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SEVERE ACCIDENT PROGRESSION PERSPECTIVES BASED ON IPE RESULTS

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ABSTRACT

Accident progression perspectives were gathered from the level 2 PRA analyses (the analysis of the accident after core damage has occurred involving the containment performance and the radionuclide release from the containment) described in the IPE submittals. Insights related to the containment failure modes, the releases associated with those failure modes, and the factors responsible for the types of containment failures and release sizes reported were obtained. Complete results are discussed in NUREG-1560[1] and summarized here.

I. BACKGROUND

Accident progression perspectives were gathered from the level 2 PRA analyses (the analysis of the accident after core damage has occurred involving the containment performance and the radionuclide release from the containment) described in the IPE submittals. Insights related to the containment failure modes, the releases associated with those failure modes, and the factors responsible for the types of containment failures and release sizes reported were obtained. Complete results are discussed in NUREG-1560[1] and summarized here.

The approach used to obtain the level 2 insights consisted of: (1) comparing results of IPE submittals for plants with similar containments to obtain average values and ranges of failure probabilities, (2) identifying the major contributors to containment failure and fission product release cited in the IPEs for particular containment types, (3) comparing the IPE results to those found in previous PRA studies such as NUREG-1150 [2], (4) establishing the reasons for the variation in the results, and (4) identifying the modifications and changes made by the licensees in response to their IPE findings.

II. GENERAL PERSPECTIVES

When the accident progression analyses in the IPEs are viewed globally, they are, for the most part, consistent with level 2 PRA analyses performed previously. Failure mechanisms identified in the past as being important are shown to be important in the IPEs also. The significance of individual containment failure mechanisms is often determined by particular features of a containment class.

The importance of early fission product releases to all risk measures (i.e., acute and latent health effects including land contamination) has been established in past PRAs which included consequence calculations. In keeping with the significance of such early releases, the level 2 analysis descriptions found in the IPE submittals emphasized the phenomena, mechanisms, and accident scenarios which could lead to early releases. These involve early structural failure of the containment, containment bypass, containment isolation failure, and for some BWRs deliberate venting of the containment.

As a group the PWR large dry containments analyzed in the IPEs have discernibly smaller conditional probabilities of early structural failure than the BWR pressure suppression containments analyzed, as indicated in Figure 1. On the other hand, containment bypass, as well as isolation failures, are, in general, more significant for the PWR containments. However, because of the considerable range in the results, these general trends are often not true for individual IPEs.

III. BWR CONTAINMENTS

The BWR plants are separated into three groups, according to the type of pressure suppression containment used: BWR Mark I Containments, BWR Mark II

Containments, and BWR Mark III Containments.

The results follow expected trends and indicate that the early Mark I containments are, in general, more likely to fail during a severe accident than the later Mark II and Mark III designs. However, the ranges of predicted failure probabilities are quite large for all containment designs and there is significant overlapping of the results. The variability in the results is attributable to a combination of factors including plant design differences such as the reactor pedestal and drywell floor configuration, drywell flooding, containment construction (steel versus concrete), and combustible gas control; modeling assumptions; and differences in recovery actions that could be taken during a severe accident. However, IPEs for plants in all three containment groups reported a significant probability of early or late structural failure conditional on core damage occurring. These results are expected because smaller pressure suppression containments have been found to have relatively high containment failure probabilities in past PRAs. The probabilities of the various failure modes are shown for each BWR pressure suppression containment group in Figure 2. In general, the factors that influence the failure modes are not the same for each group.

Twenty-two BWR units (17 IPE submittals) are housed in Mark I containments. All of the plants in the BWR 2/3 group and most of the plants in the BWR 3/4 group have Mark I containments. These containments have relatively high strength but small volumes and rely on pressure suppression pools to condense steam released from the reactor pressure vessel during an accident. The IPE results indicate a significant probability of early and/or late containment failure for most of the Mark I containments.

