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SEVERE ACCIDENT PROGRESSION PERSPECTIVES FOR MARK I CONTAINMENTS BASED ON THE IPE RESULTS*

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ABSTRACT

Based on the level 2 analyses in the IPE submittals accident progression perspectives were obtained for all containment types. These perspectives consisted of insights related to the containment failure modes, the releases associated with those failure modes, and the factors responsible for the results reported. To illustrate the types of perspectives acquired regarding severe accident progression, insights obtained for Mark I containments are discussed here.

Mark I containments have relatively high strength but small volumes and rely on pressure suppression pools to condense steam released from the reactor coolant system during an accident. In these containments those accidents that cause structural failure of the drywell shortly after the core debris melts through the reactor vessel were found to be dominant contributors to risk. The importance of individual containment failure mechanisms depended on plant-specific features and in some cases on modeling assumptions; however, the following mechanisms were found important for many Mark I containments: (1) Drywell shell melt-through caused by direct contact with the core debris (i.e., liner melt-through), and (2) Drywell failure caused by rapid pressure (and temperature) pulses at the time of reactor vessel melt-through

Drywell failure caused by gradual pressure (and temperature) buildup due to gases and steam released during core/concrete interactions is important in some IPEs. In other IPEs venting was found to be an important contributor. However, accidents that bypass containment (such as interfacing systems LOCA) or involve containment isolation failure were not important contributors to the CDF in any of the IPEs for Mark I plants. These accidents are also not important to risk (even though they can involve large fission product release) because their frequencies of occurrence are so much lower than the frequencies of early structural failure caused by other accidents that dominate the CDF.

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BACKGROUND

NRC issued Generic Letter 88-20 in November 1988 requesting that all licensees perform an Individual Plant Examination (IPE) to identify any plant-specific vulnerabilities to severe accidents, and to report the results to the Commission. There are four general purposes of the IPE program for the licensees, as stated in the Generic Letter:

1. Develop an appreciation of severe accident behavior;

2. Understand the most likely severe accident sequences that could occur at the plant;

3. Gain a more quantitative understanding of the overall probabilities of core damage and fission product releases; and

4. If necessary, reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

The Executive Director of Operations to the Office of Nuclear Regulatory Research in NRC recommended on May 12, 1993 that NRC should publish a document highlighting the significant safety insights resulting from this program and showing how the safety of reactors has been improved by the IPE initiative. The IPE Insights Program was initiated to document such safety insights. Significant insights and improvements identified from the IPE submittals are captured and are documented in a soon-to-be-published NUREG report.

The major insights to be gained through the IPE Insights program include:

- How has the IPE program affected reactor safety?

- What is driving the CDF and containment performance?
  - What are the important design and operational features that affect the CDF and containment performance?
  - How important is the role of the plant operators?
  - How much influence do the IPE methodology and assumptions have on the results?

APPROACH

To accomplish these objective, the IPEs were examined by NRC Research, with the assistance of Brookhaven National Laboratory (BNL) and Sandia National Laboratory (SNL), to determine what the collective IPE results imply about the safety of U.S. nuclear power plants. SNL concentrated on obtaining insights based on the level 1 results reported in the IPEs, while BNL focussed on the level 2 results reported. Variations and commonalities among plant results were studied to determine which factors were most influential on the results. In addition, the improvements that have been made at the plants, and the impact of these improvements were examined.

The approach used by BNL for 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, (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.

The examination of the level 2 results documented in the IPEs indicated that there was significant variability in performance results among containments of similar type for every containment group. This variability could be traced back to differences in individual features among containments of the same type, but also to differences in assumptions made in the IPE analyses. These differences in assumptions included what phenomena were considered, how containment loads were developed, how containment capability was assessed, and the operator actions credited.

The rest of this paper illustrates the perspectives and insights gained from the examination of the level 2 analyses in the IPE submittals by summarizing the perspectives obtained for one group of containments, i.e. the BWR Mark I type.

