THE US NUCLEAR REGULATORY COMMISSION'S CONTAINMENT PERFORMANCE IMPROVEMENT PROGRAM

W. Beckner, J. Mitchell, L. Soffer
E. Chow, J. Lane, J. Ridgely

ABSTRACT

The Containment Performance Improvement (CPI) program has been one of the main elements in the US Nuclear Regulatory Commission's (NRC's) integrated approach to closure of severe accident issues for US nuclear power plants. During the course of the program, results from various probabilistic risk assessment (PRA) studies and from severe accident research programs for the five US containment types have been examined to identify significant containment challenges and to evaluate potential improvements. The five containment types considered are: the boiling water reactor (BWR) Mark I containment, the BWR Mark II containment, the BWR Mark III containment, the pressurized water reactor (PWR) ice condenser containment, and the PWR dry containments (including both subatmospheric and large subtypes). The focus of the CPI program has been containment performance and accident mitigation, however, insights are also being obtained in the areas of accident prevention and accident management. Recommendations relative to BWR plants with Mark I containments were made in January 1989. One, hardening of the wetwell vent, is being implemented either voluntarily by the licensees or by invoking the backfit rule (10 CFR 50.109). Other recommended changes are being explicitly reviewed within the Individual Plant Examination (IPE) program to examine individual plants for vulnerabilities to severe accidents. These other changes include: (a) alternate water supply for drywell sprays and vessel injection, (b) enhanced reactor vessel depressurization system reliability, and (c) improved emergency procedures and training. Recommendations on the other containment types were presented to the Commission in March 1990. In general, the same containment challenges and potential improvements were examined for the BWR Mark II and Mark III plants as in the Mark I program, with the addition of improvements related to the hydrogen igniters for Mark III plants. For the PWR ice condenser and dry containments, containment by-pass and direct containment heating are issues. In addition, improvements related to hydrogen igniters have been considered for ice condenser plants. Primarily because the benefits of proposed changes are perceived to be less or because of large design differences among plants, the case for generic recommendations is not so clear cut as for the BWR Mark I plants. Therefore, the NRC staff has not identified any recommended generic improvements that would be applicable to all containments of a given type, but has identified improvements to be considered further on a plant-specific basis as part of the IPE program. Improvements to be included in the accident management program and areas requiring additional research have also been identified.

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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BACKGROUND

The ability to mitigate the consequences of severe accidents is a function of the containment systems that are provided on all U. S. light water reactors (LWRs). The containments are designed to withstand the effects of design basis accidents (pressures, temperatures, humidity, and radiation) with some margin for safety. The design basis accidents do not challenge reactor vessel integrity, but may lead to damage of some of the fuel in the reactor core. Ever since the TMI and Chernoybl accidents, interest has been focused on the ability of the current reactor and containment designs to withstand accidents beyond the design basis. These have historically been called Class 9 accidents or, today, severe accidents. The severe accident is one where, if no corrective actions are taken either by system operation or operator intervention, the core will fail, the molten core will melt through the reactor vessel, and the molten core and reactor vessel internals will be deposited directly into containment. The Containment Performance Improvement (CPI) program was established to identify generic containment challenges from severe accidents and to propose plant improvements to arrest a severe accident, prevent or delay containment failure, or to mitigate the consequences of a failed containment.

The designs of all U.S. containments consider external events (such as earthquakes and tornadoes), while the containment temperature and pressure design bases are typically determined by a postulated design basis loss-of-coolant accident (LOCA) in which operation of the emergency core cooling system (ECCS) would prevent a core melt from occurring. Despite this containment design basis which does not include core melting, radiological consequences that could only result from a substantial core melt are nevertheless postulated in accordance with the provisions of 10 CFR 100[1]. This assumption is used to assure the adequacy of certain plant features such as containment leak tightness and fission product cleanup systems, as well as the adequacy of the reactor site. While the temperature and pressure conditions associated with a core melt accident are not part of the containment design bases, there is some assurance that existing containments are capable of surviving the temperature and pressure conditions associated with some severe accidents due to the substantial safety margin in the containment design. Studies of various containment types under beyond design basis loading conditions[2] indicate survival at load levels of 2 to 3 times design basis LOCA pressures and at elevated temperature conditions. Although only a few detailed structural analyses of containments have been attempted, extrapolations from design assessments and testing on scale models of containments and penetrations at Sandia National Laboratory confirm these higher failure pressure conclusions[3]. Such confirmation, however, assumes containment isolation devices (including seals) isolate and do not fail.