Shell melt-through is found to be the most important contributor to early containment failure for Mark I containments, given core melt. This failure mechanism has a relatively high likelihood of occurring because, for most Mark I containments, the reactor pedestal and the drywell floor are at the same level and openings exist between the pedestal region and the floor. This design allows the core debris to flow across the drywell floor and fail the steel drywell shell either by direct melt-through or via creep rupture. The capability to flood the drywell floor, the design configuration of the drywell, and assumptions regarding core debris dispersal on the drywell floor determine, on a plant-specific basis, how significant shell melt-through is as a containment failure mechanism. In this regard the presence of a water pool on the drywell floor is found to mitigate shell melt-through in all of the submittals, while the design of the drywell sump and drywell floor can prevent or mitigate shell melt-through in some Mark I containments. For example, containment sumps in one plant are large enough to contain the molten core material and thus prevent it from reaching the containment boundary. Finally, the amount of core debris released to the drywell and the fluidity of the core debris assumed in the IPEs also determine whether or not shell melt-through occurs. A number of utilities are being proactive and are identifying minor hardware modifications and changes in procedures to ensure a flooded drywell floor prior to reactor vessel melt-through. Several IPEs also discuss the possibility of relaxing the restrictions on drywell spray initiation in the current EOPs, thus providing greater assurance that there would be water on the drywell floor.

High pressure and temperature loads at the time the core debris melts through the reactor vessel are also a significant contributor to early containment failure for Mark I containments. This failure mechanism occurs in Mark I containments because of their relatively small volumes. The RCS pressure at vessel melt-through, the containment failure location, and modeling assumptions regarding the rate of RCS depressurization and amount of core debris dispersed determine whether this failure mechanism is a significant contributor to early containment failure for individual Mark I containments.

Containment challenges from ATWS sequences are important in a number of IPEs for plants with Mark I containments. These sequences belong to an accident class in which containment heat removal and containment venting are inadequate. In ATWS events the energy deposited to the containment can overwhelm the normal containment heat removal mechanisms as well as the available vent paths, leading to early core damage and containment failure. The inability to remove heat from the containment causes containment failure to occur before core damage. The containment failure in turn can lead to the loss of emergency core cooling systems (due to a loss of net positive suction head for pumps drawing from the suppression pool, for instance) with resulting core damage and vessel failure. Depending on the accident progression, core damage could occur first, but containment failure follows quickly. These accidents have been found risk significant in past PRAs since core damage, vessel failure and containment failure can occur within a short time interval, thus producing conditions for significant release to the environment. However, many IPE submittals report that, by proper RPV level control and by opening the maximum number of vent paths, many ATWS scenarios can be controlled. The significance of ATWS events in the different IPEs depends on some plant specific features, such as the ability of pumps to work with saturated water, as well as on assumptions regarding power level, point in the fuel cycle, and rapidity of operator response.

Accidents with successful reactor scram but loss of containment heat removal are found to be relatively unimportant in all the Mark I IPEs. The ability to vent the containment is sometimes a major factor in reducing the importance of this class of accident. In general, venting is used in the Mark I IPE analyses to reduce releases and is sometimes credited for preventing core damage in accidents involving loss of containment heat removal. However, a few utilities state in their IPEs that their analyses indicate that the installation of a hardened vent does not significantly impact risk and therefore is only of marginal benefit. The pressure at which venting should be started is also examined in detail by several utilities. The impact of high temperatures on the structural capability of the drywell is also noted. For example, one IPE reports that at 400°F the containment could fail at pressures below the current venting pressure in the EOPs. Further analysis is recommended that could refine the vent actuation pressure.

Accidents that bypass containment are found to be not important for Mark I containments, according to the IPEs. Interfacing systems LOCA are found to be not important for BWR Mark I containments because of their relatively low frequency compared with the frequency of accidents that dominate the CDF and which can lead to early structural failure. Also accidents that involve failure to isolate containment are also found not important for Mark I containments in the IPEs because of their relatively low frequencies.

High pressure and temperature loads caused by core/concrete interactions are a significant contributor to late containment failure for Mark I containments. Gradual pressurization at high temperatures caused by non-condensible gases and steam released from the drywell floor during core/concrete interactions can fail Mark I containments several hours after vessel melt-through. The significance of this failure mechanism to late containment failure is determined by whether or not the drywell is flooded, the design configuration of the drywell, the availability of sprays or venting, and modeling assumptions regarding the quantity and temperature of core debris dispersed across the drywell floor.