BWR Mark I Perspectives

Twenty-four 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 coolant system during an accident.

Those accidents that cause structural failure of the drywell shortly after the core debris melts through the reactor vessel were found to be dominant contributors to risk. The importance of individual containment failure mechanisms depended on plant-specific features and in some cases on modeling assumptions; however, the following mechanisms were found important for many Mark I containments.

- Drywell shell melt-through caused by direct contact with the core debris (i.e., liner melt-through)

- Drywell failure caused by rapid pressure (and temperature) pulses at the time of reactor vessel melt-through

In general terms, these failure mechanisms are important to risk because of the relatively short time available for radioactivity decay, natural deposition processes, and for accident response actions. In addition, drywell failure means fission products released from the damaged core bypass the suppression pool (significant retention can occur if aerosol fission products pass through a suppression pool). The relatively short time to fission product release and the magnitude of the release means these failure mechanisms are important to all risk measures (i.e., acute and latent health effects including land contamination). These failure mechanisms can also occur for any accident class that involves release of a significant amount of core debris from the reactor vessel. A few plants identified other failure mechanisms as being important. Drywell failure caused by gradual pressure (and temperature) buildup due to gases and steam released during core/concrete interactions is important in some IPEs. In other IPEs venting was found to be an important contributor. However, accidents that bypass containment (such as interfacing systems LOCA) or involve containment isolation failure were not important contributors to the CDF in any of the IPEs for Mark I plants. These accidents are also not important to risk (even though they can involve large fission product release) because their frequencies of occurrence are so much lower than the frequencies of early structural failure caused by other accidents that dominate the CDF. Each failure mechanism is discussed in more detail below.

Liner melt-through was found to be the most important contributor to early containment failure for Mark I containments. This failure mechanism occurs frequently in Mark I containments, 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, whether liner melt-through is a significant containment failure mechanism. The most important plant features and modeling characteristics are discussed below.

- Drywell floor flooding - The presence of a water pool on the drywell floor was found to mitigate liner melt-through in all of the submittals. The benefit of water on the drywell floor prior to vessel failure as a mitigating mechanism for liner melt-through is significant and should be highlighted in future accident management plans of utilities with Mark I containments.

- Containment design configuration - The design of the drywell sump and drywell floor can prevent liner melt-through in some Mark I containments. For example, containment sumps in the Monticello plant are large enough to contain the molten core material and thus prevent it from reaching the containment boundary. In the Oyster Creek drywell, a concrete curb prevents or limits the core debris from reaching the containment shell. Also, the Brunswick containment is unique among Mark I designs because it is of concrete rather than steel construction. Thus, even if the molten core debris reaches the Brunswick containment, it would be difficult to thermally degrade such a thick concrete structure.

- Core debris characteristics - The amount of core debris released to the drywell and the fluidity of the core debris assumed in the IPEs determined whether or not liner melt-through occurred. Liner melt-through was found to be an important risk contributor if a large amount of core debris at high temperature was assumed to be released to the drywell. Under these circumstances, the core debris can flow across the floor and melt-through the shell. Liner melt-through was not important to risk if smaller quantities of core debris at lower temperatures (less able to flow across the floor) were assumed to be released into the drywell. As different modeling assumptions can produce such significantly different results (i.e., containment failure vs. no failure) any actions taken by the utilities to mitigate this failure mechanism should reflect this uncertainty. Therefore, as water can effectively mitigate liner melt-through, it is prudent to eliminate the uncertainty regarding containment failure caused by this failure mechanism by ensuring a flooded drywell floor.

