in Table B.4.

It should be noted that the target concentrations of 8000 and 10,000 ppm of the boron-10 isotope were not attained in solution because the added powder was not completely dissolved. However, these results demonstrate that supersaturated solutions with concentrations as high as 6000 ppm can be achieved in cool water with very simple means of slight mixing, such as bubbles induced by an air hose. Attempts to store such prepared solutions for possible use under accident conditions would probably be unsuccessful, however, because prolonged standing of a supersaturated solution would induce some of the less soluble phases to crystallize irreversibly.

As mentioned at the beginning of this appendix, the boron concentrations formed during these tabletop experiments were determined by titration with 0.1 N hydrochloric acid (HCL). It is recognized that this method introduces a small systematic error, due to the carbon dioxide dissolved in these slightly alkaline solutions. To obtain a practical correction factor, a representative set of HCL titration results

Table B.3 Ratio of released concentration (ppm) to the initial (well-mixed) concentration for Polybor® at two temperatures

Temperature (°F)	Initial concentration	Release concentration	Ratio C_2/C_1
78	700	456	0.65
38	1000	527	0.53

Table B.4 Effects of adding large amounts of Polybor® to cool water

Polybor® added		Boron-10 isotope	Observation
g/mL	ppm	(ppm)	
0.0965	96,520	4,000	Dissolved within 10 min
0.1449	144,860	6,000	Dissolved within 10 min
0.1934	193,440	8,000	Incompletely dissolved
0.2414	241,400	10,000	Incompletely dissolved

were compared with those obtained using the more elaborate and more accurate mannitol titration* procedure for determining the concentration of B_2O_3. It was found that the values obtained by HCL titration are low by a factor of 1.092, and this correction factor has been applied (where appropriate) to obtain the results described in this appendix.

*W. W. Scott: *Standard Methods of Chemical Analysis*, N. H. Furman, Ed., 5th ed., 1946, p. 170.

Appendix C

Small Reactor Calculations

Chapters 14 through 22 of this report address the results of calculations and analyses based upon the large boiling water reactor (BWR) Mark I containment facilities such as Peach Bottom or Browns Ferry. As explained in Chap. 16, physical considerations dictate that the effectiveness of water cooling of the outer surface of the reactor vessel bottom head in removing heat from the lower plenum debris bed would be increased as the vessel size was reduced. Accordingly, the conservative approach to evaluate this strategy based upon its hypothetical application to the large BWR facilities has been adopted. The results of this evaluation for the existing plants (without modifications to reduce the trapped gas pocket within the vessel skirt) have been provided in Chap. 19.

For this appendix, the calculations described in Chap. 19 have been repeated with only the physical dimensions of the plant changed as appropriate to represent the smallest of the BWR Mark I containment facilities, Duane Arnold (see Table 14.1). The purpose is to demonstrate the increased effectiveness of the proposed drywell flooding strategy for the smaller reactor vessels.

C.1 Accident Sequence

The Duane Arnold calculations are based upon the short-term station blackout accident sequence, which was also represented in the Peach Bottom calculations described in Sect. 19.1. A comparison of the timing of the initial events for this accident sequence as calculated for Peach Bottom (Table 19.1) and Duane Arnold is provided in Table C.1. As indicated, the predicted progressions of the vessel internal events are similar for these two facilities through the time of lower plenum debris bed dryout.

C.2 Containment Pressure and Water Level

In keeping with the goal of maintaining the Duane Arnold conditions similar to the conditions considered for the base case Peach Bottom calculations as described in Sect. 19.2, the drywell is assumed to be vented as necessary to maintain the containment pressure at 40 psia (0.276 MPa). The reactor vessel pressure then cycles between 60 and 90 psia (0.414 and 0.621 MPa) as water is evaporated from the downcomer region and the safety/relief valves (SRVs) open and close.

Table C.1 Calculated timing of events for short-term station blackout accident sequence at Peach Bottom and Duane Arnold

Event	Time (min)	
	Peach Bottom	Duane Arnold
Station blackout-initiated scram from 100% power. Independent loss of the steam turbine-driven HPCI and RCIC injection systems	0.0	0.0
Swollen water level falls below top of core	40.3	40.8
Open one SRV	77.0	77.0
ADS system actuation	80.0	80.0
Core plate dryout	80.7	80.4
Relocation of core debris begins	130.8	129.8
First local core plate failure	132.1	130.9
Collapse of fuel pellet stacks in central core	215.9	210.4
Reactor vessel bottom head dryout; structural support by control rod guide tubes fails; remainder of core falls into reactor vessel bottom head	245.8	251.2

With the drywell pressure held near 40 psia (0.276 MPa), the saturation temperature of the water surrounding the reactor vessel would be 267°F (403.7 K). Beneath the vessel support skirt, the water is assumed to cover the outer surfaces of the four lowest bottom head wall calculational nodes. For Duane Arnold, this would leave a trapped gas pocket similar to that depicted for Peach Bottom in Fig. 19.1. The smaller volume of the gas pocket at Duane Arnold and the reduced thickness of the vessel bottom head (Table 16.1) have been considered in the calculations. For both Peach Bottom and Duane Arnold, the support skirt attachment is at the upper boundary of wall node 12.

C.3 Initial Debris Bed Configuration

The initial configuration of the lower plenum debris bed for the Duane Arnold calculations is shown in Fig. C.1. This may be compared to the initial configuration for Peach Bottom shown in Fig. 19.2 and described in Sect. 19.3.

It is important to note that the initial height of the central portion of the debris bed [101 in. (2.565 m)] is greater than the radius of curvature [91.5 in. (2.324 m)] of the bottom head hemisphere for the Duane Arnold calculation (Fig. C.1). For the outer "crust" layer, the code model restricts the initial height as necessary to provide a clearance of at least 1 in. (0.0254 m) beneath the bottom of the shroud baffle, which is located at 89.5 in. (2.273 m) above vessel zero.

Reference to Fig. 19.2 reveals, however, that the initial debris height for the Peach Bottom calculation [116 in. (2.946 m)] is uniform across the bed and lies well below both the bottom head center of curvature [125.5 in. (3.188 m)] and the bottom of the shroud baffle [121.5 in. (3.086 m)]. Why would a whole-core debris bed be easily contained within the bottom head hemisphere at Peach Bottom yet produce a significant overflow at Duane Arnold?

The answer lies within the most basic considerations of geometry. Because the length of the fuel assemblies forming the core is the same at all BWR facilities, the volume of the core tends to decrease with the square of the reactor vessel radius. The volume of the bottom head hemisphere,

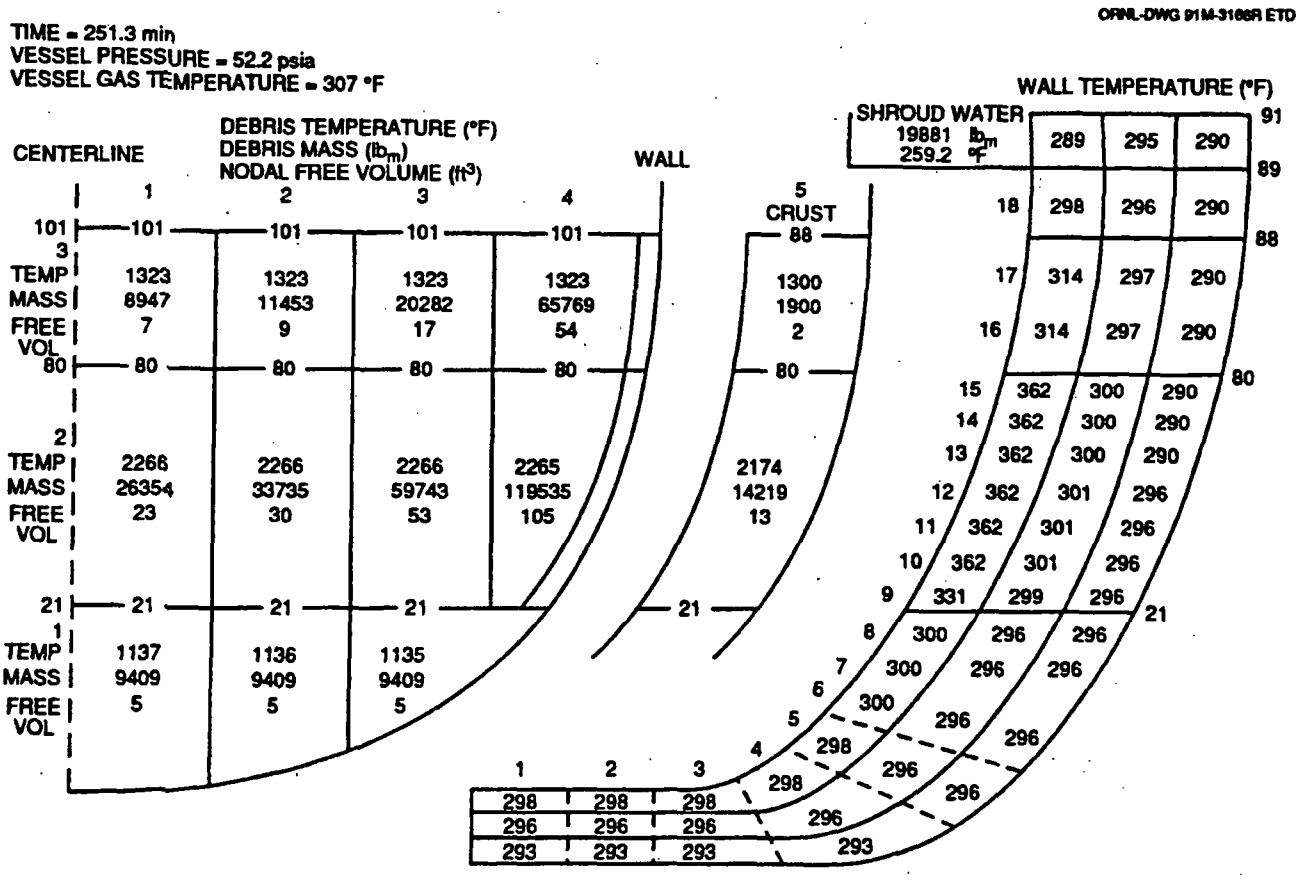


Figure C.1 Initial configuration of lower plenum debris bed and initial bottom head wall temperatures for calculations based upon Duane Arnold facility

however, decreases with the cube of the vessel radius. The effect of this unequal race is that the ratio of the core volume to the bottom head volume increases as the reactor vessel size decreases, culminating in the overflow of debris for Duane Arnold that is indicated in Fig. C.1.

C.4 Lower Plenum and Bottom Head Response

By time 360 min after scram, the calculated relocation and settling of the lower plenum debris has reconfigured the bed into the structure shown in Fig. C.2. As indicated, all of the material is now contained within the bottom head hemisphere and beneath the bottom of the shroud baffle. This situation may be compared with the Peach Bottom results for the same time after scram, which are provided in Fig. 19.4.

The reader is reminded that in both the Peach Bottom and Duane Arnold calculations, bottom head wall nodes one through four transfer heat by nucleate boiling to water on their outer surfaces. Wall nodes five through twelve transfer heat by natural convection to the atmosphere trapped in the armpit region between the reactor vessel support skirt and the vessel wall (Fig. 19.1), while wall nodes 13 through 19 transfer heat to the water surrounding the vessel wall above the skirt attachment. For both plants, the maximum predicted wall temperatures at time 360 min occur at wall node 11, just beneath the support skirt attachment and adjacent to the trapped gas pocket. In neither case are the wall temperatures threatening at this time.

For the Peach Bottom calculations, failure of the bottom head wall by creep rupture at node 11 is considered certain by time 840 min, as discussed in Sect. 19.4. For Duane Arnold, however, the calculated situation at time 840 min after scram, shown in Fig. C.3, is not threatening. The maximum average wall temperature at Peach Bottom (Fig. 19.12) is 2221°F (1490 K), while the Duane Arnold maximum average wall temperature (node eight) is 1929°F (1327 K) at this time. Because the wall tensile stresses that

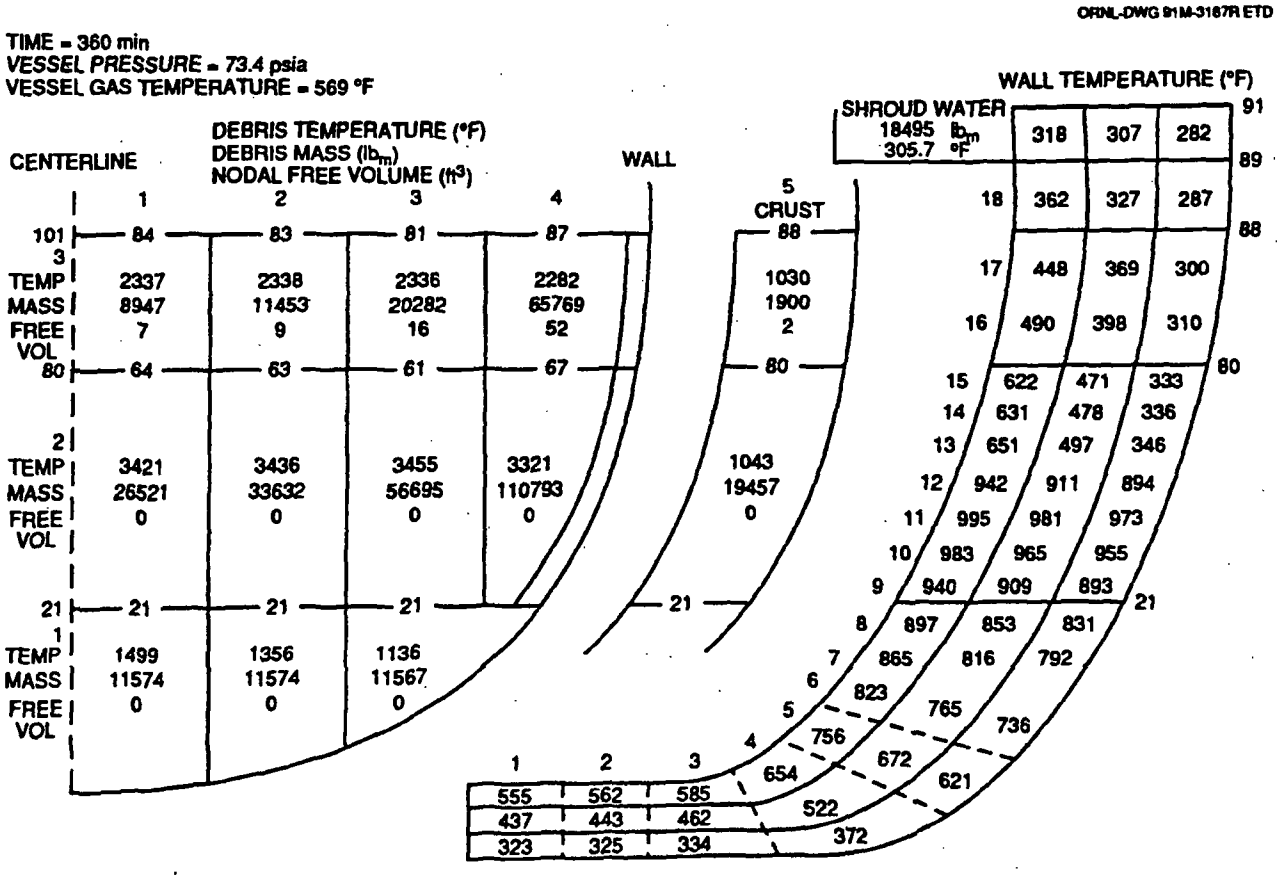


Figure C.2 Lower plenum debris bed and vessel wall response at time 360 min after scram for calculations based upon Duane Arnold facility

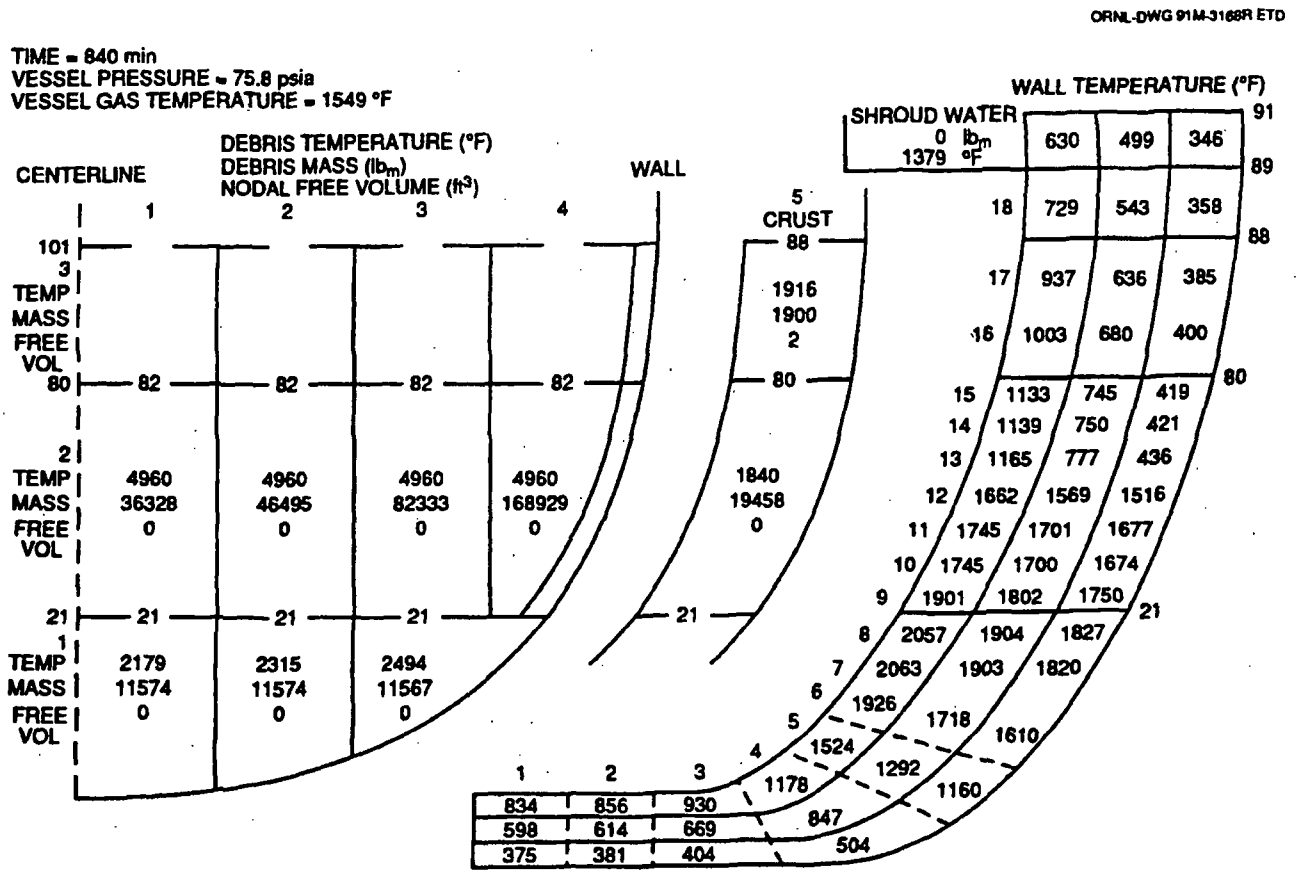


Figure C.3 Lower plenum debris bed and vessel wall response at time 840 min after scram for calculations based upon Duane Arnold facility

would be produced under severe accident conditions at these two plants are almost identical,* the lower predicted wall temperatures for the Duane Arnold calculation directly translate into an increased period of wall integrity.

Based upon the results of these calculations, creep rupture of the reactor vessel bottom head would be expected to occur for Duane Arnold at some time between 900 min (Fig. C.4) and 960 min (Fig. C.5) after scram. Wall node eight continues to endure the maximum wall temperatures; the local average is predicted to increase from 2101°F to 2193°F (1423 K to 1474 K) during this 1-h period. As indicated in Table 18.7, temperatures of this magnitude are sufficient to induce creep rupture in <1 h for wall tensile stresses in the neighborhood of 5 MPa.

*With a reactor vessel-to-drywell pressure differential of 50 psi (0.345 MPa) and considering the weight of debris, the wall stress at Duane Arnold would be 5.12 MPa as compared to 5.25 MPa for Peach Bottom.

C.5 The Effect of Vessel Size

As described in the preceding section, the same unmitigated short-term station blackout severe accident sequence leads to conditions inducing bottom head creep rupture ~2 h later for Duane Arnold than for Peach Bottom. Much of this delay is attributable to the greater heat transfer through the bottom head wall for the smaller reactor vessel.

The predicted energy transfer from the outer surface of the reactor vessel wall is provided in Table C.2 for the Duane Arnold calculation. As indicated, this energy transfer constitutes a greater proportion of the decay heat release as the calculation progresses and the wall becomes hotter. This information can be compared with the similar information for the Peach Bottom base case as listed in Table 19.7. The percentages of the current decay heat releases represented by the energy transfers through the wall for these two calculations are directly compared in Table C.3.

It should be noted from Table C.3 that a higher proportion of the decay heat is removed through the bottom head wall

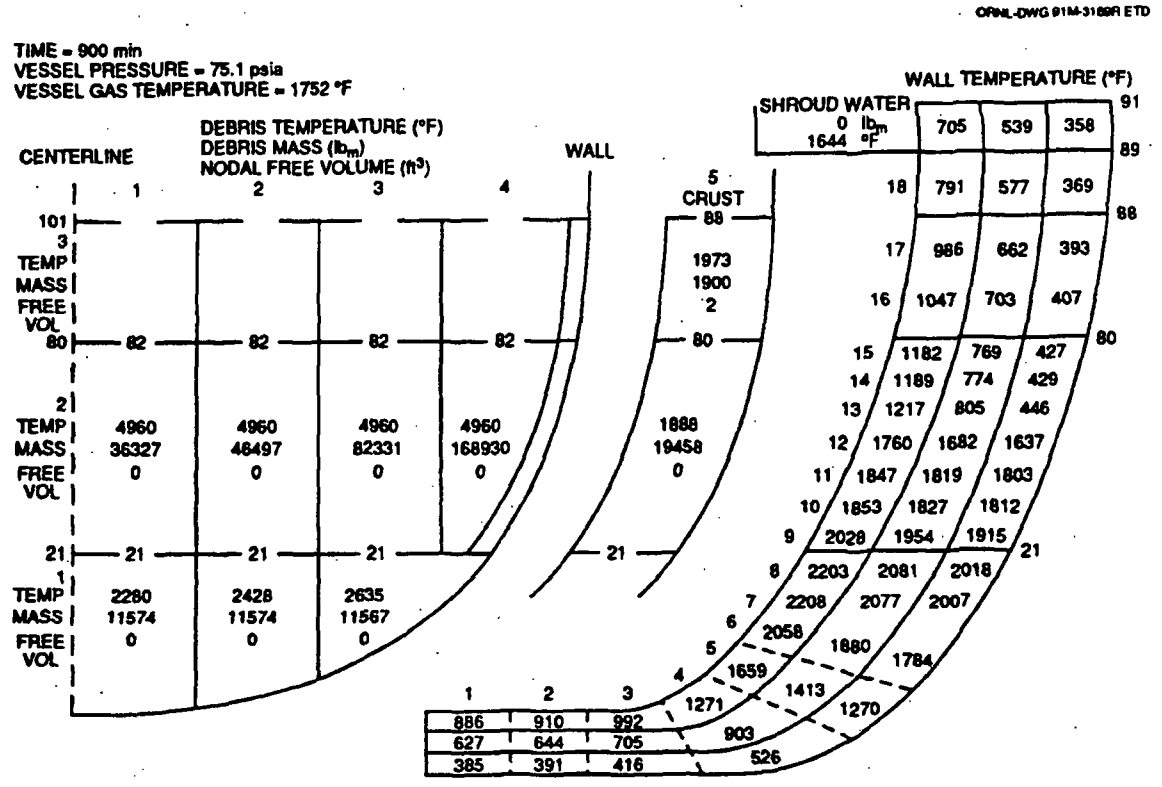


Figure C.4 Lower plenum debris bed and vessel wall response at time 900 min after scram for calculations based upon Duane Arnold facility

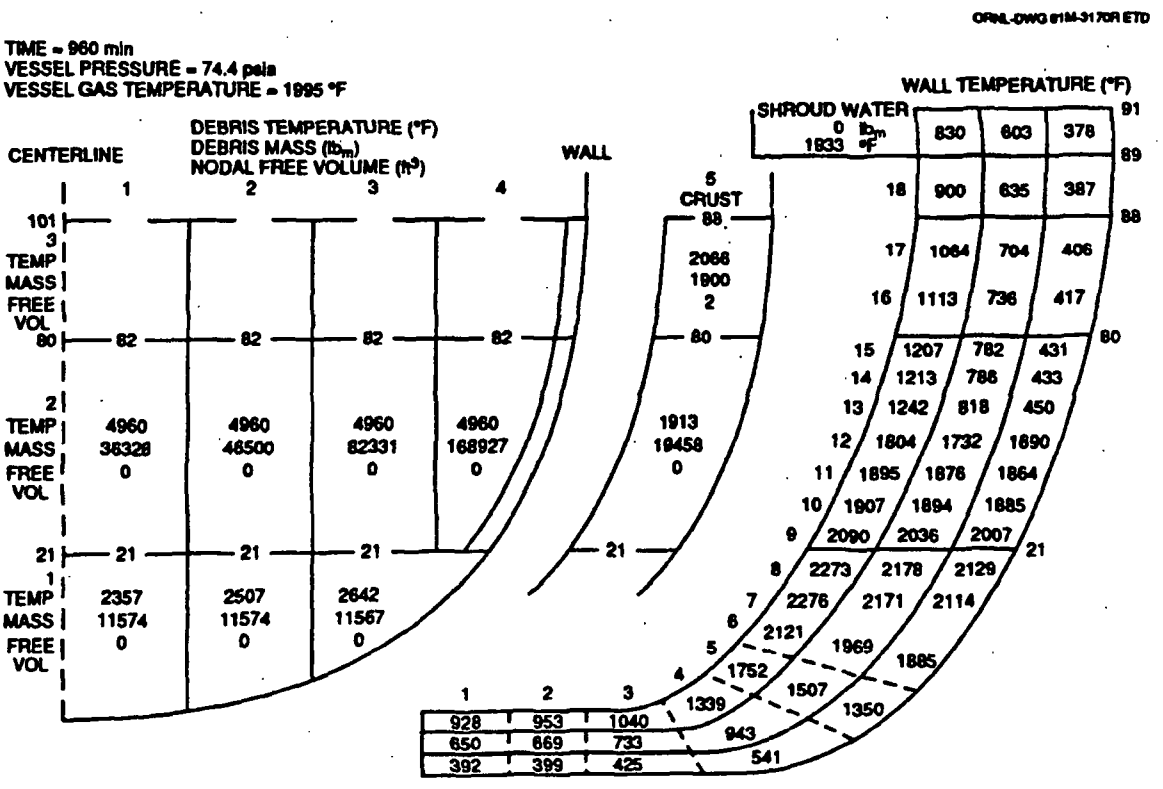


Figure C.5 Lower plenum debris bed and vessel wall response at time 960 min after scram for calculations based upon Duane Arnold facility

Table C.2 Integrated heat transfer from outer surface of the reactor vessel bottom head for Duane Arnold short-term station blackout with drywell flooding

Period (min)	Decay heat released (Btu × 10^6)	Heat transfer from vessel wall	
		Btu × 10^6	%
251.3–300	27.870	2.613	9.4
300–360	32.545	4.223	13.0
360–420	31.185	4.562	14.6
420–480	29.919	4.957	16.6
480–540	28.710	5.101	17.8
540–600	27.395	5.411	19.8
600–660	26.738	6.394	23.9
660–720	25.657	7.021	27.4
720–780	25.318	8.193	32.4
780–840	24.778	9.722	39.2
840–900	24.237	10.416	43.0

Table C.3 Comparison of heat transfer from outer surface of reactor vessel bottom head as percentage of decay heat for Peach Bottom and Duane Arnold short-term station blackout with drywell flooding

Period (min)	Heat transfer from vessel wall (percent of decay heat release during period)	
	Peach Bottom	Duane Arnold
Dryout[a]–300	7.2	9.4
300–360	9.0	13.0
360–420	10.6	14.6
420–480	12.1	16.6
480–540	15.3	17.8
540–600	20.5	19.8
600–660	25.4	23.9
660–720	32.3	27.4
720–780	37.3	32.2
780–840		39.2
840–900		43.0

[a]Lower plenum debris bed dryout occurs at 246.2 min after scram for Peach Bottom and 251.3 min after scram for Duane Arnold.

for the Duane Arnold calculation during the first 6 h after lower plenum debris bed dryout. Overall, 15.1% of the decay heat follows this pathway during the period between bed dryout and time 600 min for Duane Arnold as opposed to 12.2% for Peach Bottom. After this time, the Peach Bottom inner wall temperature is significantly higher than the Duane Arnold wall temperature; correspondingly, the heat transfer through the Peach Bottom wall becomes greater than the Duane Arnold heat transfer until bottom head creep rupture after 780 min. For Duane Arnold, more than 40% of the decay heat is predicted to be transferred through the bottom head during the period just before creep rupture.

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43. A. E. Levin, Division of Systems Technology, U. S. Nuclear Regulatory Commission, Rockville, MD 20852
44. S. L. Dingman, Sandia National Laboratory, Division 6412, Albuquerque NM 87185
45. M. Z. Podowski, Department of Nuclear Engineering and Engineering Physics, Rensselaer Polytechnic Institute, Troy NY 12180-3590
46. C. A. Kukielka, Pennsylvania Power and Light Company, A2-5, Two North Ninth Street, Allentown, PA 18104
47. Office of Assistant Manager for Energy Research and Development, DOE-OR, Oak Ridge, TN 37831
48–49. Office of Scientific and Technical Information, P. O. Box 62, Oak Ridge, TN 37831

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

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S. A. Hodge, J. C. Cleveland, T. S. Kress, M. Petek

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Oak Ridge, TN 37831-6285

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Division of Systems Research
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This report provides the results of work carried out in support of the U.S. Nuclear Regulatory Commission Accident Management Research Program to develop a technical basis for evaluating the effectiveness and feasibility of current and proposed strategies for BWR severe accident management. First, the findings of an assessment of the current status of accident management strategies for the mitigation of in-vessel events for BWR severe accident sequences are described. This includes a review of the BWR Owners' Group Emergency Procedure Guidelines (EPGs) to determine the extent to which they currently address the characteristic events of an unmitigated severe accident and to provide the basis for recommendations for enhancement of accident management procedures. Second, where considered necessary, new candidate accident management strategies are proposed for mitigation of the late-phase (after core damage has occurred) events. Finally, recommendations are made for consideration of additional strategies where warranted, and two of the four candidate strategies identified by this effort are assessed in detail. These are (1) preparation of a boron solution for reactor vessel refill should control blade damage occur during a period of temporary core dryout and (2) containment flooding to maintain the core debris within the reactor vessel if the injection systems cannot be restored.

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Bano, Mahmooda

From: Hochevar, Albert R. (INPO) [HochevarAR@INPO.org]
Sent: Monday, April 04, 2011 10:14 AM
To: Scott, Michael
Subject: RE: NISA/TEPCO MEETING

Mike,

I did – not a rumor. I need to have the team understand what is the lay of the land, see how they can help, understand the issues, and get everybody on my team to play in the same sandbox. I will cut it back.
Al

From: Scott, Michael [mailto:Michael.Scott@nrc.gov]
Sent: Monday, April 04, 2011 6:07 PM
To: Hochevar, Albert R. (INPO)
Subject: NISA/TEPCO MEETING

Al:

Rumor has is you brought several of your folks to subject meeting today. Anything you can do to help reduce the number of people at that meeting would be most appreciated. We definitely want you there, but maybe not all of your folks. Happy to discuss and be convinced otherwise.

Thanks

Mike

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Thank you.

AAA/75

Wittick, Brian

From: Wittick, Brian
Sent: Saturday, April 16, 2011 12:27 AM
To: 'Hughart, Joe'
Subject: RE: Personnel accountability check

Tim

NRC are accounted for

-----Original Message-----
From: Hughart, Joe [mailto:jhughart@ofda.gov]
Sent: Saturday, April 16, 2011 12:05 AM
To: DART_PACTSU; Casto, Chuck; Wittick, Brian; dartdoeliaison@ofda.goov; cherryRC@state.gov
Subject: Personnel accountability check

ALCON

The aftershock today is another good opportunity to run down our accountability process, particularly with the NRC transitions.

DART folks please reply directly to me (not reply all), NRC/DOE folks let me know if your personnel are all accounted for.

Tim

AAA176

Wittick, Brian

From: Wittick, Brian
Sent: Saturday, April 16, 2011 1:29 AM
To: Casto, Chuck
Cc: Reynolds, Steven; Meighan, Sean
Subject: alert notification

Chuck,

The earthquake notification system works by JMA sending a signal that is picked up by TV/radio/cellphone carriers, who rebroadcast it to their customers. For cell phones you need either a phone that is specially configured to receive the relay, or an I-phone which can download an app that handles the relay. We have not found a general entity that will send out texts or emails, which is the only way I envision the site team would be able to receive a notification.

AAA/77

Wittick, Brian

From: Wittick, Brian
Sent: Saturday, April 16, 2011 1:36 AM
To: Casto, Chuck
Cc: Reynolds, Steven
Subject: Saturday earthquake

Chuck,

The earthquake this morning was 3-4 at Fukushima

Thanks
Brian

AAA/78

Wittick, Brian

From: Wittick, Brian
Sent: Saturday, April 16, 2011 3:04 AM
To: 'Christopher.Smith@NNSA.DOE.GOV'
Subject: ALERT Notication System

Chris,

These are the two individuals from NISA for the "backpack" alert notifications.

Toshihiro Bannai bannai-toshihiro@meti.go.jp

Masaomi Koyama koyama-masaomi@meti.go.jp

Please let me know if/that they are included in the alert notification system.

Thanks,
Brian

AAA179

Wittick, Brian

From: Wittick, Brian
Sent: Saturday, April 16, 2011 4:01 AM
To: LIA02 Hoc
Subject: RE: RE: RE: IAEA ENAC reprt for April 14 , 2011

What is the ENAC login?

From: LIA02 Hoc
Sent: Thursday, April 14, 2011 7:34 AM
To: Castleman, Patrick; Hipschman, Thomas; Orders, William; Franovich, Mike; Snodderly, Michael; Wittick, Brian; Jones, Cynthia
Cc: Doane, Margaret; Mamish, Nader; Abrams, Charlotte; Schwartzman, Jennifer; Kreuter, Jane; Larson, Emily; LIA06 Hoc; LIA08 Hoc; Whitney, James; Bloom, Steven
Subject: RE: RE: IAEA ENAC reprt for April 14 , 2011

Attached is the IAEA ENAC report for April 14, 2011.

This information is being provided in response to several Commission Offices request.

Please note the sensitivity of the information.

AAA180

Wittick, Brian

From: Wittick, Brian
Sent: Monday, April 18, 2011 6:55 PM
To: LIA08 Hoc; Liaison Japan
Subject: RE: question about remote helicopter pictures

Jeff,

We do not have a copy of a helicopter video. We saw a brief clip at a meeting last Friday but were not provided a copy. We will ask for a copy at today's meetings.

Thanks
Brian

From: LIA08 Hoc
Sent: Monday, April 18, 2011 5:12 PM
To: Liaison Japan
Subject: question about remote helicopter pictures

Mike Weber asked if anyone has seen still photos or videos taken by a remotely operated helicopter at Fukushima. I think you guys have seen some video and still photos, but the images are too large to email back to us, so someone is going to hand carry when they return. Is this correct?

Thanks for any info you can provide.

Jeff Temple
Liaison Team Coordinator
301-816-5800

AAA181

Wittick, Brian

From: Wittick, Brian
Sent: Monday, April 18, 2011 9:00 PM
To: LIA08 Hoc
Cc: Reynolds, Steven
Subject: FW: SFP4 water sampling

This may answer your earlier question about availability of photos/videos.

TEPCO has footage of the water sampling of SFP4 on this page:

http://www.tepco.co.jp/en/news/110311/

This is different from the water sampling video that we obtained from TEPCO. It seems that the sampler can be seen entering the pool.

-Mike

AAA182

Merzke, Daniel

From: Merzke, Daniel
Sent: Tuesday, April 19, 2011 9:42 AM
To: Shropshire, Alan; VandenBerghe, John
Subject: FW: Request from Marty from todays EDO's alignment meeting with Task Force (Japan Event)

I'm not exactly sure which division is responsible for B.5.b, but during the alignment meeting yesterday for the Japan task force, one of the "Prompt Actions" listed was making B.5.b guidance available. Marty then asked if NSIR has a path forward, including a schedule. Is anyone familiar with releasing this guidance, and does has a schedule been drafted? Please let me know as soon as possible. Thanks.

Dan

From: Bush-Goddard, Stephanie
Sent: Monday, April 18, 2011 5:41 PM
To: Merzke, Daniel
Subject: Request from Marty from todays EDO's alignment meeting with Task Force (Japan Event)

Dan,

In today's meeting: EDO Alignment for 5/12 & 6/16 CM re: Near Term Tasking - 30 Day Quick Look & 60 Day Quick Look for the Japanese Event;

Marty wants to confirm with NSIR that we are on a path forward for the B.5.B guidance, timelines and milestones and wants to see them.

We can discuss tomorrow.