Eight BWR units (five IPE submittals) are housed in Mark II containments. Four units are of the BWR 4 type, while the other four units are BWR 5 designs. Mark II containments retain many of the features of the older Mark I containments from which they evolved: They also are characterized by relatively high strength but small volume, and in the event of an accident they depend on a pressure suppression pool to condense the steam released to the containment from the reactor coolant system. However, unlike the Mark I group, most of the Mark II containments are of concrete construction. The exception is one plant (one unit) where the containment consists of a steel shell.

As Figure 2 shows, the conditional probability of early failures varies considerably among the Mark II containments. To a large extent this variation can be attributed to variations in plant-specific containment features, specific plant features play an important role in accident progression in Mark II containments, but modeling assumptions play a role as well. Failure mechanisms found to lead to early failure of Mark II containments include:

- Containment over-pressure failure due to loss of containment heat removal or inadequate containment heat removal.
- Fuel-coolant interaction (FCI) and direct impingement of core debris on the containment boundary.
- Rapid pressure and temperature rise at the time of reactor vessel failure (important in only a few Mark II IPE analyses).

With the exception of one plant, containment venting does not play a significant role in the accident progression in the Mark II plants. Accidents that bypass containment (such as interfacing systems LOCA) or involve containment isolation failure are not important contributors to the CDF in any of the IPEs for Mark II plants. Similar to the Mark I situation, these accidents are also found not to be important because their frequencies of occurrence are so much lower than the frequencies of early structural failure caused by other accidents that dominate the CDF.

As with the Mark I IPEs, high pressure and temperature loads caused by core/concrete interactions are significant contributors to late containment failure for Mark II containments, according to these IPEs. In addition, some Mark II IPEs report that late containment failure also results when significant discharge from safety relief valves (SRV) into a hot suppression pool occurs. This assumption is based on the fact that only very limited data exists to support containment integrity at a high SRV discharge rate and elevated containment pressure and temperature. There are a number of issues with large uncertainty affecting containment failure under these conditions. These issues include the condensation phenomena in the suppression pool, the temperature profile for the quencher device used, and the effect of elevated water levels on the hydrodynamic loads.

Four single unit BWRs, described in four separate IPE submittals, are housed in Mark III containments. All four plants are a BWR 6 design. Mark III containments are significantly different from their predecessors, the Mark I and Mark II designs, and this is reflected in the different accident progression expected with these containments. The total free volume of a Mark III containment is significantly greater than that of a Mark I or Mark II. The containment volume to thermal power ratio is about four times that of Mark Is or Mark IIs while the containment design pressure and the estimated failure pressure are significantly lower than those of Mark Is and Mark IIs. Because of their relatively larger volume Mark

III containments are not inerted but rely on glow plug igniters to burn off accumulating hydrogen during a severe accident and prevent energetic hydrogen events.

Since the drywell is completely enclosed by the primary containment in the Mark III design, a release to the environment will be scrubbed by the suppression pool if the containment fails but the drywell remains intact. Early drywell failure is therefore an important consideration in the accident progression, and radionuclide release is highest when both the containment and the drywell fail. Since the drywell has a much higher design pressure than the containment, such a failure would most likely be caused by energetic events such as hydrogen combustion and the phenomena associated with vessel breach. These considerations are reflected by the IPE results.

While the causes for early containment failures are not discussed in detail in most of the IPE submittals for Mark III plants, early containment failure seems to be primarily caused by energetic events, such as fuel-coolant interactions (FCIs) or hydrogen burns. The wide spread in the conditional early failure probability among the four Mark III plants shown in Figure 2 is mainly due to the small failure probability assigned to one plant, where ATWS loads are identified in the IPE as the only mechanism capable of causing an early containment failure. While the dismissal of other failure mechanisms may be partly attributable to design differences between this plant and other Mark IIIs, modeling assumptions of the IPE analysis play a significant role as well.

A venting scheme considered in one Mark III plant produces a significant contribution to the frequency of radionuclide release. Venting of the primary system using the Main Steam Isolation Valves (MSIVs) results in an early release and is the most severe release mode in this IPE. According to the analysis, MSIV venting is directed by the BWR emergency procedure guidelines for containment flooding in response to loss of RPV level indication. The procedure requires that a vent path to the RPV be established as containment flooding proceeds beyond the top of the drywell weir wall. This vent path is realized by bypassing the containment interlocks and, regardless of potential releases, opening the MSIVs. This results in a release that bypasses the containment. The licensee suggest in the IPE submittal that this procedure be revisited.