One class of containments, the Mark I, has been used with 24 licensed boiling water reactors (BWR) reactors. Although all LWRs have containments designed to safely attenuate the energy that would be released in a LOCA, Mark I containments have among the smallest internal volumes. It is because of this relatively small internal volume that Mark Is have been perceived as being the most likely to fail during a severe accident and thus were considered first in the CPI program. This relatively small volume is offset, for some accidents, by a pressure suppression water pool which is designed to reduce containment pressure by condensing steam. The suppression pool is not effective, however, in preventing a pressure rise due to releases of non-condensible gases such as hydrogen and concrete ablation products produced during a severe accident. In addition, Mark I containments have a steel shell that may be vulnerable to failure upon contact with molten core material following a severe accident. As a result, for many severe accidents Mark I containments may be viewed as potentially more susceptible to containment failure than other containment types. The "Reactor Safety Study" (WASH-1400)[4] found that, for the Peach Bottom BWR Mark I nuclear plant, even though the core melt probability was relatively low, the containment could be severely challenged if a large core melt occurred. This conclusion has been reinforced by similar findings in the first and second drafts of the "Reactor Risk Reference Document" (NUREG-1150)[5,6].

Since there has been some concern over the ability of Mark I containments to withstand severe accidents, the question has also been raised as to the ability of the other containment types to withstand severe accidents. Thus, similar evaluations have been made for the BWR Mark II and Mark III containments and the pressurized water reactor (PWR) ice condenser and dry (both atmospheric and sub-atmospheric) containments. Each of the containment types is discussed below with the current technical findings.

BWR MARK I RECOMMENDED IMPROVEMENTS

Probabilistic risk assessment (PRA) studies for BWRs indicate that accidents initiated by transients rather than LOCAs dominate the total core damage frequency estimates. The principal accident sequences for BWRs consist of Long-term Loss of Decay Heat Removal (TW), Station Blackout (SBO), and Anticipated Transient Without Scram (ATWS). WASH-1400 estimated a total core melt frequency of $\sim 10^{-5}$ per reactor year and indicated that TW is the dominant core damage accident sequence for Peach Bottom. NUREG-1150, however, estimated a total core melt frequency of $\sim 10^{-6}$ per reactor year and indicated that the dominant contribution to core melt frequency at Peach Bottom is due to SBO. The TW sequence frequency estimate from the later study of Peach Bottom was greatly reduced by consideration of containment venting procedures. NUREG-1150 assumed that these venting actions could be successfully used to remove decay heat from the containment and thus prevent core melt due to TW sequences. For those Mark I plants for which TW has been eliminated as the dominant contributor, the residual risk is largely due to ATWS and SBO sequences. Available PRA studies of Mark I plants indicate that the estimated likelihood of core damaging accidents varies between $\sim 10^{-4}$ and $\sim 10^{-6}$ per reactor year.

Much of the focus of concerns relative to the ability of a Mark I containment to survive a core melt accident centered on the containment shell melt issue. Significant technical disagreement existed over whether or not molten core material on the drywell floor would fail the containment shell. This subject was an important issue discussed at an NRC sponsored workshop held on February 24-26, 1988 in Baltimore, Maryland[7]. This workshop was attended by national laboratory staff performing research on this subject, representatives of the nuclear industry and utilities and interested members of the public. While a variety of calculations and experimental results were presented relative to the issue of shell melt, it was clear that additional research was required to determine the likely impact of molten core material on the containment shell. In addition, the efficacy of, as well as the need for, methods to control core debris and prevent contact with the shell were questioned. A consensus did start to emerge, however, concerning the usefulness of water on the drywell floor. While one could not positively conclude that water would prevent or even delay failure of the shell due to contact with molten core debris, there was agreement that water would help mitigate the consequences of such an accident by providing scrubbing of fission products and thus a reduction in releases to the environment, irrespective containment failure. While complete resolution of this issue was not achieved because of continuing uncertainty about the ability of water to prevent shell failure, an interim means of addressing this issue (by increasing the likelihood of having a water pool over the core material) was identified which appeared to provided a reduction in the consequences of a major core melt accident in a Mark I containment. This consensus was important because it allowed the staff to focus on other challenges and potential improvements to the Mark I containment.