-Steph

Stephanie Bush-Goddard, Ph. D. | Executive Technical Assistant | EDO | U.S. NRC
11555 Rockville Pike, Rockville, MD 20852 | ☎ (301) 415-1136 | ✉: Stephanie.Bush-Goddard@nrc.gov

AAA/83

Rihm, Roger

From: Rihm, Roger
Sent: Tuesday, April 19, 2011 2:58 PM
To: Landau, Mindy
Subject: FW: QUESTIONS FOR THE RECORD: 3/16/11- "The FY2012 Department of Energy and Nuclear Regulatory Commission Budgets"
Attachments: QFR- Chairman Gregory B. Jaczko.pdf

FYI

From: Decker, David
Sent: Tuesday, April 19, 2011 2:29 PM
To: Champ, Billie; McKelvin, Sheila; Mike, Linda
Cc: Powell, Amy; Schmidt, Rebecca; Rihm, Roger; Rothschild, Trip
Subject: QUESTIONS FOR THE RECORD: 3/16/11- "The FY2012 Department of Energy and Nuclear Regulatory Commission Budgets"

Billie, Sheila and Linda,
Here are some Questions for the Record associated with the March 16th hearing with the House Energy and Commerce Subcommittees on Energy and Power, and Environment and the Economy that have come in.

David

AAA/84

FRED UPTON, MICHIGAN
CHAIRMAN

HENRY A. WAXMAN, CALIFORNIA
RANKING MEMBER

ONE HUNDRED TWELFTH CONGRESS

Congress of the United States
House of Representatives
COMMITTEE ON ENERGY AND COMMERCE

2125 RAYBURN HOUSE OFFICE BUILDING
WASHINGTON, DC 20515–6115

Majority (202) 225–2927
Minority (202) 225–3641

April 18, 2011

The Honorable Gregory B. Jaczko
Chairman
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Dear Chairman Jaczko:

Thank you for appearing before the Subcommittee on Energy and Power and the Subcommittee on Environment and the Economy on March 16, 2011, to testify at the joint hearing entitled "The FY2012 Department of Energy and Nuclear Regulatory Commission Budgets."

Pursuant to the Rules of the Committee on Energy and Commerce, the hearing record remains open for ten business days to permit Members to submit additional questions to witnesses, which are attached. The format of your responses to these questions should be as follows: (1) the name of the Member whose question you are addressing, (2) the complete text of the question you are addressing in bold, and then (3) your answer to that question in plain text.

To facilitate the printing of the hearing record, please respond to these questions by the close of business on Monday, May 2, 2011. Your responses should be e-mailed to the Legislative Clerk, in Word or PDF format, at Allison.Busbee@mail.house.gov.

Thank you again for your time and effort preparing and delivering testimony before the Subcommittees.

Sincerely,

Ed Whitfield
Chairman
Subcommittee on Energy and Power

John Shimkus
Chairman
Subcommittee on Environment and the Economy

cc: The Honorable Bobby L. Rush, Ranking Member,
Subcommittee on Energy and Power

The Honorable Gene Green, Ranking Member,
Subcommittee on Environment and the Economy

Attachment

The Honorable Bobby Rush

1. What are the qualifications for becoming a NRC inspector and approximately how many qualified inspectors are there currently in the U.S.?

2. What are the protocols for deploying Resident Inspectors? Where and how are they assigned geographically and logistically?

3. What are the protocols for deploying Regional Inspectors? Where and how are they assigned geographically and logistically?

4. What is the rationale for deploying Resident Inspectors at facilities geographically and logistically?

5. What is the rationale for deploying Regional Inspectors at facilities geographically and logistically?

6. What are the term limits for both Resident Inspectors and Regional Inspectors?

7. How do Resident Inspectors receive data for the plants they are assigned? Is this data collected independently or obtained from facility managers?

8. Does the NRC have independent sensors or monitoring equipment at each nuclear facility?

9. If an emergency takes place at a facility does the NRC have the authority to intervene independently or must it wait for permission from facility managers?

10. Does the NRC require each nuclear reactor at each facility to have emergency backup power generators both underground and also off-site?

11. Does the NRC require each nuclear reactor at each facility to have hardened vents installed in the containment areas?

The Honorable Jim Matheson

1. I understand that the NRC is currently updating rulemaking and guidance regarding storage for blended radioactive waste and other "unique" waste streams, like depleted uranium. I have long had concerns about these unique waste streams and whether the Class A storage site in Clive, Utah is appropriate to accept these wastes. I am pleased that the State of Utah decided to require a site-specific performance analysis for these types of waste before they are allowed to be stored there, and I hope that the State remains firm in requiring this analysis.

    a. When will NRC have their Branch Technical Guidance on site-specific performance analysis ready?

    b. Given the increase in unique waste streams that were not included in the low-level waste classification system as defined in Federal code at 10 CFR 61.55, do you believe this classification system should be revised? If so, how long would this process take?

Rihm, Roger

From: Rihm, Roger
Sent: Tuesday, April 19, 2011 3:27 PM
To: Landau, Mindy
Subject: RE: QFRs from EW Appopriations Hearing

We will, but they haven't really gotten into the system yet. I think the first two belong to EDO offices.

From: Landau, Mindy
Sent: Tuesday, April 19, 2011 3:25 PM
To: Rihm, Roger
Subject: RE: QFRs from EW Appopriations Hearing

Are we tasking these?

From: Rihm, Roger
Sent: Tuesday, April 19, 2011 2:56 PM
To: Landau, Mindy
Subject: FW: QFRs from EW Appopriations Hearing

FYI. A Few Qs and As from a Senate hearing. David Decker says we should have ques coming from other hearings as well.

From: Decker, David
Sent: Tuesday, April 19, 2011 2:33 PM
To: McKelvin, Sheila; Champ, Billie
Cc: Powell, Amy; Schmidt, Rebecca; Rihm, Roger; Rothschild, Trip
Subject: QFRs from EW Appopriations Hearing

One more set of Questions for the Record coming in. This small set is from the March 30th hearing with the Senate Appropriations Subcommittee on Energy and Water Development.

AAA/85

Merzke, Daniel

From: Merzke, Daniel
Sent: Tuesday, April 19, 2011 3:57 PM
To: Virgilio, Martin
Subject: Composite Paper

Marty, I talked to Roy Zimmerman today and explained the PLE paper and composite paper, and the various meetings associated with each. He tells me the composite paper has all the comments incorporated, and it's ready to be sent out for interagency comment. He wants your approval. We could go on forever tweaking it, but I recommend getting other stakeholders involved.

Dan

AAA/86

Merzke, Daniel

From: Merzke, Daniel
Sent: Wednesday, April 20, 2011 3:43 PM
To: Virgilio, Martin
Cc: Milligan, Patricia; Andersen, James
Subject: Summary of Meeting w/ Patty Bubar re: Re-Entry Criteria

Marty, I met with Patty and Andrea this afternoon to try to answer questions they had on the PLE re-entry criteria paper, the EPA PAG Manual, and the Fukushima re-entry paper. Trish was unavailable, so I did my best to address their questions. One issue that Patty said she would speak to you about is on the Fukushima re-entry paper. CMR Magwood is concerned that it looks like an NRC policy paper, which includes criteria like infrastructure, which is outside NRC expertise. Patty made a recommendation that the paper have some context. The suggestion was that the paper should start out by stating this is a staff-generated paper developed at the request of the U.S. Ambassador/GOJ/whoever, to recommend criteria for U.S. citizens to re-enter the area inside 50 miles, or words to that effect. That will make it clear that it is not a Commission endorsement of the recommendations being made. Andrea also thought it should be clear that it is an interagency paper, not just an NRC product. I was asked if the Commission would see the paper before it goes out, and I couldn't answer that.

Patty said that they've spoken to Pete Lyons at DOE concerning the PLE re-entry criteria to get his support. Apparently NNSA briefed him that EPA said that the criteria being recommended were just a starting point, so it was downplayed considerably. She said he's willing to support us, but he doesn't appear to be sufficiently informed. Patty asked if staff is trying to work any coalitions with other agencies to build a consensus on optimization. I told her Scott Morris told me they're working on doing just that, but I didn't have any specifics.

The other question was on the timeline for sending the EPA letter stating our non-concurrence on the draft PAG Manual. They think it's important that it be done prior to the Deputies meeting on the PLE document. The need will become more manifest based on the results of your meeting with EPA tomorrow. It was ticketed to be due next week, which will be prior to the Deputies meeting.

Dan

AAA/87

Rihm, Roger

From: Rihm, Roger
Sent: Wednesday, April 20, 2011 5:48 PM
To: Peterson, Gordon; Landau, Mindy; RidsEdoMailCenter Resource
Cc: Golder, Jennifer; Smolik, George; Allwein, Russell; Ojeda, Jennifer; Decker, David
Subject: RE: Post Hearing Questions from the March 31, 2011 Hearing on the FY 2012 Budget Request

We have turned your completed Qs and As over to David Decker in OCA who is compiling them from the various offices. There is time to revise one of your responses. Please let David know directly what your timetable is for supplying the revision.

Roger S. Rihm
Communications and Performance Improvement Staff
Office of the Executive Director for Operations
US NRC
301.415.1717
roger.rihm@nrc.gov

From: Peterson, Gordon
Sent: Wednesday, April 20, 2011 5:44 PM
To: Landau, Mindy; RidsEdoMailCenter Resource
Cc: Golder, Jennifer; Smolik, George; Allwein, Russell; Rihm, Roger; Ojeda, Jennifer
Subject: RE: Post Hearing Questions from the March 31, 2011 Hearing on the FY 2012 Budget Request

Mindy,

It just came to my attention that the amount to be reprogrammed will change. This directly impacts at least one of the questions. Is it possible to hold the document until the exact amount can be confirmed?

Thanks,
Gordon

Gordon S. Peterson, Deputy Director
Division of Planning and Budget
Office of the Chief Financial Officer
Nuclear Regulatory Commission
301-415-7348
gordon.peterson@nrc.gov

From: Ojeda, Jennifer
Sent: Wednesday, April 20, 2011 1:37 PM
To: Rihm, Roger; Landau, Mindy; RidsEdoMailCenter Resource
Cc: Golder, Jennifer; Peterson, Gordon; Smolik, George; Allwein, Russell
Subject: Post Hearing Questions from the March 31, 2011 Hearing on the FY 2012 Budget Request

Attached are the questions and answers OCFO was responsible for handling from the March 31, 2011 Hearing on the FY 2012 budget request. Please let me know if you have any questions.

AAA/88

Wittick, Brian

From: Wittick, Brian
Sent: Wednesday, April 20, 2011 7:13 PM
To: Bloom, Steven
Subject: RE: Looking for Video of Kashiwazaki-Kariwa SFP Sloshing

Steve,

See if you can find what you are looking for here: http://www.tepco.co.jp/en/news/110311

If not let me know; we will have to ask for it separately.

Thanks
Brian

From: Bloom, Steven
Sent: Wednesday, April 20, 2011 4:15 PM
To: Wittick, Brian
Subject: FW: Looking for Video of Kashiwazaki-Kariwa SFP Sloshing

Brian,

Do you know of anyone in Tokyo who may have this video. Please let me know.

Steve

Steven Bloom, International Relations Officer
International Cooperation and Assistance Branch (ICA)
301-415-2431
O-4F4
M/S O-4E21

From: Abrams, Charlotte
Sent: Wednesday, April 20, 2011 3:42 PM
To: Hasselberg, Rick
Cc: Pavlechko, Frank; Foggie, Kirk; Bloom, Steven
Subject: RE: Looking for Video of Kashiwazaki-Kariwa SFP Sloshing

I don't have it, but am copying Kirk Foggie, our Japan desk officer, who may be able to assist – also, Steve Bloom, who is currently with us assisting on Japan.

From: Hasselberg, Rick
Sent: Wednesday, April 20, 2011 3:38 PM
To: Abrams, Charlotte
Cc: Pavlechko, Frank
Subject: FW: Looking for Video of Kashiwazaki-Kariwa SFP Sloshing

Charlotte/Frank,

AAA/89

Would either of you happen to know where I could locate a copy of this short video clip? I'm revising our Incident Response Training related to Spent Fuel Pools and hydrogen generation. Thanks.

Rick

Rick Hasselberg
Sr. Emergency Response Coordinator
NRC Reactor Safety Team
Office of Nuclear Security and Incident Response
M/S T-4A43
Office - 301-415-6417

From: Hasselberg, Rick
Sent: Wednesday, April 20, 2011 3:26 PM
To: Young, Francis
Cc: Andrews, Tom
Subject: Looking for Video of Kashiwazaki-Kariwa SFP Sloshing

Skip/Tom,

I saw this view clip of this one time. Do either of you happen to know where I can get a copy of that? Thanks!

Rick

Wittick, Brian

From: Wittick, Brian
Sent: Wednesday, April 20, 2011 4:43 AM
To: 'nakagawa'
Subject: RE: RE: Lunch with Mr.Mitsumata of METI

Nakagawa-san

The 25th at 1215 is a good time. Thank you for the additional information you provided in support of the meeting. We look forward to an enjoyable conversation.

Best regards,
Brian

From: nakagawa [mailto:nakagawa@ruby.famille.ne.jp]
Sent: Tuesday, April 19, 2011 8:27 AM
To: Wittick, Brian
Subject: Fwd: RE: Lunch with Mr.Mitsumata of METI

Brian san

Let's agree on 25th. 12:15 at the lobby of Okura hotel.

Nakagawa

-------- Original Message --------
Subject:RE: Lunch with Mr.Mitsumata of METI
Date: Tue, 19 Apr 2011 06:18:01 -0400
From: Wittick, Brian <Brian.Wittick@nrc.gov>
To:'??air' <nakagawa@ruby.famille.ne.jp>

Dear Nakagawa-san,

It is always a pleasure to hear from you. I spoke to Chuck Casto and he will be honored to attend lunch with Mitsumata-san. Request clarification of which days are acceptable.

We look forward to our visit.

Best regards,
Brian Wittick

-----Original Message-----
From: 中川air [mailto:nakagawa@ruby.famille.ne.jp]
Sent: Monday, April 18, 2011 11:34 PM
To: Wittick, Brian
Subject: Lunch with Mr.Mitsumata of METI

Brian san

Could you be so kind to try the lunch between Casto san. you and Mitumata san of Director of Nuclear Energy Policy Planning Division and me either on 25 (Mon) or 27 (Tue)?

AAA/90

Nakagawa

Merzke, Daniel

From: Merzke, Daniel
Sent: Wednesday, April 20, 2011 4:03 PM
To: Dudek, Michael
Subject: Composite Paper

Michael, when you get a chance, could you send me the latest draft of the composite paper. I should have reviewed it a while ago. Thanks.

Dan

AAA/91

Merzke, Daniel

From: Merzke, Daniel
Sent: Wednesday, April 20, 2011 9:11 AM
To: Snodderly, Michael
Subject: RE: Request for Temporary Instruction
Attachments: TI-183.pdf

Mike, see attached. I actually sent this to you on the 12th, too. Unfortunately, I neglected to label the document or subject line for easy reference.

Dan

From: Snodderly, Michael
Sent: Wednesday, April 20, 2011 8:58 AM
To: Merzke, Daniel
Subject: FW: Request for Temporary Instruction

Per Greg's out of office email.

From: Snodderly, Michael
Sent: Wednesday, April 20, 2011 8:45 AM
To: Bowman, Gregory
Cc: Franovich, Mike; Castleman, Patrick; Orders, William; Marshall, Michael; Hipschman, Thomas; Gilles, Nanette; Sosa, Belkys; Davis, Roger
Subject: Request for Temporary Instruction

Commissioner Apostolakis would like a copy of the Temporary Instruction or guidance that was provided to Resident Inspectors in response to the Japan event.

Thanks,

Mike Snodderly
Technical Assistant for Reactors
to Commissoner Apostolakis
U. S. Nuclear Regulatory Commission

Phone: 301-415-2241
Email: michael.snodderly@nrc.gov

AAA/92

TEMPORARY INSTRUCTION 2515/183

FOLLOWUP TO THE FUKUSHIMA DAIICHI NUCLEAR STATION FUEL DAMAGE EVENT

CORNERSTONE: INITIATING EVENTS AND MITIGATING SYSTEMS

APPLICABILITY: This Temporary Instruction (TI) applies to all holders of operating licenses for nuclear power reactors, except plants which have permanently ceased operations.

2515/183-01 OBJECTIVES

The objective of this TI is to independently assess the adequacy of actions taken by licensees in response to the Fukushima Daiichi nuclear station fuel damage event. The inspection results from this TI will be used to evaluate the industry's readiness for a similar event and to aid in determining whether additional regulatory actions by the U.S. Nuclear Regulatory Commission are warranted. Therefore, the intent of this TI is to be a high-level look at the industry's preparedness for events that may exceed the design basis for a plant. If necessary, a more specific followup inspection will be performed at a later date.

2515/183-02 BACKGROUND

On March 11, 2011, the Tohoku-Taiheiyou-Oki Earthquake occurred near the east coast of Honshu, Japan. This magnitude 9.0 earthquake and the subsequent tsunami caused significant damage to at least four of the six units of the Fukushima Daiichi nuclear power station as the result of a sustained loss of both the offsite and on-site power systems. Efforts to restore power to emergency equipment have been hampered or impeded by damage to the surrounding areas due to the tsunami and earthquake. The following background information is current as of March 18, 2011.

Units 1 through 3, which had been operating at the time of the earthquake, scrammed automatically, inserting their neutron absorbing control rods to ensure immediate shutdown of the fission process. Following the loss of electric power to normal and emergency core cooling systems and the subsequent failure of back-up decay heat removal systems, water injection into the cores of all three reactors was compromised, and reactor water levels could not be maintained. Tokyo Electric Power Company (TEPCO), the operator of the plant, resorted to injecting sea water and boric acid into the reactor vessels of these three units, in an effort to cool the fuel and ensure the reactors remained shutdown. However, the fuel in the reactor cores became partially uncovered. Hydrogen gas built up in Units 1 and 3 as a result of exposed, overheated fuel reacting with water. Following gas venting from the primary containment to relieve

pressure, hydrogen explosions occurred in both units and damaged the secondary containments. It appears that primary containments for Units 1 and 3 remained functional, but the primary containment for Unit 2 may have been damaged. TEPCO cut a hole in the side of the Unit 2 secondary containment to prevent hydrogen buildup following a sustained period when there was no water injection into the core.

In addition, problems were encountered with monitoring and maintaining Units 3 and 4 spent fuel pool (SFP) water levels. Efforts continue to supply seawater to the SFPs for Units 1 through 4 using various methods. At this time, the integrity of the SFPs for Units 3 and 4 is unknown.

Fukushima Daiichi Units 4 through 6 were shutdown for refueling outages at the time of the earthquake. The fuel assemblies for Unit 4 had been offloaded from the reactor core to the SFP. The SFPs for Units 5 and 6 appear to be intact.

The damage to Fukushima Daiichi nuclear power station appears to have been caused by initiating events that may have exceeded the design basis for the facilities.

2515/183-03 INSPECTION REQUIREMENTS AND GUIDANCE

NRC inspection staff should assess the licensee's activities and actions to assess its readiness to respond to an event similar to the Fukushima Daiichi nuclear plant fuel damage event. These inspections should occur at the operating power reactor facilities. Licensee emergency preparedness will not be assessed by this TI.

This TI may be completed all at once or in phases as the licensee verifies its capability to respond to such an event. The inspector(s) should coordinate the inspection effort with the licensee in accordance with the licensee's verification schedule.

The events at the Fukushima Daiichi plant appear to be caused by factors directly impacting nuclear safety that may have exceeded the design basis for the facility. While details on the full extent of damage to these units remain unknown, the damage poses a significant challenge to the nuclear safety of these units. Immediate actions by the U.S. industry are appropriate to assess and take corrective actions to address potential vulnerabilities that would challenge response to events that are beyond site design bases.

03.01 Assess the licensee's capability to mitigate conditions that result from beyond design basis events, typically bounded by security threats, committed to as part of NRC Security Order Section B.5.b issued February 25, 2002, and severe accident management guidelines and as required by Title 10 of the Code of Federal Regulations (10 CFR) 50.54(hh). Use Inspection Procedure (IP) 71111.05T, "Fire Protection (Triennial)," Section 02.03 and 03.03 as a guideline. If IP 71111.05T was recently performed at the facility the inspector should review the inspection results and findings to identify any other potential areas of inspection. Particular emphasis should be placed on strategies related to the spent fuel pool. The inspection should include, but not be limited to, an assessment of any licensee actions to:

a. Verify through test or inspection that equipment is available and functional. Active equipment shall be tested and passive equipment shall be walked down and inspected. It is not expected that permanently installed equipment that is tested under an existing regulatory testing program be retested.

b. Verify through walkdowns or demonstration that procedures to implement the strategies associated with B.5.b and 10 CFR 50.54(hh) are in place and are executable. Licensees may choose not to connect or operate permanently installed equipment during this verification.

c. Verify the training and qualifications of operators and the support staff needed to implement the procedures and work instructions are current for activities related to Security Order Section B.5.b and severe accident management guidelines as required by 10 CFR 50.54 (hh).

d. Verify that any applicable agreements and contracts are in place and are capable of meeting the conditions needed to mitigate the consequences of these events.

e. Review any open corrective action documents to identify vulnerabilities that may not have yet been addressed.

03.02 Assess the licensee's capability to mitigate station blackout (SBO) conditions, as required by 10 CFR 50.63, "Loss of All Alternating Current Power," and station design, is functional and valid. Refer to TI 2515/120, "Inspection of Implementation of Station Blackout Rule Multi-Plant Action Item A-22" as a guideline. It is not intended that TI 2515/120 be completely reinspected. The inspection should include, but not be limited to, an assessment of any licensee actions to:

a. Verify through walkdowns and inspection that all required materials are adequate and properly staged, tested, and maintained.

b. Demonstrate through walkdowns that procedures for response to an SBO are executable.

03.03 Assess the licensee's capability to mitigate internal and external flooding events required by station design. Refer to IP 71111.01, "Adverse Weather Protection," Section 02.04, "Evaluate Readiness to Cope with External Flooding" as a guideline. The inspection should include, but not be limited to, an assessment of any licensee actions to verify through walkdowns and inspections that all required materials and equipment are adequate and properly staged. These walkdowns and inspections shall include verification that accessible doors, barriers, and penetration seals are functional.

03.04 Assess the thoroughness of the licensee's walkdowns and inspections of important equipment needed to mitigate fire and flood events to identify the potential that the equipment's function could be lost during seismic events possible for the site. Assess the licensee's development of any new mitigating strategies for identified vulnerabilities (e.g., entered it in to the corrective action program and any immediate

actions taken). As a minimum, the licensee should have performed walkdowns and inspections of important equipment (permanent and temporary) such as storage tanks, plant water intake structures, and fire and flood response equipment; and developed mitigating strategies to cope with the loss of that important function. Use IP 71111.21, "Component Design Basis Inspection," Appendix 3, "Component Walkdown Considerations," as a guideline to assess the thoroughness of the licensee's walkdowns and inspections.

2515/183-04 REPORTING REQUIREMENTS

The inspection results, including both observations and findings, of this TI should be in a stand-alone report. NOTE: This TI will be updated with a template which will provide specific guidance on reporting and documenting observations and findings.

The inspection report containing the results should be forwarded to NRR/DIRS/IRIB, Attention: Tim Kobetz via e-mail at timothy.kobetz@nrc.gov. Mr. Kobetz can also be reached at (301) 415-1932. The inspection results from this TI will be used to evaluate industry's readiness for a similar event and to aid in determining whether additional NRC regulatory actions are warranted.

2515/183-05 COMPLETION SCHEDULE

This TI is to be initiated upon issuance. Inspection activities are to be completed by April 29, 2011 and the inspection report issued by May 13, 2011.

2515/183-06 EXPIRATION

The TI will expire on June 30, 2012.

2515/183-07 CONTACT

Any technical questions regarding this TI should be addressed to Tim Kobetz at 301-415-1932 or timothy.kobetz@nrc.gov.

2515/183-08 STATISTICAL DATA REPORTING

All direct inspection effort expended on this TI is to be charged to 2515/183 with an IPE code of TI. All indirect inspection effort expended on this TI for preparation and documentation should be attributed to activity codes TIP and TID respectively.

2515/183-9 RESOURCE ESTIMATE

The estimated average time to complete the TI inspection requirements is 40 hours per site. Where applicable, inspectors should credit the baseline inspection program for samples reviewed during this TI assessment.

2515/183-10 TRAINING

No additional training is required.

END

ATTACHMENT 1

Revision History for TI 2515/183
FOLLOWUP TO FUKUSHIMA DAIICHI NUCLEAR STATION FUEL DAMAGE EVENT

Commitment Tracking Number	Issue Date	Description of Change	Training Needed	Training Completion Date	Comment Resolution Accession Number
N/A	ML11077A007 03/23/11	Researched commitments for 4 years and found none. This is a new document issued for inspections related to the industry response to the Fukushima Daiichi Nuclear Station Fuel Damage Event.	No	N/A	N/A

Merzke, Daniel

From: Merzke, Daniel
Sent: Tuesday, April 12, 2011 11:03 AM
To: Hipschman, Thomas; Castleman, Patrick; Snodderly, Michael; Orders, William; Franovich, Mike
Subject: FW:
Attachments: ML11077A007.pdf

Attached is the TI issued for inspection of NRC licensees as a result of the event at Fukushima Daiichi. Subsequent inspection requirements will be determined by the lessons learned task force.

Dan

From: Kobetz, Timothy
Sent: Tuesday, April 12, 2011 10:57 AM
To: Merzke, Daniel
Subject:

Per your request.

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TEMPORARY INSTRUCTION 2515/183

FOLLOWUP TO THE FUKUSHIMA DAIICHI NUCLEAR STATION FUEL DAMAGE EVENT

CORNERSTONE: INITIATING EVENTS AND MITIGATING SYSTEMS

APPLICABILITY: This Temporary Instruction (TI) applies to all holders of operating licenses for nuclear power reactors, except plants which have permanently ceased operations.

2515/183-01 OBJECTIVES

The objective of this TI is to independently assess the adequacy of actions taken by licensees in response to the Fukushima Daiichi nuclear station fuel damage event. The inspection results from this TI will be used to evaluate the industry's readiness for a similar event and to aid in determining whether additional regulatory actions by the U.S. Nuclear Regulatory Commission are warranted. Therefore, the intent of this TI is to be a high-level look at the industry's preparedness for events that may exceed the design basis for a plant. If necessary, a more specific followup inspection will be performed at a later date.

2515/183-02 BACKGROUND

On March 11, 2011, the Tohoku-Taiheiyou-Oki Earthquake occurred near the east coast of Honshu, Japan. This magnitude 9.0 earthquake and the subsequent tsunami caused significant damage to at least four of the six units of the Fukushima Daiichi nuclear power station as the result of a sustained loss of both the offsite and on-site power systems. Efforts to restore power to emergency equipment have been hampered or impeded by damage to the surrounding areas due to the tsunami and earthquake. The following background information is current as of March 18, 2011.

Units 1 through 3, which had been operating at the time of the earthquake, scrammed automatically, inserting their neutron absorbing control rods to ensure immediate shutdown of the fission process. Following the loss of electric power to normal and emergency core cooling systems and the subsequent failure of back-up decay heat removal systems, water injection into the cores of all three reactors was compromised, and reactor water levels could not be maintained. Tokyo Electric Power Company (TEPCO), the operator of the plant, resorted to injecting sea water and boric acid into the reactor vessels of these three units, in an effort to cool the fuel and ensure the reactors remained shutdown. However, the fuel in the reactor cores became partially uncovered. Hydrogen gas built up in Units 1 and 3 as a result of exposed, overheated fuel reacting with water. Following gas venting from the primary containment to relieve

pressure, hydrogen explosions occurred in both units and damaged the secondary containments. It appears that primary containments for Units 1 and 3 remained functional, but the primary containment for Unit 2 may have been damaged. TEPCO cut a hole in the side of the Unit 2 secondary containment to prevent hydrogen buildup following a sustained period when there was no water injection into the core.

In addition, problems were encountered with monitoring and maintaining Units 3 and 4 spent fuel pool (SFP) water levels. Efforts continue to supply seawater to the SFPs for Units 1 through 4 using various methods. At this time, the integrity of the SFPs for Units 3 and 4 is unknown.

Fukushima Daiichi Units 4 through 6 were shutdown for refueling outages at the time of the earthquake. The fuel assemblies for Unit 4 had been offloaded from the reactor core to the SFP. The SFPs for Units 5 and 6 appear to be intact.

The damage to Fukushima Daiichi nuclear power station appears to have been caused by initiating events that may have exceeded the design basis for the facilities.

2515/183-03 INSPECTION REQUIREMENTS AND GUIDANCE

NRC inspection staff should assess the licensee's activities and actions to assess its readiness to respond to an event similar to the Fukushima Daiichi nuclear plant fuel damage event. These inspections should occur at the operating power reactor facilities. Licensee emergency preparedness will not be assessed by this TI.

This TI may be completed all at once or in phases as the licensee verifies its capability to respond to such an event. The inspector(s) should coordinate the inspection effort with the licensee in accordance with the licensee's verification schedule.

The events at the Fukushima Daiichi plant appear to be caused by factors directly impacting nuclear safety that may have exceeded the design basis for the facility. While details on the full extent of damage to these units remain unknown, the damage poses a significant challenge to the nuclear safety of these units. Immediate actions by the U.S. industry are appropriate to assess and take corrective actions to address potential vulnerabilities that would challenge response to events that are beyond site design bases.

03.01 Assess the licensee's capability to mitigate conditions that result from beyond design basis events, typically bounded by security threats, committed to as part of NRC Security Order Section B.5.b issued February 25, 2002, and severe accident management guidelines and as required by Title 10 of the Code of Federal Regulations (10 CFR) 50.54(hh). Use Inspection Procedure (IP) 71111.05T, "Fire Protection (Triennial)," Section 02.03 and 03.03 as a guideline. If IP 71111.05T was recently performed at the facility the inspector should review the inspection results and findings to identify any other potential areas of inspection. Particular emphasis should be placed on strategies related to the spent fuel pool. The inspection should include, but not be limited to, an assessment of any licensee actions to:

a. Verify through test or inspection that equipment is available and functional. Active equipment shall be tested and passive equipment shall be walked down and inspected. It is not expected that permanently installed equipment that is tested under an existing regulatory testing program be retested.

b. Verify through walkdowns or demonstration that procedures to implement the strategies associated with B.5.b and 10 CFR 50.54(hh) are in place and are executable. Licensees may choose not to connect or operate permanently installed equipment during this verification.

c. Verify the training and qualifications of operators and the support staff needed to implement the procedures and work instructions are current for activities related to Security Order Section B.5.b and severe accident management guidelines as required by 10 CFR 50.54 (hh).

d. Verify that any applicable agreements and contracts are in place and are capable of meeting the conditions needed to mitigate the consequences of these events.

e. Review any open corrective action documents to identify vulnerabilities that may not have yet been addressed.

03.02 Assess the licensee's capability to mitigate station blackout (SBO) conditions, as required by 10 CFR 50.63, "Loss of All Alternating Current Power," and station design, is functional and valid. Refer to TI 2515/120, "Inspection of Implementation of Station Blackout Rule Multi-Plant Action Item A-22" as a guideline. It is not intended that TI 2515/120 be completely reinspected. The inspection should include, but not be limited to, an assessment of any licensee actions to:

a. Verify through walkdowns and inspection that all required materials are adequate and properly staged, tested, and maintained.

b. Demonstrate through walkdowns that procedures for response to an SBO are executable.

03.03 Assess the licensee's capability to mitigate internal and external flooding events required by station design. Refer to IP 71111.01, "Adverse Weather Protection," Section 02.04, "Evaluate Readiness to Cope with External Flooding" as a guideline. The inspection should include, but not be limited to, an assessment of any licensee actions to verify through walkdowns and inspections that all required materials and equipment are adequate and properly staged. These walkdowns and inspections shall include verification that accessible doors, barriers, and penetration seals are functional.

03.04 Assess the thoroughness of the licensee's walkdowns and inspections of important equipment needed to mitigate fire and flood events to identify the potential that the equipment's function could be lost during seismic events possible for the site. Assess the licensee's development of any new mitigating strategies for identified vulnerabilities (e.g., entered it in to the corrective action program and any immediate

actions taken). As a minimum, the licensee should have performed walkdowns and inspections of important equipment (permanent and temporary) such as storage tanks, plant water intake structures, and fire and flood response equipment; and developed mitigating strategies to cope with the loss of that important function. Use IP 71111.21, "Component Design Basis Inspection," Appendix 3, "Component Walkdown Considerations," as a guideline to assess the thoroughness of the licensee's walkdowns and inspections.

2515/183-04 REPORTING REQUIREMENTS

The inspection results, including both observations and findings, of this TI should be in a stand-alone report. NOTE: This TI will be updated with a template which will provide specific guidance on reporting and documenting observations and findings.

The inspection report containing the results should be forwarded to NRR/DIRS/IRIB, Attention: Tim Kobetz via e-mail at timothy.kobetz@nrc.gov. Mr. Kobetz can also be reached at (301) 415-1932. The inspection results from this TI will be used to evaluate industry's readiness for a similar event and to aid in determining whether additional NRC regulatory actions are warranted.

2515/183-05 COMPLETION SCHEDULE

This TI is to be initiated upon issuance. Inspection activities are to be completed by April 29, 2011 and the inspection report issued by May 13, 2011.

2515/183-06 EXPIRATION

The TI will expire on June 30, 2012.

2515/183-07 CONTACT

Any technical questions regarding this TI should be addressed to Tim Kobetz at 301-415-1932 or timothy.kobetz@nrc.gov.

2515/183-08 STATISTICAL DATA REPORTING

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2515/183-9 RESOURCE ESTIMATE

The estimated average time to complete the TI inspection requirements is 40 hours per site. Where applicable, inspectors should credit the baseline inspection program for samples reviewed during this TI assessment.

2515/183-10 TRAINING

No additional training is required.

END

ATTACHMENT 1

Revision History for TI 2515/183
FOLLOWUP TO FUKUSHIMA DAIICHI NUCLEAR STATION FUEL DAMAGE EVENT

Commitment Tracking Number	Issue Date	Description of Change	Training Needed	Training Completion Date	Comment Resolution Accession Number
N/A	ML11077A007 03/23/11	Researched commitments for 4 years and found none. This is a new document issued for inspections related to the industry response to the Fukushima Daiichi Nuclear Station Fuel Damage Event.	No	N/A	N/A

From: Flory, Shirley
To: Rini, Brett; Ibarra, Jose; Rivera-Lugo, Richard; Eisenberg, Wendy; Sangimino, Donna-Marie; Dehn, Jeff; Case, Michael; Coe, Doug; Correia, Richard; Gibson, Kathy; Richards, Stuart; Scott, Michael; Sheron, Brian; Uhle, Jennifer; Valentin, Andrea
Subject: FW: Additional Information on a Lapse of Appropriation
Date: Friday, April 08, 2011 8:58:50 AM
Attachments: Office Report.pdf

From: Taylor, Renee
Sent: Friday, April 08, 2011 8:53 AM
To: Ellmers, Glenn; Ash, Darren; Boger, Bruce; Boyce, Thomas (OIS); Brenner, Eliot; Brown, Milton; Burns, Stephen; Carpenter, Cynthia; Casto, Chuck; Cohen, Miriam; Collins, Elmo; Dapas, Marc; Dean, Bill; Doane, Margaret; Droggitis, Spiros; Dyer, Jim; Greene, Kathryn; Grobe, Jack; Hackett, Edwin; Haney, Catherine; Hayden, Elizabeth; Holahan, Gary; Howard, Patrick; Johnson, Michael; Kelley, Corenthis; Leeds, Eric; Mamish, Nader; McCrary, Cheryl; McCree, Victor; Miller, Charles; Moore, Scott; Pederson, Cynthia; Plisco, Loren; Poole, Brooke; Powell, Amy; Reyes, Luis; Satorius, Mark; Schaeffer, James; Schmidt, Rebecca; Sheron, Brian; Stewart, Sharon; Uhle, Jennifer; Virgilio, Martin; Weber, Michael; Wiggins, Jim; Williams, Barbara; Zimmerman, Roy; Campbell, Andy; Holahan, Patricia; Dorman, Dan; Muessle, Mary; Wert, Leonard; Tracy, Glenn; Taylor, Renee; Krupnick, David; Evans, Michele
Cc: Akstulewicz, Brenda; Andersen, James; Bellosi, Susan; Belmore, Nancy; Boyd, Lena; Buckley, Patricia; Casby, Marcia; Cianci, Sandra; Crawford, Carrie; Flory, Shirley; Garland, Stephanie; Higginbotham, Tina; Hudson, Sharon; Landau, Mindy; Matakas, Gina; Miles, Patricia; Pulley, Deborah; Rihm, Roger; Riner, Janet; Ronewicz, Lynn; Ross, Robin; Salus, Amy; Tannenbaum, Anita; Taylor, Renee; Thomas, Loretta; Walker, Dwight; Warner, MaryAnn; Wright, Darlene; Wyatt, Melissa; Cannady, Ashley; Lockhart, Denise; Perez-Ortiz, Aracelis; Riddick, Nicole; King, Shannon; Penny, Melissa; Sprogeris, Patricia; Banks, Eleasah; Nagel, Cheri; Hasan, Nasreen; Call, Michel; Thaggard, Mark; Young, Gary; Moore, Mary; Daniels, Stanley; Kreuter, Jane; Schumann, Stacy; Rihm, Roger; Schwarz, Sherry; Boyer, Rachel
Subject: Additional Information on a Lapse of Appropriation

For OD's and RA's

Since the Congress has not yet reached alignment on FY 2011 funding, we need to continue to take actions towards a lapse in appropriation. As you have heard from the Chairman and the EDO, the NRC will utilize carryover to continue to run at full staff for at least the next week. OCFO will monitor funds usage daily, and we will let you know (likely within a few days) if and when it will become necessary to furlough non-excepted staff.