Principal contributors to late failures in Mark III containments are late combustible gas burns and phenomena associated with core/concrete interaction. As for the other BWR containments, containment bypass as well as containment isolation failures are small for most of the IPEs of plants using Mark III containments.

IV. PWR CONTAINMENTS

For the purpose of identifying containment performance perspectives from the IPEs submitted for PWR plants, the PWRs are separated into two groups according to containment type: large dry containments, including those which operate with a subatmospheric internal pressure, and ice condenser containments. In addition to the PWRs one early BWR, Big Rock Point, is housed in a large dry containment. Containment performance results for all the PWRs in the two groups are shown in Figure 3. The results indicate that in both PWR groups most of the containments have relatively low conditional probabilities of early failure.

A large variability exists for both containment groups in the contributions of the different failure modes. This variability is due to plant specific design features, but also due to the modeling assumptions made in the different IPE analyses. The uncertainty of the phenomena associated with HPME, for instance, is reflected in the variation in likelihood and in magnitude for HPME loads found in the IPEs. Differences in assigning credit for recovery of the core in-vessel after core damage also plays a role in broadening the range of the containment failure results reported.

Sixty-four PWR reactor units and one BWR unit, described in forty-three submittals, are housed in large dry containments. For seven of the PWR units (four submittals) the containments are kept at an internal pressure that is a somewhat below atmospheric pressure. All of these containments rely on structural strength and large internal volume to maintain containment integrity during an accident.

In general, only very severe and rapid pressure loads will fail these containments early, and, with a few notable exceptions, the probability of early containment failure for plants in this group is quite small. Important factors for early containment failure are found to be the following:

- Phenomena associated with HPME.
- In a few cases, specific design features leading to unique and significant failure modes.
- Containment bypass, especially steam generator tube rupture, an important source of significant early release.

The most important challenges to containment integrity before or at vessel breach are those associated with high pressure melt ejection (HPME). The containment loads associated with HPME are generated by the addition of mass and energy to the containment atmosphere from a number of sources. This combined load is referred to as the DCH load in some IPEs. There are significant uncertainties related to the containment pressure loads that can be produced from the energetic events associated with HPME. The pressure of the reactor coolant system (RCS) at vessel breach is obviously a factor, as is the geometry of the reactor cavity and the presence or absence of water in the cavity. These parameters, plus some additional assumptions, will determine what the estimated pressure rise at vessel breach will be. However, the estimated containment pressure load before vessel breach also plays an important role in determining the early failure probability. The containment pressure capability curve, particularly the shape of the distribution assumed at the lower pressure end of the curve, is also important. Since a point estimate (rather than a distribution) is used in most of the IPEs, a single pressure load estimate is usually obtained and compared with the containment pressure capability to determine the failure probability.

In some IPEs the probability of early containment structural failure is determined to be not credible. In one group of PWR IPE submittals, which use similar analysis methods, the estimated early containment pressure loads are less than the containment pressure capability, and therefore early containment structural failure is assumed not to occur. It is argued in these IPEs that early containment failure modes, such as those discussed above, are not expected to challenge the containment.

The predicted containment pressure loads are higher in those IPEs that reported relatively higher early containment failure probabilities (i.e., from 0.05 to 0.10) than the IPEs that predict no early containment failure. Usually in these analyses the containment failure pressure is reached when the pressure prior to vessel breach, the "base" pressure, is combined with the pressure rise at vessel breach. Depending on the individual submittal, the higher pressure loads may be due to a high containment base pressure before vessel breach, or a bigger pressure rise at HPME, or both. The primary cause for a high base pressure is usually the loss of containment heat removal with successful core injection.

In a number of IPEs specific containment features lead to unique and significant failure modes. For instance, the large probability value of early containment failure in one IPE (0.32) arises from the location of the engineered safeguards (ESF) sump. The IPE postulates a flow of molten core debris from the reactor cavity into the ESF sump and subsequently into the ESF recirculation piping. In the IPE analysis the debris is assumed to melt through the pipe wall eventually and enter the Auxiliary Building. The maximum failure area is presumed to be twice the area of an ESF recirculation pipe (there are two pipes), resulting in a large containment failure area.