While the CPI program was initially concerned with containment performance given a severe accident, the NRC staff pursued a balanced approach of considering improvements to Mark I plants to both prevent severe accidents and mitigate the consequences. Six potential Mark I containment and plant improvements have been examined: (1) hydrogen control, (2) alternate water supply for reactor vessel injection and containment drywell sprays, (3) containment pressure relief capability (venting), (4) enhanced reactor pressure vessel depressurization system reliability, (5) core debris controls, and (6) procedures and training[8]. Each of these was evaluated to determine the potential benefits in terms of reducing the core melt frequency, containment failure probability, and offsite consequences.

Hydrogen Control

Although BWR Mark Is are required to be operated with an inerted containment atmosphere to prevent hydrogen combustion in the event of a severe accident, plant technical specifications permit de-inerting to commence 24 hours prior to plant shutdown, and do not require inerting to be completed until 24 hours after plant startup, in order to permit plant personnel access. The containment could also eventually become de-inerted by containment leakage in the event of a severe accident, such as a long-term station blackout. Therefore, two potential improvements with regard to hydrogen control were evaluated. These were: (1) elimination of the two 24 hour de-inerted periods and (2) providing a backup supply of nitrogen.

Since the time spent de-inerted is so short compared with the time spent inerted during normal operation, eliminating this time of de-inerting was judged to not significantly reduce risk. Absent containment failure, only the slow process of air ingress through containment leakage paths could cause containment de-inerting. Since offsite supplies of nitrogen could readily be obtained during this period, an onsite backup supply of nitrogen would not significantly reduce risk. Therefore, the staff concluded that additional Mark I improvements to control hydrogen beyond the existing hydrogen control rule and the procedures in Revision 4 of the Emergency Procedure Guidelines would have no significant benefit and are not warranted.

Alternate Water Supply

Another proposed improvement is to employ a backup or alternate supply of water and a pumping capability that is independent of normal and emergency AC power (e.g., diesel fire pumps). The needed valves would be required to be operated manually operated or would be provided with backup power. By connecting this source of water to the low pressure residual heat removal system as well as to the existing drywell sprays, water could be delivered either into the reactor vessel or to the drywell, by use of an appropriate valving arrangement. An alternate source of water injection into the reactor vessel, combined with other improvements discussed below, would greatly reduce the likelihood of core melt due to station blackout or loss of long-term decay heat removal, as well as provide significant accident management capability. Water for the drywell sprays would also provide significant mitigative capability to cool core debris, to cool the containment steel shell to possibly delay or prevent its failure, and to scrub particulate fission products. This improvement was judged to be useful in reducing risk.

A review of some BWR Mark I facilities indicates that most plants have one or more diesel driven pumps which could be used to provide an alternate water supply. The flow rate using this backup water system may be significantly less than the design flow rate for the drywell sprays. The potential benefits of modifying the spray headers to assure a spray were compared with having the water run out of the spray nozzles. Fission product removal from the atmosphere by the spray in the small crowded volume was judged to be small compared with the benefit of having a water pool on top of the core debris. Therefore, modifications to the spray nozzles were not considered warranted.

Containment Pressure Relief Capability (Venting)

The TW sequence is unusual in that the containment failure precedes core melt and, in fact, containment failure leads to core melt. This sequence is important because it could be a relatively likely sequence (in the absence of effective means to deal with the event) and could result in unmitigated releases due to a failed containment early in the accident. The TW sequence involves loss of long term containment heat removal. The core is effectively cooled, but decay heat transferred to the suppression pool can not be removed. Absent any recovery actions, the suppression pool would heat up and the containment would eventually be pressurized, resulting in containment failure (20-30 hours). Low pressure injection could be lost either from high containment pressure forcing vessel relief valves to close and thus preventing the use of low pressure sources of core cooling water or from damage to cooling water pumps as a result of containment failure (e.g., cavitation of pumps from inadequate net positive suction head (NPSH)). High pressure injection could also be lost due to inadequate cooling of the pump or drive turbine. Any of these methods of losing injection could lead to core degradation. One potential means of effectively dealing with a TW sequence is to vent the containment atmosphere to the environment to remove decay heat and prevent containment failure due to high pressure. Containment venting under these conditions would be a "clean vent" in that (1) the core would still be undamaged, (2) the only fission products released would be activity associated with the reactor coolant, and (3) all material released to the environment would have been scrubbed by the suppression pool.