In order to maximize the time of full NRC staffing, we ask that you cancel immediately all conference and travel related training for next week. (All TTC classes will be cancelled until FY 2011 funding is restored. Regional classes and PDC will continue but people should not travel to attend them.) We also ask that you cancel all non-critical travel (including rotations) and limit attendance at events where we are not mandated or it is not critical for us to attend. For your convenience, I have attached planned international travel from OIP and OCFO will send out general travel as well today. If the travel begins later next week or the following week, the office and traveler can wait to cancel until they would incur a hotel penalty which will vary based on the entity. We will also be sending an EDO announcement to staff with this information.

We also ask that you only use overtime during a lapse in funding for mission critical work, and that with the exception of exceptional needs that you not obligate any additional funds on purchase

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cards, lab agreements after Friday, April 8th until FY 2011 funding is restored.

In the event of a lapse in appropriations, we will be sending you your office "mark" for excepted staff on Monday. This will be in addition to excepted staff that for operations center shift work, Japan site work, and the Japan Task force work which will be provided separately. (Operations center work will move to dedicated shifts as of April 18th.) We will likely need your lists by Tuesday. One thing to remember is that excepted staff should be available for work and not on leave for at least a 3-4 few weeks of a shutdown.

HR and ADM will be reaching out to you with additional guidance and information as required. Please continue to refer staff to our EDO update and to the EDO sharepoint site for information.
http://portal.nrc.gov/edo/staff/Lists/Announcements/DispForm.aspx?ID=16&Source=http%3A%2F%2Fportal%2Enrc%2Egov%2Fedo%2Fstaff%2Fdefault%2Easpx

Thank you for all of your efforts and your flexibility during these difficult times.
Mary

Mary Muessle
Assistant for Operations - Acting
Office of the Executive Director for Operations
U.S. Nuclear Regulatory Commission
301-415-1703 office
301-415-2700 fax

NRC International Travel By Office From 4/4/2011 To 5/13/2011

Office	LastName	FirstName	From	To	Country	Purpose
Contractor						
	Akin	Lili	4/18/2011	4/23/2011	France	NRC has been invited to participate in WAIAGE Expert Meeting on "Post-Tensioning Methodologies for containment building: greased or cement grouted tendons-consequences on monitoring, periodic testing and modeling activities." SNL and the NRC will be leading discussions in monitoring and testing, and participating in the modeling activity discussions. This work is of particular interest to the NRC because of new plant applications requesting use of said tendons
	Chu	Tsong-Lun	4/16/2011	4/23/2011	South Korea	Kick off meeting for NRC/Brookhaven National Laboratory/Korea Atomic Energy Research Institute (KAERI) collaboration on quantitative softward relability methods. Traveler will meet with KAERI staff to establish logistics for the collaboration and to learn about previous related work performed by KAERI
	Humphries	Larry	4/9/2011	4/13/2011	Italy	Attend 3rd European MELCOR User Group (EMUG) meeting and give presentations on code development status
	Phillips	Jesse	4/9/2011	4/13/2011	Italy	Attend 3rd European MELCOR User Group (EMUG) meeting and give presentations on SNAP
	Pickett	David	5/9/2011	5/14/2011	Cyprus	PLACEHOLDER PENDING TRAVEL COMING TO OIP
	Prelewicz	Daniel	4/25/2011	4/30/2011	Argentina	Attend the Spring 2011 CAMP Meeting in San Carlos de Bariloche, Argentina, and present a paper on "Status of the RELAPS Code and User Problems"
FSME						
	Gnugnoli	Giorgio	5/7/2011	5/15/2011	Austria	To serve on the U.S. delegation to the 4th Organizational Meeting of the Joint Convention Contracting Parties. The traveler will perform technical reviews, support U.S.G. senior executives, and represent U.S. interests at the Organizational Meeting
	McConnell	Keith	4/15/2011	4/20/2011	Norway	Consultancy on three year work plan for "Regulatory Supervision of Legacy Sites". Meeting sponsored by Norwegian Radiation Protection Authority
	Miller	Charles	4/23/2011	4/30/2011	France	To attend the 122nd Session of the Nuclear Energy Agency (NEA) Steering Committee as a member of the U.S. Delegation
	Orlando	Dominick	5/2/2011	5/7/2011	Austria	Participate in IAEA meeting to complete work on a project to develop safety assessment guidance for the decommissioning of nuclear facilities

Office	LastName	FirstName	From	To	Country	Purpose
NMSS						
	Fedors	Randall	4/9/2011	4/15/2011	Finland	Participate in Development of Coupled Models and their Validation against Experiments (DECOVALEX) Workshop research progress on tasks and planning for next phase. DECOVALEX is an international collaboration focused on deep geological disposal environment. Also meet with representatives of the Finnish regulator STUK to discuss their prep for a license application and discuss waste disposal program regulatory and technical issues in general
	Gonzalez	Hipolito	4/24/2011	4/30/2011	Austria	Participate in IAEA technical and consultancy meeting on very long term storage of used nuclear fuel. The purpose of the meeting is to share and discuss information relevant to successful implementation of VLLTS, and to provide an opportunity for mutually beneficial collaboration. The meeting will also identify needs and interests of member states relevant to VLTS, and familiarize participants with areas that need to be addressed for successful implementation of VLTS. The consultancy group will collect information provided and TecDoc on VLTS
	Pham	Thomas	4/24/2011	4/30/2011	Austria	Participate in a technical meeting organized by the IAEA on developing a guidance on nuclear material control and accounting (MC&A). The travel is the NRC rep and matter expert for the MC&A and invited US expert in this IAEA consultancy and technical meeting
	Rubenstone	James	4/10/2011	4/15/2011	Finland	Participate in Development of Coupled Models and their Validation against Experiments (DECOVALEX) Workshop research progress on tasks and planning for the next DECOVALEX phase. DECOVALEX is an international collaboration focused on deep geological disposal environment. Also meet with representatives of both the Chinese implementer and oversight groups for their waste disposal program.
	Smith	Shawn	4/4/2011	4/8/2011	France	Attend the Reversibility and Retrievability (R&R) Working Group Meeting

Office	LastName	FirstName	From	To	Country	Purpose
NRO						
	Akstulewicz	Frank	4/30/2011	5/11/2011	France	To participate at the ICAPP Conference and support plenary session and technical sessions. The presentation will (1) provide a brief overview of the NRC's licensing process for new reactors; (2) provide a discussion of the regulatory, technical & environmental issues that have created challenges for the NRC in completing its required licensing activities; (3) provide a status of staff's consideration of a graded approach to the review of small modular reactor design applications; and (4) provide status on the regulatory activities conducted in the US to oversee construction activities
	Chokshi	Nilesh	4/9/2011	4/16/2011	Austria	Attend IAEA Working Groups on Seismic Design and Qualifications of SSCs and Seismic Safety Evaluation and Seismic PRA
	Coffin	Stephanie	4/17/2011	4/21/2011	France	To participate in the meeting of the Infrastructure Development Working Group (IDWG) in support of International Framework for Nuclear Energy Cooperation (IFNEC)
	Dube	Donald	4/9/2011	4/16/2011	Austria	Participate in IAEA meeting on Safety Goals for Nuclear Installations. The objective is to provide an international forum for presentations and discussions on the current practices pertaining to the establishment and use of safety goals for nuclear installations
	Holahan	Gary	4/25/2011	5/6/2011	France	Participate in MDEP Steering Technical committee. And to participate in the International Congress on Advances in Power Plans (ICAPP)
	Magruder	Stewart	4/30/2011	5/9/2011	France	To participate at the ICAPP Conference and support plenary session and technical sessions. The presentation will (1) provide a brief overview of the NRC's licensing process for new reactors; (2) provide a discussion of the regulatory, technical & environmental issues that have created challenges for the NRC in completing its required licensing activities; (3) provide a status of staff's consideration of a graded approach to the review of small modular reactor design applications; and (4) provide status on the regulatory activities conducted in the US to oversee construction activities
	Magruder	Stewart	4/30/2011	5/9/2011	France	To participate at the ICAPP Conference and support plenary session and technical sessions. The presentation will (1) provide a brief overview of the NRC's licensing process for new reactors; (2) provide a discussion of the regulatory, technical & environmental issues that have created challenges for the NRC in completing its required licensing activities; (3) provide a status of staff's consideration of a graded approach to the review of small modular reactor design applications; and (4) provide status on the regulatory activities conducted in the US to oversee construction activities

Office	LastName	FirstName	From	To	Country	Purpose
NRO						
	Rasmussen	Richard	4/10/2011	4/15/2011	Canada	Primary purpose of the trip is to ensure that the NUPIC inspections remain as an acceptable alternative to the NRC's vendor inspection/audit program. The trip will also support the Multinational Design Evaluation Program (MDEP) by inviting a member of the Canadian Nuclear Regulator (CNSC) to observe the NRC's NUPIC oversight process
	Rivera-Varona	Aida	4/30/2011	5/6/2011	France	To participate at the ICAPP Conference and support plenary session and technical sessions. The presentation will (1) provide a brief overview of the NRC's licensing process for new reactors; (2) provide a discussion of the regulatory, technical & environmental issues that have created challenges for the NRC in completing its required licensing activities; (3) provide a status of staff's consideration of a graded approach to the review of small modular reactor design applications; and (4) provide status on the regulatory activities conducted in the US to oversee construction activities
	Smith	Stacy	4/10/2011	4/16/2011	Canada	Primary purpose of the trip is to ensure that the NUPIC inspections remain as an acceptable alternative to the NRC's vendor inspection/audit program. The trip will also support the Multinational Design Evaluation Program (MDEP) by inviting a member of the Canadian Nuclear Regulator (CNSC) to observe the NRC's NUPIC oversight process
	Tappert	John	4/30/2011	5/6/2011	France	To participate at the ICAPP Conference and support plenary session and technical sessions. The presentation will (1) provide a brief overview of the NRC's licensing process for new reactors; (2) provide a discussion of the regulatory, technical & environmental issues that have created challenges for the NRC in completing its required licensing activities; (3) provide a status of staff's consideration of a graded approach to the review of small modular reactor design applications; and (4) provide status on the regulatory activities conducted in the US to oversee construction activities
	Terao	David	4/16/2011	4/21/2011	France	Participate as the NRC representative in a meeting of MDEP's Codes and Standards Working Group
	Thadani	Ashok	4/9/2011	4/16/2011	Austria	Participate in IAEA meeting on Safety Goals for Nuclear Installations. The objective is to provide an international forum for presentations and discussions on the current practices pertaining to the establishment and use of safety goals for nuclear installations
	Williams	Donna	4/25/2011	5/6/2011	France	To participate in MDEP Steering Technical Committee and to participate in the International Congress on Advnces in Power Plans

Office	LastName	FirstName	From	To	Country	Purpose
NRR						
	Adams	John	4/30/2011	5/5/2011	France	Participate in the second NEA/CNRA scoping Task Group meeting concerning the safety of research reactors
	Eads	Johnny	5/6/2011	5/15/2011	Vietnam	Presenter and conference attendee for the IAEA regional workshop on radiation protection and safety for research reactors
	Hiser	Allen	4/12/2011	4/16/2011	Austria	Participate in IAEA International GALL activity as a member of the Clearing Group
	Karwoski	Kenneth	4/16/2011	4/30/2011	Japan	Participate in technical discussions on a variety of steam generator issues including inspection, repair, and analysis. Participants will promote nuclear safety by presenting the US regulatory perspective. Knowledge about nuclear safety will be transferred to other participating countries.
	Kobetz	Timothy	4/12/2011	4/20/2011	Austria	Annual WGIP working group meeting preparing a document Best Inspection Commendable and to participate/facilitate in workshops
	Murphy	Emmett	4/16/2011	4/24/2011	Japan	Participate in technical discussions on a variety of steam generator issues including inspection, repair, and analysis. Participants will promote nuclear safety by presenting the U.S. regulatory perspective. Knowledge about nuclear safety will be transferred to other participating countries.
	Nguyen	John	5/6/2011	5/15/2011	Vietnam	Presenter and conference attendee for IAEA regional workshop on radiation protection and safety for research reactors.
NSIR						
	Jones	Cynthia	4/9/2011	4/17/2011	Austria	Invited to revise the training materials for Train-the-Trainers course and IAEA staff for course on the Security of Radioactive Sources and Nuclear Security Recommendations on Radioactive Materials and associated facilities. Traveler will provide technical expertise and include revisions recommended by NRC in the conduct of previous courses for IAEA on Security and Radioactive Sources
	Milligan	Patricia	4/10/2011	4/15/2011	France	Peer review of ESN emergency planning document, "Transition From the Urgent Phase"
OIP						
	Stahl	Eric	5/2/2011	5/6/2011	France	To Accompany Commissioner Apostolakis to the 2011 International Congress of Advance in Nuclear Power Plants and to Nuclear Site Visits in Southern France.

Office	LastName	FirstName	From	To	Country	Purpose
RES						
	Burke	John	5/7/2011	5/13/2011	France	Participate in the update of the CSNI Knowledge Base Report on ECCS - suction strainer clogging
	Calvo	Antony	4/24/2011	4/30/2011	Argentina	Will conduct the Spring CAMP meeting and make technical presentations about the current status of the NRC computer codes that are available through the CAMP Program
	Esmaili	Hossein	4/9/2011	4/13/2011	Italy	Attend the 3rd European MELCOR User Group Meeting (EMUG) at ENEA in Bologna, Italy
	Graves	Herman	4/18/2011	4/24/2011	France	Participate in the OECD/NEA International Workshop on Post-tensioning Methods used in Concrete Containments." Traveler serves as the US repr to the OECD/NEA concrete task group and will discuss US design philosophy and summarize US in-service experience for plants licensed by the NRC. In addition, Mr. Graves is the lead project manager for several regulatory guides in this area
	Hoxie	Christopher	4/24/2011	4/30/2011	Argentina	Will conduct the Spring CAMP meeting and make technical presentations about the current status of the NRC computer codes that are available through the CAMP Program
	Marshall	Shawn	4/12/2011	3/16/2011	Hungary	Traveler will view and discuss the results from the recently completed test investigating upper-head voiding, participate in planning discussions for the upcoming ROSA/PKL counterpart test, and discuss closeout activities for the PKL2 Program
	Mulheim	Michael	4/16/2011	4/23/2011	Austria	Meet with the IAEA on new guidance on I&C systems at Research and Test Reactors.
	Ramadan	Liliana	5/8/2011	5/13/2011	France	Participate in the final CSNI DIDELSYS workshop
	Raynaud	Patrick	4/7/2011	4/15/2011	France	Participate in (1) Technical Advisory Group discussion re test conditions and parameters to be used in anticipated CABRI Water Loop experiments; (2) on-site review of CABRI instrumentation capability prior to completion of reactor modifications; (3) in bilateral discussions with IRSN on CSNI activity on "Mechanical Testing of Fuel Cladding" and (4) provide non-financial support (i.e., encouragement) to IRSN in their efforts to continue experimental work at the CABRI facility
	Richards	Stuart	4/24/2011	4/28/2011	France	Attend CSNI Program Review Group Meeting
	Velazquez-Lozada	Alexander	4/9/2011	4/15/2011	Sweden	Participate and present TRACE results in the 2nd workshop of the OECD/NRC PSBT Benchmark to be held on KTH Stockholm, Sweden

Office	LastName	FirstName	From	To	Country	Purpose
RES						
	Voglewede	John	5/11/2011	5/13/2011	Hungary	Participate in Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA) Working Group on Fuel Safety. Participate in OECD/Halden meeting on proposed fuel safety experimental program. Attend OECD/Halden workshop on VVER fuel behavior in Halden Reactor Project experiments
	Whitman	Josh	4/24/2011	4/30/2011	Argentina	Travelers will conduct the Spring CAMP meeting and make technical presentations about the current status of the NRC computer codes that are available through the CAMP Program
RI						
	Bellamy	Ronald	5/7/2011	4/14/2011	China	IAEA Expert Mission to Beijing, China.
TTC						
	Miller	Mark	4/12/2011	4/21/2011	Austria	IAEA

From:	Rini, Brett
To:	Case, Michael; Richards, Stuart; Correia, Richard; Coe, Doug; Gibson, Kathy; Scott, Michael; Valentin, Andrea
Cc:	Sheron, Brian; Uhle, Jennifer
Subject:	List of Issues and Research Areas from Japanese Event
Date:	Monday, April 11, 2011 5:27:35 PM
Attachments:	Potential Long term Issues Rev1.docx

Division Directors,

Please find attached a list of possible issues and research areas to follow-up on as a result of the Japanese earthquake. I compiled the input I received from your divisions along with a document that Brian sent me and classified the recommendations into various areas (e.g., electrical, severe accidents, external events).

Please review the attached document and let me know if you have any additional thoughts or changes.

Thanks,

Brett

Brett A. Rini
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AAA/95

Attachment Potential Long term Issues Rev1_1.docx (33876 Bytes) cannot be converted to PDF format.

From:	Marksberry, Don
To:	RST01 Hoc; Tinkler, Charles; Schaperow, Jason; Esmaili, Hossein; Helton, Donald; Salay, Michael; Thorp, John; Garmon, David
Cc:	Lee, Richard; Demoss, Gary; Coyne, Kevin; Stutzke, Martin; Salley, MarkHenry; Siu, Nathan; Joy L Rempe; Correia, Richard; Coe, Doug
Subject:	Plant Status Chronologies of Units 1, 2, and 3
Date:	Thursday, April 14, 2011 9:50:36 AM
Attachments:	RES (4-14-2011) Fukushima Daiichi Chronology, Units 1,2,3.xlsx

Here is this morning's edition of the plant status chronology and data tables for Units 1, 2, and 3. It does not include radiological information.

Please note that the info sources are official press releases from TEPCO and NISA. No other sources or speculations were included.

NOTES:

4/14/11 The TEPCO reference column for Unit 2 was inadvertently shifted a line or two. I believe this has been corrected.

4/14/11 We are considering stopping the updates of the parameter spreadsheets, since other more current sources are becoming readily available. More to come....

4/14/11 INL-DOE has started a timeline for all 6 units at Fukushima Daiichi. We will assess whether our efforts can be combined with INL's to avoid duplication and conflicts within the Federal community. More to come....

4/13/11 Earlier event descriptions were revised to match the press release text a little closer. Also, select Article 15 reports and emergency declarations (from TEPCO to NISA) were added for those that may imply change in unit status.

Don

Don Marksberry

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AAA/96

Plant Parameters: **Unit 3**

Revision 4/14/2011 (as of 10:00 a.m. EDT)

Source: NISA Press Releases	Unit	3/14	3/15	3/16	3/17	3/17	3/18	3/18	3/18	3/19	3/19	3/19	3/20	3/20	3/20
		19:30	11:42	12:40	3:20	11:10	2:45	12:35	21:05	6:10	11:15	17:25	4:30	11:00	16:00
RPV Injection.....Water Source	Salt/Fresh														
via Feed Water Line	L/min														
via Fire Extinguishing Line	L/min														
via Fire Extinguishing Line	m3/h														
RPV Level															
Fuel Range A	mm	-1900	-1900	-1900	-1950	-1950	-2000	-2000	-1900	-1200	-1950	-1850	-1950	-1950	-1650
Fuel Range B	mm	-2300	-2300	-2300	-2300	-2300	-2300	-2300	-2300	-2300	-2300	-2300	-2300	2350	-2000
~~RRV Pressure (see note)~~		~~Note: 1~~	~~Note: 1~~	~~Note: 1~~	~~Note: 1~~	~~Note: 1~~									
See NISA Press Release dated 4/11/2011															
~~Channel A~~	~~MPa g~~	~~0.183~~	~~0.244~~	~~0.059~~	~~0.023~~	~~0.014~~	~~-0.005~~	~~-0.018~~	~~-0.016~~	~~0.045~~					
~~Channel B~~	~~MPa g~~	~~0.183~~	~~0.244~~	~~0.065~~	~~0.032~~	~~0.023~~	~~0.014~~	~~0.000~~	~~0.016~~	~~0.005~~	~~-0.023~~	~~0.086~~	~~0.216~~	~~0.189~~	~~0.162~~
~~Channel C~~	~~MPa g~~										~~-0.023~~	~~-0.050~~	~~0.180~~	~~0.149~~	~~0.119~~
~~Channel A~~	~~MPa abs~~	~~0.284~~	~~0.345~~	~~0.160~~	~~0.124~~	~~0.115~~	~~0.096~~	~~0.083~~	~~0.085~~	~~0.146~~					
~~Channel B~~	~~MPa abs~~	~~0.284~~	~~0.345~~	~~0.166~~	~~0.133~~	~~0.124~~	~~0.115~~	~~0.101~~	~~0.117~~	~~0.106~~	~~0.078~~	~~0.187~~	~~0.317~~	~~0.290~~	~~0.263~~
~~Channel C~~	~~MPa abs~~										~~0.078~~	~~0.051~~	~~0.281~~	~~0.250~~	~~0.220~~
Drywell Pressure	MPa abs	0.335	0.415	0.230	0.200	0.190	0.155	0.155	0.160	0.185	0.160	0.210	0.340	0.310	0.290
Suppression Chamber Pressure	MPa abs	0.500	dwn/scale	dwn/scale	dwn/scale	dwn/scale	dwn/scale	dwn/scale	dwn/scale	dwn/scale	dwn/scale	dwn/scale	Note 2	Note 3	Note 4
RPV Temperature		Note 1: Both readings are noted in the press releases as coming from channel A											Note 2: "Down scale 100"		
Feedwater Nozzle Temp	C												Note 3: "400 overscale"		
RPV Bottom Head Temp	C												Note 4: "800 down scale"		
Containment Atm Monitoring System															
Drywell	Sv/h														
Suppression Chamber	Sv/h														
Notes:															
NISA News releases started reported gage pressure, then converted readings to absolute pressure. NISA separate parameter tables report *gage pressure. Conversion in this table in BLUE.*															
Standard atmospheric pressure = 101.325 kPa = 0.101325 Mpa Absolute pressure = 0.101325 MPa + 0.06 MPa = 0.161325 Mpa															

Plant Parameters: **Unit 3**

Revision 4/14/2011 (as of 10:00 a.m. EDT)

Source: NISA Press Releases	Unit	3/21 4:00	3/21 12:15	3/21 14:55	3/22 5:30	3/22 10:35	3/23 4:00	3/23 9:10	3/24 2:40	3/24 10:20	3/24 18:00	3/25 6:10	3/25 16:10	3/25 18:02	3/25 20:05
RPV Injection.....Water Source	Salt/Fresh								Salt	Salt	Salt			Fresh	Fresh
via Feed Water Line	L/min														
via Fire Extinguishing Line	L/min								Note 8	Note 8	Note 8			240	240
via Fire Extinguishing Line	m3/h														Note 7
RPV Level															
Fuel Range A	mm	-1650	-1600	-1550	-1575	-1575	-1900	-1800	-1800	-1900	-1850	-1900	-1900		
Fuel Range B	mm	-1950	-2000	-2025	-2350	-2350	-2300	-2300	-2300	-2300	-2300	-2300	-2300		
~~RRV Pressure (see note)~~															
See NISA Press Release dated 4/11/2011															
~~Channel A~~	~~MPa g~~	~~0.214~~	~~0.043~~	~~0.045~~	~~0.038~~	~~0.036~~	~~0.036~~	~~0.034~~	~~0.041~~	~~0.036~~	~~0.038~~	~~0.038~~	~~0.036~~		
~~Channel B~~	~~MPa g~~														
~~Channel C~~	~~MPa g~~	~~-0.027~~	~~-0.083~~	~~-0.088~~	~~-0.101~~	~~-0.101~~	~~-0.101~~	~~-0.104~~	~~-0.097~~	~~-0.099~~	~~-0.101~~	~~-0.097~~	~~-0.099~~		
~~Channel A~~	~~MPa abs~~	~~0.315~~	~~0.144~~	~~0.146~~	~~0.139~~	~~0.137~~	~~0.137~~	~~0.135~~	~~0.142~~	~~0.137~~	~~0.139~~	~~0.139~~	~~0.137~~		
~~Channel B~~	~~MPa abs~~														
~~Channel C~~	~~MPa abs~~	~~0.074~~	~~0.018~~	~~0.013~~	~~0.000~~	~~0.000~~	~~0.000~~	~~-0.003~~	~~0.004~~	~~0.002~~	~~0.000~~	~~0.004~~	~~0.002~~		
Drywell Pressure	MPa abs	0.160	0.120	0.110	0.100	0.100	0.100	0.100	[illegible]	0.107	0.107	0.1074	0.1075		
Suppression Chamber Pressure	MPa abs	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.199	0.200	0.1937	0.1895		
RPV Temperature									Note 5,6	Note 6	Note 6	Note 6	Note 6		
Feedwater Nozzle Temp	C								80.7	14.1	65.6	42.8	-33.4		
RPV Bottom Head Temp	C								185.4	185.5	155.7	111.6	111.0		
Containment Atm Monitoring System															
Drywell	Sv/h								57.9	55.9	53.3	51.0	38.8		
Suppression Chamber	Sv/h								1.66	1.62	1.45	1.5	1.31		

Notes:

NISA News releases started reported gage pressure, then converted readings to absolute pressure. NISA separate parameter tables report gage pressure. Conversion in this table in BLUE.

Standard atmospheric pressure = 101.325 kPa = 0.101325 Mpa
Absolute pressure = 0.101325 MPa + 0.06 MPa = 0.161325 Mpa

Note 5: Time of temperatures 2:20
Note 6: Feedwater nozzle temperature reading under survey
Note 7: 240 to 250 l/hr
Note 8: "measurement instrument malfunction"

Plant Parameters: **Unit 3**															
Revision 4/14/2011 (as of 10:00 a.m. EDT)															
Source: NISA Press Releases	Unit	3/26	3/26	3/26	3/26	3/26	3/27	3/27	3/27	3/28	3/28	3/28	3/29	3/29	3/29
		5:05	7:00	10:00	11:15	18:00	4:45	10:10	14:00	5:30	10:30	20:32	4:45	12:00	14:39
RPV Injection.....Water Source	Salt/Fresh		Fresh			Fresh			Fresh			Fresh			Fresh
via Feed Water Line	L/min														
via Fire Extinguishing Line	L/min		230			220			210			200			116
via Fire Extinguishing Line	m3/h														
RPV Level															
Fuel Range A	mm	-1850		-1800	-1850		-1850	-1900		-1900	-1850		-1900	-1850	
Fuel Range B	mm	-2300		-2300	-2300		-2300	-2300		-2300	-2250		-2300	-2250	
~~RRV Pressure (see note)~~															
See NISA Press Release dated 4/11/2011															
~~Channel A~~	~~MPa g~~	~~0.038~~		~~0.038~~	~~0.038~~		~~0.029~~	~~0.032~~		~~0.029~~	~~0.025~~		~~0.034~~	~~0.029~~	
~~Channel B~~	~~MPa g~~														
~~Channel C~~	~~MPa g~~	~~-0.101~~		~~-0.101~~	~~-0.101~~		~~-0.099~~	~~-0.099~~		~~-0.090~~	~~-0.095~~		~~-0.090~~	~~-0.095~~	
~~Channel A~~	~~MPa abs~~	~~0.139~~		~~0.139~~	~~0.139~~		~~0.130~~	~~0.133~~		~~0.130~~	~~0.126~~		~~0.135~~	~~0.130~~	
~~Channel B~~	~~MPa abs~~														
~~Channel C~~	~~MPa abs~~	~~0.000~~		~~0.000~~	~~0.000~~		~~0.002~~	~~0.002~~		~~0.011~~	~~0.006~~		~~0.011~~	~~0.006~~	
Drywell Pressure	MPa abs	0.1069		0.1066	0.1068		0.1073	0.1076		0.1083	0.1076		0.1085	0.1075	
Suppression Chamber Pressure	MPa abs	0.1848		0.1839	0.1836		0.1804	0.1806		0.1794	0.1792		0.1792	0.1796	
RPV Temperature		Note 6		Note 6	Note 6		Note 6	Note 6		Note 6			Note 6	Note 6	
Feedwater Nozzle Temp	C	26.1		37.6	33.7		1 /	13.6		38			61.5	62.2	
RPV Bottom Head Temp	C	102.5		106.1	100.4		106.8	121.6		123.0			120.9	121.1	
Containment Atm Monitoring System															
Drywell	Sv/h	36.6		36.1	36.1		34.2	33.7		31.5			29.2	28.6	
Suppression Chamber	Sv/h	1.41		1.4	1.4		1.34	1.31		1.24			1.18	1.16	
Notes:															
NISA News releases started reported gage pressure, then converted readings to absolute pressure. NISA separate parameter tables report gage pressure. Conversion in this table in BLUE.															
Standard atmospheric pressure = 101.325 kPa = 0.101325 Mpa Absolute pressure = 0.101325 MPa + 0.06 MPa = 0.161325 Mpa															

Plant Parameters: **Unit 3**															
Revision 4/14/2011 (as of 10:00 a.m. EDT)															
Source: NISA Press Releases	Unit	3/30	3/30	3/31	3/31	4/1	4/1	4/2	4/2	4/3	4/3	4/3	4/3	4/4	4/4
		3:50	13:30	4:40	9:40	5:45	11:45	1:30	12:10	2:30	10:30	12:18	17:32	3:30	9:30
RPV Injection.....Water Source	Salt/Fresh											Fresh	Fresh		
via Feed Water Line	L/min														
via Fire Extinguishing Line	L/min														
via Fire Extinguishing Line	m3/h											8	7		
RPV Level															
Fuel Range A	mm	-1850	-1850	-1900	-1850	-1900	-1900	-1850	-1850	-1850	-1850			-1800	-1800
Fuel Range B	mm	-2250	-2250	-2300	-2250	-2250	-2250	-2250	-2250	-2250	-2250			-2250	-2250
~~RRV Pressure (see note)~~															
See NISA Press Release dated 4/11/2011															
~~Channel A~~	~~MPa g~~	~~0.023~~	~~0.018~~	~~0.020~~	~~0.016~~	~~0.018~~	~~0.016~~	~~0.025~~	~~0.014~~	~~0.011~~	~~0.011~~			~~0.007~~	~~0.005~~
~~Channel B~~	~~MPa g~~														
~~Channel C~~	~~MPa g~~	~~-0.092~~	~~-0.095~~	~~-0.09~~	~~-0.092~~	~~-0.086~~	~~-0.088~~	~~-0.086~~	~~-0.095~~	~~-0.088~~	~~-0.083~~			~~-0.081~~	~~-0.083~~
~~Channel A~~	~~MPa abs~~	~~0.124~~	~~0.119~~	~~0.121~~	~~0.117~~	~~0.119~~	~~0.117~~	~~0.126~~	~~0.115~~	~~0.112~~	~~0.112~~			~~0.108~~	~~0.106~~
~~Channel B~~	~~MPa abs~~														
~~Channel C~~	~~MPa abs~~	~~0.009~~	~~0.006~~	~~0.011~~	~~0.009~~	~~0.015~~	~~0.013~~	~~0.015~~	~~0.006~~	~~0.013~~	~~0.018~~			~~0.020~~	~~0.018~~
Drywell Pressure	MPa abs	0.1071	0.1064	0.1073	0.1066	0.1071	0.1068	0.1055	0.1050	0.1061	0.1062			0.1073	0.1069
Suppression Chamber Pressure	MPa abs	0.1780	0.1775	0.1764	0.1755	0.1755	0.1757	0.1748	0.1748	0.1748	0.1750			0.1754	0.1757
RPV Temperature			Note 6	Note 6	Note 6	Note 6	Note 6	Note 6	Note 6	Note 6	Note 6			Note 6	Note 6
Feedwater Nozzle Temp	C	75.3	76.7	88.5	88.6	92.6	90.2	90.8	92.3	90.6	90.6			89.8	90.0
RPV Bottom Head Temp	C	116	115.4	114.2	114.4	116.3	117.8	119.4	117.8	115.9	114.3			114.1	113.7
Containment Atm Monitoring System															
Drywell	Sv/h	27.6	26.8	25.7	25.4	24.5	24.2	24.0	23.2	22.2	21.7			21.5	21.1
Suppression Chamber	Sv/h	1.11	1.09	1.05	1.04	1.00	0.98	0.955	0.935	0.911	0.897			0.871	0.864
Notes:															
NISA News releases started reported gage pressure, then converted readings to absolute pressure. NISA separate parameter tables report gage pressure. Conversion in this table in BLUE.															
Standard atmospheric pressure = 101.325 kPa = 0.101325 Mpa Absolute pressure = 0.101325 MPa + 0.06 MPa = 0.161325 Mpa															

Plant Parameters: **Unit 3**															
Revision 4/14/2011 (as of 10:00 a.m. EDT)															
Source: NISA Press Releases	Unit	4/5	4/5	4/6	4/6	4/7	4/7	4/8	4/8	4/9	4/9	4/10			
		5:40	10:20	0:00	12:30	6:00	12:00	1:30	12:00	0:00	12:25	6:00			
RPV Injection.....Water Source	Salt/Fresh														
via Feed Water Line	L/min														
via Fire Extinguishing Line	L/min														
via Fire Extinguishing Line	m3/h														
RPV Level															
Fuel Range A	mm	-1850	-1850	-1850	-1800	-1850	-1900	-2000	-1850	-1950	-1700	-1900			
Fuel Range B	mm	-2250	-2250	-2250	-2200	-2250	-2250	-2250	-2250	-2250	-2250	-2250			
~~RRV Pressure (see note)~~															
See NISA Press Release dated 4/11/2011															
~~Channel A~~	~~MPa g~~	~~0.011~~	~~0.005~~	~~0.009~~	~~0.005~~	~~0.002~~	~~0.000~~	~~-0.002~~	~~-0.004~~	~~-0.002~~	~~-0.004~~	~~-0.009~~			
~~Channel B~~	~~MPa g~~														
~~Channel C~~	~~MPa g~~	~~-0.081~~	~~-0.083~~	~~-0.081~~	~~-0.086~~	~~-0.079~~	~~-0.081~~	~~-0.081~~	~~-0.079~~	~~-0.085~~	~~-0.081~~	~~-0.077~~			
~~Channel A~~	~~MPa abs~~	~~0.112~~	~~0.106~~	~~0.110~~	~~0.106~~	~~0.103~~	~~0.101~~	~~0.099~~	~~0.097~~	~~0.099~~	~~0.097~~	~~0.092~~	~~0.101~~	~~0.101~~	~~0.101~~
~~Channel B~~	~~MPa abs~~														
~~Channel C~~	~~MPa abs~~	~~0.020~~	~~0.018~~	~~0.020~~	~~0.015~~	~~0.022~~	~~0.020~~	~~0.020~~	~~0.022~~	~~0.016~~	~~0.020~~	~~0.024~~	~~0.101~~	~~0.101~~	~~0.101~~
Drywell Pressure	MPa abs	0.1078	0.1071	0.1069	0.1069	0.1075	0.1059	0.1061	0.1052	0.1054	0.1055	0.1061			
Suppression Chamber Pressure	MPa abs	0.1733	0.1733	0.1733	0.1731	0.1729	0.1720	0.1726	0.1722	0.1727	0.1722	0.1717			
RPV Temperature		Note 6	Note 6	Note 6	Note 6	Note 6	Note 6	Note 6	Note 6	Note 6	Note 6	Note 6			
Feedwater Nozzle Temp	C	84.7	84.7	84.4	78.8	83.4	88.3	88.2	58.8	99.2	96.6	91.7			
RPV Bottom Head Temp	C	113.7	114.1	114.1	115.0	115.8	112.3	110.8	110.7	107.5	109.2	110.8			
Containment Atm Monitoring System															
Drywell	Sv/h	21.0	20.6	20.9	19.5	19.6	19.3	19.0	18.8	18.6	18.4	18.1			
Suppression Chamber	Sv/h	0.839	0.833	0.815	0.799	0.777	0.768	0.748	0.738	Note 9	0.717	0.703			
										Note 9: 0.028 "Under investigation of the change of the sit					
Notes:															
NISA News releases started reported gage pressure, then converted readings to absolute pressure. NISA separate parameter tables report gage pressure. Conversion in this table in BLUE.															
Standard atmospheric pressure = 101.325 kPa = 0.101325 Mpa Absolute pressure = 0.101325 MPa + 0.06 MPa = 0.161325 Mpa															

Plant Parameters: **Unit 3**						
Revision 4/14/2011 (as of 10:00 a.m. EDT)						
Source: NISA Press Releases	Unit					
RPV Injection.....Water Source	Salt/Fresh					
via Feed Water Line	L/min					
via Fire Extinguishing Line	L/min					
via Fire Extinguishing Line	m3/h					
RPV Level						
Fuel Range A	mm					
Fuel Range B	mm					
~~RRV Pressure (see note)~~						
See NISA Press Release dated 4/11/2011						
~~Channel A~~	~~MPa g~~					
~~Channel B~~	~~MPa g~~					
~~Channel C~~	~~MPa g~~					
~~Channel A~~	~~MPa abs~~	~~0.101~~	~~0.101~~	~~0.101~~	~~0.101~~	~~0.101~~
~~Channel B~~	~~MPa abs~~					
~~Channel C~~	~~MPa abs~~	~~0.101~~	~~0.101~~	~~0.101~~	~~0.101~~	~~0.101~~
Drywell Pressure	MPa abs					
Suppression Chamber Pressure	MPa abs					
RPV Temperature						
Feedwater Nozzle Temp	C					
RPV Bottom Head Temp	C					
Containment Atm Monitoring System						
Drywell	Sv/h					
Suppression Chamber	Sv/h					
		uation"				
Notes:						
NISA News releases started reported gage pressure, then converted readings to absolute pressure. NISA separate parameter tables report gage pressure. Conversion in this table in BLUE.						
Standard atmospheric pressure = 101.325 kPa = 0.101325 Mpa Absolute pressure = 0.101325 MPa + 0.06 MPa = 0.161325 Mpa						

Coyne, Kevin

From: Coyne, Kevin
Sent: Thursday, March 17, 2011 2:10 PM
To: Beasley, Benjamin
Subject: RE: RES support for commission meeting on Monday 3/21.