Containment bypass, especially SGTR, is an important source of early release in many IPEs for plants with large dry containments. Containment bypass failures include those from interfacing-system LOCA (ISLOCA), steam generator tube rupture (SGTR), or temperature-induced SGTR. The probability of ISLOCA and SGTR is determined in the CDF analyses of the IPE. Temperature-induced SGTR is calculated as part of the accident progression analysis. It occurs if one or more steam generator tubes have a creep rupture due to the flow of high temperature hot gases from the core when the RCS is at system pressure. For those IPEs where containment bypass has a significant contribution, SGTR is normally the dominant contributor.

Isolation failure is assumed to be negligible in some PWR IPEs for plants with large dry containments, and assumed to have a large conditional probability in others. A large probability of isolation failure is most likely in those IPEs which assume a lack of operator actions to locally or remotely close the isolation valves if no containment isolation signal is provided.

The IPE results for large dry containments show that the dominant late containment failure mode is containment over pressurization, which occurs when containment heat removal capability (CHR) is lost.

Nine PWR units, described in five IPE submittals, are housed in ice condenser containments. All of these plants utilize a Westinghouse four loop reactor system design. Ice condenser containments have smaller volumes as well as smaller volume to thermal power ratios than other PWR containments. Their containment strength is also less than that of other types. To avoid excessive containment pressure these pressure suppression containments rely on the capability of the ice condenser system to absorb energy released accidentally from the reactor coolant system. Similar to BWR Mark III containments, ice condenser containments rely on glow plug igniters to burn off accumulating hydrogen during a severe accident and thus prevent energetic hydrogen events. Seven of the nine ice condenser units have a cylindrical steel containment surrounded by a concrete secondary containment. The remaining two units feature reinforced concrete containments with steel liners, and lack secondary containments.

Figure 3 shows the containment failure probabilities for this group. Among the five ice condenser IPE analyses the most important causes of early containment failure are:

- Direct impingement of core debris on the containment in the seal table room.
- Rapid steam generation, DCH, and hydrogen burns.
- Over pressurization when containment heat removal is not available.

Although the majority of the ice condenser IPEs used data from the NUREG-1150 Sequoyah analysis in their accident progression models, additional plant specific models result in lower failure probabilities than found in NUREG-1150. The primary cause of late containment failure for these containments is found to be overpressure failure in the IPEs. Draining of the refueling water storage tank into the failed vessel, and therefore the reactor cavity, with subsequent boil-off and ice melt contributes to this failure mode. Containment bypass is dominated by ISLOCA and SGTR initiators, but one IPE finds induced SGTR to be dominant due to the restart of the reactor coolant pumps (RCPs) when inadequate core cooling conditions exist.

The early failure conditional probabilities for the ice condenser IPEs are on average smaller than the values obtained for the large dry and subatmospheric IPEs. The conditional probabilities of isolation failure and early failure found in the IPEs for the ice condenser containments are, on average, smaller than the values obtained from the IPEs for plants with large dry and subatmospheric containments. This smaller failure probability for ice condenser containments as a group is somewhat surprising. The containment volume to reactor thermal power ratios for ice condenser containments are a factor of two to three less than those for large dry containments and subatmospheric containments. The ultimate containment pressure capabilities for ice condenser containments are also smaller than those for large dry and subatmospheric containments (e.g., 80 psig versus 130 psig). No single reason for the lower (on average) ice condenser failure probabilities is apparent from the IPE submittals. Modeling assumptions such as the availability of the ice condenser and its availability to absorb the energy produced by phenomena like DCH play a role and are discussed below. However, it must also be remembered that there are only five IPEs for ice condenser plants, a relatively small sample, while there are forty-five IPEs for plants with either a large dry or subatmospheric containment. Therefore, much greater variation in the likelihood of early failure can be found in this larger group.

V. CONCLUSIONS

Differences in containment designs account for much of differences in failure probabilities indicated in Figure 1. This is true for the variations between containment classes but also for differences between individual plants in the same containment class. In a significant number of cases unique, plant specific containment features were identified in the analyses as leading to important failure mechanisms. However, differing assumptions in the accident progression modeling also play a major role in explaining the significant range in the results obtained. Since there is still considerable uncertainty regarding the loads imposed on containments by the phenomena postulated in a level 2 PRA analysis, differences in modeling assumptions are not surprising.