The CPI program considered venting of Mark I containments due to the potential for eliminating what could otherwise be a dominant risk contributor for Mark I plants, but also investigated concerns with both the ability and desirability of venting using existing hardware and procedures. Venting procedures are contained in Emergency Procedure Guidelines (EPG) developed by the BWR Owners Group (BWROG) for all BWRs and have been implemented to various degrees in plant-specific Emergency Operating Procedures (EOP). Inspections by the Nuclear Regulatory Commission (NRC) staff of venting procedures as implemented at a number of plants raised concerns about the adequacy of the procedures and thus questions about the likelihood that operators could successfully vent the containment if required. In addition, the adequacy of existing plant hardware to perform the venting was also questionable. Mark I containments typically have a number of vent lines of varying sizes located on both the wetwell and drywell. These lines are normally used to ventilate the containment and in many cases are connected to the standby gas treatment system. The larger vent lines usually have sheet metal ventilation duct work for part of the vent path. Venting through such a path during a TW sequence could fail this duct work and release steam to the reactor building. The consequences of such a release of steam to the reactor building were not explicitly evaluated, but viewed by the NRC staff as highly undesirable because of the potential for further damage of essential equipment, personnel injury, and greatly complicated recovery from the accident. The NRC staff concluded that venting via a sheet metal ductwork path, as currently implemented at some Mark I plants, is likely to greatly hamper or complicate post-accident recovery activities, and is therefore viewed by the staff as yielding reduced imporvements in safety. The NUREG-1150 study did assume that venting could be successfully performed at Peach Bottom, but only after considering the long time periods that would be available for operator actions and after determining that venting using existing "hard pipes" would be adequate to remove decay heat.

The CPI program evaluated the impact of improving venting capability, including both procedures and hardware, so as to ensure that operators could vent, if required, and so as to not fail the vent path within the reactor building. No credit for preventing TW sequences was given without these capabilities. It was estimated that TW would be the dominant core melt sequence for BWRs with Mark I containments without venting and that the probability of core melt would be between $\sim 10^{-4}$ and $\sim 10^{-5}$ per reactor year. The higher number was estimated from studies performed for Unresolved Safety Issue A-45, "Decay Heat Removal"[9,10] and the lower number estimated from the NUREG-1150 studies without the assumption of venting. With proper procedures and hardware to ensure a high probability of successful venting, it was assumed that the TW sequence could be virtually eliminated. Under these assumptions, venting improvements were found to be a most effective means of reducing risk for Mark I plants.

Enhanced Depressurization Capability

The Automatic Depressurization System (ADS) consists of relief valves which can be automatically or manually operated to depressurize the reactor coolant system. Actuation of the ADS valves requires DC power. In an extended station blackout after station batteries have been depleted, the ADS would not be available and the reactor would re-pressurize. With enhanced reactor vessel depressurization system reliability, depressurization of the reactor coolant system would have a greater degree of assurance. Together with a low pressure alternate source of water injection into the reactor vessel, the major benefit of enhanced reactor vessel depressurization reliability would be to provide an additional source of core cooling which could significantly reduce the likelihood of severe accidents, especially those at high pressure, such as from the short-term station blackout.

Another important benefit is in the area of accident mitigation. Reduced reactor pressure would greatly reduce the possibility of core debris being expelled under high pressure, given a core melt and failure of the reactor pressure vessel. With the reactor at low pressure, the molten debris will pour out of the failed reactor vessel as compared with being sprayed out of the vessel which could result in a challenge to the containment. In order to increase reliability of the reactor vessel depressurization system, additional assurance of power for the ADS valves may be necessary.

Core Debris Control

Core debris controls, in the form of curbs in the drywell and/or curbs or weir walls in the torus room under the wetwell have been proposed in the past to prevent containment shell melt through and to retain sufficient water to permit fission product scrubbing. However, the technical feasibility for such controls has not been established and the design and installation costs, as well as the occupational exposure during installation, could be significant. There is a growing consensus that water in the containment (from an alternate supply to the drywell sprays) may help mitigate risk by fission product scrubbing and possibly by preventing or delaying containment shell melt by core debris, thereby realizing the improvement envisioned for the core debris controls. Because of the uncertainty in effectiveness, high potential cost, and the potential for water in the drywell to prevent or mitigate containment failure, core debris controls were not recommended.

Procedures and Training

A major element of the Mark I containment performance improvement evaluation involves emergency procedures and training. Current emergency operating procedures (EOPs) are symptom-based procedures that originated from requirements of TMI Task Action Plan[11] item I.C.1. Plant-specific EOPs are generally implemented based on generic Emergency Procedure Guidelines (EPGs) developed by the BWR Owners Group. As part of the balanced approach to examining potential BWR Mark I plant improvements, both the generic EPGs and the plant-specific implementation of EOPs and training have been examined. Revision 4 of the BWR Owners Group EPGs[12] has recently been reviewed by the NRC staff. Revision 4 to the BWR Owners Group EPGs is generally an improvement over earlier versions in that they have been simplified and all open items from previous versions have been resolved. Although the BWR EPGs do not deal fully with severe accidents, the BWR EPGs extend well beyond the design bases and include many actions appropriate for severe accident management.