Ben –

Are you, John, and/or Marty able to support the Commission meeting?

From: Beasley, Benjamin
Sent: Thursday, March 17, 2011 11:14 AM
To: Wilson, George
Cc: Kauffman, John; Killian, Lauren; Manoly, Kamal; Coyne, Kevin; Stutzke, Martin
Subject: FW: RES support for commission meeting on Monday 3/21.

George,

As I mentioned on the phone call, we took the liberty of drafting a key message for the GI-199 Comm Plan. It is provided in John's message below.

I will talk to Kevin Coyne (acting director) and Marty about support for the Commission briefing. Let us know if you need anything else for the briefing or the Comm Plan.

Ben

From: Kauffman, John
Sent: Thursday, March 17, 2011 10:15 AM
To: Beasley, Benjamin
Subject: RE: RES support for commission meeting on Monday 3/21.

Ben,

For GI-199 and the Fukushima Daiichi earthquake and tsunami a key message could be (this is from Annie's document (answers 3 and 22 combined)),

> US plants are designed for appropriate earthquake shaking levels and are safe. Currently the NRC is conducting a program called Generic Issue 199, which is reviewing the adequacy of the earthquake design of US NPPs in central and eastern North America based on the latest data and analysis techniques. The NRC will look closely at all aspects of the response of the plants in Japan to the earthquake and tsunami to determine if any actions need to be taken in US plants and if any changes are necessary to NRC regulations.

Key messages from the GI-199 Communications Plan (slightly tweaked) are:

> (1) In August 2010, the Safety/Risk Assessment for GI-199 was completed. That assessment found that operating nuclear power plants are safe: Plants have adequate safety margin for seismic issues. The NRC's Safety/Risk Assessment confirmed that overall seismic risk estimates remain small and that adequate protection is maintained.
>
> (2) Though still small, some seismic hazard estimates have increased: Updates to seismic data and models indicate increased seismic hazard estimates for some operating nuclear power plant sites in the Central and Eastern United States.

AAA/97

(3) Assessment of GI-199 will continue: Plants are safe (see key message 1), but the NRC has separate criteria for evaluating whether plant improvements may be imposed. The NRC's Safety/Risk Assessment used readily available information and found that for about one-quarter of the currently operating plants, the estimated core damage frequency change is large enough to warrant further attention. Action may include obtaining additional, updated information and developing methods to determine if plant improvements to reduce seismic risk are warranted.

DRAFT

Fukushima Units 1 to 3
Risk Analysis

US Nuclear Regulatory Commission

April 26, 2011

DRAFT

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APPA/QL

4-26

Assumptions

- Analysis is on per reactor bases
  - Each unit is expected to have similar risk profile – it is recognized that Units 1, 2 and 3 have varying amounts of core damage and different status of reactor pressure vessels (RPV) and containments
- No analysis has been completed for spent fuel pools yet
- End state is large release (LR) caused by inadequate cooling of core material in reactor and/or containment
- Assumed time to LR is 10 hours
  - Based on estimated time without cooling that core material in RPV could melt through RPV bottom head
- Analysis performed using SAPHIRE code (version 8.0.7.13)
- Human Reliability Analysis (HRA) based on SPAR-H methodology

Initiating Events Evaluated

- "Normal" water supply source (1.9E-2/year) – see next slide
- Pump failure (5E-1/year)
- Hoses connecting pumps and injection piping (1E-1/year)
- Injection pipe failure (1.1E-2/year)
  - Feedwater pipe on Unit 1
  - Low pressure core injection or recirculation piping on Units 2 and 3
- RCS vent path (1E-2/year)
  - This is significant contributor and
  - If RPV is breached then failure probability is zero
  - If RPV is not breached then probability of unknown vent failing is difficult to estimate
- Loss of "normal" AC power (10/year)
  - Based on 1 event in 6 weeks
- Second tsunami (1.8E-3/year)
  - Based on 2 event in 1100 years

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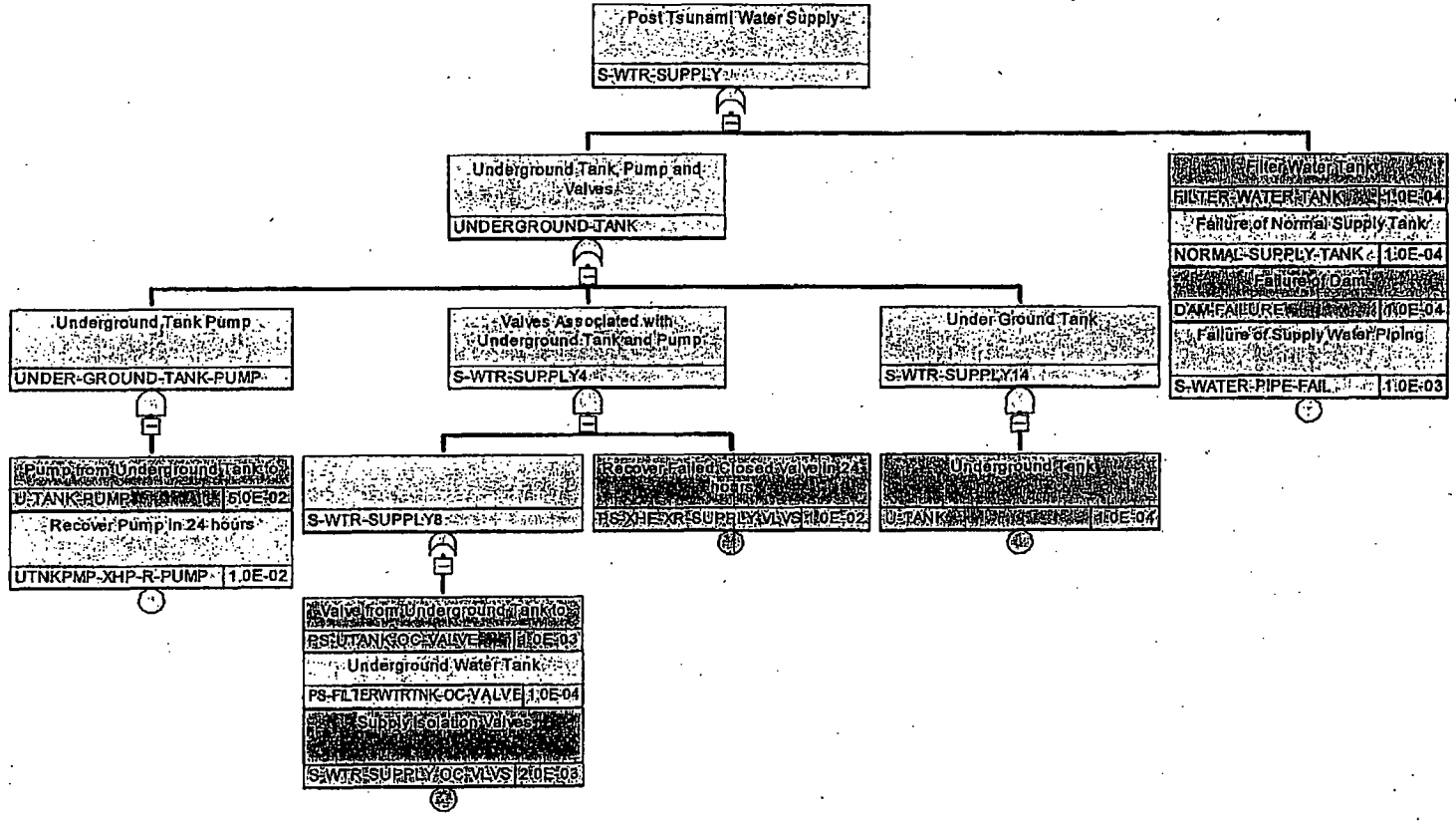
"Normal" Water Supply from Dam to Pumps

- This is frequency (events per year) estimation of losing supply water to currently running injection pumps
- Equipment associated with initiating event frequency (IEF) estimation includes:
  - Dam
  - Underground tank
  - Underground tank pump and valves
  - Filter tank
  - Supply tank
- Result: 1.9E-3 per year

FT for Estimating IEF for Loss of Water Supply



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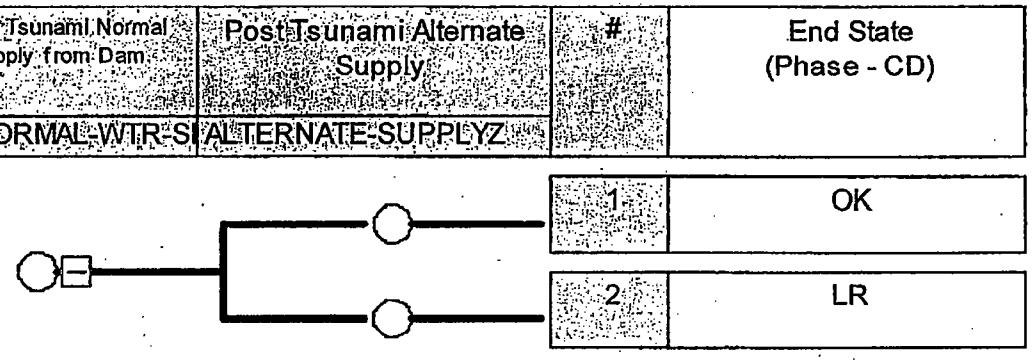
Failure of Normal Water Supply



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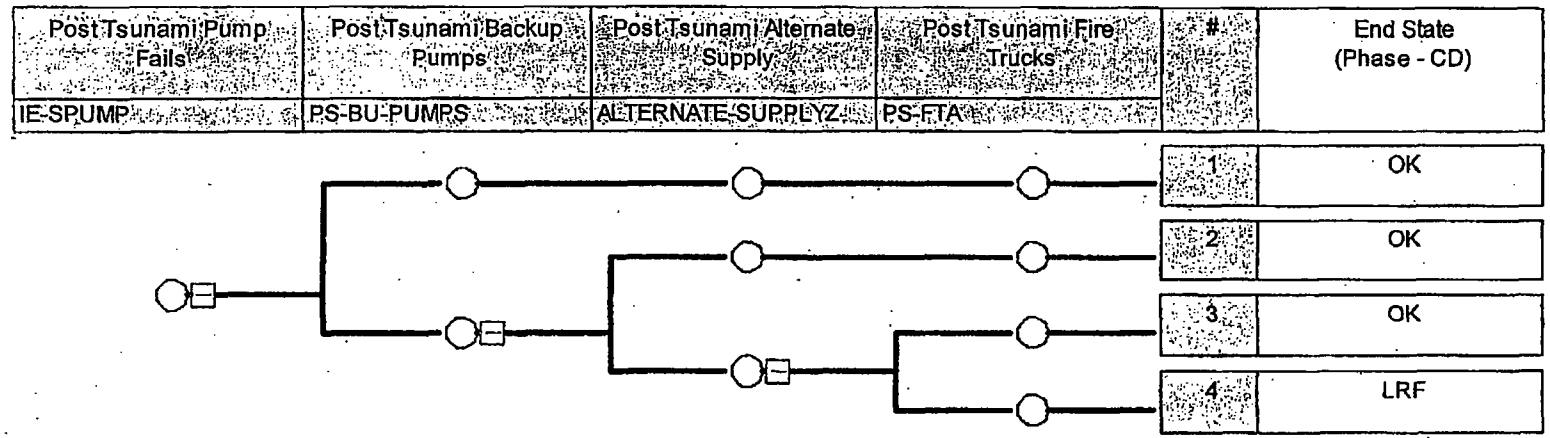
Loss of Running Injection Pump



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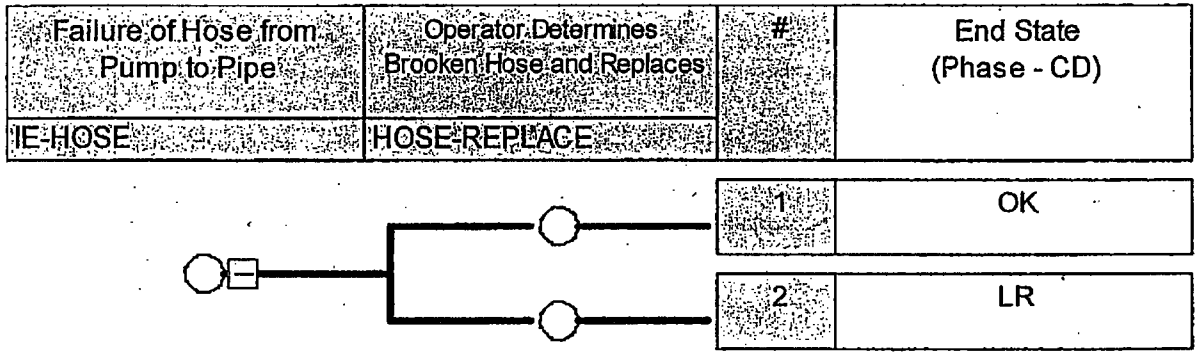
Failure of Hose from Pump to Pipe



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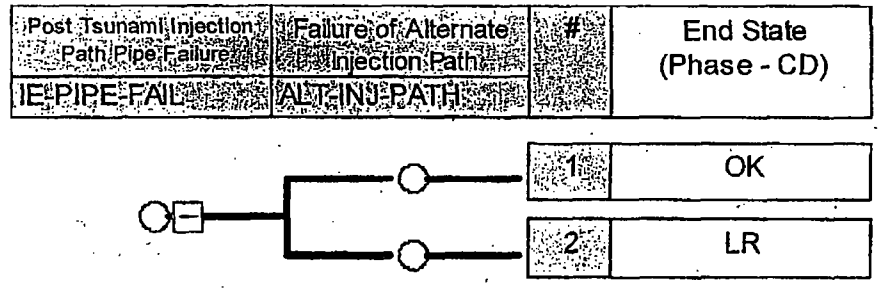
Failure of Pipe to RPV



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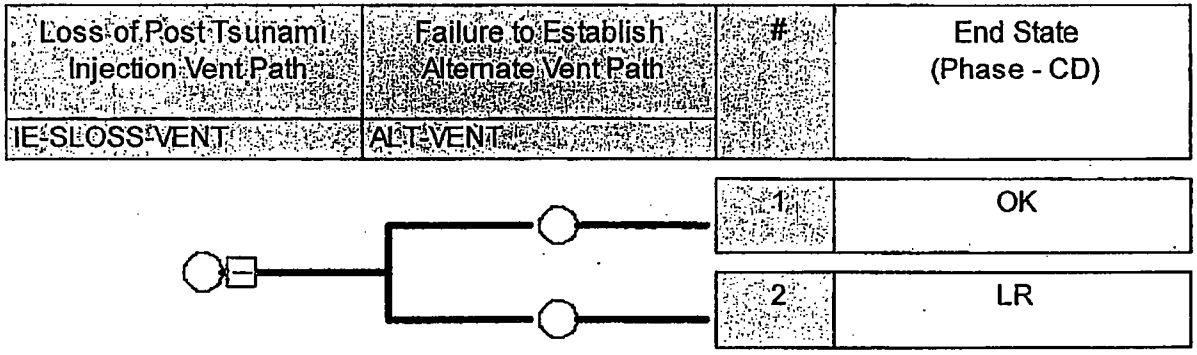
Failure of RPV Vent Path



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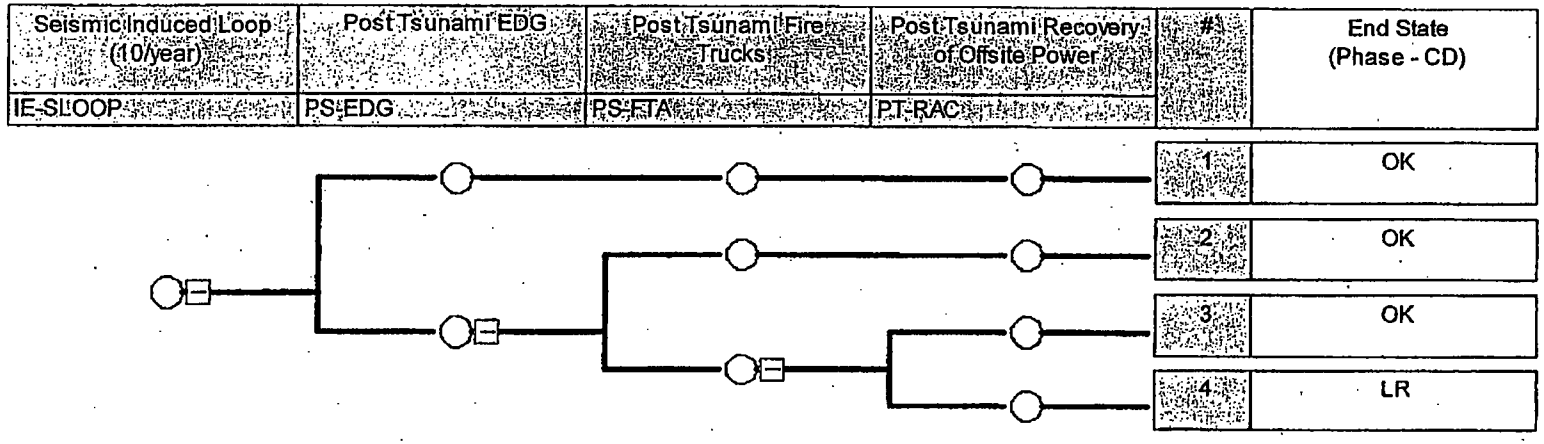
Failure of Electrical Power



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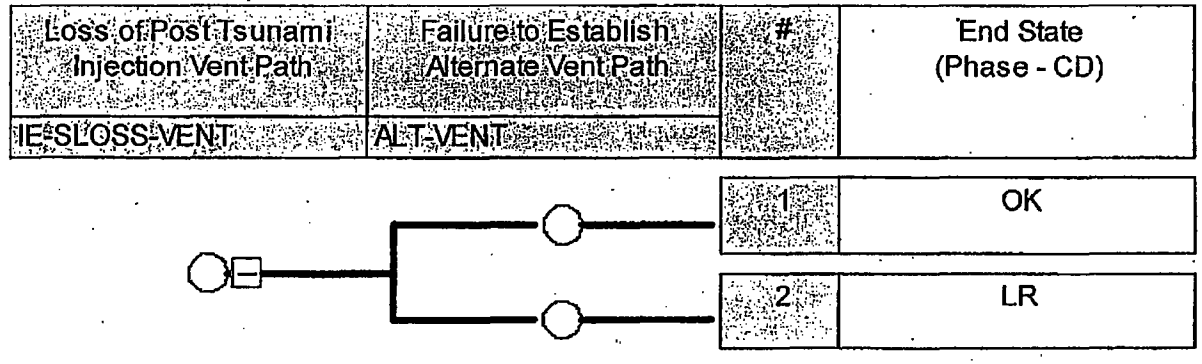
Second Tsunami



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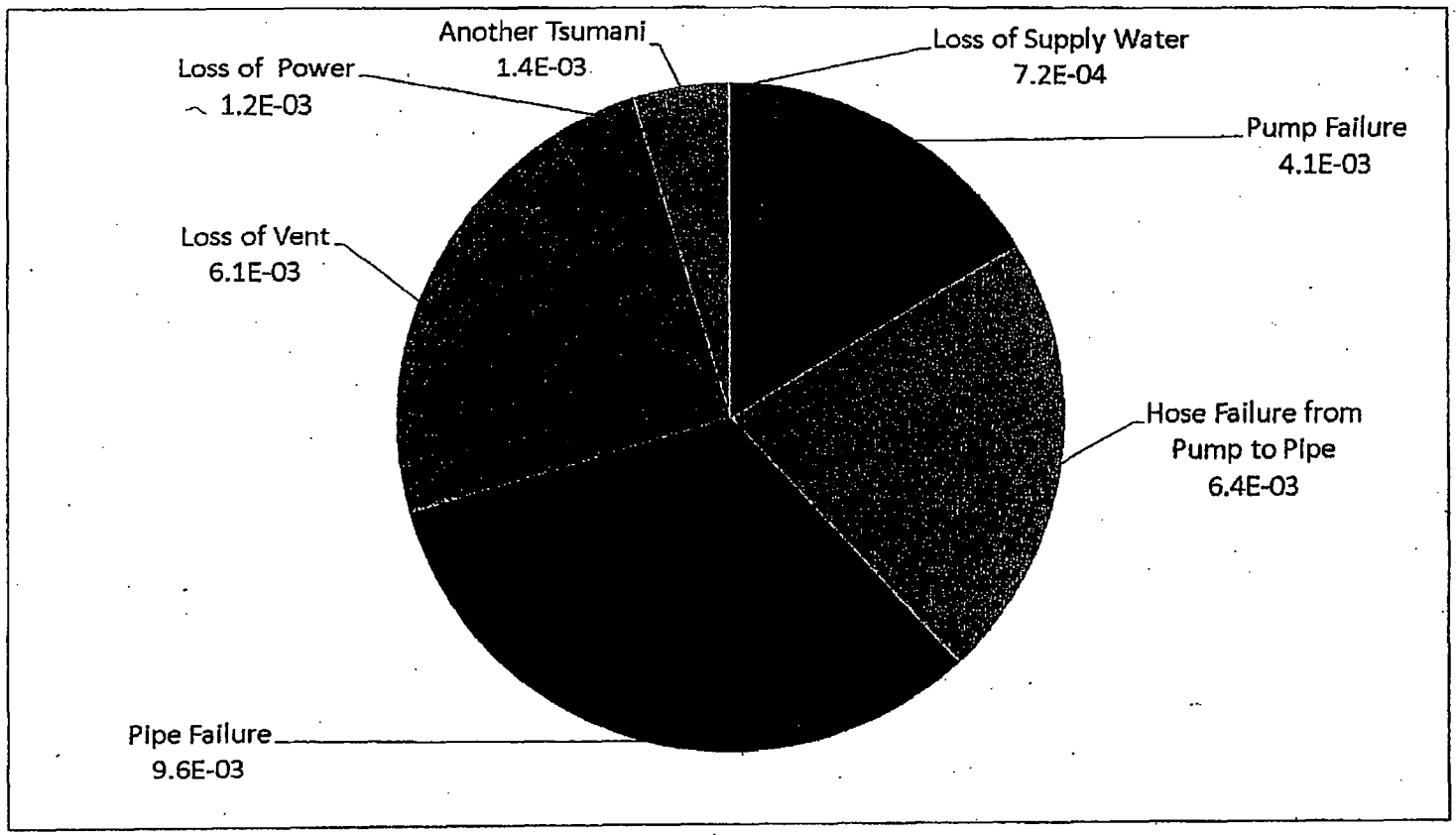
Results: Risk of Large Release from Each Reactor = 3.1% per year



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Insights from Results

- Precision of results is not important
- Risk results do give insights to vulnerabilities and where to allocate resources

Largest Risk Contributors

- Injection pipe failure
- Injection hose failure
- Human actions
- Vent failure

- See next slide for ranking

Risk Significant Contributors

Name	Probability	Risk Reduction Factor	Description	Fix
IE-PIPE-FAIL	1.1E-2	1.50	Post Tsunami Injection Path Pipe Failure	Add injection path through independent pipe
IE-HOSE	1.0E-1	1.30	Failure of Hose from Pump to Pipe	Protect hoses from wear, falling objects, trucks, etc.
PS-XHE-XM-PSFT	1.5E-1	1.30	Operator Fails to Align Post Tsunami Fire Truck	Improve operator training, create procedures, reduce stress
PS-XHE-XM-PSFT-BU	7.5E-1	1.30	Operator Fails to Align Backup Fire Trucks	Improve operator training, create procedures, reduce stress
IE-SLOSS-VENT	1.0E-2	1.20	Loss of Post Tsunami Injection Vent Path	Reduce probability of lossing existing vent path
IE-SPUMP	5.0E-1	1.20	Post Tsunami Pump Fails	Improve reliability of pumps
PS-XHE-XD-HOSE-	5.5E-2	1.20	Operator Identifies Brooken Hose, How to Isolate ad Replaces	Improve operator training, create procedures, reduce stress
PS-XHE-XM-ALT-SUPPLY	3.8E-1	1.20	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps	Improve operator training, create procedures, reduce stress
PS-XHE-XM-PIPEALIGN	7.5E-1	1.20	Operator Fails to Align Alternate Pipe Path	Improve operator training, create procedures, reduce stress
ALT-VENT1	5.0E-1	1.10	Alternate Vent Path Available	Establish a backup vent path as necessary
IE-SLOOP	1.0E+1	1.10	Seismic Induced Loop (10/year)	No corrective fix possible - refine number
OEP-XHE-XL-NR10H	5.1E-2	1.10	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10	Improve operator training, create procedures, reduce stress
PS-XHE-XM-BUP	7.5E-2	1.10	Operator Fails to Start Backup Pump	Improve operator training, create procedures, reduce stress
PS-XHE-XM-PSEDG-	1.5E-1	1.10	Operator Fails to Start Post Tsunami Mobile EDG	Improve operator training, create procedures, reduce stress
PS-XHE-XM-PSEDG-	1.5E-1	1.10	Operator Fails to Start Post Tsunami Stationary EDG	Improve operator training, create procedures, reduce stress

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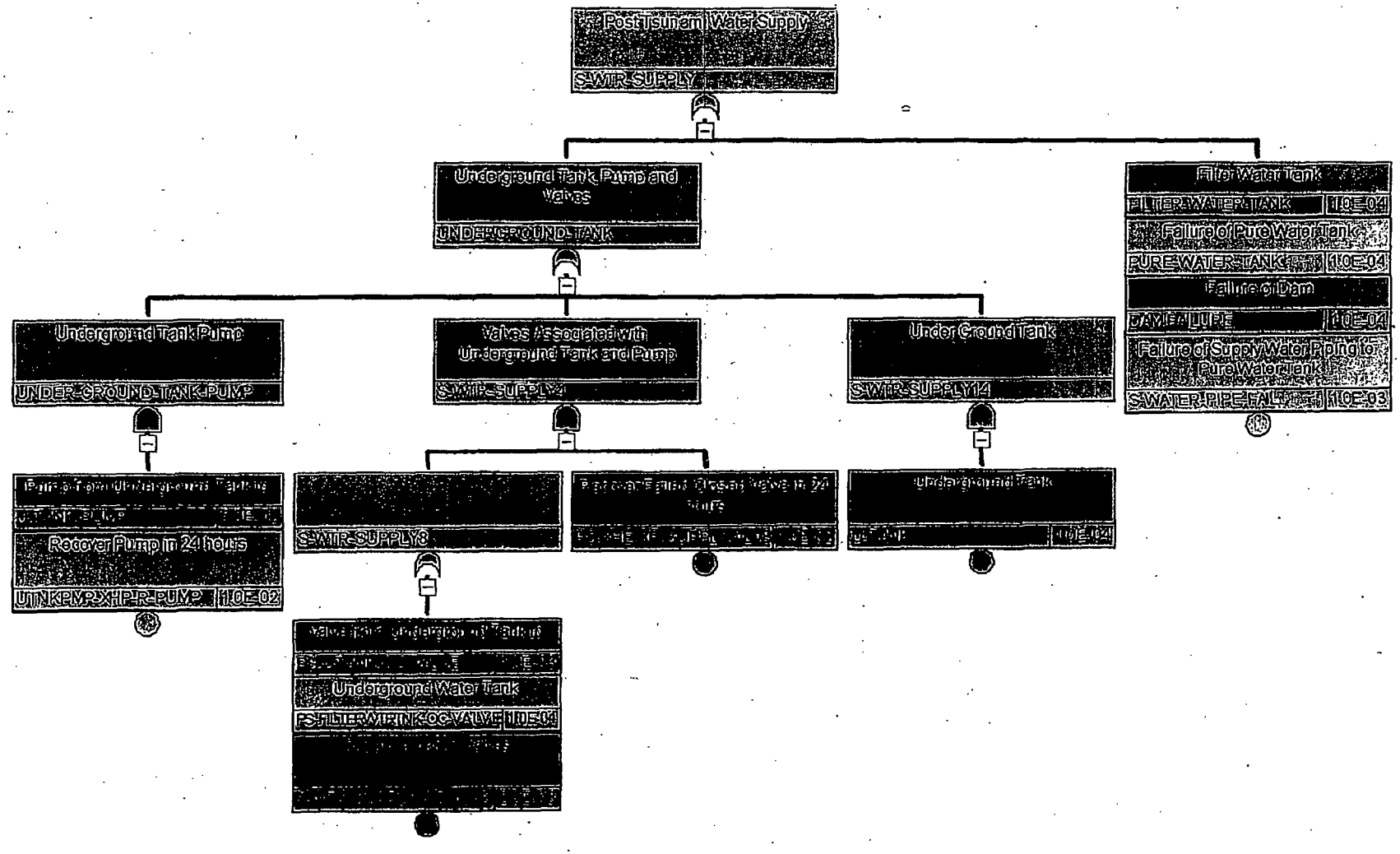
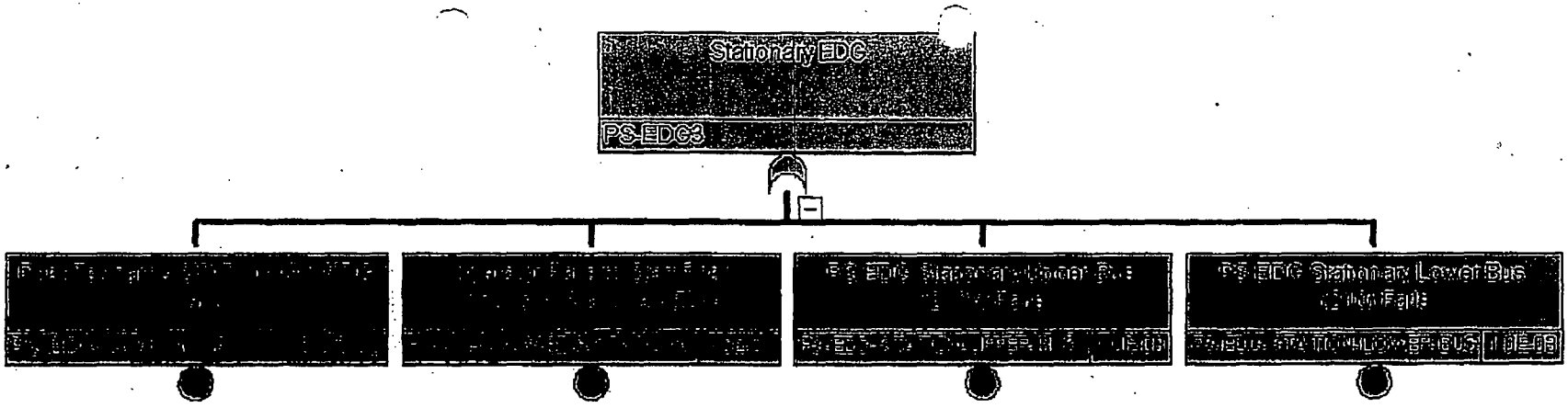

Post Tsunami Water Supply
S-WTR-SUPPLY
Underground Tank Pump and Valves
UNDERGROUND-TANK
Filter Water Tank
FILTER-WATER-TANK
1.0E-04
Failure of Pure Water Tank
PURE-WATER-TANK
1.0E-04
Failure of Dam
DAM-FAILURE
1.0E-04
Failure of Supply Water Piping to Pure Water Tank
S-WATER-PIPE-FAILURE
1.0E-03
Underground Tank Pump
UNDER-GROUND-TANK-PUMP
Valves Associated with Underground Tank and Pump
S-WTR-SUPPLY2
Under Ground Tank
S-WTR-SUPPLY14
Recover Pump in 24 hours
UTNKPMP-XHPR-PUMP
1.0E-02
S-WTR-SUPPLY8
Underground Water Tank
PS-FILTER-WTR-TNK-OC-VALVE
1.0E-04

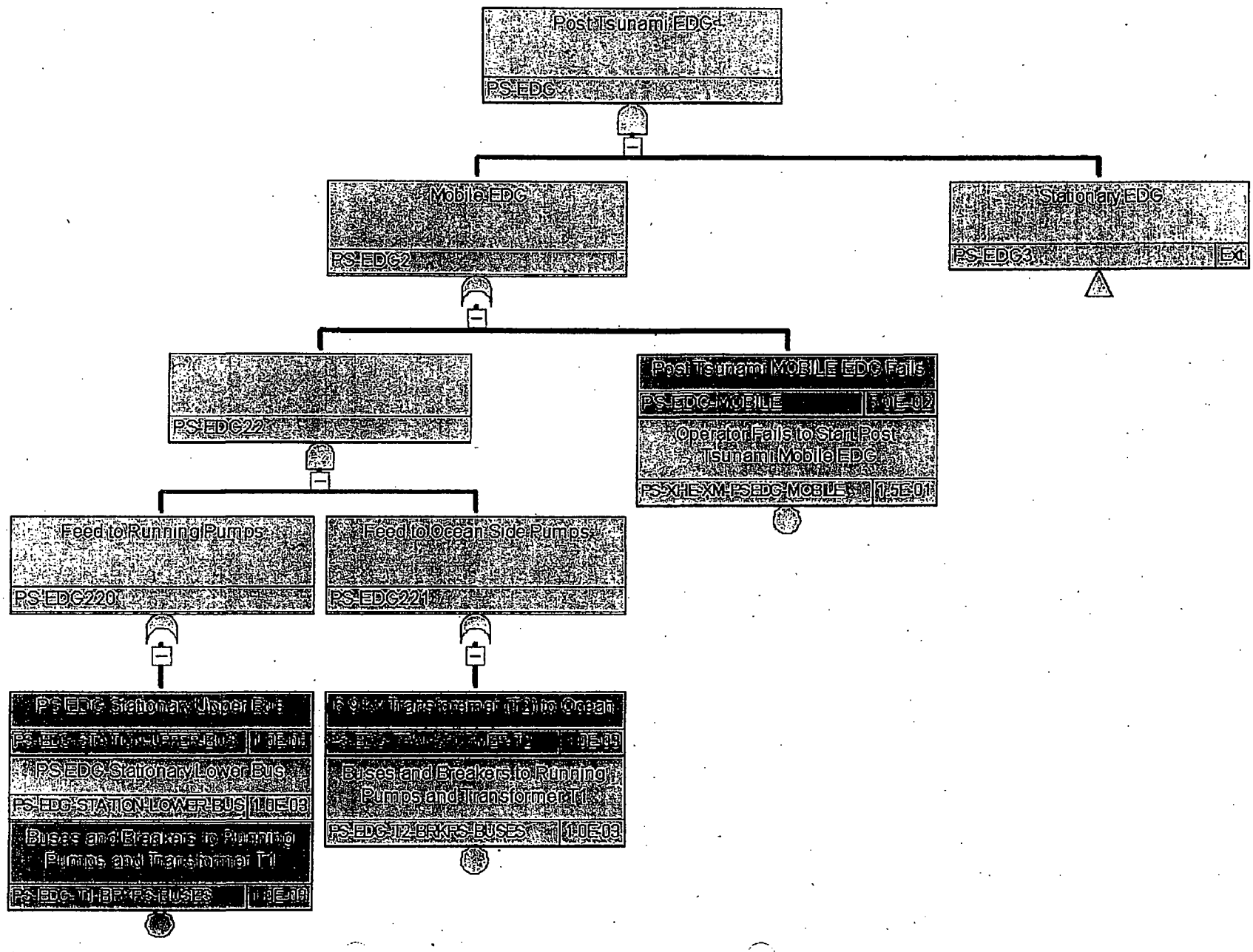
Post Tsunami Recovery of Offsite Power (3.5E-11)