ACKNOWLEDGMENTS

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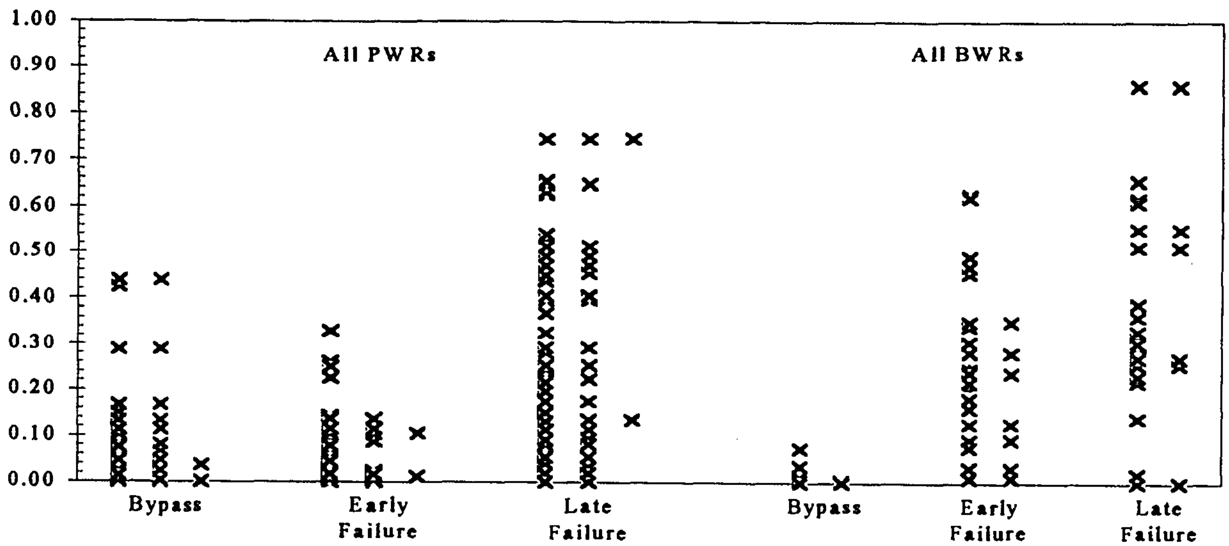

Figure 1 Conditional Containment Failure Probabilities for all PWRs and all BWRs as Reported in the IPEs