The improvement to EPGs is only as beneficial as the plant specific EOP implementation and the training that operators receive on use of the improved procedures. Licensees have been encouraged to implement Revision 4 of the EPGs and the NRC has reiterated the need for proper implementation and training of operators.

Station Blackout Rule

The NRC staff also viewed acceleration of staff review and implementation of SBO improvements required by recent revisions to NRC regulations as useful for Mark I plants.

Benefit of Improvements

The overall recommendations of the Mark I CPI program include: 1) improved venting hardware and procedures, 2) alternate supplies of water to the vessel and containment sprays, 3) enhanced depressurization system reliability, 4) improved procedures and training, and 5) accelerated implementation of the SBO rule. These recommended improvements form a set in that, taken as a whole, they complement each other in either prevention or mitigation. These improvements would reduce the likelihood of core melt due to SBO and TW sequences and may delay core melt from ATWS sequences. Given a severe accident, mitigation benefits of the above improvements are also considered to be significant. Mitigation of fission product releases would be realized for all accident sequences, including ATWS scenarios. Venting would be effective in preventing containment failure arising from slow over-pressurization. Venting via the suppression pool would provide significant scrubbing of non-noble gas fission products if no shell failure occurs. Water in the drywell may be effective in preventing or at least delaying failure of the containment shell by molten core debris. Finally, even if shell failure were to occur, and there were a water layer atop the core debris, combined with the drywell spray, fission product releases to the environment would be reduced.

Hardened vents are being evaluated using formal backfit procedures for all Mark I plants not making these improvements voluntarily. In addition, the staff has performed plant specific inspections of the implementation of venting procedures at all Mark I plants to ensure that they have been correctly implemented. The other improvements will be evaluated on a plant-specific basis by licensees for each Mark I as part of the IPE.

BWR MARK II RECOMMENDED IMPROVEMENTS

There are nine Mark II plants on six sites. While the containment designs are similar, there are six different designs of the drywell floor area which comprise three classes as shown in Figure 1. The first class has downcomers in the in-pedestal area from underneath the reactor into the suppression pool (Shoreham and Nine Mile Point 2). The second class has a deep recessed cavity which could contain more than the entire reactor core and internals without flowing through the pedestal doorway (Washington Nuclear Project No. 2 (WNP-2) and LaSalle). WNP-2 has water and LaSalle has a dry concrete plug below the cavity. The third class has a relatively flat in-pedestal floor (Limerick and Susquehanna). Limerick has an in-pedestal drain system (and thus drain lines) while Susquehanna has ex-pedestal drain lines. As with Mark I plants, PRAs for Mark IIs indicate that the dominant severe accident sequences are SBO and ATWS. The TW sequence, although not identified as a significant severe accident sequence in published literature, would be expected to be longer than 30 hours before start of core degradation due to the larger volume of the Mark II.

The improvements recommended for Mark I plants discussed above are also generally applicable for Mark II plants. Venting as a means to prevent core melt as a result of the TW sequence is equally applicable to Mark II plants, although the likelihood of a TW sequence may be less. The issue of venting Mark II plants is more complex than for Mark I plants, however. For SBO sequences, the containment is expected to be well below the primary containment pressure limit at which venting would be initiated up until the time of vessel failure. Following vessel failure and subsequent deposition of molten core materials on the containment floor, there is significant uncertainty in containment response which will also vary depending on the different designs of the containment floor as discussed above. In designs with a deep recessed cavity under the vessel, the molten core material is expected to be contained in the cavity. It is unlikely that early containment failure would occur simply from overpressurization due to core-concrete interactions (CCI), at least not before the core material eroded through the concrete floor. For other Mark II designs, molten core material could flow directly down the in-pedestal downcomers or, in other designs, flow across the drywell floor and down the ex-pedestal downcomers. Significant uncertainty exists concerning the amount of core material that would flow down the downcomers and the steam that would be produced as the molten core material contacted the water in the suppression pool. However, it is possible that steam would be produced so rapidly that there would not be sufficient time to prevent containment overpressure failure by venting. In addition, the response of the downcomers to the molten core material is also uncertain. It is possible that the downcomers, or alternatively drain lines, might fail from contact with molten core materials, resulting in suppression pool bypass and a subsequent unscrubbed vent. Both issues of downcomer failure and core material interaction with suppression pool water are areas of continuing research. While the staff believes that venting improvements would be of benefit for Mark II plants, the benefits of venting may be less than for Mark I containments because of a possible reduced likelihood of a TW sequence and because of the uncertainty in containment response following core melt and subsequent vessel failure. Therefore, the NRC staff did not recommend requiring a generic backfit of the hardened vent for Mark II plants, but recommended evaluation on a plant-specific basis as part of the IPE.