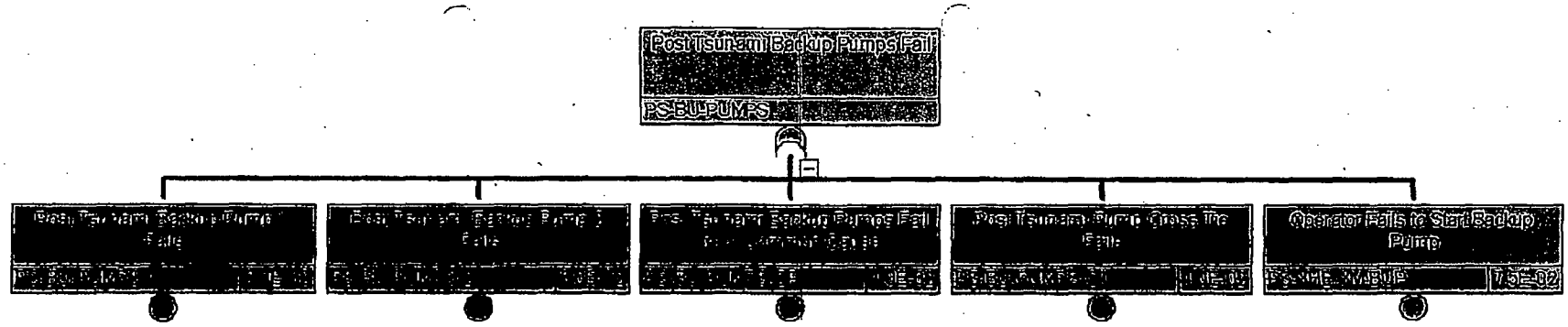
PT-RAC

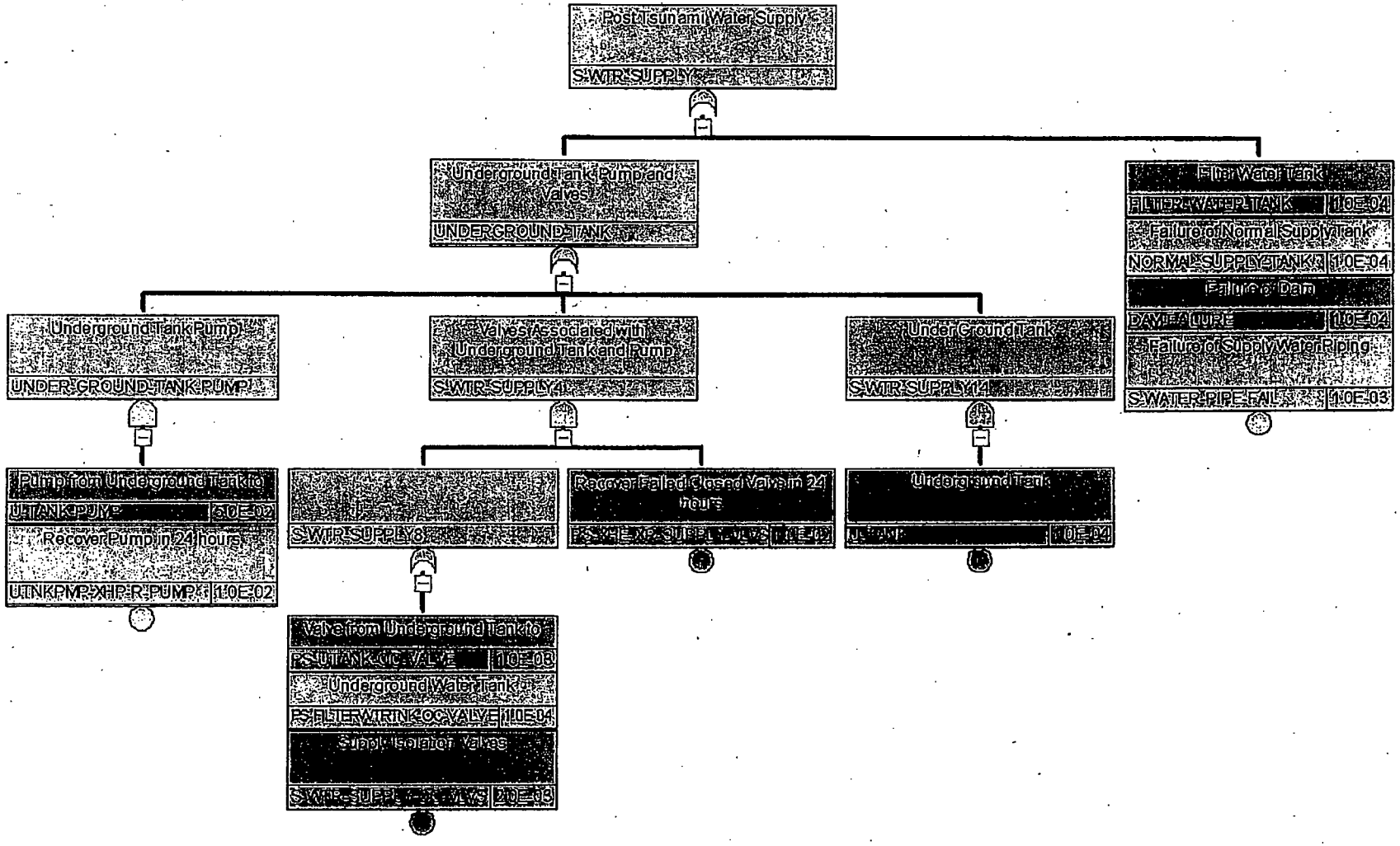
OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS

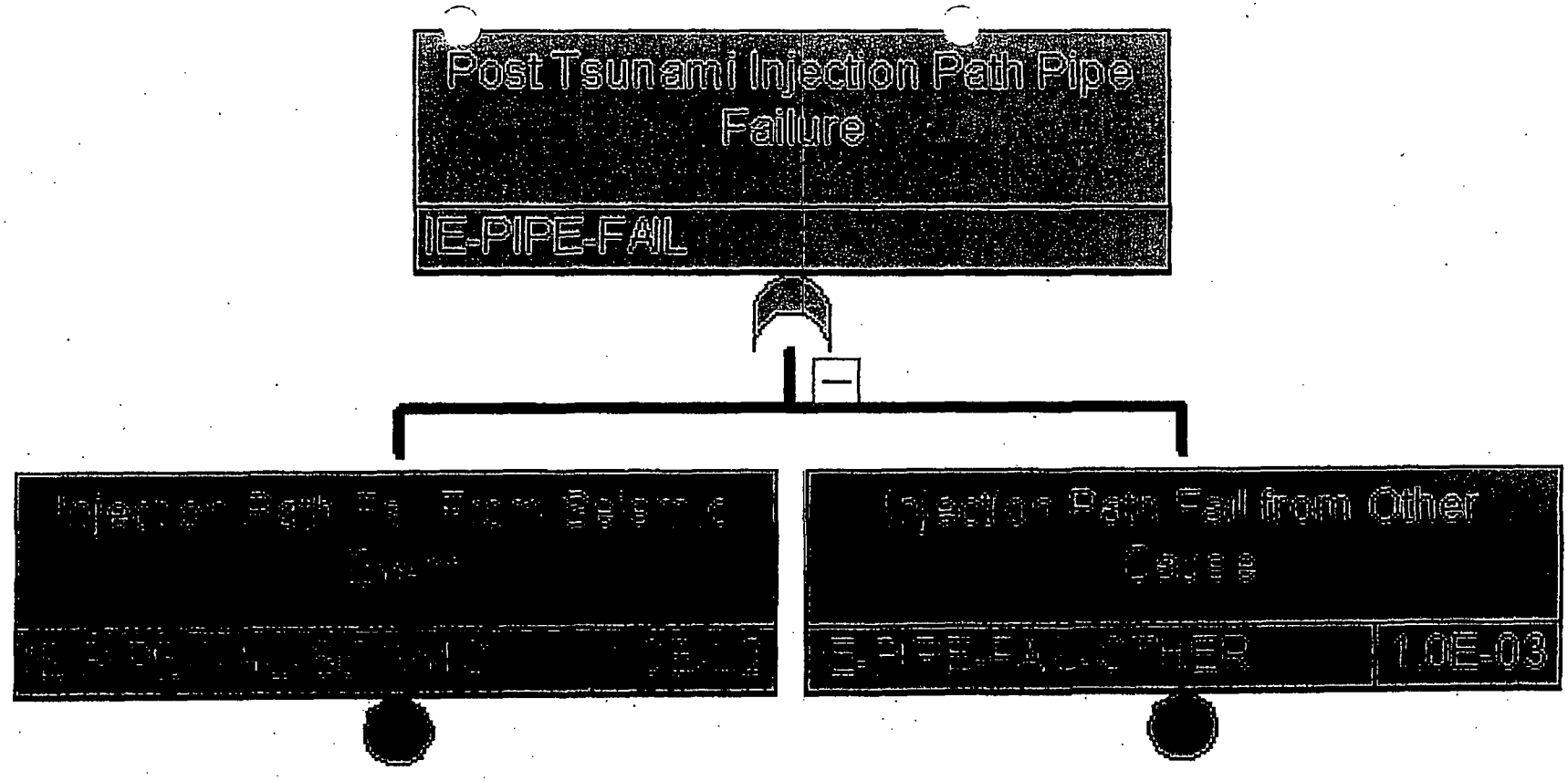
OEP-XHE-XL-NR10H	5.1E-02

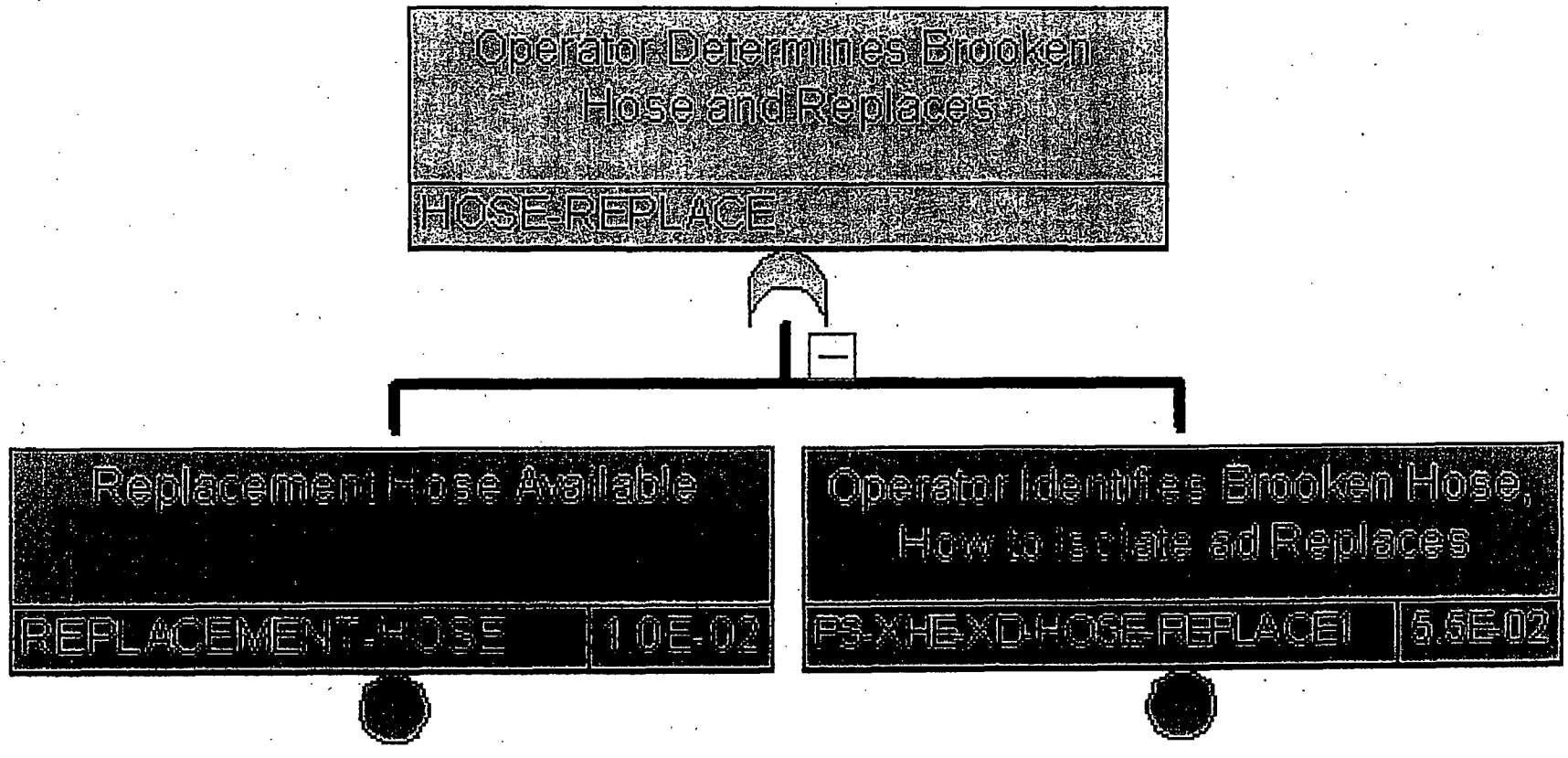

Stationary EDG
PS-EDG3
PS-EDG Stationary Lower Bus

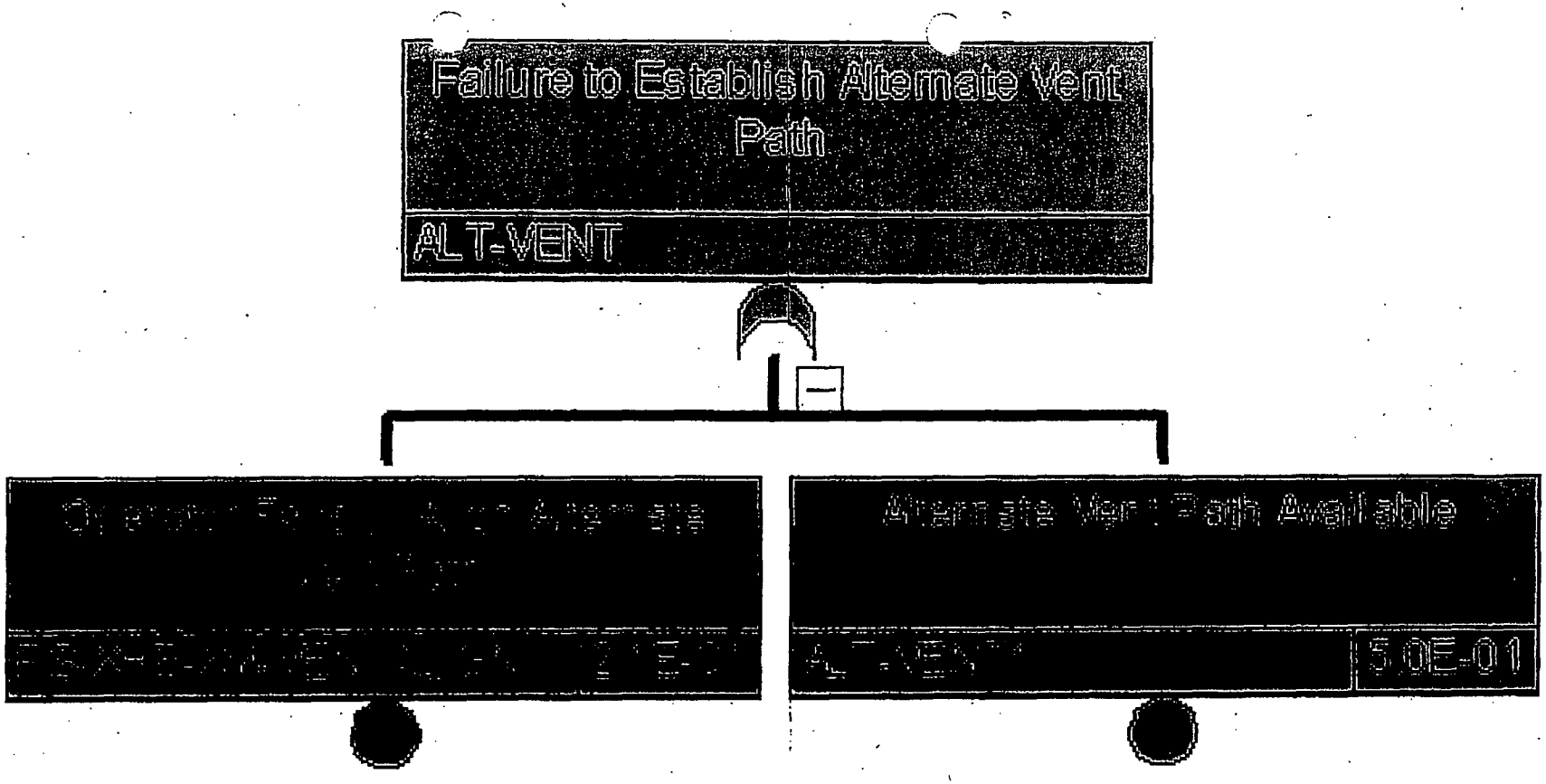

Post Tsunami EDC
PS-EDC
Mobile EDC
PS-EDC2
Stationary EDC
PS-EDC3
Ex
PS-EDC22
Post Tsunami MOBILE EDC Fails
PS-EDC-MOBILE
5.0E-02
Operator Fails to Start Post Tsunami Mobile EDC
PS-XHE-XM-PSEDC-MOBLE
1.5E-01
Feed to Running Pumps
PS-EDC220
Feed to Ocean Side Pumps
PS-EDC221
PS-EDC Stationary Upper Bus
PS-EDC-STATION-UPPER-BUS
1.0E-03
PS-EDC Stationary Lower Bus
PS-EDC-STATION-LOWER-BUS
1.0E-03
Buses and Breakers to Running Pumps and Transformer T1
PS-EDC-T1-BRKRS-BUSES
1.0E-03
Buses and Breakers to Running Pumps and Transformer T1
PS-EDC-T2-BRKRS-BUSES
1.0E-03

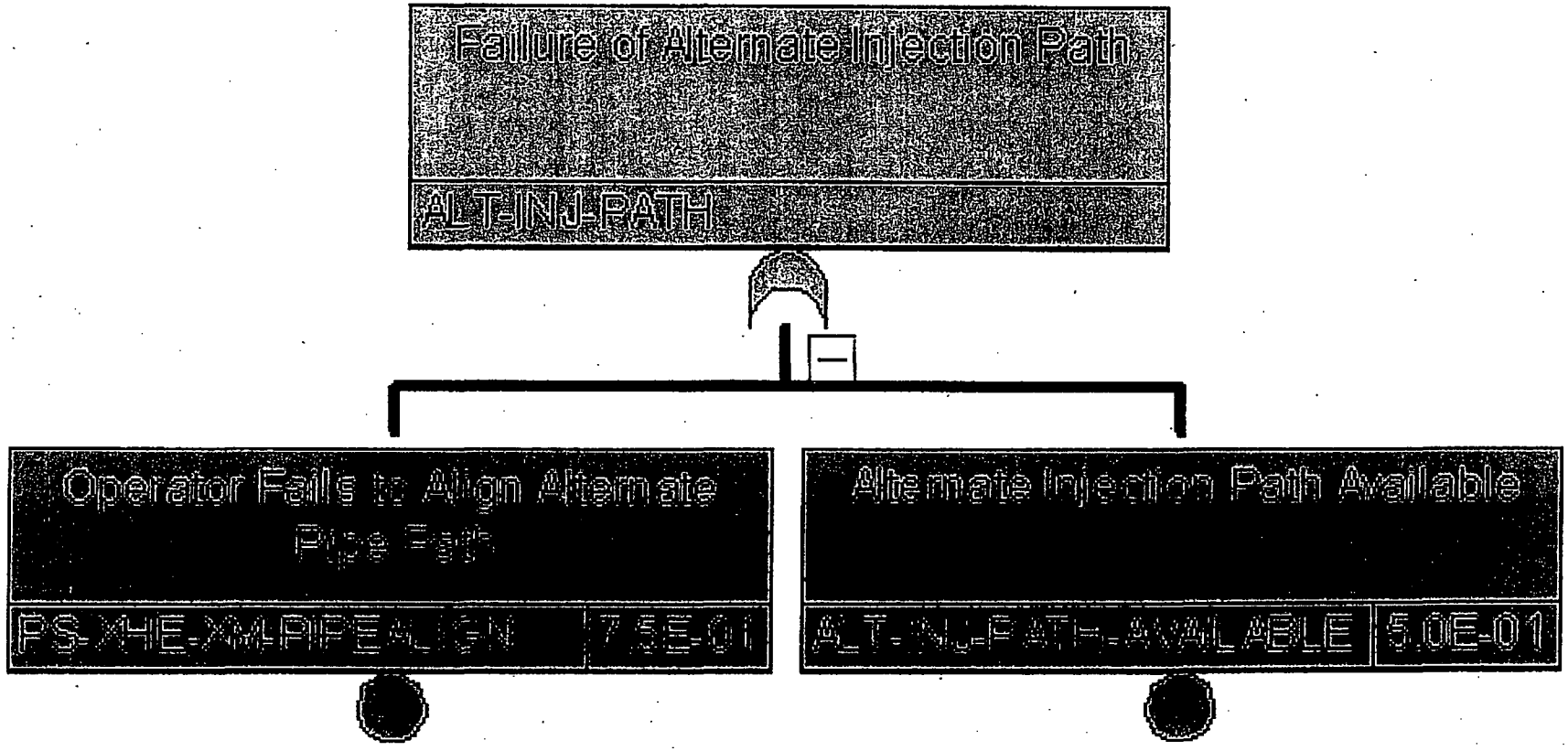

Post Tsunami Backup Pumps Fail
PS-BUPUMPS
Operator Fails to Start Backup Pump
7.5E-02


Post Tsunami Water Supply
S-WTR-SUPPLY
Underground Tank Pump and Valves
UNDERGROUND-TANK
Filter Water Tank
FILTER-WATER-TANK
1.0E-04
Failure of Normal Supply Tank
NORMAL-SUPPLY-TANK
1.0E-04
Failure of Dam
DAM-FAILURE
1.0E-04
Failure of Supply Water Piping
S-WATER-PIPE-FAIL
1.0E-03
Underground Tank Pump
UNDER-GROUND-TANK-PUMP
Valves Associated with Underground Tank and Pump
S-WTR-SUPPLY4
Under Ground Tank
S-WTR-SUPPLY14
Pump from Underground Tank to
UTANK-PUMP
6.0E-02
Recover Pump in 24 hours
UTNKPMP-XHR-R-PUMP
1.0E-02
S-WTR-SUPPLY8
Recover Failed Closed Valve in 24 hours
1.0E-02
Underground Tank
1.0E-04
Valve from Underground Tank to
PS-UTANK-OC-VALVE
1.0E-03
Underground Water Tank
PS-FILTER-WTRNK-OC-VALVE
1.0E-04
Supply Isolation Valves
2.0E-03


Post Tsunami Injection Path Pipe Failure
IE-PIPE-FAIL
Injection Path Fail from Other Causes
IPBREAKOTHER
1.0E-03

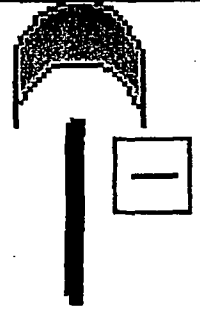

Operator Determines Brooken Hose and Replaces
HOSE-REPLACE
Replacement Hose Available
REPLACEMENT-HOSE
1.0E-02
Operator Identifies Brooken Hose, How to Isolate and Replaces
P3-XHEXD-HOSE-REPLACE1
5.5E-02


Failure to Establish Alternate Vent Path
ALT-VENT
Alternate Vent Path Available
5.0E-01


Failure of Alternate Injection Path
ALT-INJ-PATH
Operator Fails to Align Alternate Pipe Path
PS-XHE-XM-PIPEALIGN
7.5E-01
Alternate Injection Path Available
ALT-INJ-PATH-AVALABLE
5.0E-01

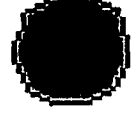
Recover from Second Tsunami

TSUNAMI-RECOVERY



Recover from Second Tsunami

TSUNAMI-EX-RECOVER | 7.8E-01



Cut Set Report - ET-Group (ET)			Fukushima Daiichi In ...Jeff Mitman\Desktop\Fuku Model\	
04/26/2011 9:56:01 AM				
#	Prob/Freq	Total %	Cut Set	Description
Total	3.1E-2	1E02	Displaying 100 of 434 Cut Sets.	
1	8.2E-3	27	0-SPIPE-FAIL :2	
	1.1E-2		IE-PIPE-FAIL	Post Tsunami Injection Path Pipe Failure
	7.5E-1		PS-XHE-XM-PIPEALIGN	Operator Fails to Align Alternate Pipe Path
		EndState	->LR	Added through Event Tree Add
2	5.5E-3	18	0-SPIPE-FAIL :2	
	1.1E-2		IE-PIPE-FAIL	Post Tsunami Injection Path Pipe Failure
	5.0E-1		ALT-INJ-PATH-AVAILABLE	Alternate Injection Path Available
		EndState	->LR	Added through Event Tree Add
3	5.5E-3	18	0-HOSE-FAIL :2	
	1.0E-1		IE-HOSE	Failure of Hose from Pump to Pipe
	5.5E-2		PS-XHE-XD-HOSE-REPLACE1	Operator Identifies Brooken Hose, How to Isolate ad Replaces
		EndState	->LR	Added through Event Tree Add
4	5.0E-3	16	0-SLOSS-VENT :2	
	1.0E-2		IE-SLOSS-VENT	Loss of Post Tsunami Injection Vent Path
	5.0E-1		ALT-VENT1	Alternate Vent Path Available
		EndState	->LR	Added through Event Tree Add
5	2.1E-3	6.9	0-SLOSS-VENT :2	
	1.0E-2		IE-SLOSS-VENT	Loss of Post Tsunami Injection Vent Path
	2.1E-1		PS-XHE-XM-VENTALIGN	Operator Fails to Align Alternate Vent Path
		EndState	->LR	Added through Event Tree Add
6	1.6E-3	5.2	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
7	1.4E-3	4.4	0-TSUNAMI :2	
	1.8E-3		IE-TSUNAMI	Tsunami
	7.5E-1		TSUNAMI-XHE-XL-RECOVER	Recover from Second Tsunami
		EndState	->LR	Added through Event Tree Add
8	1.3E-3	4.2	0-SLOOP :4	
	1.0E+1		IE-SLOOP	Seismic Induced Loop (10/year)
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.5E-1		PS-XHE-XM-PSEDG-MOBILE	Operator Fails to Start Post Tsunami Mobile EDG
	1.5E-1		PS-XHE-XM-PSEDG-STATIONA	Operator Fails to Start Post Tsunami Stationary EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck

	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LR	Added through Event Tree Add
9	1.1E-3	3.4	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
10	1.1E-3	3.4	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
11	1.0E-3	3.3	0-HOSE-FAIL :2	
	1.0E-1		IE-HOSE	Failure of Hose from Pump to Pipe
	1.0E-2		REPLACEMENT-HOSE	Replacement Hose Available
		EndState	->LR	Added through Event Tree Add
12	7.1E-4	2.3	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
		EndState	->LR	Added through Event Tree Add
13	4.3E-4	1.4	0-SLOOP :4	
	1.0E+1		IE-SLOOP	Seismic Induced Loop (10/year)
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	5.0E-2		PS-EDG-MOBILE	Post Tsunami MOBILE EDG Fails
	1.5E-1		PS-XHE-XM-PSEDG-STATIONA	Operator Fails to Start Post Tsunami Stationary EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LR	Added through Event Tree Add
14	4.3E-4	1.4	0-SLOOP :4	
	1.0E+1		IE-SLOOP	Seismic Induced Loop (10/year)
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	5.0E-2		PS-EDG-STATIONARY	Post Tsuname STATIONARY EDG Fails
	1.5E-1		PS-XHE-XM-PSEDG-MOBILE	Operator Fails to Start Post Tsunami Mobile EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LR	Added through Event Tree Add
15	2.1E-4	0.69	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails

	1.0E-2		PS-BU-PUMPS-CT	Post Tsunami Pump Cross Tie Fail
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
16	2.1E-4	0.69	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		PS-BU-PUMPS-CF	Post Tsunami Backup Pumps Fail from Common Cause
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
17	1.4E-4	0.47	0-SLOOP :4	
	1.0E+1		IE-SLOOP	Seismic Induced Loop (10/year)
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	5.0E-2		PS-EDG-MOBILE	Post Tsunami MOBILE EDG Fails
	5.0E-2		PS-EDG-STATIONARY	Post Tsuname STATIONARY EDG Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LR	Added through Event Tree Add
18	8.6E-6	0.03	0-SLOOP :4	
	1.0E+1		IE-SLOOP	Seismic Induced Loop (10/year)
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-EDG-STATION-UPPER-BUS	PS EDG Stationary Upper Bus (210V) Fails
	1.5E-1		PS-XHE-XM-PSEDG-MOBILE	Operator Fails to Start Post Tsunami Mobile EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LR	Added through Event Tree Add
19	8.6E-6	0.03	0-SLOOP :4	
	1.0E+1		IE-SLOOP	Seismic Induced Loop (10/year)
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-EDG-STATION-LOWER-BUS	PS EDG Stationary Lower Bus (210) Fails
	1.5E-1		PS-XHE-XM-PSEDG-MOBILE	Operator Fails to Start Post Tsunami Mobile EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LR	Added through Event Tree Add
20	3.2E-6	0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-3		POWER-NORMAL	Normal Power Supply Fails
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSEDG-STATIONA	Operator Fails to Start Post Tsunami Stationary EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck

	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
21	2.9E-6	0.01	0-SLOOP :4	
	1.0E+1		IE-SLOOP	Seismic Induced Loop (10/year)
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	5.0E-2		PS-EDG-MOBILE	Post Tsunami MOBILE EDG Fails
	1.0E-3		PS-EDG-STATION-UPPER-BUS	PS EDG Stationary Upper Bus (210V) Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LR	Added through Event Tree Add
22	2.9E-6	0.01	0-SLOOP :4	
	1.0E+1		IE-SLOOP	Seismic Induced Loop (10/year)
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	5.0E-2		PS-EDG-MOBILE	Post Tsunami MOBILE EDG Fails
	1.0E-3		PS-EDG-STATION-LOWER-BUS	PS EDG Stationary Lower Bus (210) Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LR	Added through Event Tree Add
23	2.1E-6	0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-4		PS-CF-OCEANSIDEPUMPS-FS	Common Cause Failure to Run of Ocean Side Pumps
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
24	2.1E-6	0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
	5.0E-2		PS-XHE-XR-X3PUMP	Operator Fails to Recover Pump to Ocean Side Tank in 24 Hours
	1.0E-2		S-X3-FS-PUMP	Pump from Filtered Tank to Ocean Side Tanks
		EndState	->LRF	Added through Event Tree Add
25	2.1E-6	0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-3		POWER-NORMAL	Normal Power Supply Fails
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	1.5E-1		PS-XHE-XM-PSEDG-STATIONA	Operator Fails to Start Post Tsunami Stationary EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
26	2.1E-6	0.01	0-SPUMP-FAIL :4	

	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-3		POWER-NORMAL	Normal Power Supply Fails
	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	1.5E-1		PS-XHE-XM-PSEDG-STATIONA	Operator Fails to Start Post Tsunami Stationary EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
27	1.4E-6	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	5.0E-3		POWER-NORMAL	Normal Power Supply Fails
	1.5E-1		PS-XHE-XM-PSEDG-STATIONA	Operator Fails to Start Post Tsunami Stationary EDG
		EndState	->LR	Added through Event Tree Add
28	1.4E-6	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	5.0E-4		PS-CF-OCEANSIDEPUMPS-FS	Common Cause Failure to Run of Ocean Side Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
29	1.4E-6	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	5.0E-4		PS-CF-OCEANSIDEPUMPS-FS	Common Cause Failure to Run of Ocean Side Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
30	1.4E-6	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-4		BACKUP-FT-CF-FS	Common Cause Failure of All Fire Trucks
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
		EndState	->LRF	Added through Event Tree Add
31	1.4E-6	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
	5.0E-2		PS-XHE-XR-X3PUMP	Operator Fails to Recover Pump to Ocean Side Tank in 24 Hours
	1.0E-2		S-X3-FS-PUMP	Pump from Filtered Tank to Ocean Side Tanks
		EndState	->LRF	Added through Event Tree Add
32	1.4E-6	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails

	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
	5.0E-2		PS-XHE-XR-X3PUMP	Operator Fails to Recover Pump to Ocean Side Tank in 24 Hours
	1.0E-2		S-X3-FS-PUMP	Pump from Filtered Tank to Ocean Side Tanks
		EndState	->LRF	Added through Event Tree Add
33	1.1E-6	< 0.01	0-SLOOP :4	
	1.0E+1		IE-SLOOP	Seismic Induced Loop (10/year)
	1.0E-4		BACKUP-FT-CF-FS	Common Cause Failure of All Fire Trucks
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.5E-1		PS-XHE-XM-PSEDG-MOBILE	Operator Fails to Start Post Tsunami Mobile EDG
	1.5E-1		PS-XHE-XM-PSEDG-STATIONA	Operator Fails to Start Post Tsunami Stationary EDG
		EndState	->LR	Added through Event Tree Add
34	1.1E-6	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-3		POWER-NORMAL	Normal Power Supply Fails
	5.0E-2		PS-EDG-STATIONARY	Post Tsuname STATIONARY EDG Fails
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
35	9.5E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	5.0E-2		PS-XHE-XR-X3PUMP	Operator Fails to Recover Pump to Ocean Side Tank in 24 Hours
	1.0E-2		S-X3-FS-PUMP	Pump from Filtered Tank to Ocean Side Tanks
		EndState	->LR	Added through Event Tree Add
36	9.5E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	5.0E-4		PS-CF-OCEANSIDEPUMPS-FS	Common Cause Failure to Run of Ocean Side Pumps
		EndState	->LR	Added through Event Tree Add
37	9.4E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-4		BACKUP-FT-CF-FS	Common Cause Failure of All Fire Trucks
	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
		EndState	->LRF	Added through Event Tree Add
38	9.4E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-4		BACKUP-FT-CF-FS	Common Cause Failure of All Fire Trucks
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
		EndState	->LRF	Added through Event Tree Add

39	7.0E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-3		POWER-NORMAL	Normal Power Supply Fails
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	5.0E-2		PS-EDG-STATIONARY	Post Tsuname STATIONARY EDG Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
40	7.0E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-3		POWER-NORMAL	Normal Power Supply Fails
	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	5.0E-2		PS-EDG-STATIONARY	Post Tsuname STATIONARY EDG Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
41	4.8E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	5.0E-3		POWER-NORMAL	Normal Power Supply Fails
	5.0E-2		PS-EDG-STATIONARY	Post Tsuname STATIONARY EDG Fails
		EndState	->LR	Added through Event Tree Add
42	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-4		FILTER-WATER-TANK	Filter Water Tank
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
43	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
	1.0E-4		U-TANK	Underground Tank
		EndState	->LRF	Added through Event Tree Add
44	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks

		EndState	->LRF	Added through Event Tree Add
45	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
46	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
47	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
48	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
49	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
50	4.2E-7	< 0.01	0-SPUMP-FAIL :4	

	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
51	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
52	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
53	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-3		POWER-NORMAL	Normal Power Supply Fails
	1.0E-2		PS-BU-PUMPS-CT	Post Tsunami Pump Cross Tie Fails
	1.5E-1		PS-XHE-XM-PSEDG-STATIONA	Operator Fails to Start Post Tsunami Stationary EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
54	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-3		POWER-NORMAL	Normal Power Supply Fails
	1.0E-2		PS-BU-PUMPS-CF	Post Tsunami Backup Pumps Fail from Common Cause
	1.5E-1		PS-XHE-XM-PSEDG-STATIONA	Operator Fails to Start Post Tsunami Stationary EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
55	3.8E-7	< 0.01	0-SLOOP :4	
	1.0E+1		IE-SLOOP	Seismic Induced Loop (10/year)
	1.0E-4		BACKUP-FT-CF-FS	Common Cause Failure of All Fire Trucks

	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	5.0E-2		PS-EDG-MOBILE	Post Tsunami MOBILE EDG Fails
	1.5E-1		PS-XHE-XM-PSEDG-STATIONA	Operator Fails to Start Post Tsunami Stationary EDG
		EndState	->LR	Added through Event Tree Add
56	3.8E-7	< 0.01	0-SLOOP :4	
	1.0E+1		IE-SLOOP	Seismic Induced Loop (10/year)
	1.0E-4		BACKUP-FT-CF-FS	Common Cause Failure of All Fire Trucks
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	5.0E-2		PS-EDG-STATIONARY	Post Tsuname STATIONARY EDG Fails
	1.5E-1		PS-XHE-XM-PSEDG-MOBILE	Operator Fails to Start Post Tsunami Mobile EDG
		EndState	->LR	Added through Event Tree Add
57	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
58	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
59	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
60	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks

		EndState	->LRF	Added through Event Tree Add
61	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
62	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
63	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
64	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
65	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
66	2.8E-7	< 0.01	0-SPUMP-FAIL :4	

	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
67	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
68	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
69	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
70	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
71	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS

	1.0E-2		OCEANSIDE-FS-PUM[illegible]3	Ocean Side Backup Electrical Pum[illegible]2B-FS
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
72	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
73	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
74	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
75	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
	1.0E-4		U-TANK	Underground Tank
		EndState	->LRF	Added through Event Tree Add
76	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
	1.0E-4		U-TANK	Underground Tank

		EndState	->LRF	Added through Event Tree Add
77	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-4		FILTER-WATER-TANK	Filter Water Tank
	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
78	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-4		FILTER-WATER-TANK	Filter Water Tank
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
79	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		PS-BU-PUMPS-CT	Post Tsunami Pump Cross Tie Fails
	5.0E-4		PS-CF-OCEANSIDEPUMPS-FS	Common Cause Failure to Run of Ocean Side Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
80	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		PS-BU-PUMPS-CF	Post Tsunami Backup Pumps Fail from Common Cause
	5.0E-4		PS-CF-OCEANSIDEPUMPS-FS	Common Cause Failure to Run of Ocean Side Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
81	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		PS-BU-PUMPS-CF	Post Tsunami Backup Pumps Fail from Common Cause
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
	5.0E-2		PS-XHE-XR-X3PUMP	Operator Fails to Recover Pump to Ocean Side Tank in 24 Hours
	1.0E-2		S-X3-FS-PUMP	Pump from Filtered Tank to Ocean Side Tanks
		EndState	->LRF	Added through Event Tree Add
82	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		PS-BU-PUMPS-CT	Post Tsunami Pump Cross Tie Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks

	5.0E-2		PS-XHE-XR-X3PUMP	Operator Fails to Recover Pump to Ocean Side Tank in 24 Hours
	1.0E-2		S-X3-FS-PUMP	Pump from Filtered Tank to Ocean Side Tanks
		EndState	->LRF	Added through Event Tree Add
83	2.1E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-5		PS-CF-OCEANSIDEPUMPS-FR	Common Cause Failure to Run of Ocean Side Pumps
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
84	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
		EndState	->LR	Added through Event Tree Add
85	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
		EndState	->LR	Added through Event Tree Add
86	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
		EndState	->LR	Added through Event Tree Add
87	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
		EndState	->LR	Added through Event Tree Add
88	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
		EndState	->LR	Added through Event Tree Add
89	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
		EndState	->LR	Added through Event Tree Add
90	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam

