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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. <u>Keep the Reactor Vessel Depressurized</u>. 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

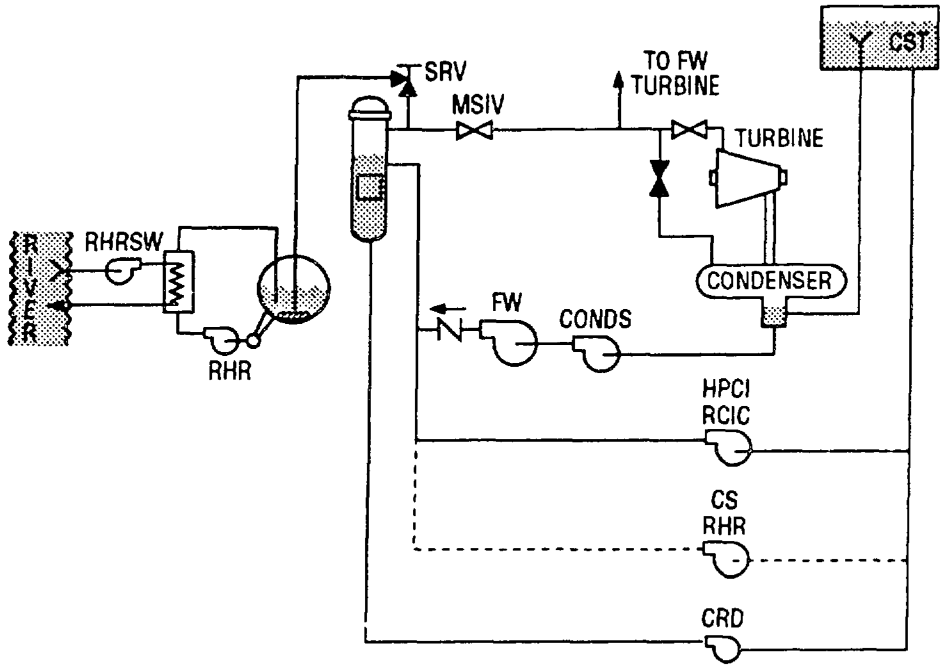

Fig. 1. Reactor vessel injection can be provided from the condensate storage tank by the motor-driven core spray (CS) or residual heat removal (RHR) system pumps.

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.

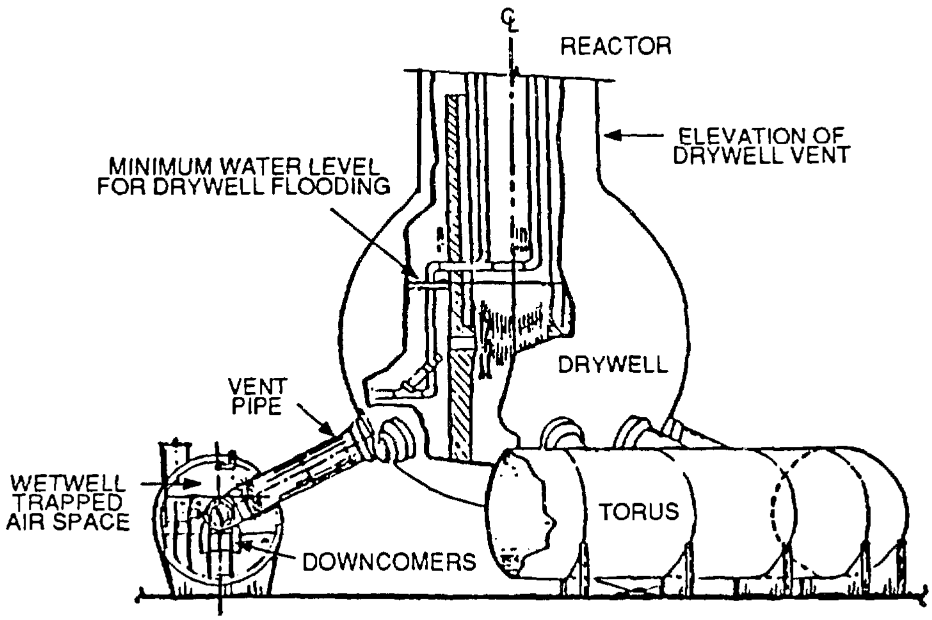

Fig. 2. Containment flooding to cover the reactor vessel bottom head in the BWR Mark I containment design.