The staff has recommended[13] that the same potential improvements recommended for Mark I plants also be evaluated on a plant-specific basis by licensees for Mark II plants. However, it was recommended that venting improvements should be evaluated on a plant-specific basis as part of the IPE. This change from the recommendations for Mark I plants was due to the apparent reduced benefit of venting for Mark II plants and the differences among Mark II designs.

BWR MARK III RECOMMENDED IMPROVEMENTS

The NUREG-1150 PRA for Grand Gulf (the only Mark III analysis available) indicates that the dominant core damage sequences are SBO and ATWS. However, the probability of core melt for the Grand Gulf plant has been estimated to be very low ($\sim 10^{-6}$). The same improvements recommended for Mark II containments are also generally applicable to Mark III containment types and were recommended for evaluation as part of the IPE program[13]. However, the volume of the Mark III containment is significantly larger than either the Mark I or Mark II containments. Thus, the likelihood that, for example, the venting improvement would be needed is believed to be less. Unlike the Mark I and Mark II containments, the Mark III is not inerted and makes use of hydrogen igniters to control hydrogen concentrations in containment during a severe accident. The dominant containment failure mode for the Mark III plant studied by NUREG-1150 PRA is from hydrogen combustion and steam explosions in the drywell since the dominate accident sequence is SBO during which the igniters would not be available. Thus the NRC staff also recommended[13] that backup power to the hydrogen igniters also be evaluated as part of the IPE program.

PWR DRY CONTAINMENT RECOMMENDED IMPROVEMENTS

The second draft of NUREG-1150 indicates that the early containment failure modes are containment overpressurization due to direct containment heating (DCH) effects (DCH results from ejection of melt as the vessel fails at high pressure including the effects of hydrogen generation and combustion), in-vessel steam explosion leading to ejection of the vessel upper head and impacting on the containment dome (alpha mode failure), and containment isolation failure. While these are the possible containment failure modes, NUREG-1150 indicated that the conditional early containment failure probability given a core melt was very low. With respect to late containment failure, NUREG-1150 indicates that the late containment failure modes are non-condensible gas overpressurization and basemat melt through or leakage. The likelihood of a late containment failure is estimated to vary from a few percent to about 25%. For the Surry plant, (a subatmospheric containment) containment bypass was found to be the dominant contributor to risk. However, this area was not investigated further within the CPI program because of other ongoing activities for resolution of Generic Issue 105, "Interfacing LOCAs at LWRs". DCH was not estimated to be important for the dry containments studied in the second draft of NUREG-1150 mainly because the primary system may be depressurized as a result of temperature-induced failure of the pressurizer surge line. However, this is an area of large uncertainty and the importance of DCH to risk is the subject of continuing research. Depressurization to avoid DCH is being investigated as part of the accident management research program.

Past concerns about possible containment failure due to hydrogen combustion during a severe accident resulted in Generic Issue 121, "Hydrogen Control for Large Dry PWR Containments." NUREG-1150 did not identify hydrogen combustion as a significant threat to the containment for the two PWR plants investigated. However, the NRC staff does not know whether or not this conclusion can be extended to all PWR containments. Therefore, hydrogen combustion for dry containments was studied further in order to resolve this generic issue.