	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
		EndState	->LR	Added through Event Tree Add
91	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
		EndState	->LR	Added through Event Tree Add
92	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
		EndState	->LR	Added through Event Tree Add
93	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-4		U-TANK	Underground Tank
		EndState	->LR	Added through Event Tree Add
94	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-4		FILTER-WATER-TANK	Filter Water Tank
		EndState	->LR	Added through Event Tree Add
95	1.9E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-4		BACKUP-FT-CF-FS	Common Cause Failure of All Fire Trucks
	1.0E-2		PS-BU-PUMPS-CF	Post Tsunami Backup Pumps Fail from Common Cause
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
		EndState	->LRF	Added through Event Tree Add
96	1.9E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-4		BACKUP-FT-CF-FS	Common Cause Failure of All Fire Trucks
	1.0E-2		PS-BU-PUMPS-CT	Post Tsunami Pump Cross Tie Fails
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
		EndState	->LRF	Added through Event Tree Add
97	1.4E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-2		PS-BU-PUMPS-2	Post Tsunami Backup Pump 2 Fails
	5.0E-5		PS-CF-OCEANSIDEPUMPS-FR	Common Cause Failure to Run of Ocean Side Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
98	1.4E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails

	5.0E-2		PS-BU-PUMPS-1	Post Tsunami Backup Pump 1 Fails
	5.0E-5		PS-CF-OCEANSIDEPUMPS-FR	Common Cause Failure to Run of Ocean Side Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
99	1.4E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-3		POWER-NORMAL	Normal Power Supply Fails
	1.0E-2		PS-BU-PUMPS-CT	Post Tsunami Pump Cross Tie Fails
	5.0E-2		PS-EDG-STATIONARY	Post Tsuname STATIONARY EDG Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add
100	1.4E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-3		POWER-NORMAL	Normal Power Supply Fails
	1.0E-2		PS-BU-PUMPS-CF	Post Tsunami Backup Pumps Fail from Common Cause
	5.0E-2		PS-EDG-STATIONARY	Post Tsuname STATIONARY EDG Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
		EndState	->LRF	Added through Event Tree Add

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Fukushima Units 1 to 3
Risk Analysis

US Nuclear Regulatory Commission

May 2, 2011

AAA/99

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Assumptions

- Analysis is on per reactor bases
  - Each unit is expected to have similar risk profile – it is recognized that Units 1, 2 and 3 have varying amounts of core damage and different status of reactor pressure vessels (RPV) and containments
- No analysis has been completed for spent fuel pools yet
- End state is large release (LR) caused by inadequate cooling of core material in reactor and/or containment
- Assumed time to LR is 10 hours
  - Based on estimated time without cooling that core material in RPV could melt through RPV bottom head
- Analysis performed using SAPHIRE code (version 8.0.7.13)
- Human Reliability Analysis (HRA) based on SPAR-H methodology

Changes from 04-29-2011 Version

- Added typhoon initiator
- Refined electrical system model by adding more detail
- Revised some basic event probabilities
- Created bases document
- Numbers have changed somewhat
- Risk insights have not changed

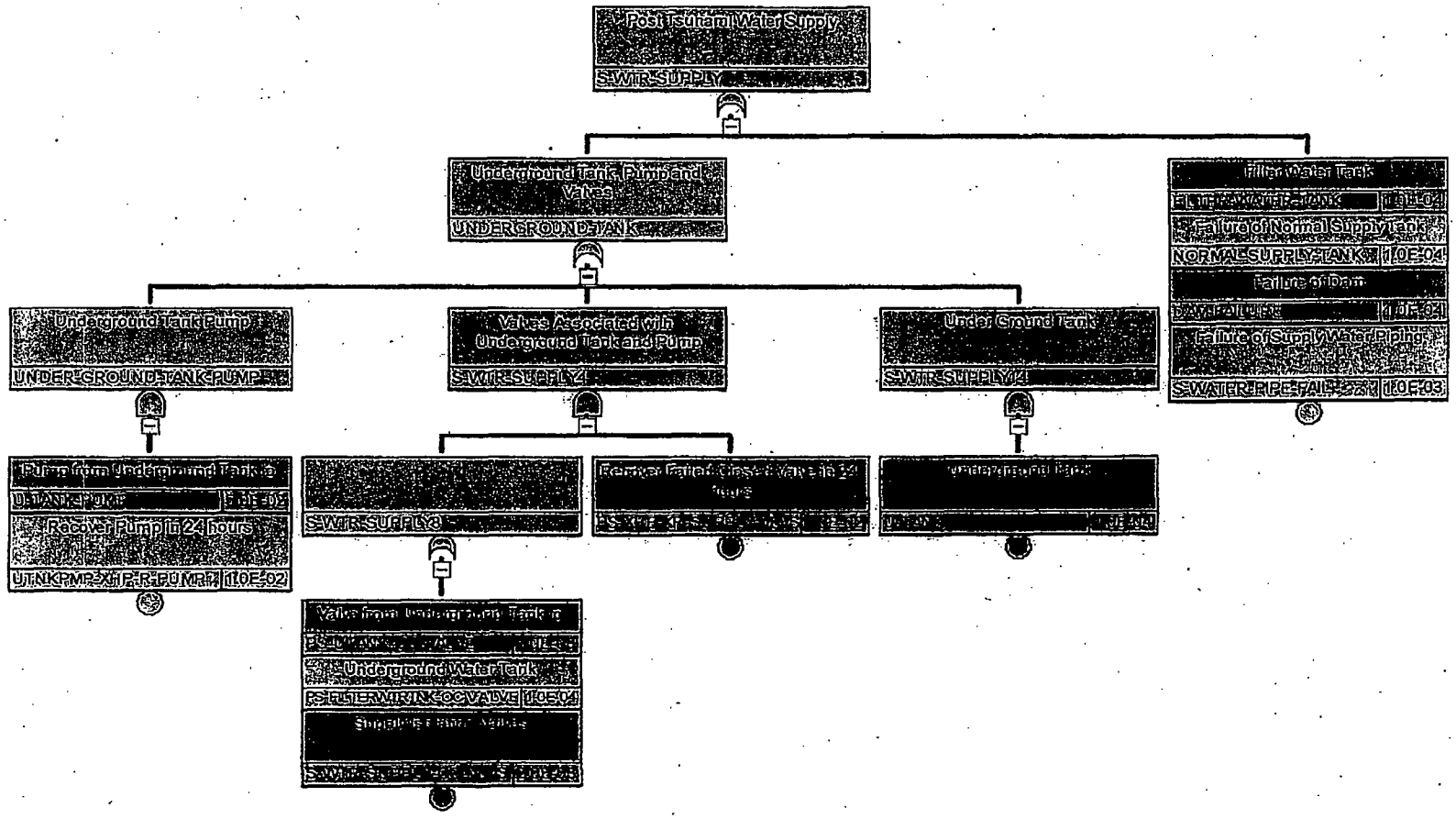
Initiating Events Evaluated

- "Normal" water supply source (1.9E-2/year) – see next slide
- Pump failure (5E-1/year)
- Hoses connecting pumps and injection piping (1E-1/year)
- Injection pipe failure (1.1E-2/year)
  - Feedwater pipe on Unit 1
  - Low pressure core injection or recirculation piping on Units 2 and 3
- RCS vent path (1E-2/year)
  - This is significant contributor and
  - If RPV is breached then failure probability is zero
  - If RPV is not breached then probability of unknown vent failing is difficult to estimate
- Loss of "normal" AC power (10/year)
  - Based on 1 event in 6 weeks
- Second tsunami (1.8E-3/year)
  - Based on 2 event in 1100 years
- Typhoon (2.5E-1/year)
  - Based on 16 events in 63 years (http://www.accuweather.com/blogs/news/story/47259/japan-typhoon-season-will-disa.asp)

"Normal" Water Supply from Dam to Pumps

- This is frequency (events per year) estimation of losing supply water to currently running injection pumps
- Equipment associated with initiating event frequency (IEF) estimation includes:
  - Dam
  - Underground tank
  - Underground tank pump and valves
  - Filter tank
  - Supply tank
- Result: 1.9E-3 per year

FT for Estimating IEF for Loss of Water Supply



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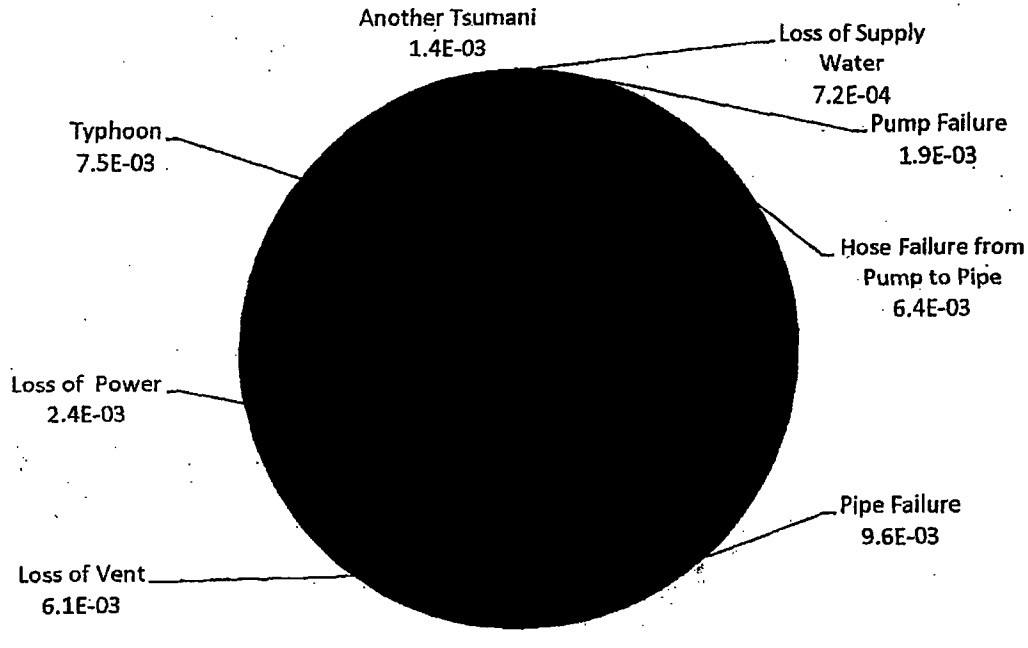
Insights from Results

- Precision of results is not important
- Risk results do give insights to vulnerabilities and where to allocate resources



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Largest Risk Contributors

- Injection pipe failure
- Injection hose failure
- Human actions
- Vent failure

- See next slide for ranking

Risk Significant Contributors

Fukushima Daiichi

05/02/2011 8:23:32 AM

Name	Prob-ability	RRR	Description	Fix
IE-PIPE-FAIL	1.1E-2	1.40	Post Tsunami Injection Path Pipe Failure	Add injection path through independent pipe
IE-TYPHOON	2.5E-1	1.30	Typhoon	No corrective fix possible - refine number
TYPHOON-LARGE-SF	1.0E-1	1.30	Split Fraction - Assumed only 10% of Storms Cause Damage	No corrective fix possible - refine number
TYPHOON-XHE-XL-RECOVER	3.0E-1	1.30	Operators recover after Typhoon	Improve operator training, create procedures, reduce stress
IE-HOSE	1.0E-1	1.20	Failure of Hose from Pump to Pipe	Protect hoses from, wear, falling objects, trucks, etc.
IE-SLOSS-VENT	1.0E-2	1.20	Loss of Post Tsunami Injection Vent Path	Reduce probability of lossing existing vent path
PS-XHE-XD-HOSE-REPLACE1	5.5E-2	1.20	Operator Identifies Broken Hose, How to Isolate ad Replaces	Improve operator training, create procedures, reduce stress
ALT-VENT1	5.0E-1	1.10	Alternate Vent Path Available	Establish a backup vent path as necessary
IE-SLOOP	1.0E+1	1.10	Seismic Induced Loop	No corrective fix possible - refine number
IE-SPUMP	5.0E-1	1.10	Post Tsunami Pump Fails	Improve reliability of pumps
OEP-XHE-XL-NR10H	5.1E-2	1.10	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS	Improve operator training, create procedures, reduce stress
PS-XHE-XM-ALT-SUPPLY	3.8E-1	1.10	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps	Improve operator training, create procedures, reduce stress
PS-XHE-XM-PIPEALIGN	7.5E-1	1.10	Operator Fails to Align Alternate Pipe Path	Improve operator training, create procedures, reduce stress
PS-XHE-XM-PSEDG-MOBILE	1.5E-1	1.10	Operator Fails to Start Post Tsunami Mobile (750kVA) EDG	Improve operator training, create procedures, reduce stress
PS-XHE-XM-PSEDG-STATIONA	1.5E-1	1.10	Operator Fails to Start Post Tsunami Stationary EDG	Improve operator training, create procedures, reduce stress
PS-XHE-XM-PSFT	1.5E-1	1.10	Operator Fails to Align Post Tsunami Fire Truck	Improve operator training, create procedures, reduce stress
PS-XHE-XM-PSFT-BU	7.5E-1	1.10	Operator Fails to Align Backup Fire Trucks	Improve operator training, create procedures, reduce stress

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Bases for the Fukushima Large Release Risk Model

Below are the Event Trees (ET) and the support Fault Trees (FT) for the model.

Figure 1: Hose from Pump to Pipe Fails

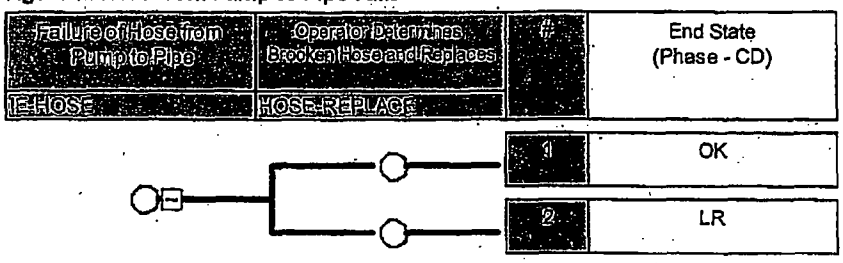


Figure 1a: FT for Hose Replacement

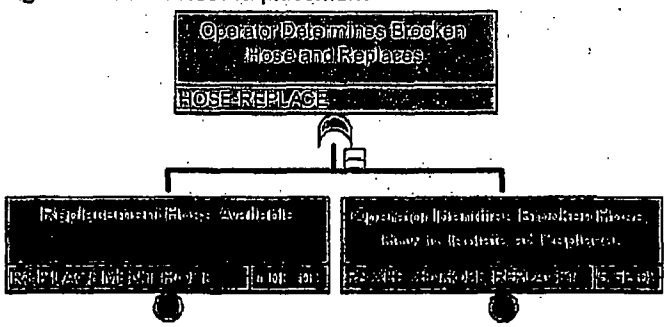


Figure 2: ET Loss of Offsite Power (LOOP) Caused by Seismic Event

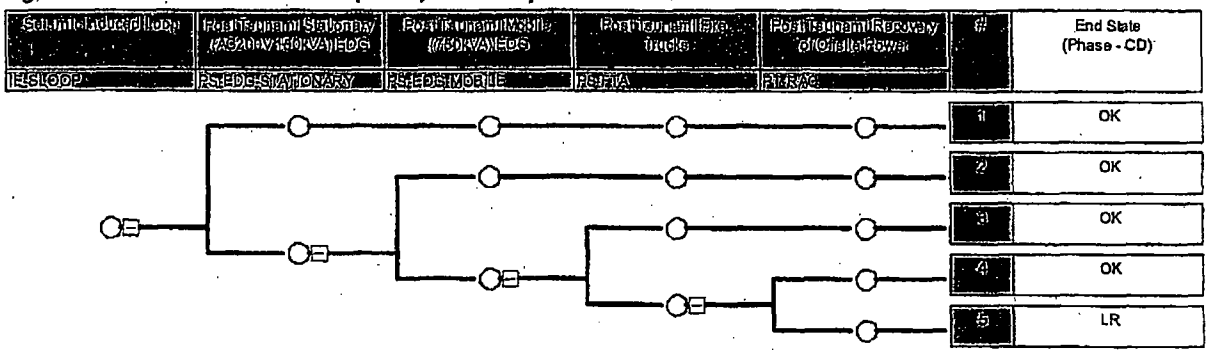


Figure 2a: FT Post Stationary Tsunami EDG

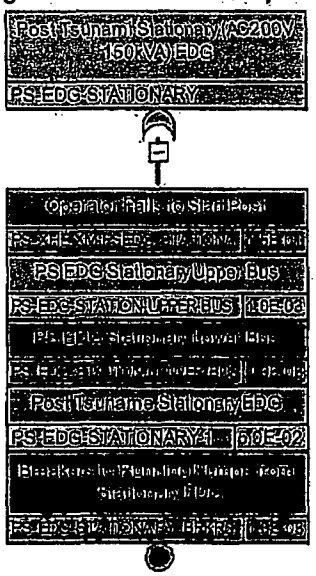


Figure 2b: FT Post Tsunami Mobile EDG

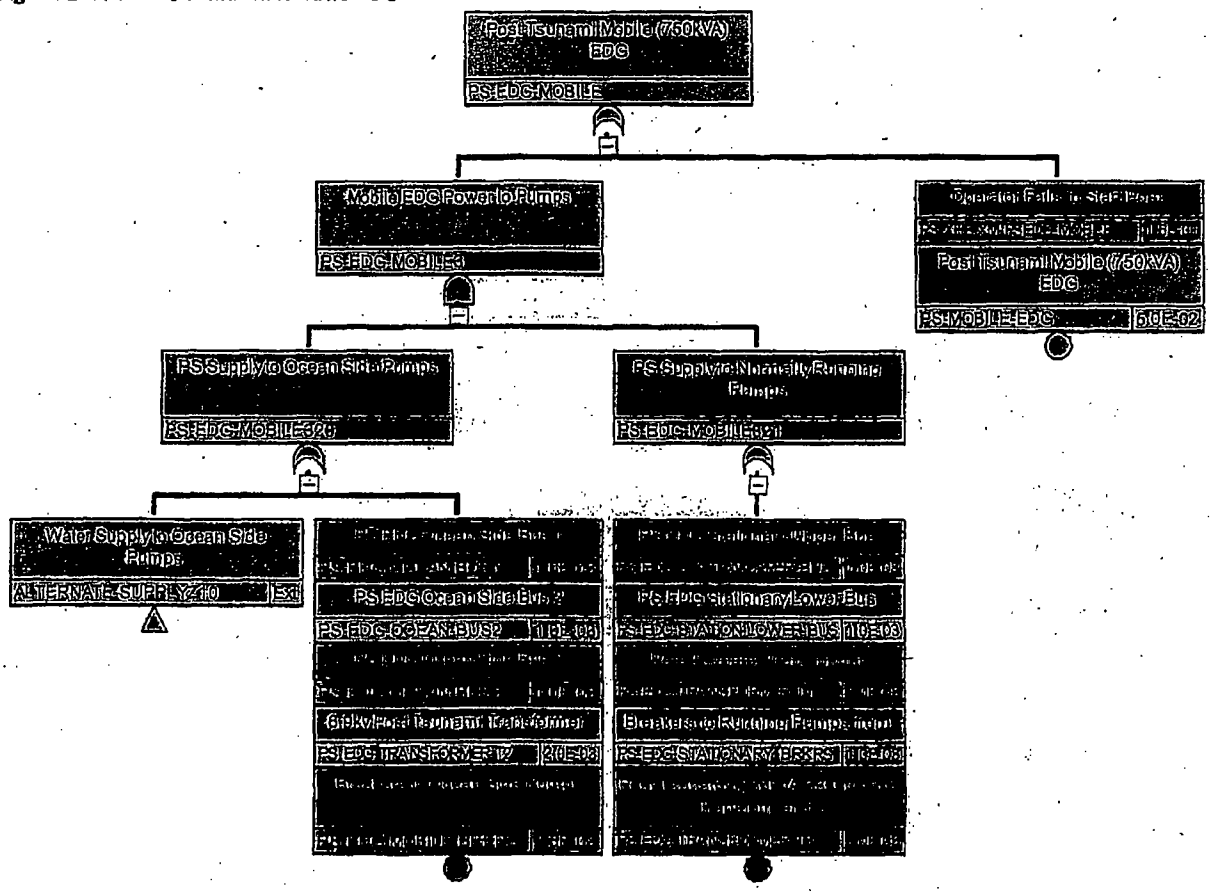


Figure 2b1: FT Alternate Water Supply (transfer)

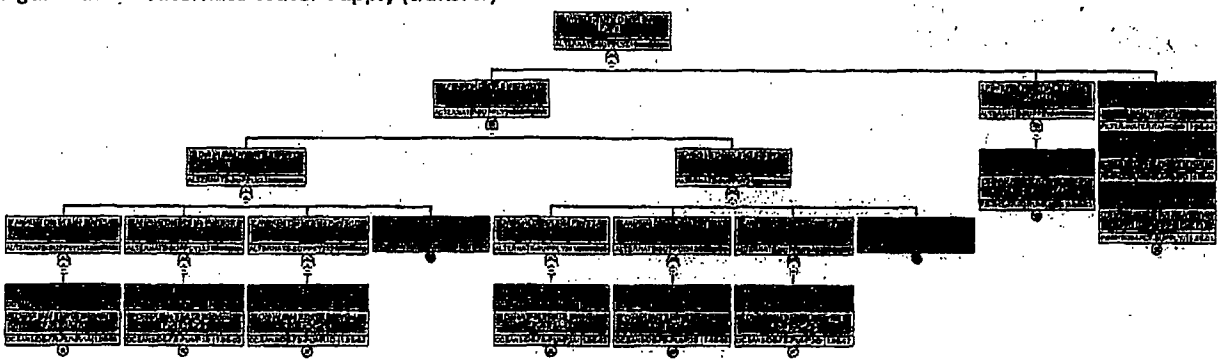


Figure 2c: FT Post Tsunami Fire Trucks

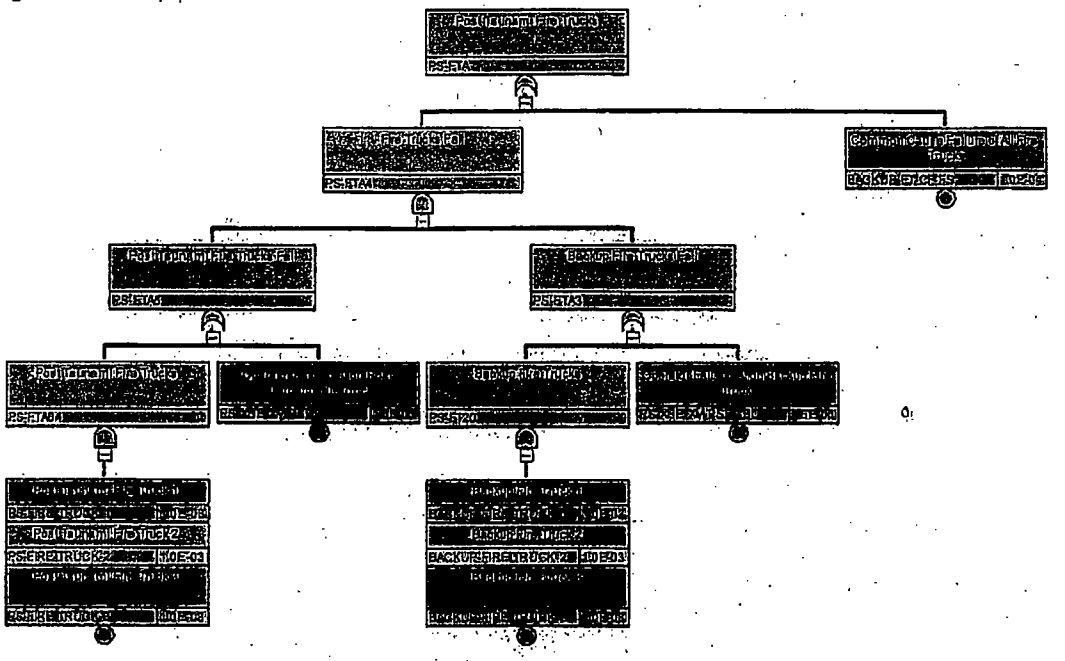


Figure 2c: FT Post Tsunami Recovery of Offsite Power

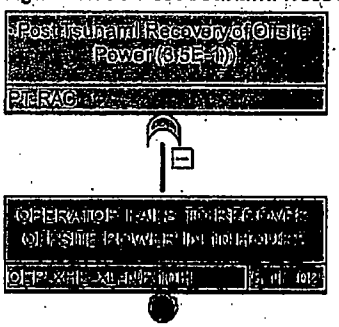


Figure 3: ET Loss of Vent Path

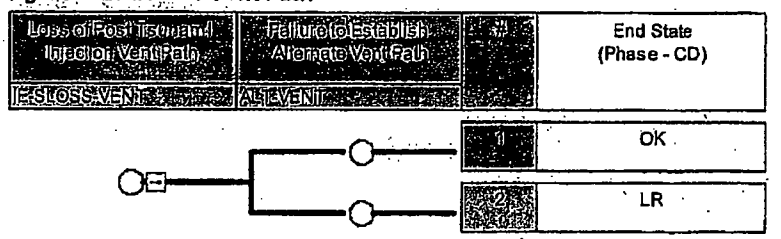


Figure 3a: FT Failure to Establish Alternate Vent Path

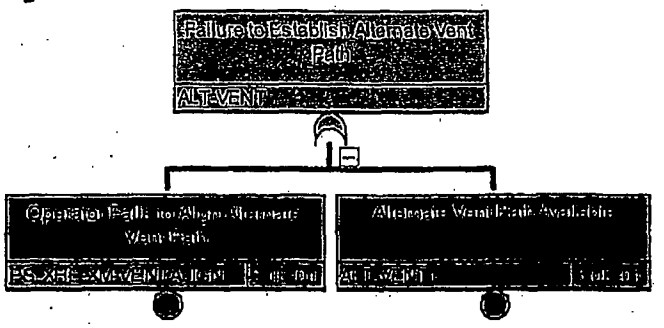


Figure 4: ET Loss of Water Supply to Running Pumps

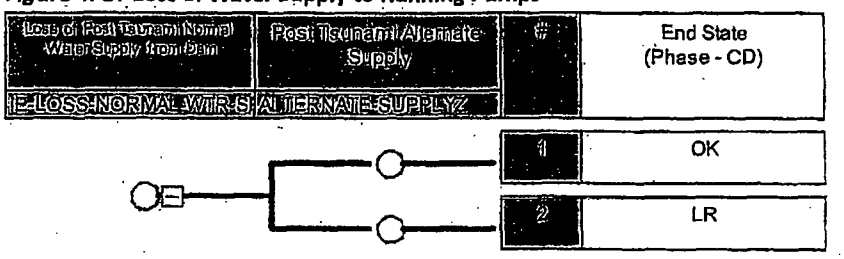


Figure 4a: FT Post Tsunami Alternate Water Supply

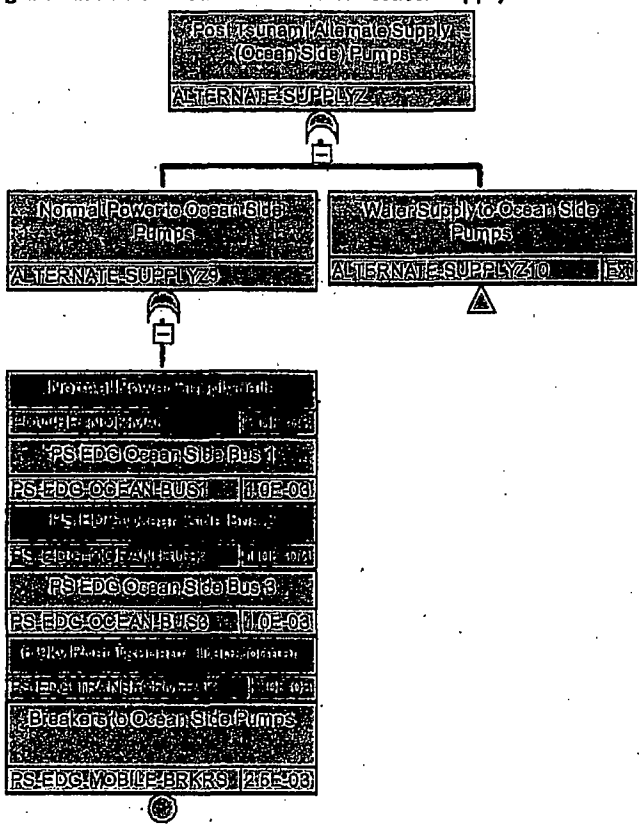


Figure 4a1: FT Water Supply to Ocean Side Pumps (transfer)

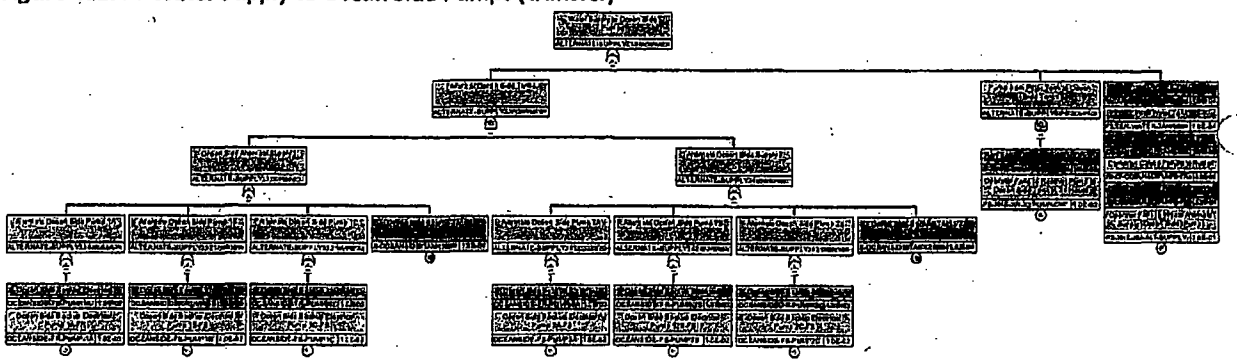


Figure 5: Pipe to RPV Fails

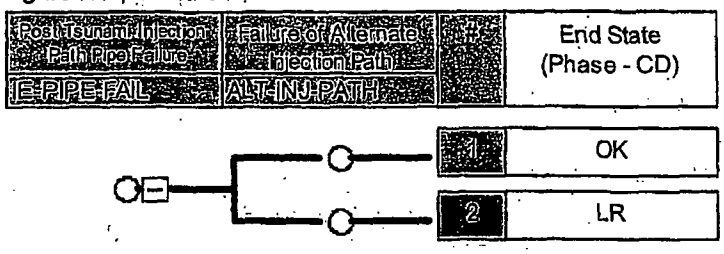


Figure 5a: Failure of Alternate Injection Path

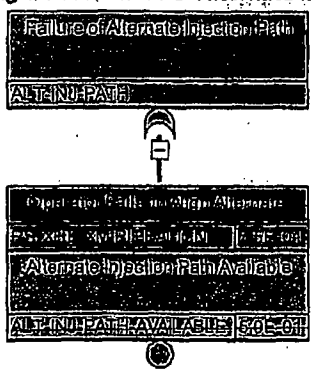


Figure 6: ET Running Pump Fails

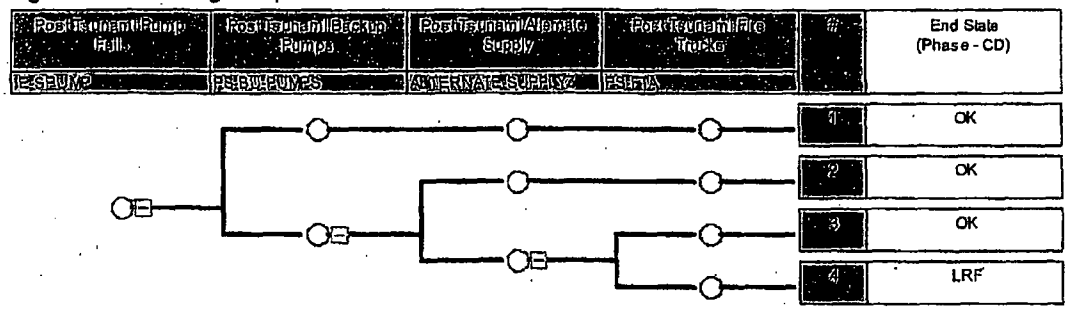


Figure 6a: FT Post Tsunami Backup Pumps

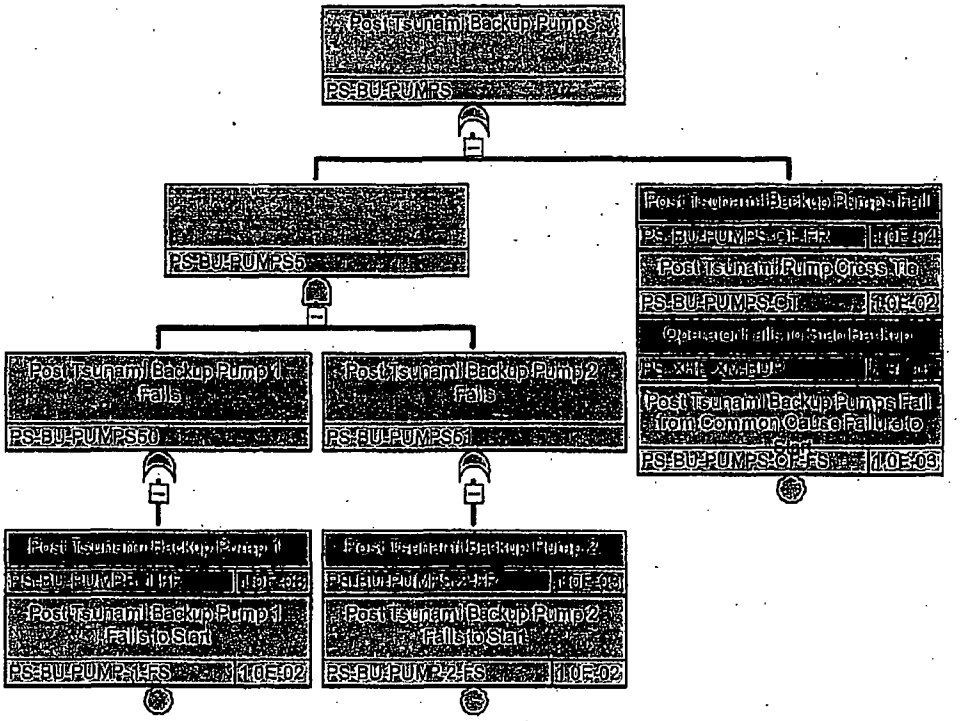


Figure 6b: FT Post Tsunami Alternate Supply

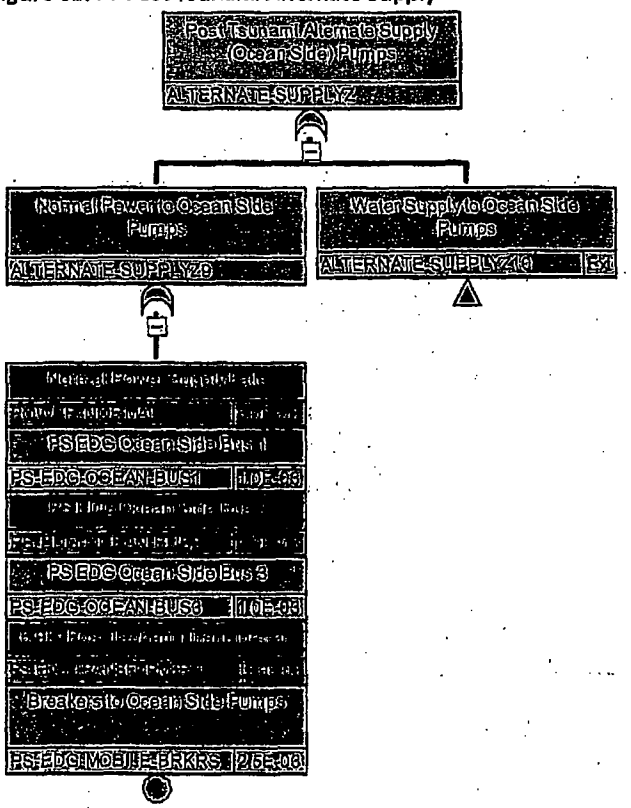


Figure 6b1: FT Post Tsunami Backup Pumps (transfer)