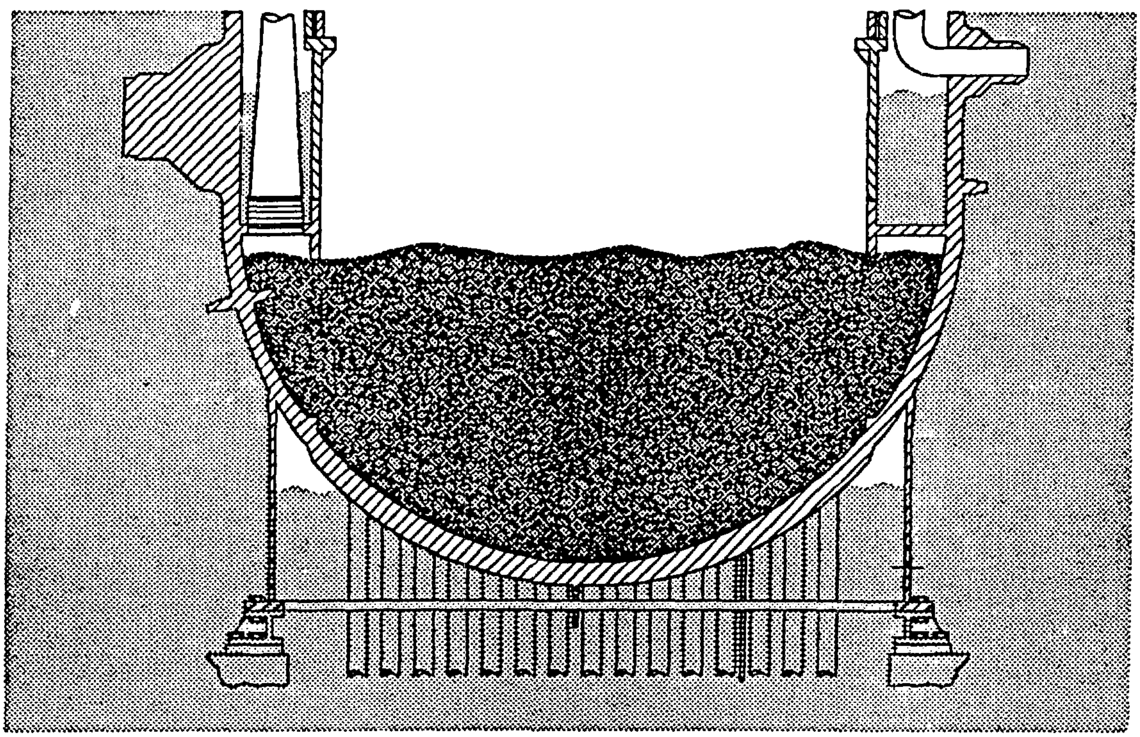

Fig. 3. An illustration of the quenched debris bed within the BWR lower plenum immediately after bed dryout.

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).