Deflagration is the most likely mode of hydrogen combustion in a dry containment. Hydrogen combustion on a global basis is not believed to be a significant threat to large dry containments. However, less firm conclusions have been reached for the smaller subatmospheric containments. Since the subatmospheric containments operate at about 10 psia and have less air to dilute the hydrogen, they may develop detonable mixtures of hydrogen on a global basis. For example, based on an assumption of the 75% metal/water reaction, the hydrogen volume concentrations in dry hydrogen-air mixture are estimated to be about 17% for subatmospheric containments whereas the concentrations are estimated to be about 10% to 13% for most atmospheric containments. Furthermore, depending on the degree of compartmentalization and the release point of the hydrogen from the primary vessel, higher local concentrations of hydrogen could be formed. High local concentrations and flame acceleration in the presence of obstacles could be a mechanism for deflagration-to-detonation transition (DDT). Since local detonable mixtures of hydrogen could be formed in either type of dry containment during a severe accident, the containment and important equipment, if any is nearby, could be damaged following a local detonation or a deflagration with accelerated flames. Again, NUREG-1150 did not identify any significant threat from local hydrogen detonations for the plants investigated, but the staff does not know if this conclusion can be extended to all PWR plants and recommended plant-specific evaluations. It should be noted that currently available computer codes have been shown to overestimate mixing of hydrogen in the containment and may not be adequate to evaluate the potential for high local concentrations of hydrogen[14]. Thus, any analyses should be supplemented by judgement as to the adequacy of the results and consideration of the impact of higher than predicted hydrogen concentration due to stratification. Given an estimate of local concentration of hydrogen and a knowledge of compartment configuration, NUREG/CR-5275[15] provides a discussion of one method that has been used to evaluate qualitatively the potential for local hydrogen detonation.

Therefore, the NRC staff has recommended[13] that owners of dry containments examine locations of possible hydrogen evolution and evaluate the potential for damage to the containment and important equipment due to localized detonations as part of the IPE program. The NRC staff believes that consideration of hydrogen control under the IPE and accident management research program represents an acceptable resolution of GI-121 and will make a recommendation concerning resolution of this issue in the near future.

PWR ICE CONDENSER RECOMMENDED IMPROVEMENTS

The second draft of NUREG-1150 provides the most up-to-date insights into the important contributors to core damage and potential containment challenges facing the ice condenser plants. This study calculated a total mean core damage frequency from internal events of $\sim 10^{-5}$ per year. The Sequoyah risk analysis indicates that containment bypass, including interfacing systems (IS) LOCA and steam generator tube rupture (SGTR), dominates early fatality risk. Bypass events again emerge, along with station blackout events, to dominate latent cancer fatality risk. (Bypass, itself, accounts for over 80 percent of mean early fatality and latent cancer fatality risk.) In recognition of these challenges, the NRC has established separate programs to examine IS-LOCA and testing of steam generator tubes.

In contrast to estimates for the dry containment, the NUREG-1150 second draft still predicts that DCH is an important contributor to mean risk at Sequoyah. DCH accounts for about 8 percent of early fatality risk and 23 percent of latent cancer fatality risk (however, it must be recognized that significant phenomenological uncertainty still exists regarding DCH). The CPI program investigated a number of possible improvements to preserve containment integrity following a high pressure melt ejection. The improvements investigated included venting, inerting, and hydrogen igniter operation under station blackout conditions. The staff looked at these improvements separately and in some combinations. The results indicated that predicted containment pressures could be reduced somewhat but not enough to conclusively say that the ultimate containment pressure capability would not be challenged. Therefore, no recommended improvements to prevent containment failure due to DCH emerged.

The NRC's accident management research program has examined this issue and has concluded that full depressurization could significantly reduce, if not eliminate, DCH. The CPI program has made use of the ongoing accident management work on this subject and has evaluated its impact on potential containment improvements for ice condensers. Under the assumption of successful depressurization, that is, DCH not present, the staff looked at the associated containment challenge associated with hydrogen production resulting from the primary system depressurization. The results indicate that, in the absence of hydrogen mitigation (because of, for instance, loss of AC power for the hydrogen igniters in a station blackout) hydrogen concentrations high enough to support local detonations are predicted. As a result of the small containment volume, these local detonations could fail the containment. Postulated improvements include providing backup power to the igniters in the event of loss of emergency AC power, and the addition of igniters in the ice beds to preclude a detonation in that region. Computer predictions of ice condenser performance with igniters fully functional indicate that operation of existing igniters could be expected to prevent containment failure from hydrogen detonation. Some uncertainty still exists regarding the need for and benefits available from installation of additional igniters in the ice beds themselves. Nevertheless, it appears that the most important conclusion to be drawn from the CPI program for ice condensers is that for depressurization to be successful in preventing containment failure, the igniter system should be functional. Therefore, the NRC staff recommended that backup power to the hydrogen igniters be evaluated as part of the IPE[13].

SUMMARY AND CONCLUSIONS

The NRC has evaluated important containment challenges for all U.S. containment types, making use of the latest information from PRA studies and severe accident research. For Mark I containments, the staff has recommended improvements that should significantly reduce risk from Mark Is by both reducing the likelihood of a severe accident and improving containment performance given a core melt. These improvements are either being evaluated under the formal backfit procedures for U.S. Mark I plants or being evaluated on a plant-specific basis as part of the IPE program. No improvements were found for other containment types that the staff would recommend for generic backfit on all containments of a given type. However, a number of insights concerning containment challenges during severe accidents and potential improvements have been identified that have been recommended for further evaluation as part of the IPE program.