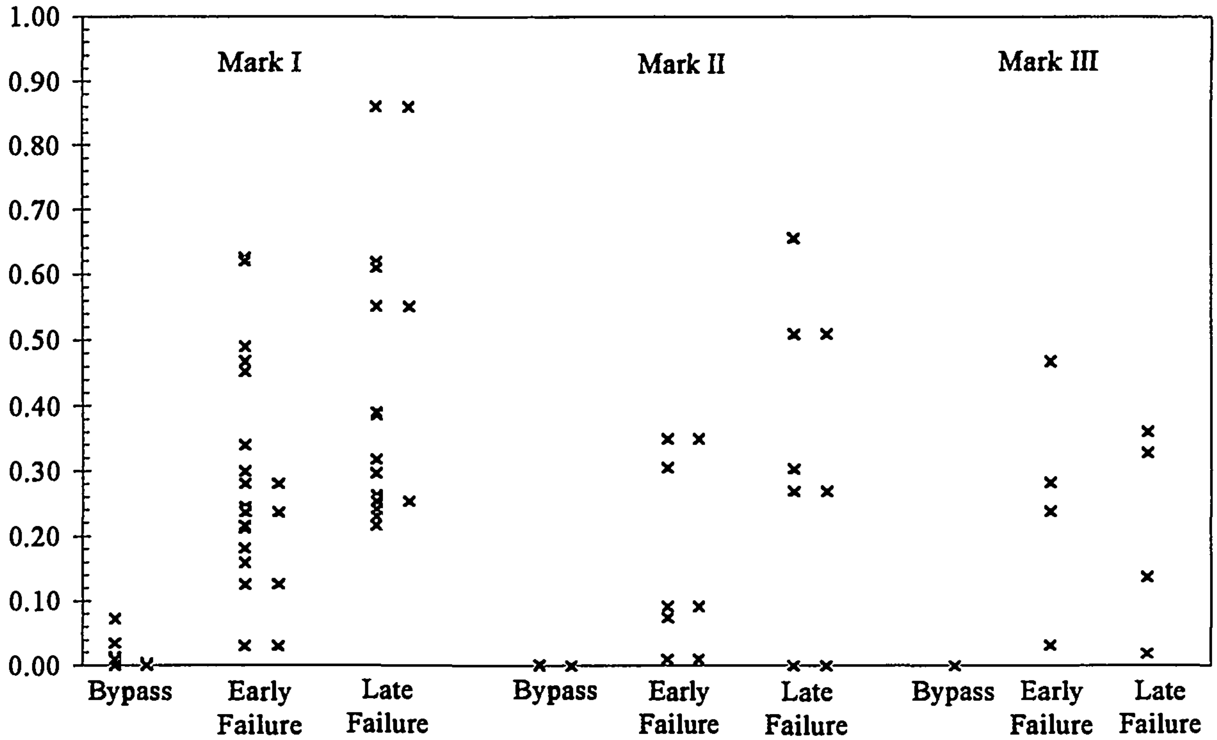

Figure 2 Conditional Containment Failure Probabilities for BWRs by Containment Type as Reported in the IPEs

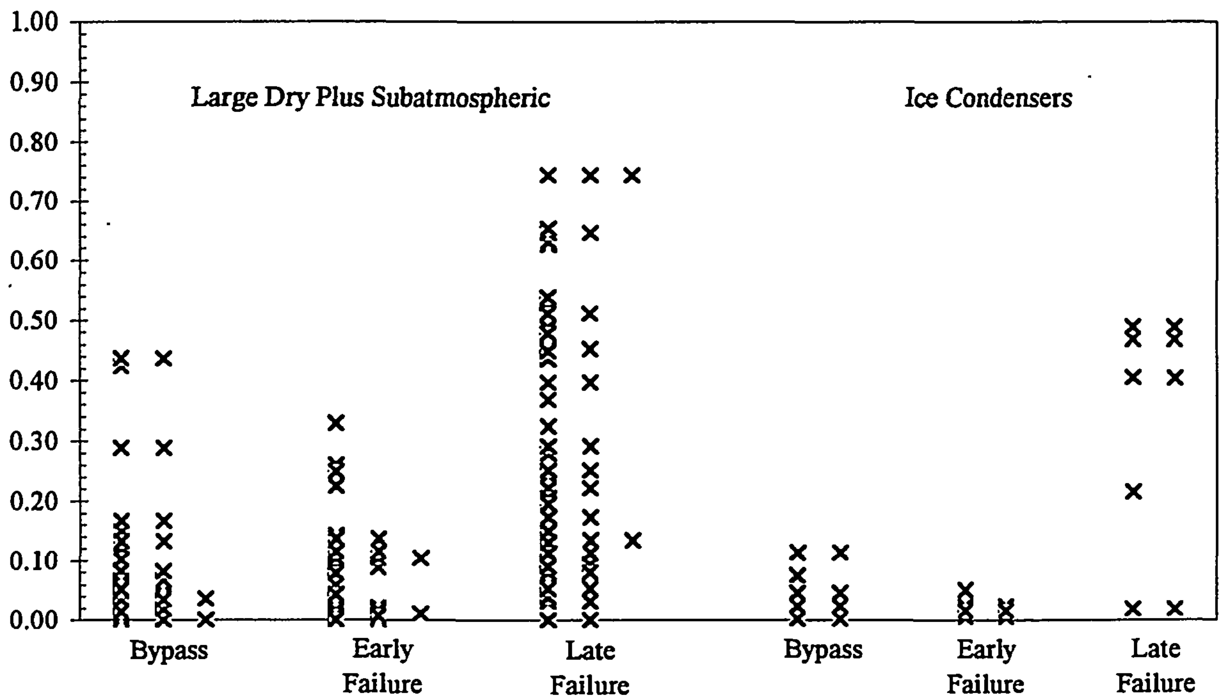

Figure 3 Conditional Containment Failure Probabilities for PWRs by Containment Type as Reported in the IPEs