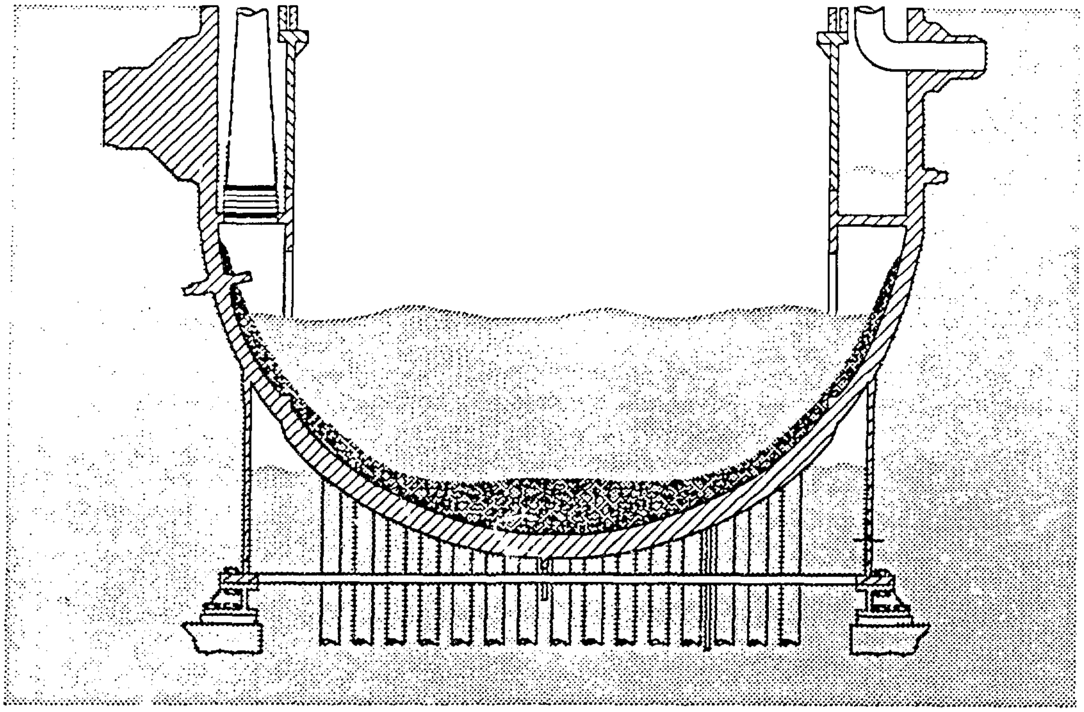

Fig. 4. A molten pool is predicted to form and spread radially from the upper center of the quenched debris.

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	Time to Failure 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.

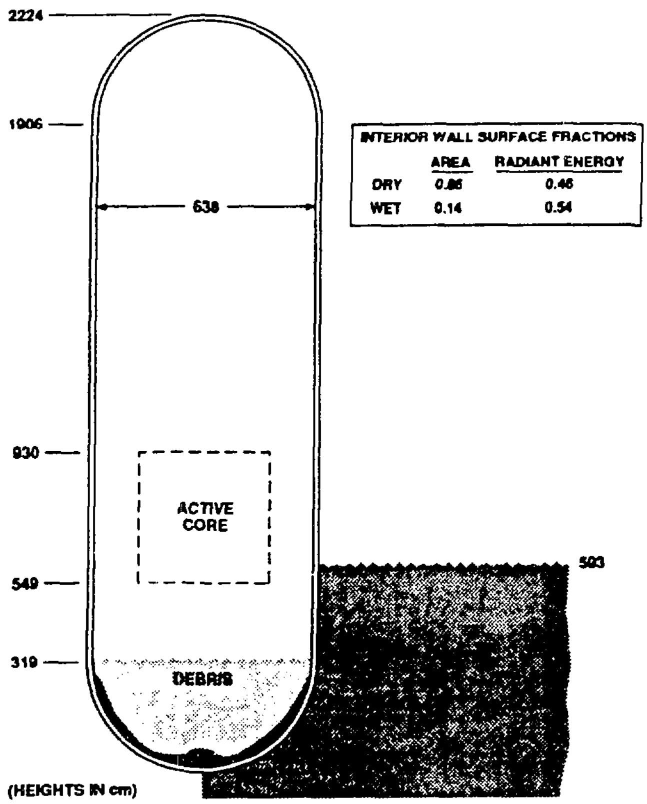

Fig. 5. The calculated minimum coverage of the upper reactor vessel necessary to preclude melting of the inner surface of the wall.

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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* This case corresponds to the last entry in Table 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.

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