REFERENCES

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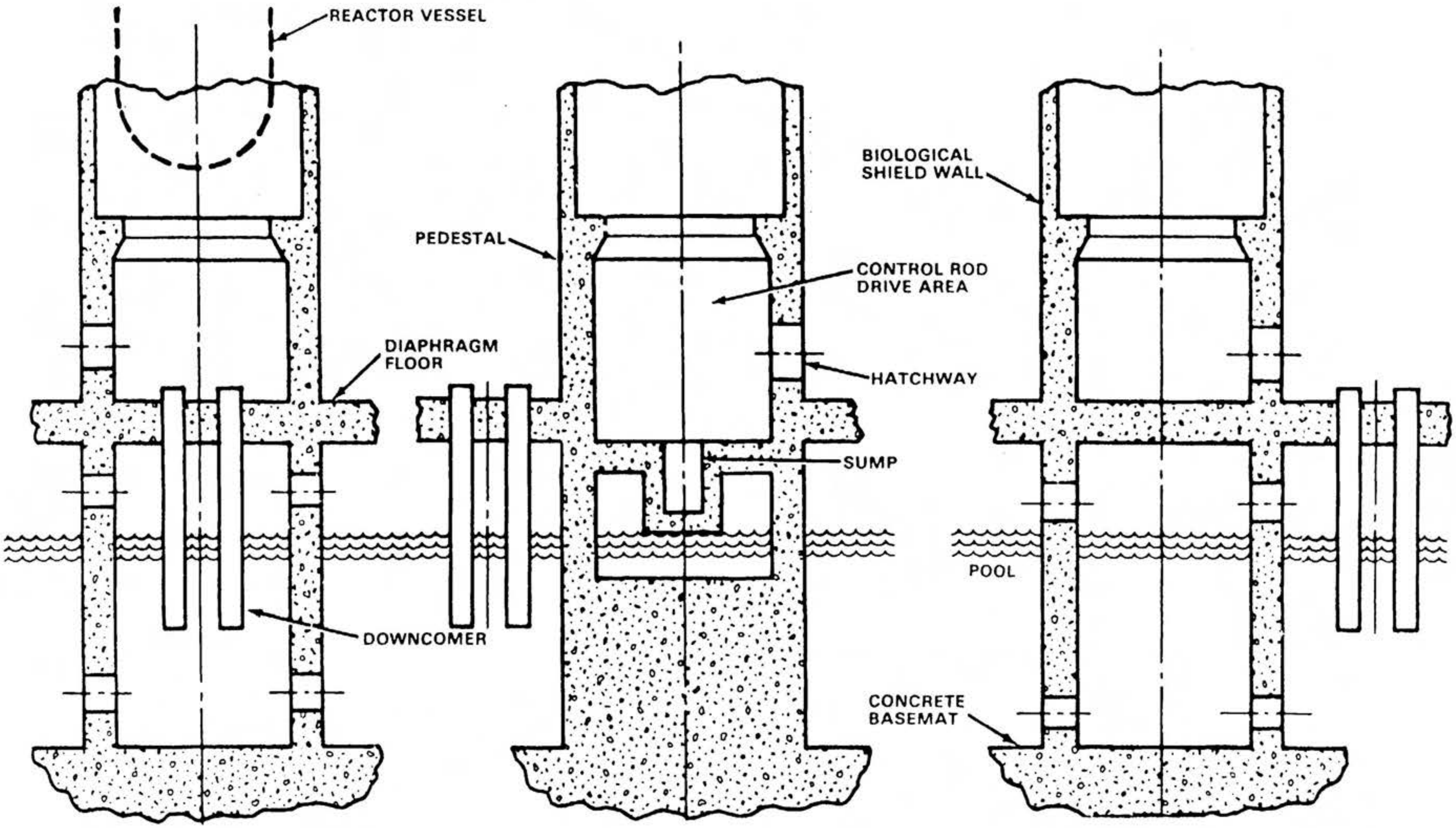

SHOREHAM
NINE MILE POINT 2

WNP-2
LASALLE 1/2

LIMERICK 1/2
SUSQUEHANNA 1/2

FIGURE 1 - VARIATIONS IN MARK II PEDESTAL CONFIGURATIONS