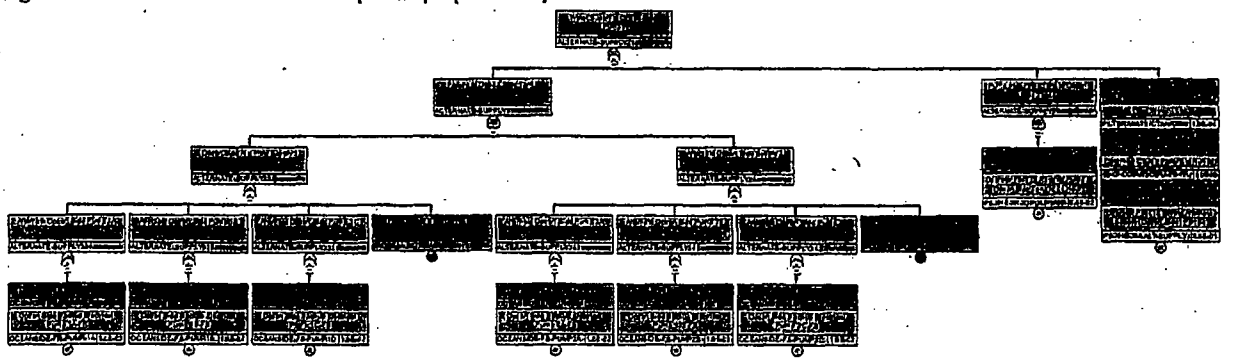


Figure 6c: FT Post Tsunami Fire Trucks

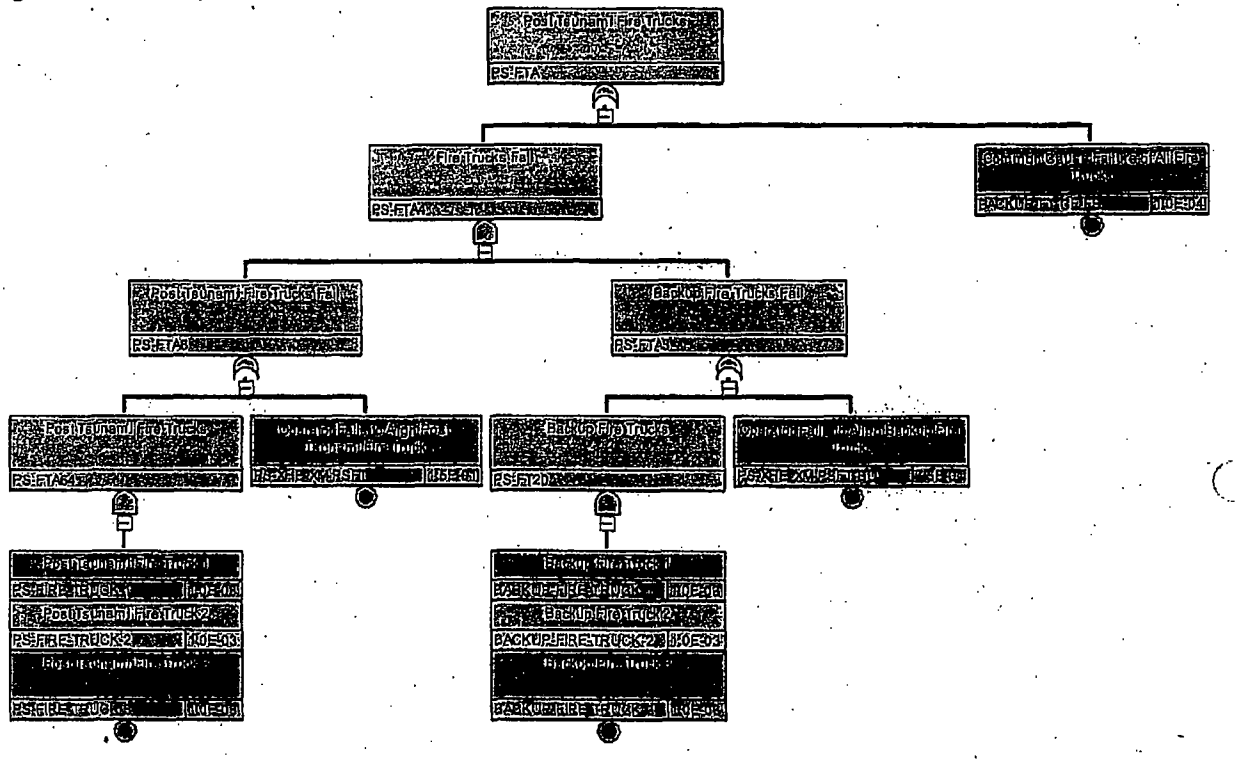


Figure 7: ET Second Tsunami

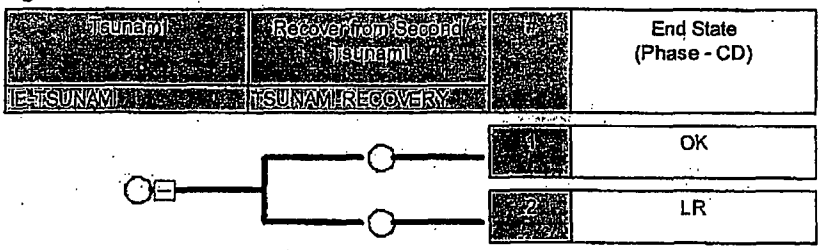


Figure 7a: FT Recover from Second Tsunami

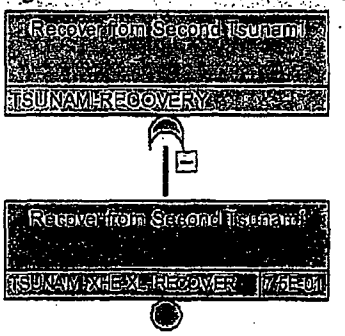


Figure 8: ET Typhoon Causes Loss of Running Pump

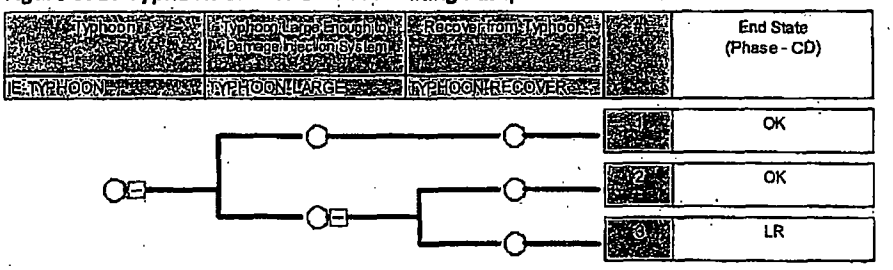


Figure 8a: FT Split Fraction of Typhoons Large to Cause Loss of Running Pumps

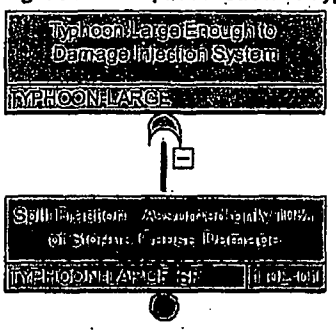
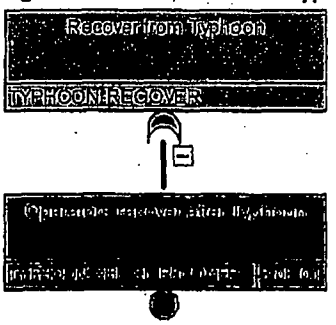


Figure 8b: FT Recover from Typhoon



Initiating Events	Description	Failure Frequency (events per Year)	Bases
IE-HOSE	Failure of Hose from Pump to Pipe	1.0E-01	It is assumed that these are standard canvas fire hoses. The hoses have been replaces once I assume because of wear. The hoses should not be subject to failure from a seismic event by any kind of cracking or stress failure. However, they are subject to fretting wear caused by a seismic event or other cause. They are also subject to vehicular damage or from falling objects both inside and outside the buildings. I don't have any bases for this number other than to say I believe it is a large number.
IE-PIPE-FAIL	Post Tsunami Injection Path Pipe Failure	1.0E-02	EPRI Pipe Rupture Frequency for Internal Flooding HELB PRA Rev. 2 Draft 1021089 Table 5-16 (Failure Rates and Cumulative Rupture Frequencies for BWR NPS ≤ 10" FWC) gives a pipe rupture frequency of 5.62e-6 per foot of pipe per reactor year. I estimated a total of 100 feet of pipe between injection point and reactor on Unit 1. This gives a pipe break frequency of ~6E-4 per year. This is based on pipe being at NOP, which it is not the case here, thus this number is conservative based on non-NOP conditions. I increased the probability by about an order of magnitude to 1E-2 based on uncertainty. This uncertainty is from potential degradation caused by a very large seismic event, the tsunami and finally the explosions. I also used the same frequency for Units 2 and 3 which use fire water and LPCI piping.
IE-SLOOP	Seismic Induced LOOP	1.0E+01	Fukushima Dalichi has experience one seismic induced LOOP since the tsunami. This was a quake of ~7.1 magnitude on April 11th. A cording to the Gutenberg-Richter Law additional large aftershocks can be expected. I choose a value of 10 based on readings on this topic.

IE-SLOSS-VENT	Loss of Post Tsunami Injection Vent Path	1.0E-02	All of the injection systems currently being used or available to be used are low pressure systems. Thus if the RCS vent is lost it will repressurize and the injection systems will not be able to inject. Currently how the RCSs are being vented is unclear. Early in the accidents it was assumed that SRV(s) has stuck open, thus the injection path could easily be lost. However, it is now hypothesized that the reactor recirculation pump seals have failed and also that the RPVs proper may have developed leaks through the bottom head penetrations. If these "holes" exist are they large enough to prevent repressurization? Actual status remains unknown. But this input remains to risk important to ignore, therefore, this initiator was added to the analysis. Without better information a failure rate could not be estimated. Therefore, I arbitrarily chose a value of 1% for this IEF.
IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam	1.9E-03	This was calculated using a fault tree (S-WTR-SUPPLY) composed of two tanks, a pump, several valves (assumed), piping and a dam. All tanks in this analysis were assumed to fail at 1E-4 based on these being non-safety related and therefore not as sound as a safety related CST (which has a failure probability of ~E-6). Piping was assumed to fail at 1E-3. The pump was assumed to fail at 5E-2. All valves in the lineup are currently in their required position and therefore, were assumed to fail at 1E-5. There are also human recovery actions (time for recovery assumed to be 24 hours as the tanks are assumed to contain 24 hours worth of inventory. The recovery HEPs were calculated using SPAR-H with values of 1E-2. The dam was assumed to fail at 1E-4 based on previous DRA analysis.
IE-SPUMP	Post Tsunami Pump Fails	5.0E-01	There are two continuously running commercial grade pumps currently running at the site. Either one of which if it fails will cause a loss of injection. IEEE-500, Chapter 11 Table 11.1.2.4 - All Service Composite of Standby, Alternating and Continuous Service Pumps recommends a fail to run probability of 1.2E-5 per hour and gives a failure probability of 5.8E-4 per hour for high probability. On an annual basis these are 0.1 and 5 respectively. These pumps are not nuclear grade, are being run throttled significantly and are in a harsh environment. Based on these factors I chose a failure probability of 0.5.
IE-TSUNAMI	Tsunami	1.8E-03	Fukushima has experience two large tsunamis since ~900 AD. Thus, 2/1100 years = 1.8E-3.

IE-TYPHOON	Typhoon near Fukushima Daiichi	2.5E-01	Web page reports 16 typhoons in last 63 years within 100 miles of Fukushima Daiichi. These are Category 1 or 2 storms. Link is: http://www.accuweather.com/blogs/news/story/47259/japan-typhoon-season-will-disa.asp

Component	Description	Prob. Of Failure	Bases
PS-BU-PUMPS-1-FR	Post Tsunami Backup Pump 1 Fails to Run	1.0E-03	IEEE-500, Chapter 11, Table 11.1.2.4.1 - All service, composite of standby, alternating and continuous service, recommends a failure to run rated of 12E-6 per hour. I gave the pump a mission time of 48 hours. This yields a probability of failure of 6E-4 per 48 hours. However, I increased the value to 1E-3 based on the pumps not being nuclear grade and the pumps being outside, subject to weather and being in less than nominal conditions.
PS-BU-PUMPS-2-FR	Post Tsunami Backup Pump 2 Fails to Run	1.0E-03	
PS-BU-PUMPS-CF-FR	Post Tsunami Backup Pumps Fail from Common Cause Failure to Run	1.0E-04	I simply set this as one order of magnitude less than the failure to run probability. I assume this is conservative and if it contributes to the dominate cutsets it will warrant additional refinement.
PS-BU-PUMP-1-FS	Post Tsunami Backup Pump 1 Fails to Start	1.0E-02	IEEE-500, Chapter 11, Table 11.1.2.4.1 - All service, composite of standby, alternating and continuous service, recommends a failure to start probability of 4.73E-3 per demand. However, I increased the value to 1E-2 based on the pumps not being nuclear grade and the pumps being outside, subject to weather and being in less than nominal conditions.
PS-BU-PUMP-2-FS	Post Tsunami Backup Pump 2 Fails to Start	1.0E-02	
PS-BU-PUMPS-CF-FS	Post Tsunami Backup Pumps Fail from Common Cause Failure to Start	1.0E-03	I simply set this as one order of magnitude less than the failure to run probability. I assume this is conservative and if it contributes to the dominate cutsets it will warrant additional refinement.
PS-EDG-STATIONARY-1	Post Tsunami Stationary EDG Fails	5.0E-02	IEEE-500, Chapter 4, Table 4.2.1.3 gives a recommended failure probability for a diesel driven generator of 2E-5 per hour or 1E-3 for a mission time of 48 hours (the time I'm assuming to recover offsite power). It also gives a high value of 1.9E-3 per hour or 1E-1 for a 48 hour mission time. However, these are commercial grade diesels not nuclear grade and they are in a harsh environment. Therefore, I've chosen a failure probability of 5E-2 for the 48 hour mission time

PS-EDG-MOBILE	Post Tsunami MOBILE EDG Fail	5.0E-02	IEEE-500, Chapter 4, Table 4.2.1.3 gives a recommended failure probability for a diesel driven generator of 2E-5 per hour or 1E-3 for a mission time of 48 hours (the time I'm assuming to recover offsite power). It also gives a high value of 1.9E-3 per hour or 1E-1 for a 48 hour mission time. However, these are commercial grade diesels not nuclear grade and they are in a harsh environment. Therefore, I've chosen a failure probability of 5E-2 for the 48 hour mission time
PS-EDG-STATION-UPPER-BUS	PS EDG Stationary Upper Bus (210V) Fails	1.0E-03	Brunswick SPAR model gives a failure probability for AC bus (ACP-BAC-LP-E6) of 9.6E-5 for a mission time of 24 hours. This yields a probability of 1.9E-5 for my 48 hour mission time. But this is for a nuclear grade bus inside a building not a commercial grade bus without protective relaying outside. Therefore, I increased the probability to 1E-3.
PS-EDG-STATION-LOWER-BUS	PS EDG Stationary Lower Bus (210) Fails	1.0E-03	
PS-EDG-OCEAN-BUS1	PS EDG Ocean Side Bus 1	1.0E-03	
PS-EDG-OCEAN-BUS2	PS EDG Ocean Side Bus 2	1.0E-03	
PS-EDG-OCEAN-BUS3	PS EDG Ocean Side Bus 3	1.0E-03	
PS-EDG-STATIONARY-BRKRS	Breakers to Running Pumps from Stationary EDG	1.0E-03	Brunswick SPAR model gives a failure probability for AC breaker of (ACP-CRB-CO-AU9) with a mission time of 3.6E-6. Converting this to a mission time of 48 hours yields 7.2E-6. However, this is for a nuclear grade breaker inside a building not a commercial breaker outside. Therefore, I used a value of 3E-4 per breaker. However, there are 4 breakers in this circuit. Therefore, I used a value of 1E-3 for this BE.
PS-EDG-MOBILE-BRKRS	Breakers to Ocean Side Pumps	2.5E-03	Brunswick SPAR model gives a failure probability for AC breaker of (ACP-CRB-CO-AU9) with a mission time of 3.6E-6. Converting this to a mission time of 48 hours yields 7.2E-6. However, this is for a nuclear grade breaker inside a building not a commercial breaker outside. Therefore, I used a value of 3E-4 per breaker. However, there are 10 breakers in this circuit. Therefore, I used a value of 2.5E-3 for this BE.
PS-EDG-TRANSFORMER-T1	Post Tsunami Transformer (440v/210v) T1 to Running Pumps	2.2E-03	Brunswick SPAR Model gives a failure probability for an AC transformer (ACP-TFM-FC-E1E5) of 2.2E-5 for a mission time of 24 hours. Converting that to a mission time of 48 hours yields 4.3 E-5. However, this is for a nuclear grade transformer in a normal environment. This transformer is a commercial grade transformer in a harsh environment. Therefore, I increased the probability to 2E-3.
PS-EDG-TRANSFORMER-T2	6.9 Kv Transformer (T2) to Ocean Side Pumps	2.2E-03	
PS-EDG-TRANSFORMER-T3	Post Tsunami (350kVA 6600/440V) Transformer T3	2.2E-03	
TYPHOON-LARGE-SF	Split Fraction - Assumed only 10% of Storms Cause Damage	1.0E-01	I simply assumed that 10% of Typhoons that reached the site would be strong enough to stop the running pumps.

REPLACEMENT-HOSE	Replacement Hose Available	1.0E-02	I simply assumed that there was a 1% chance that there would not be sufficient correct replacement hose available.
OCEANSIDE-FR-PUMP-1A	Ocean Side Backup Electrical Pump 1A-FR	1.0E-03	IEEE-500, Chapter 11, Table 11.1.2.4.1 - All service, composite of standby, alternating and continuous service, recommends a failure to run rated of 12E-6 per hour. I gave the pump a mission time of 48 hours. This yields a probability of failure of 6E-4 per 48 hours. However, I increased the value to 1E-3 based on the pumps not being nuclear grade and the pumps being outside, subject to weather and being in less than nominal conditions.
OCEANSIDE-FR-PUMP-1B	Ocean Side Backup Electrical Pump 1B-FR	1.0E-03	
OCEANSIDE-FR-PUMP-1C	Ocean Side Backup Electrical Pump 1C-FR	1.0E-03	
OCEANSIDE-FR-PUMP-2A	Ocean Side Backup Electrical Pump 2A-FR	1.0E-03	
OCEANSIDE-FR-PUMP-2B	Ocean Side Backup Electrical Pump 2B-FR	1.0E-03	
OCEANSIDE-FR-PUMP-2C	Ocean Side Backup Electrical Pump 2C-FR	1.0E-03	
PS-CF-OCEANSIDEPUMPS-FR	Common Cause Failure to Run of Ocean Side Pumps	1.0E-04	I simply set this as one order of magnitude less than the failure to run probability. I assume this is conservative and if it contributes to the dominate cutsets it will warrant additional refinement.
OCEANSIDE-FS-PUMP-1A	OCEANSIDE-FS-PUMP-1A	1.0E-02	IEEE-500, Chapter 11, Table 11.1.2.4.1 - All service, composite of standby, alternating and continuous service, recommends a failure to start probability of 4.73E-3 per demand. However, I increased the value to 1E-2 based on the pumps not being nuclear grade and the pumps being outside, subject to weather and being in less than nominal conditions.
OCEANSIDE-FS-PUMP-1B	OCEANSIDE-FS-PUMP-1B	1.0E-02	
OCEANSIDE-FS-PUMP-1C	OCEANSIDE-FS-PUMP-1C	1.0E-02	
OCEANSIDE-FS-PUMP-2A	OCEANSIDE-FS-PUMP-2A	1.0E-02	
OCEANSIDE-FS-PUMP-2B	OCEANSIDE-FS-PUMP-2B	1.0E-02	
OCEANSIDE-FS-PUMP-2C	OCEANSIDE-FS-PUMP-2C	1.0E-02	
PS-CF-OCEANSIDEPUMPS-FS	Common Cause Failure to Start of Ocean Side Pumps	1.0E-03	I simply set this as one order of magnitude less than the failure to run probability. I assume this is conservative and if it contributes to the dominate cutsets it will warrant additional refinement.
FILTER-WATER-TANK	Filter Water Tank	1.0E-04	A safety grade tank such as a condensate storage tank (CST) has a failure probability on the order of 1E-6. The tanks in question at Fukushima Dalichi are not safety grade tanks and are in a harsh environment. Therefore, I chose a failure frequency of 1E-4. This may be conservative but the tanks are not showing up as risk significant so refining this value is not warrented at this time.
U-TANK	Underground Tank	1.0E-04	
S-OCEANSIDE-TANK1	Ocean Side Backup Tank 1	1.0E-04	
S-OCEANSIDE-TANK2	Ocean Side Backup Tanks 2	1.0E-04	
PURE-WATER-TANK	Failure of Pure Water Tank	1.0E-04	
BACKUP-FIRE-TRUCK-1	Backup Fire Truck 1	1.0E-03	Don't have a good bases for these numbers, therefore, I based on my experience with cars. However, the document found at the below link (Scottsdale Apparatus Replacement) gives an "operational down time" rate of 20% for a fleet of fire trucks. However, this seams high. (http://www.usfa.dhs.gov/pdf/efop/efo41147.pdf)
BACKUP-FIRE-TRUCK-2	Backup Fire Truck 2	1.0E-03	
BACKUP-FIRE-TRUCK-3	Backup Fire Truck 2	1.0E-03	
BACKUP-FIRE-TRUCK-1	Backup Fire Truck 1	1.0E-03	
BACKUP-FIRE-TRUCK-2	Backup Fire Truck 2	1.0E-03	
BACKUP-FIRE-TRUCK-3	Backup Fire Truck 3	1.0E-03	

Cut Set Report - ET-				
11 9:51:55 AM				
#	Prob/ Freq	Total %	Cut Set	Description
Tota	3.6E-2	100	Displaying 100 of 595 Cut Sets.	
1	8.2E-3	23	0-SPIPE-FAIL :2	
	1.1E-2		IE-PIPE-FAIL	Post Tsunami Injection Path Pipe Failure
	7.5E-1		PS-XHE-XM-PIPEALIGN	Operator Fails to Align Alternate Pipe Path
2	7.5E-3	21	0-TYPHOON :3	
	2.5E-1		IE-TYPHOON	Typhoon
	1.0E-1		TYPHOON-LARGE-SF	Split Fraction - Assumed only 10% of Storms Cause Damage
	3.0E-1		TYPHOON-XHE-XL-RECOVER	Operators recover after Typhoon
3	5.5E-3	15	0-SPIPE-FAIL :2	
	1.1E-2		IE-PIPE-FAIL	Post Tsunami Injection Path Pipe Failure
	5.0E-1		ALT-INJ-PATH-AVAILABLE	Alternate Injection Path Available
4	5.5E-3	15	0-HOSE-FAIL :2	
	1.0E-1		IE-HOSE	Failure of Hose from Pump to Pipe
	5.5E-2		PS-XHE-XD-HOSE-REPLACE1	Operator Identifies Brooken Hose, How to Isolate ad Replaces
5	5.0E-3	14	0-SLOSS-VENT :2	
	1.0E-2		IE-SLOSS-VENT	Loss of Post Tsunami Injection Vent Path
	5.0E-1		ALT-VENT1	Alternate Vent Path Available
6	2.1E-3	5.9	0-SLOSS-VENT :2	
	1.0E-2		IE-SLOSS-VENT	Loss of Post Tsunami Injection Vent Path
	2.1E-1		PS-XHE-XM-VENTALIGN	Operator Fails to Align Alternate Vent Path
7	1.6E-3	4.4	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
8	1.4E-3	3.8	0-TSUNAMI :2	
	1.8E-3		IE-TSUNAMI	Tsunami
	7.5E-1		TSUNAMI-XHE-XL-RECOVER	Recover from Second Tsunami
9	1.3E-3	3.6	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.5E-1		PS-XHE-XM-PSEDG-MOBILE	Operator Fails to Start Post Tsunami Mobile (750kVA) EDG
	1.5E-1		PS-XHE-XM-PSEDG-STATIONA	Operator Fails to Start Post Tsunami Stationary EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
10	1.0E-3	2.8	0-HOSE-FAIL :2	
	1.0E-1		IE-HOSE	Failure of Hose from Pump to Pipe
	1.0E-2		REPLACEMENT-HOSE	Replacement Hose Available
11	7.1E-4	2	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
12	4.3E-4	1.2	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	5.0E-2		PS-MOBILE-EDG	Post Tsunami Mobile (750kVA) EDG
	1.5E-1		PS-XHE-XM-PSEDG-STATIONA	Operator Fails to Start Post Tsunami Stationary EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
13	4.3E-4	1.2	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	5.0E-2		PS-EDG-STATIONARY-1	Post Tsuname Stationary EDG Fails
	1.5E-1		PS-XHE-XM-PSEDG-MOBILE	Operator Fails to Start Post Tsunami Mobile (750kVA) EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
14	2.1E-4	0.59	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails

	1.0E-2		PS-BU-PUMPS-CT	Post Tsunami Pump Cross Tie Fails
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
15	1.4E-4	0.4	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	5.0E-2		PS-EDG-STATIONARY-1	Post Tsuname Stationary EDG Fails
	5.0E-2		PS-MOBILE-EDG	Post Tsunami Mobile (750kVA) EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
16	2.1E-5	0.06	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-EDG-STATION-UPPER-BUS	PS EDG Stationary Upper Bus (210V) Fails
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
17	2.1E-5	0.06	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOU
	1.0E-3		PS-EDG-STATION-LOWER-	PS EDG Stationary Lower Bus (210) Fails
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
18	2.1E-5	0.06	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-EDG-STATIONARY-BRKRS	Breakers to Running Pumps from Stationary EDG
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
19	2.1E-5	0.06	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-3		PS-BU-PUMPS-CF-FS	Post Tsunami Backup Pumps Fail from Common Cause Failure to
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
20	2.1E-5	0.06	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-3		POWER-NORMAL	Normal Power Supply Fails
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
21	1.1E-5	0.03	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	2.5E-3		PS-EDG-MOBILE-BRKRS	Breakers to Ocean Side Pumps
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
22	9.5E-6	0.03	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	5.0E-3		POWER-NORMAL	Normal Power Supply Fails
23	8.6E-6	0.02	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-EDG-STATION-UPPER-BUS	PS EDG Stationary Upper Bus (210V) Fails
	1.5E-1		PS-XHE-XM-PSEDG-MOBILE	Operator Fails to Start Post Tsunami Mobile (750kVA) EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
24	8.6E-6	0.02	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop

	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-EDG-STATION-LOWER-	PS EDG Stationary Lower Bus (210) Fails
	1.5E-1		PS-XHE-XM-PSEDG-MOBILE	Operator Fails to Start Post Tsunami Mobile (750kVA) EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
25	8.6E-6	0.02	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-EDG-STATIONARY-BRKRS	Breakers to Running Pumps from Stationary EDG
	1.5E-1		PS-XHE-XM-PSEDG-MOBILE	Operator Fails to Start Post Tsunami Mobile (750kVA) EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
26	8.4E-6	0.02	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	2.0E-3		PS-EDG-TRANSFORMER-T2	6.9kv Post Tsunami Transformer (750k VA)T2
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
27	6.4E-6	0.02	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	2.0E-3		PS-EDG-TRANSFORMER-T3	Post Tsunami (350kVA 6600/440V) Transformer T3
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSEDG-STATIONA	Operator Fails to Start Post Tsunami Stationary EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
28	6.4E-6	0.02	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	2.0E-3		PS-EDG-TRANSFORMER-T1	Post Tsunami Transformer (440v/210v) T1 to Running Pumps
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSEDG-STATIONA	Operator Fails to Start Post Tsunami Stationary EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
29	4.7E-6	0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	2.5E-3		PS-EDG-MOBILE-BRKRS	Breakers to Ocean Side Pumps
30	4.2E-6	0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-3		PS-CF-OCEANSIDEPUMPS-FS	Common Cause Failure to Start of Ocean Side Pumps
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
31	4.2E-6	0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-3		PS-EDG-OCEAN-BUS3	PS EDG Ocean Side Bus 3
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
32	4.2E-6	0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-3		PS-EDG-OCEAN-BUS1	PS EDG Ocean Side Bus 1
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
33	4.2E-6	0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-3		PS-EDG-OCEAN-BUS2	PS EDG Ocean Side Bus 2
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
34	3.8E-6	0.01	0-SLOSS-WTR-SUPPLY :2	

	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	2.0E-3		PS-EDG-TRANSFORMER-T2	6.9kv Post Tsunami Transformer (750k VA)T2
35	2.9E-6	0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-EDG-STATION-UPPER-BUS	PS EDG Stationary Upper Bus (210V) Fails
	5.0E-2		PS-MOBILE-EDG	Post Tsunami Mobile (750kVA) EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
36	2.9E-6	0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-EDG-STATION-LOWER-	PS EDG Stationary Lower Bus (210) Fails
	5.0E-2		PS-MOBILE-EDG	Post Tsunami Mobile (750kVA) EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
37	2.9E-6	0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-EDG-STATIONARY-BRKRS	Breakers to Running Pumps from Stationary EDG
	5.0E-2		PS-MOBILE-EDG	Post Tsunami Mobile (750kVA) EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
38	2.8E-6	0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-3		POWER-NORMAL	Normal Power Supply Fails
	1.0E-2		PS-BU-PUMPS-CT	Post Tsunami Pump Cross Tie Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
39	2.1E-6	0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	5.0E-2		PS-EDG-STATIONARY-1	Post Tsuname Stationary EDG Fails
	2.0E-3		PS-EDG-TRANSFORMER-T1	Post Tsunami Transformer (440v/210v) T1 to Running Pumps
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
40	2.1E-6	0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HO
	5.0E-2		PS-EDG-STATIONARY-1	Post Tsuname Stationary EDG Fails
	2.0E-3		PS-EDG-TRANSFORMER-T3	Post Tsunami (350kVA 6600/440V) Transformer T3
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
41	2.1E-6	0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
	5.0E-2		PS-XHE-XR-X3PUMP	Operator Fails to Recover Pump to Ocean Side Tank in 24 Hours
	1.0E-2		S-X3-FS-PUMP	Pump from Filtered Tank to Ocean Side Tanks
42	2.1E-6	0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-4		PS-BU-PUMPS-CF-FR	Post Tsunami Backup Pumps Fail from Common Cause Failure to Run
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
43	2.1E-6	0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		PS-BU-PUMP-1-FS	Post Tsunami Backup Pump 1 Fails to Start
	1.0E-2		PS-BU-PUMP-2-FS	Post Tsunami Backup Pump 2 Fails to Start

	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
44	1.9E-6	0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-3		PS-EDG-OCEAN-BUS1	PS EDG Ocean Side Bus 1
45	1.9E-6	0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-3		PS-EDG-OCEAN-BUS2	PS EDG Ocean Side Bus 2
46	1.9E-6	0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-3		PS-EDG-OCEAN-BUS3	PS EDG Ocean Side Bus 3
47	1.9E-6	0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-3		PS-CF-OCEANSIDEPUMPS-FS	Common Cause Failure to Start of Ocean Side Pumps
48	1.4E-6	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-4		BACKUP-FT-CF-FS	Common Cause Failure of All Fire Trucks
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
49	1.4E-6	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		PS-BU-PUMPS-CT	Post Tsunami Pump Cross Tie Fails
	2.5E-3		PS-EDG-MOBILE-BRKRS	Breakers to Ocean Side Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
50	1.1E-6	< 0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	1.0E-4		BACKUP-FT-CF-FS	Common Cause Failure of All Fire Trucks
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.5E-1		PS-XHE-XM-PSEDG-MOBILE	Operator Fails to Start Post Tsunami Mobile (750kVA) EDG
	1.5E-1		PS-XHE-XM-PSEDG-STATIONA	Operator Fails to Start Post Tsunami Stationary EDG
51	1.1E-6	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		PS-BU-PUMPS-CT	Post Tsunami Pump Cross Tie Fails
	2.0E-3		PS-EDG-TRANSFORMER-T2	6.9kv Post Tsunami Transformer (750k VA)T2
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
52	9.5E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	5.0E-2		PS-XHE-XR-X3PUMP	Operator Fails to Recover Pump to Ocean Side Tank in 24 Hours
	1.0E-2		S-X3-FS-PUMP	Pump from Filtered Tank to Ocean Side Tanks
53	5.6E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		PS-BU-PUMPS-CT	Post Tsunami Pump Cross Tie Fails
	1.0E-3		PS-CF-OCEANSIDEPUMPS-FS	Common Cause Failure to Start of Ocean Side Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
54	5.6E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		PS-BU-PUMPS-CT	Post Tsunami Pump Cross Tie Fails
	1.0E-3		PS-EDG-OCEAN-BUS2	PS EDG Ocean Side Bus 2
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
55	5.6E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		PS-BU-PUMPS-CT	Post Tsunami Pump Cross Tie Fails
	1.0E-3		PS-EDG-OCEAN-BUS1	PS EDG Ocean Side Bus 1
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
56	5.6E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails

	1.0E-2		PS-BU-PUMPS-CT	Post Tsunami Pump Cross Tie Fails
	1.0E-3		PS-EDG-OCEAN-BUS3	PS EDG Ocean Side Bus 3
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
57	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-4		PS-CF-OCEANSIDEPUMPS-FR	Common Cause Failure to Run of Ocean Side Pumps
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
58	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-4		FILTER-WATER-TANK	Filter Water Tank
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
59	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
	1.0E-4		U-TANK	Underground Tank
60	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
61	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
62	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
63	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
64	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
65	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump

	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
66	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
67	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
68	4.2E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
	7.5E-2		PS-XHE-XM-BUP	Operator Fails to Start Backup Pump
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
69	3.8E-7	< 0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	1.0E-4		BACKUP-FT-CF-FS	Common Cause Failure of All Fire Trucks
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	5.0E-2		PS-MOBILE-EDG	Post Tsunami Mobile (750kVA) EDG
	1.5E-1		PS-XHE-XM-PSEDG-STATIONA	Operator Fails to Start Post Tsunami Stationary EDG
70	3.8E-7	< 0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	1.0E-4		BACKUP-FT-CF-FS	Common Cause Failure of All Fire Trucks
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	5.0E-2		PS-EDG-STATIONARY-1	Post Tsuname Stationary EDG Fails
	1.5E-1		PS-XHE-XM-PSEDG-MOBILE	Operator Fails to Start Post Tsunami Mobile (750kVA) EDG
71	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	5.0E-3		POWER-NORMAL	Normal Power Supply Fails
	1.0E-3		PS-BU-PUMPS-CF-FS	Post Tsunami Backup Pumps Fail from Common Cause Failure to
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
72	2.8E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		PS-BU-PUMPS-CT	Post Tsunami Pump Cross Tie Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
	5.0E-2		PS-XHE-XR-X3PUMP	Operator Fails to Recover Pump to Ocean Side Tank in 24 Hours
	1.0E-2		S-X3-FS-PUMP	Pump from Filtered Tank to Ocean Side Tanks
73	2.1E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		PS-BU-PUMP-1-FS	Post Tsunami Backup Pump 1 Fails to Start
	1.0E-3		PS-BU-PUMPS-2-FR	Post Tsunami Backup Pump 2 Fails to Run
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
74	2.1E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-2		PS-BU-PUMP-2-FS	Post Tsunami Backup Pump 2 Fails to Start
	1.0E-3		PS-BU-PUMPS-1-FR	Post Tsunami Backup Pump 1 Fails to Run
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks

75	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
76	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
77	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2C	Ocean Side Backup Electrical Pump 2C-FS
78	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
79	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
80	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2B	Ocean Side Backup Electrical Pump 12B-FS
81	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1A	Ocean Side Backup Electrical Pump 1A-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
82	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1B	Ocean Side Backup Electrical Pump 1B-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
83	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-2		OCEANSIDE-FS-PUMP1C	Ocean Side Backup Electrical Pump 1C-FS
	1.0E-2		OCEANSIDE-FS-PUMP2A	Ocean Side Backup Electrical Pump 2A-FS
84	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-4		PS-CF-OCEANSIDEPUMPS-FR	Common Cause Failure to Run of Ocean Side Pumps
85	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-4		U-TANK	Underground Tank
86	1.9E-7	< 0.01	0-SLOSS-WTR-SUPPLY :2	
	1.9E-3		IE-LOSS-NORMAL-WTR-SPPLY	Loss of Post Tsunami Normal Water Supply from Dam
	1.0E-4		FILTER-WATER-TANK	Filter Water Tank
87	1.9E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-4		BACKUP-FT-CF-FS	Common Cause Failure of All Fire Trucks
	1.0E-2		PS-BU-PUMPS-CT	Post Tsunami Pump Cross Tie Fails
	3.8E-1		PS-XHE-XM-ALT-SUPPLY	Operator Fails to Initiate Alternate Supply (Ocean Side) Pumps
88	1.4E-7	< 0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	2.5E-3		PS-EDG-MOBILE-BRKRS	Breakers to Ocean Side Pumps
	1.0E-3		PS-EDG-STATION-UPPER-BUS	PS EDG Stationary Upper Bus (210V) Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
89	1.4E-7	< 0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	2.5E-3		PS-EDG-MOBILE-BRKRS	Breakers to Ocean Side Pumps
	1.0E-3		PS-EDG-STATION-LOWER-	PS EDG Stationary Lower Bus (210) Fails

	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
90	1.4E-7	< 0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	2.5E-3		PS-EDG-MOBILE-BRKRS	Breakers to Ocean Side Pumps
	1.0E-3		PS-EDG-STATIONARY-BRKRS	Breakers to Running Pumps from Stationary EDG
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
91	1.4E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-3		PS-BU-PUMPS-CF-FS	Post Tsunami Backup Pumps Fail from Common Cause Failure to
	2.5E-3		PS-EDG-MOBILE-BRKRS	Breakers to Ocean Side Pumps
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
92	1.3E-7	< 0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	1.0E-4		BACKUP-FT-CF-FS	Common Cause Failure of All Fire Trucks
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	5.0E-2		PS-EDG-STATIONARY-1	Post Tsuname Stationary EDG Fails
	5.0E-2		PS-MOBILE-EDG	Post Tsunami Mobile (750kVA) EDG
93	1.1E-7	< 0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-EDG-STATION-UPPER-BUS	PS EDG Stationary Upper Bus (210V) Fails
	2.0E-3		PS-EDG-TRANSFORMER-T2	6.9kv Post Tsunami Transformer (750k VA)T2
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
94	1.1E-7	< 0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-EDG-STATION-LOWER-	PS EDG Stationary Lower Bus (210) Fails
	2.0E-3		PS-EDG-TRANSFORMER-T2	6.9kv Post Tsunami Transformer (750k VA)T2
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
95	1.1E-7	<.0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-EDG-STATIONARY-BRKRS	Breakers to Running Pumps from Stationary EDG
	2.0E-3		PS-EDG-TRANSFORMER-T2	6.9kv Post Tsunami Transformer (750k VA)T2
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
96	1.1E-7	< 0.01	0-SPUMP-FAIL :4	
	5.0E-1		IE-SPUMP	Post Tsunami Pump Fails
	1.0E-3		PS-BU-PUMPS-CF-FS	Post Tsunami Backup Pumps Fail from Common Cause Failure to
	2.0E-3		PS-EDG-TRANSFORMER-T2	6.9kv Post Tsunami Transformer (750k VA)T2
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
97	5.7E-8	< 0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-EDG-OCEAN-BUS3	PS EDG Ocean Side Bus 3
	1.0E-3		PS-EDG-STATION-UPPER-BUS	PS EDG Stationary Upper Bus (210V) Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
98	5.7E-8	< 0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-EDG-OCEAN-BUS1	PS EDG Ocean Side Bus 1
	1.0E-3		PS-EDG-STATION-UPPER-BUS	PS EDG Stationary Upper Bus (210V) Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks

99	5.7E-8	< 0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-EDG-OCEAN-BUS2	PS EDG Ocean Side Bus 2
	1.0E-3		PS-EDG-STATION-UPPER-BUS	PS EDG Stationary Upper Bus (210V) Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks
100	5.7E-8	< 0.01	0-SLOOP :5	
	1.0E+1		IE-SLOOP	Seismic Induced Loop
	5.1E-2		OEP-XHE-XL-NR10H	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 10 HOURS
	1.0E-3		PS-CF-OCEANSIDEPUMPS-FS	Common Cause Failure to Start of Ocean Side Pumps
	1.0E-3		PS-EDG-STATION-UPPER-BUS	PS EDG Stationary Upper Bus (210V) Fails
	1.5E-1		PS-XHE-XM-PSFT	Operator Fails to Align Post Tsunami Fire Truck
	7.5E-1		PS-XHE-XM-PSFT-BU	Operator Fails to Align Backup Fire Trucks

Raphael, Mary Jean

From: Bowers, Anthony
Sent: Monday, November 21, 2011 5:05 PM
To: Foggie, Kirk
Cc: Raphael, Mary Jean
Subject: FW: ACTION - FW: (NISA) A request of official permission to release the documents
Attachments: Apr26-2011NISA-NRC expert meeting(Risk Analysis).pdf; May2-2011NISA-NRC expert meeting(Risk Analysis).pdf; MAy18-2011NISA-NRC expert meeting(SAMG).pdf; Jul7-2011NISA-NRC expert meeting(SAMG).pdf; Aug8-2011NISA-NRC expert meeting(SAMG).pdf

Importance: High

Kirk,

These documents are okay for release. Please let me or Mary Jean know if you need anything additional.

Tony

-----Original Message-----
From: Norton, Charles
Sent: Monday, November 21, 2011 2:46 PM
To: Bowers, Anthony
Cc: Mitchell, Matthew; Foster, Jack; Orenak, Michael; Mitman, Jeffrey; Taylor, Robert
Subject: FW: ACTION - FW: (NISA) A request of official permission to release the documents
Importance: High

Tony

These documents include the documents we discussed with Jeff Mitman and other documents. These documents contain no proprietary information, SUNSI, SGI, PII or anything else that should preclude either the NRC or the Japanese releasing them.

Chuck Norton

-----Original Message-----
From: Mitchell, Matthew
Sent: Monday, November 21, 2011 1:38 PM
To: Foster, Jack; Norton, Charles
Cc: Orenak, Michael
Subject: ACTION - FW: (NISA) A request of official permission to release the documents
Importance: High

Jack,

Please take the lead on this.

Mike - please make a task tracker with due date of COB Tuesday, November 22.

Matt M.

-----Original Message-----
From: Taylor, Robert
Sent: Monday, November 21, 2011 12:49 PM

AAA/100

To: Mitchell, Matthew
Cc: Norton, Charles
Subject: FW: (NISA) A request of official permission to release the documents

Probably belongs in your shop. Note that Norton was cc'd on it.

-----Original Message-----
From: Bowers, Anthony
Sent: Monday, November 21, 2011 10:35 AM
To: Skeen, David
Cc: Norton, Charles; Taylor, Robert; RST01_F Resource
Subject: FW: (NISA) A request of official permission to release the documents

Dave,

I spoke with Kirk Foggie from OIP this morning regarding Japan's request to release NRC information provided in the attached documents. (See below e-mail) I have reviewed the attached information and do not see any harm in releasing this information to the public under FOIA.

Is there any way for your staff to perform a quick review of the information provided in the attached documents and make a release or withhold determination prior to responding back to Japan. This request should only take about 15 minutes or so.

Please give me a call if you wish to discuss further.

Tony
301-415-5313

-----Original Message-----
From: Foggie, Kirk
Sent: Monday, November 21, 2011 8:59 AM
To: Bowers, Anthony
Cc: Raphael, Mary Jean
Subject: FW: (NISA) A request of official permission to release the documents

FYI.

-----Original Message-----
From: hino-yuji@meti.go.jp [mailto:hino-yuji@meti.go.jp]
Sent: Friday, November 18, 2011 7:11 AM
To: Foggie, Kirk
Cc: nishide-aki@meti.go.jp; tajiri-tomoyuki@meti.go.jp; yamasaki-takeshi@meti.go.jp
Subject: (NISA) A request of official permission to release the documents

Dear Mr. Kirk

Thank you for your cooperation.

We received attached five documents at NISA-NRC expert meeting in Tokyo.

Please hear your answer by November 25.

(See attached file: Apr26-2011NISA-NRC expert meeting(Risk Analysis).pdf)

(See attached file: May2-2011NISA-NRC expert meeting(Risk Analysis).pdf)

(See attached file: MAy18-2011NISA-NRC expert meeting(SAMG).pdf)

(See attached file: Jul7-2011NISA-NRC expert meeting(SAMG).pdf)

(See attached file: Aug8-2011NISA-NRC expert meeting(SAMG).pdf)

Sincerely

原子力安全・保安院
Nuclear and Industrial Safety Agency (NISA)

原子力安全技術基盤課
Nuclear Safety Standard Division

日野　裕司 (Yuji Hino)
TEL：03-3501-0621
PHS：71545
FAX：03-3580-5971

----- 転送者: nishide-aki/MITI-LAN 転送日: 2011/11/18 08:19 -----
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| 送信元: |
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|"Foggie, Kirk" <Kirk.Foggie@nrc.gov>
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| 宛先: |
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|"'nishide-aki@meti.go.jp'" <nishide-aki@meti.go.jp>
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|"shimada-taro@meti.go.jp" <shimada-taro@meti.go.jp>, "hino-yuji@meti.go.jp" <hino-yuji@meti.go.jp> |

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|RE: (NISA) A request of official permission to release the documents
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Dear Mr. Nishide,

I would be glad to help you with this issue. In the U.S. we have a formal process and criteria for releasing Federal government-owned information to the public. The guidance come under the Freedom of Information Act (FOIA).
I can direct you to this documentation so you will understand how the U.S. Government handles these types of request. Additionally, if you can provide me an electronic copy of the handouts you received I can make help you make a determination on what can be released to the public.

Best Regards.

Kirk

-----Original Message-----
From: nishide-aki@meti.go.jp [mailto:nishide-aki@meti.go.jp]
Sent: Thursday, November 17, 2011 2:56 AM
To: Foggie, Kirk
Cc: shimada-taro@meti.go.jp; hino-yuji@meti.go.jp
Subject: (NISA) A request of official permission to release the documents

Mr. Kirk R. Foggie

My name is Aki Nishide, from Policy Planning and Coordination Division, Nuclear and Industrial Safety Agency.
I was informed of your email address from Mr. Hara, International Affairs Office, NISA.

Here, I am writing to ask you to introduce the right person to ask for the official permission pertaining to the release of documents upon a request of disclosure by the public.

We are requested to disclose the document about Fukushima Dai-ichi Nuclear Power Station ,received from U.S.Gov, during March 11.2011 to Aug 31.2011.

We received some documents that satisfy the requirement from NRC, in Japan-NRC Meeting which was held to discuss about the accident at Fukushima Dai-ichi Nuclear Power Station, TEPCO, in Tokyo.

We would like to know if there is a problem disclosing these document, and also would like to know whether we should black out some of the names and the titles upon disclosure.

We have to notify disclosure or non-disclosure by the end of this month , so we have to hear the answer by November 25.

We would appreciate if you introduce the right person to ask about this topic.

Thank you in advance for your cooperation.

Best Regards,
Aki Nishide
Policy Planning and Coordination Division Nuclear and Industrial Safety Agency

Phone: +81-3-3501-1568
Fax :+81-3-3580-8490
E-mail: nishide-aki@meti.go.jp

Raphael, Mary Jean

From: Foggie, Kirk
Sent: Monday, November 21, 2011 8:59 AM
To: Bowers, Anthony
Cc: Raphael, Mary Jean
Subject: FW: (NISA) A request of official permission to release the documents
Attachments: Apr26-2011NISA-NRC expert meeting(Risk Analysis).pdf; May2-2011NISA-NRC expert meeting(Risk Analysis).pdf; MAy18-2011NISA-NRC expert meeting(SAMG).pdf; Jul7-2011NISA-NRC expert meeting(SAMG).pdf; Aug8-2011NISA-NRC expert meeting(SAMG).pdf

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Sent: Friday, November 18, 2011 7:11 AM
To: Foggie, Kirk
Cc: nishide-aki@meti.go.jp; tajiri-tomoyuki@meti.go.jp; yamasaki-takeshi@meti.go.jp
Subject: (NISA) A request of official permission to release the documents

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(See attached file: Jul7-2011NISA-NRC expert meeting(SAMG).pdf)

(See attached file: Aug8-2011NISA-NRC expert meeting(SAMG).pdf)

Sincerely

原子力安全・保安院
Nuclear and Industrial Safety Agency (NISA)

原子力安全技術基盤課
Nuclear Safety Standard Division

日野　裕司 (Yuji Hino)

AAA/101

TEL：03-3501-0621
PHS：71545
FAX：03-3580-5971

----- 転送者: nishide-aki/MITI-LAN 転送日: 2011/11/18 08:19 -----
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| 送信元: |
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|"Foggie, Kirk" <Kirk.Foggie@nrc.gov>
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| 宛先: |
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|"'nishide-aki@meti.go.jp'" <nishide-aki@meti.go.jp>
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| Cc: |
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|"shimada-taro@meti.go.jp" <shimada-taro@meti.go.jp>, "hino-yuji@meti.go.jp" <hino-yuji@meti.go.jp> |
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|2011/11/17 23:18
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| 件名: |
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|RE: (NISA) A request of official permission to release the documents
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I can direct you to this documentation so you will understand how the U.S. Government handles these types of request. Additionally, if you can provide me an electronic copy of the handouts you received I can make help you make a determination on what can be released to the public.

Best Regards.

Kirk

-----Original Message-----
From: nishide-aki@meti.go.jp [mailto:nishide-aki@meti.go.jp]
Sent: Thursday, November 17, 2011 2:56 AM
To: Foggie, Kirk
Cc: shimada-taro@meti.go.jp; hino-yuji@meti.go.jp
Subject: (NISA) A request of official permission to release the documents

Mr. Kirk R. Foggie

My name is Aki Nishide, from Policy Planning and Coordination Division, Nuclear and Industrial Safety Agency.
I was informed of your email address from Mr. Hara, International Affairs Office, NISA.

Here, I am writing to ask you to introduce the right person to ask for the official permission pertaining to the release of documents upon a request of disclosure by the public.

We are requested to disclose the document about Fukushima Dai-ichi Nuclear Power Station ,received from U.S.Gov, during March 11.2011 to Aug 31.2011.

We received some documents that satisfy the requirement from NRC, in Japan-NRC Meeting which was held to discuss about the accident at Fukushima Dai-ichi Nuclear Power Station, TEPCO, in Tokyo.

We would like to know if there is a problem disclosing these document, and also would like to know whether we should black out some of the names and the titles upon disclosure.

We have to notify disclosure or non-disclosure by the end of this month , so we have to hear the answer by November 25.

We would appreciate if you introduce the right person to ask about this topic.

Thank you in advance for your cooperation.

Best Regards,
Aki Nishide
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Questionnaire to NRC about Fukushima Accident

Long-term Measures of Station Blackout

Is there any good example for measures of station blackout specified in the guideline of severe accident management or others, when the situation of vast damages of social infrastructures experienced during huge natural disaster?

Plant procedures do address station blackout (SBO) irrespective of the cause of the SBO. The SAMG's do not address damage to supporting infrastructures such as bridges, roads, or fuel oil suppliers outside the nuclear facility as part of the recovery from SBO. Plant personnel in the emergency response facilities would work with off-site agencies and other resources to restore or compensate for lost power sources.

Various countermeasures to obtain power sources have been considered in US, is there any nuclear power station with a small scale power station nearby the plant which could provide required power sources even if during natural disaster?

A number of the US nuclear power plants have alternative off-site power sources such as dams or gas turbines. Availability of these sources may be affected by the nature of the natural disaster.

Containment Venting

Requirements of containment vent for the BWR, the generic letter 1989-16 "Installation of a hardened wetwell vent" issued in 1989 and requested to provide an estimated schedule of installation of hardened vent by NRC. In Japan, also requested to install a hardened wetwell vent, and in effect hardened wetwell vents were installed voluntarily by utilities.

According to the Federal register 12276 issued March 3, 1993, it can understand that some plants have a low pressure vent made of ductwork. Do those plants have a low pressure vent?

The drywell may be vented to minimize pressure fluctuations This is accomplished through air operated ventilation purge valves. The low pressure or "soft" vent would be considered the normal containment ventilation purge path system, where the discharge may be routed to the standby gas treatment system (in the event of a reactor building isolation) so that release of gases from the primary containment is controlled, with the effluents being filtered and monitored before discharge through the main stack.

What requirements do you request for structural integrity and seismic resistance of the hardened vent?

The Augmented Primary Containment Vent System (APCVS) also known as the "Hardened Vent", was installed to prevent primary containment pressure from exceeding the primary containment pressure limit (PCPL) by a transient event requiring reactor shutdown followed by a complete and sustained failure of decay heat removal capability. The vent is referred to as hardened because it utilizes steel piping rather than the soft of ducting in normal ventilation

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systems. As your question noted, GL 89-16, has many of the requirements or expectations for installation of "hardened vents."

The augmented primary containment vent system is non-safety related but seismically supported as related to the secondary containment boundary. APCVS has no active functions during normal plant operation or design basis events.
The event for which APCVS was installed is beyond the design basis of the plant. In response to Generic Letter 89-16, normally the selected vent path would be from the pressure suppression chamber only, to take advantage of the scrubbing effect of the suppression pool.
The system is designed to prevent containment pressure from exceeding the primary containment pressure limit (PCPL).

The vent is sized such that under conditions of constant heat input at a rate of approximately 1% of rated thermal power and containment pressure equal to the PCPL, the exhaust flow through the vent is sufficient to prevent the containment pressure from increasing. The design precludes possible sources of ignition for combustible gases.

Existing radiation monitoring capability in the main stack will alert control room operators of radioactive releases during venting. Venting from one unit does not compromise the safety of the other unit. System design should preclude backflow from the venting unit to the other unit.

With respect to a known facility, a dual unit station, the APCVS for both units are tied together, and a common line runs to the chimney. It is not postulated that simultaneous event sequences in both units would require simultaneous venting of both units. Although extremely unlikely, simultaneous venting of both units would be precluded administratively, through procedures and communication between units.

Seawater Injection into Core and Spent Fuel

This time, seawater injection was conducted as a ultimate measure. How do they treat seawater injection during severe accident management? Is there any regulatory requirement against this kind of situation? (This question is to know, because the severe accident management guideline is issued by US utilities and not open to the public.)

The SAMGs (page RF-5) refer to the use of fire water or RHR service water crosstie to flood the containment, if necessary. Depending on the site specific situation, this could be seawater. Seawater injection would be considered a last resort. There is no regulatory requirement against the use of seawater. The SAMGs were designed to fill the vessel and containment to a point that the fuel materials were covered in as short a time frame as possible with any source of water available to preclude release of radionuclide's from the fuel.

Ultimate Heat Sink

During recovery of accident at Fukushima, it takes a long time to recovery of loss of heat sink, to establish long-term cooling for the core and the spent fuel pool. Is there any good example of a

specific measure of severe accident management against this kind of situation?

The Fukushima event is a significant beyond design basis event. US BWRs similar in design to Fukushima plants would find themselves in a similar condition with loss of all available plant water injection inventory options. Some plants that employ an Isolation Condenser system would need to provide water within 6-10 hrs using a B5b type diesel pump or simply a fire-truck Plants that employ a HPCI /RCIC may have procedures that allow operation of the steam turbine governor manually without battery power. BWROG Emergency Procedure Guidelines and Severe Accident Guidelines provide options and strategies to deal with loss of inventory situations [e.g. RHR service water crosstie with fire system to ultimate heat sink, (seawater, lake, river etc.)].

If no injection system is available or is inadequate to restore RPV level above minimum for steam cooling RPV water level, EOPs direct operators to PRIMARY CONTAINMENT FLOODING., that initiates RPV and PCV Containment Flooding SAG. If the containment pressure approached the PCPL, operators are instructed to vent the primary containment (preference through the Torus for scrubbing), regardless of off-site dose consequences.

NRC Regulatory Requirements of Hydrogen Control for SFP and BWR Reactor Building

Breach of the reactor building of unit #4 is suspected to cause the possible hydrogen generation from the spent fuel in the spent fuel pool. There is no regulatory requirement for this concern in Japan. Is there any regulatory requirement or guideline in US?

If this question is dealing with hydrogen generation requirements in spent fuel pools, NRC does not have regulations that address this, however there are specific plant procedures including b5b strategies to keep the spent fuel pools full of water.

It seems to have been considered possibility of breach of the spent fuel pool building during study of severe accident in US. What studies have you conducted and how reflect the result in US?

Studies have been performed that consider this. We will work with NRC HQ to obtain any further information.

According to the report of "Technical Study of Spent Fuel Pool Accident" issued in 2001, it describes as follows. What is the current situation of this kind of problem?

We will work with NRC HQ to obtain any further information.

Re-criticality of SFP (possible)

NRC requirements for protecting Re-criticality of spent fuel pool are 10 CFR50.68(b)(4) and SPR9.1.1 revision2, 2007. We would like to know any other information available.

NRC RST with members of the consortium have reviewed this issue and determined that the SFP has a low likely hood of a re-criticality in a spent fuel pool loss of water inventory events.

NRC Japan Site Team recommendations for improving the September 2011 NISA report to the International Atomic Energy Agency (IAEA)

At the request of NISA, the US NRC has reviewed the Government of Japan (GOJ) June 7, 2011, report to the IAEA to identify potential improvements that NISA may wish to consider in the September update to the GOJ report to the IAEA. The US NRC staff commends the GOJ on the issuance of the June 7, 2011, report to the IAEA. The report has helped the United States and others in the international community understand the events that occurred at Fukukshima Daiichi following the March 11 earthquake and tsunami. Although the June 7, 2011, report was both transparent and thorough, the USNRC has identified the following suggestions that may enhance international understanding of the event and aid in the development of lessons learned.

1. Update/revise previous chronologies for each unit, as appropriate. If possible, include operator actions (based upon eye witness accounts and/or operator debriefs), equipment operation successes and failures, procedure use (successes and failures), corporate involvement, and simplified one line diagrams for systems used to mitigate the accident.
2. Include detailed chronologies of events and operator/staff actions to address the problems and conditions at Daini to stabilize the four units following the earthquake and tsunami – focusing on lessons learned from their successes, as well as any additional problems they encountered may benefit the international community.
3. Inclusion of air, water, and soil sample analyses (complete radionuclide analysis) and radiation survey results could help illustrate the release and deposition characteristics and confirmatory core melt signatures.
4. Additional information (forensic analysis) gained from the debris cleanup and removal efforts (extent of damage and undamaged areas and systems) would be helpful.
5. Detailed chronology of the discussions and decisions made between the government and TEPCo for both Daiichi and Daini, if available.
6. Detailed discussion of the communications (what communications networks/circuits continued to function and which did not as a result of the earthquake and tsunami) at the site and between the site and TEPCO corporate and/or NISA (immediately following the event and days that followed).
7. More details regarding the actions taken by the operators to align primary containment vent pathways (procedures/guidance and successes and failures).
8. More detailed description of the conditions at Units 5 &6, chronology of the events at Unit 5 and 6, including available temperature and pressure data/logs would be informtive.
9. Details of control room habitability and staffing through-out the accident and during the following days.

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10 Additional information on radiation survey results (time and location) and how these surveys impacted operator actions or decision making during the earlier stages of the accident.

11 Provide a root cause analysis of the accident and discuss the potential organizational factors before, during, and after the event that may have been contributors, to help the international community develop lessons learned from the event.

Japanese Emergency Measures Comparison to U.S. Response to Events at Fukushima

United States Nuclear Regulatory Commission (NRC) actions in response to the accident at Fukushima include:

- **Temporary Instruction (TI) 2515/183** - TI 183 independently assessed the adequacy of actions taken by licensees in response to the Fukushima Daiichi nuclear station fuel damage event.

- **Temporary Instruction (TI) 2515/184** - TI 184 determined that severe accident management guidelines (SAMGs) are available and maintained at all United States power reactors. The TI also determined the nature and extent of licensee implementation of SAMG training and exercises.

- **NRC Task Force** - The NRC has named six senior managers and staff to its task force to examine the agency's regulatory requirements, programs, processes, and implementation in light of information from the Fukushima Daiichi site in Japan, following the March earthquake and tsunami. The objective of this task force is to conduct a methodical and systematic review to recommend whether the agency should make near-term improvements to our regulatory system. This task force will also identify a framework and topics for review and assessment for the longer-term effort.

These temporary instructions and the task force efforts will be evaluated to determine the United States nuclear industry's readiness for a similar event and to aid in determining whether additional regulatory actions by the NRC are warranted.

The table below outlines point by point the United States regulation and guidance that existed prior to the lessons learned from the Fukushima nuclear accident as compared to the measures identified by the Japanese Nuclear and Industrial Safety Agency (NISA).

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Japanese Emergency Measures Comparison to U.S. Response to Events at Fukushima

No.	Time frame	Major Category	Sub-category	Japanese Action	Existing U.S readiness	Regulation/Guidance
1	Short Term Measures	Measures against Loss of All AC Power Supply	Emergency Response Plan Drafted	Confirm Emergency Response Plans have been drafted in consideration of damage by a tsunami.	Yes	• 10 CFR 50.54 (hh)(2) (Conditions of Licenses) • 10 CFR 50.63 (Station Blackout Rule) • NUREG 1776 (Regulatory Effectiveness of the Station Blackout
2	Short Term Measures	Measures against Loss of All AC Power Supply	Emergency Response Plan Drafted	Confirm places required for operations are accessible, and have multiple routes thereto; as well, the procedures and authorizations for the vents and seawater injection have all been clarified.	Yes	• 10 CFR 50.54 (hh)(2) (Conditions of Licenses)
3	Short Term Measures	Measures against Loss of All AC Power Supply	Securing Power Supply in case of Emergency	Confirm that the NPS has the required capacity and number of power cars, in order to maintain observation functions and be a driving force for the valves in the I&C system and the MCR.	Yes - for a single unit failure.	• 10 CFR 50.54 (hh)(2) (Conditions of Licenses) • 10CFR 50.63 (Station Blackout Rule) • NUREG 1776 (Regulatory Effectiveness of the Station Blackout Rule)

Japanese Emergency Measures Comparison to U.S. Response to Events at Fukushima

No.	Time frame	Major Category	Sub-category	Japanese Action	Existing U.S readiness	Regulation/Guidance
4	Short Term Measures	Measures against Loss of All AC Power Supply	Securing Power Supply in case of Emergency	Confirm that cables are the right length to connect with the power cars.	Yes	• 10 CFR 50.54 (hh)(2) (Conditions of Licenses)
5	Short Term Measures	Measures against Loss of All AC Power Supply	Securing Power Supply in case of Emergency	Confirm that the storage location for the above equipment is sufficiently high so a tsunami would have no impact on it.	Yes	• 10 CFR 50.54 (hh)(2) (Conditions of Licenses)
6	Short Term Measures	Measures against Loss of All AC Power Supply	Securing Heat Removal Function in Case of Emergency	Confirm that NPS has the required additional pressure capacity and number of cars of fire engines and pump trucks for water injection to remove decay heat.	Yes	• 10 CFR 50.54 (hh)(2) (Conditions of Licenses)
7	Short Term Measures	Measures against Loss of All AC Power Supply	Securing Heat Removal Function in Case of Emergency	Confirm that a water source can be secured and that the proper length of hose is available for water injection	Yes	• 10 CFR 50.54 (hh)(2) (Conditions of Licenses)
8	Short Term Measures	Measures against Loss of All AC Power Supply	Securing Heat Removal Function in Case of Emergency	Confirm that the storage location of the above equipment is sufficiently high so a tsunami would have no impact on it and that the fire department within the NPS is on standby.	Yes	• 10 CFR 50.54 (hh)(2) (Conditions of Licenses)

Japanese Emergency Measures Comparison to U.S. Response to Events at Fukushima

No.	Time frame	Major Category	Sub-category	Japanese Action	Existing U.S readiness	Regulation/Guidance
9	Short Term Measures	Measures against Loss of All AC Power Supply	Securing Heat Removal Function in Case of Emergency	Confirm that an immediate response can be implemented with clarity in the operation procedures for the vents, order of command, etc	Yes	• 10 CFR 50.54 (hh)(2) (Conditions of Licenses)
10	Short Term Measures	Measures against Loss of All AC Power Supply	Implementation of checks for machines and equipment as well as drills	Confirm that checks for machines and equipment used in emergencies are completed	Yes	• 10 CFR 50.54 (hh)(2) (Conditions of Licenses)
11	Short Term Measures	Measures against Loss of All AC Power Supply	Implementation of checks for machines and equipment as well as drills	Confirm that drills for measures against loss of all AC power supply due to a tsunami are implemented and the implementation procedures are established	Yes, drills and demonstrations have been performed for 10 CFR 50.54 (hh) (2) requirements.	• 10 CFR 50.54 (hh)(2) (Conditions of Licenses)
12	Short Term Measures	Measures against Loss of All AC Power Supply	Changes to the Operational Safety Programs	Confirm that the requested matters related to maintenance activities of a nuclear reactor facility when the power supply is lost are stipulated in the operational safety programs.	Yes	Individual Plant Emergency Plan Procedures

Japanese Emergency Measures Comparison to U.S. Response to Events at Fukushima

No.	Time frame	Major Category	Sub-category	Japanese Action	Existing U.S readiness	Regulation/Guidance
13	Short Term Measures	Measures against Inundation of Buildings	None	Confirm that measures against inundation are planned so that machines and equipment used in measures against loss of all AC power, etc. will not be impacted by a tsunami	Yes	• Reg Guide 1.102 (Flood Protection for Nuclear Power Plants) • Inspection Procedure IP 71111.06, Flood Protection • 10 CFR 50 App B, Criterion III (Design Control) • GL 88-20 (Individual Plant
14	Short Term Measures	Measures against Inundation of Buildings	None	Note – the Japanese indicated that most of these short term issues will be implemented by mid-May.	N/A	N/A
15	Mid- to Long-term Measures	Measures to Improve Reliability due to Speeding Up the Cold Shutdown	Securing back-up equipment for Seawater pump motors, etc. (implementation within about 1 year)	Confirm that a plan has been drafted to secure back-up equipment for the seawater pump motors as well as replacement pumps for the seawater pumps (submersible pumps, portable pumps) required for the recovery of the NPS's RHR system in order to bring about a quick cold shutdown.	No	

Japanese Emergency Measures Comparison to U.S. Response to Events at Fukushima

No.	Time frame	Major Category	Sub-category	Japanese Action	Existing U.S readiness	Regulation/Guidance
16	Mid- to Long-term Measures	Measures to Improve Reliability due to Speeding Up the Cold Shutdown	Establishment of air-cooling type Emergency Generators (implementation in about 1-2 years)	Confirm a plan has been drafted for a large-size generator having the driving force capacity for the heat exchanger pumps in order to remove decay heat is built on high ground so that a tsunami could not impact it easily.	There is no requirement to have air cooled Emergency Diesel Generators at United Stated Nuclear Power Stations.	
17	Mid- to Long-term Measures	Measures to Protect Against Tsunami	Building tide embankments, building seawalls around buildings, increasing water tightness around buildings (implementation in 2-3 years)	Confirm the plans to improve even more the reliability of the emergency safety measures by strengthening the water tightness of buildings, and reinforcing the tide embankments and seawalls, in order to ensure that the machines and equipment important to safety of the reactor will not be impacted by a tsunami	No	

Japanese Emergency Measures Comparison to U.S. Response to Events at Fukushima

No.	Time frame	Major Category	Sub-category	Japanese Action	Existing U.S readiness	Regulation/Guidance
18	Additional measures added on June 7th	Secure the working environment in the Main Control Room	None	Measures should be established so that the Main Control Room has a secure working environment with radiation protection, etc. by enabling the emergency ventilation and air conditioning facilities (recirculation system) to run on power supplied by a power supply vehicle in case all AC power supply is lost during an emergency	No • The station blackout rule requires that a station must be able to withstand for a specified duration and recover from a station blackout. This would include control room habitability. • The coping times established for US reactors in response to a station blackout, however, do not approach the times experienced during the Fukushima event. • US plants are not required to provide for a separate emergency electrical feed connection directly to the control room HVAC equipment.	• 10 CFR 50.63 (Station Blackout Rule)

Japanese Emergency Measures Comparison to U.S. Response to Events at Fukushima

No.	Time frame	Major Category	Sub-category	Japanese Action	Existing U.S readiness	Regulation/Guidance
19	Additional measures added on June 7th	Secure the means of communication inside the NPS premises in case of emergency.	None	Measures should be established so that work inside the NPS premises can be done quickly and efficiently, by securing a reliable means of communication inside the NPS premises in case all AC power supply is lost during an emergency.	No, long-term loss of all AC could potentially affect reliable internal communication at United States Nuclear Power Stations.	
20	Additional measures added on June 7th	Secure supplies and equipment such as high-level radiation protective gear, and develop a system for radiation dose management	None	Measures should be established so that radiation protection and radiation dose management of workers are secured in case of emergency, by companies' stocking and sharing between each other supplies and equipment such as high-level radiation protective gears and personal dosimeters. A system should be maintained so that personnel capable of radiation dose management can be increase in case of an emergency.	No	

Japanese Emergency Measures Comparison to U.S. Response to Events at Fukushima

No.	Time frame	Major Category	Sub-category	Japanese Action	Existing U.S readiness	Regulation/Guidance
21	Additional measures added on June 7th	Establish measures to prevent hydrogen explosion	None	Measures should be established to prevent the destruction of facilities due to hydrogen explosion caused by core damage, etc. by preventing large amounts of hydrogen to build up inside reactor buildings, etc. in case of core damage, etc. during an emergency.	No. There is no contingency for hydrogen mitigation in the reactor buildings of Boiling Water Reactors. There is guidance for hydrogen control in primary containment.	US plants EOPs and SAMG's are designed to prevent and mitigate the destruction of facilities by requiring the operators to vent or energize hydrogen recombiners if certain conditions are met.
22	Additional measures added on June 7th	Deploy heavy machinery for removing rubble	None	Measures should be established so that work inside the premises can be carried out swiftly, by deploying heavy machinery such as wheel loaders to quickly remove rubble caused by a tsunami, etc. in case of an emergency.	There are no requirements to have equipment staged or available for removing rubble at United States Nuclear Power Stations.	

Japanese Emergency Measures Comparison to U.S. Response to Events at Fukushima

No.	Time frame	Major Category	Sub-category	Japanese Action	Existing U.S readiness	Regulation/Guidance
23				New regulatory requirements: To establish an organization for initial activation of fire-fighting, preparing such as notification, personnel, material, evaluation and its measures. To be stipulated the above items in the operational safety programs by the utilities. NISA just confirmed by inspection.	If recently performed at the facility, review Inspection Procedure 71111.05T "Fire Protection (Triennial),"Inspection results and findings to identify any other potential areas of inspection. Use Section 02.03 and 03.03 as a guideline to conduct your inspections.	TI-2115/183 Follow up to the Fukushima Daiichi Nuclear Station Fuel Damage Event
24				NISA just confirmed by inspection. In accordance with emergency safety measures dated on March 30, 2011, NISA confirmed the utility's check of machines and equipment used in emergency.	Verify through test or inspection that B.5b equipment is available and functional. Active equipment shall be tested and passive equipment shall be walked down and inspected.	TI-2115/183 Follow up to the Fukushima Daiichi Nuclear Station Fuel Damage Event

Japanese Emergency Measures Comparison to U.S. Response to Events at Fukushima

No.	Time frame	Major Category	Sub-category	Japanese Action	Existing U.S readiness	Regulation/Guidance
25				NISA added the regulatory requirements of emergency measures during loss of power into Commercial NPP regulations, Article 11.3. Based on this requirement, utility added the measures into operational safety programs. NISA confirmed this revised operational safety programs of utility.	Verify through walkdowns or demonstration that procedures to implement the strategies associated with B.5.b and 10 CFR 50.54(hh) are in place and are executable.	TI-2115/183 Follow up to the Fukushima Daiichi Nuclear Station Fuel Damage Event
26				(ditto)	Verify the training and qualifications of operators and the support staff needed to implement the procedures and work instructions are current for activities related to Security Order Section B.5.b and severe accident management guidelines as required by 10 CFR 50.54 (hh).	TI-2115/183 Follow up to the Fukushima Daiichi Nuclear Station Fuel Damage Event

Japanese Emergency Measures Comparison to U.S. Response to Events at Fukushima

No.	Time frame	Major Category	Sub-category	Japanese Action	Existing U.S readiness	Regulation/Guidance
27				No regulatory requirements. (There are no agreements or MOU for accident mitigation between local authorities and utilities.)	Verify that any applicable agreements and contracts with outside organizations are in place and are capable of meeting the conditions needed to mitigate the consequences of these events.	TI-2115/183 Follow up to the Fukushima Daiichi Nuclear Station Fuel Damage Event
28				No regulatory requirements.- There is no rule such as 10CFR50.59 used in USA.- Long-term measures committed by utilities will be followed up.	Review any open corrective action documents to identify vulnerabilities that may not have yet been addressed.	TI-2115/183 Follow up to the Fukushima Daiichi Nuclear Station Fuel Damage Event

Japanese Emergency Measures Comparison to U.S. Response to Events at Fukushima

No.	Time frame	Major Category	Sub-category	Japanese Action	Existing U.S readiness	Regulation/Guidance
29				NISA added the regulatory requirements of emergency measures during loss of power into Commercial NPP regulations, Article 11.3. Based on this requirement, utility added the measures into operational safety programs. NISA confirms this revised operational safety programs of utility.	Assess the licensee's capability to mitigate station blackout (SBO) conditions, as required by 10 CFR 50.63, "Loss of All Alternating Current Power," and station design, is functional and valid. 1. Verify through walkdowns and inspection that all required materials are adequate and properly staged, tested, and maintained. 2. Demonstrate through walkdowns that procedures for response to an SBO are executable.	TI-2115/183 Follow up to the Fukushima Daiichi Nuclear Station Fuel Damage Event

Japanese Emergency Measures Comparison to U.S. Response to Events at Fukushima

No.	Time frame	Major Category	Sub-category	Japanese Action	Existing U.S readiness	Regulation/Guidance
30				As for Tsunami, NISA confirmed utility's inundation measures by site inspection.	Assess the licensee's capability to mitigate internal and external flooding events required by station design.	TI-2115/183 Follow up to the Fukushima Daiichi Nuclear Station Fuel Damage Event

Japanese Emergency Measures Comparison to U.S. Response to Events at Fukushima

No.	Time frame	Major Category	Sub-category	Japanese Action	Existing U.S readiness	Regulation/Guidance
31				NISA added the regulatory requirements of emergency measures during loss of power into Commercial NPP regulations, Article 11.3.Based on this requirement, utility added the measures into operational safety programs.NISA confirmed this revised operational safety programs of utility. In accordance with emergency safety measures dated on March 30, 2011, NISA confirmed the utility's check of machines and equipment used in emergency. (Procedures for fire provided from chapter 15 to chapter 21 in emergency operation procedure (EOP). These are cable spreading room, generator, DG room, oil equipment at outside/inside building, 6.9kV switchgear, 480V power center. Procedures for flooding event had not included in the EOP)	Assess the thoroughness of the licensee's walkdowns and inspections of important equipment needed to mitigate fire and flood events to identify the potential that the equipment's function could be lost during seismic events possible for the site. Assess the licensee's development of any new mitigating strategies for identified vulnerabilities. As a minimum, the licensee should have performed walkdowns and inspections of important equipment (permanent and temporary) such as storage tanks, plant water intake structures, and fire and flood response equipment; and developed mitigating strategies to cope with the loss of that important function.	TI-2115/183 Follow up to the Fukushima Daiichi Nuclear Station Fuel Damage Event

Japanese Emergency Measures Comparison to U.S. Response to Events at Fukushima

No.	Time frame	Major Category	Sub-category	Japanese Action	Existing U.S readiness	Regulation/Guidance
32				(NISA reviews the AM report submitted by utilities and confirms adequacy of the AM implementation system, adequacy of the AM measures and procedures and training on AM. Rulemaking for SA is under discussion, and the results will be reflected into a Japanese version of SAMG.)	Assess the availability and readiness of the licensee's ability to access and implement the SAMGs at their facility. This was done by reviewing SAMGs procedures, training, control of the documents, location and access, etc. In addition, operator interviews were conducted to assess the level of knowledge and familiarity with the SAMG procedures and use.	TI-125/184 Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)