A number of utilities were proactive and identified minor hardware modifications and changes in procedures to ensure a flooded drywell floor prior to reactor vessel melt-through. The availability of alternate water sources to the drywell spray header, such as water from a diesel driven fire pump during a station black-out, was shown to significantly reduce the likelihood of early failure in the Browns Ferry IPE. Another example is the Monticello plant where connections are available which enable the operators to use RHR service water for containment spray. The Nine Mile Point 1 submittal mentions the potential benefit of supplying the drywell sprays from external sources such as the containment spray raw water pumps. Peach Bottom has the capability of supplying the sprays with water from an external pond or the Emergency Cooling Tower. Several IPEs, such as Duane Arnold and Monticello, also discussed 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 is 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. High pressures and temperatures occur in containment when the RCS depressurizes as the core debris melts through the reactor vessel. Hydrogen (from clad oxidation) and steam are the driving force for pressurization. If the pressure pulse exceeds the ultimate pressure capability of the containment, then failure will occur at the weakest location either in the wetwell or the drywell.

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 determined whether this failure mechanism is a significant contributor to early containment failure for individual Mark I containments. The most important accident characteristics, design features and modeling assumptions are discussed below.

- RCS pressure at time of vessel melt-through - Containment failure via this mechanism is prevented if the RCS is depressurized before the core debris melts through the reactor vessel. The importance of this failure mechanism to risk therefore depends on the importance of accident classes in which the RCS is at high pressure (such as transient events with failure of the ADS). Enhancing the depressurization capability of the RCS was explored by a number of utilities but adverse effects were identified which need to be carefully considered.

- Containment failure location - The containment failure location can significantly influence the importance of this failure mechanism to risk. If failure occurs in the wetwell, then significant retention of the aerosol fission products occurs in the suppression pool making this failure mechanism less risk significant. Conversely, if failure occurs in the drywell, then the fission products are released without the benefit of pool scrubbing and the risk is much higher.

- RCS depressurization characteristics - The rate of RCS depressurization, steam generation, and characteristics of core debris dispersal determine the risk significance of the failure mechanism. If rapid depressurization is assumed (caused by a large opening in the reactor vessel) then high pressure pulses can occur that have a high likelihood of containment failure. In addition, if a large amount of high temperature core debris is assumed to be released and dispersed into the containment atmosphere then it can directly heat it and containment failure is very likely to occur. Containment failure does not occur if lower depressurization rates combined with less core debris dispersal are assumed. Again, different modeling assumptions give very different results and these uncertainties need to be factored into any strategy designed to prevent or mitigate this failure mechanism.

Ways of preventing or mitigating the pressure (and temperature) loads at vessel melt-through are enhanced RCS depressurization capability, containment venting and spray operation. Of these possible actions, RCS depressurization is potentially the most effective. Containment vents of sufficient capacity to mitigate pressure loads at the time of vessel melt-through (with the RCS at high pressure) do not exist in most Mark I containments and would not be practical to install, and spray operation cannot effectively mitigate all pressure loads associated with RCS depressurization during severe accidents.

A number of utilities explored controlled depressurization of the RCS prior to melt-through of the reactor vessel as a mitigation strategy for rapid over pressure failure of Mark I containments. Enhancement of the emergency depressurization capability was also an issue raised as part of the NRC's containment performance improvements (CPI) program. Although some utilities recognized the benefit of this strategy a number of potential adverse effects were also noted. For example, if low pressure injection systems are not available, then depressurization causes loss of coolant inventory which can significantly reduce the time to fuel damage and vessel melt-through. This in turn reduces the time available for other recovery actions. Given the uncertainty associated with pressure loads and the potential adverse effects some utilities recommended further study prior to implementing this strategy.

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 noncondensible gases and steam released from the drywell floor during core/concrete interactions can fail Mark I containments several hours after vessel melt-through. This failure mechanism occurs because of the relatively small volume of Mark I containments. Failure can occur either in the wetwell or in the drywell. Generally, this failure mechanism is less risk significant than the two early failure mechanisms discussed above because of the longer time available for radioactive decay, natural deposition processes and for accident response. However, even for late failures, if the failure location is in the drywell then, significant fission product release can still occur making this failure mechanism important to longer term risk measures (i.e., latent health effects and land contamination).

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. The most important accident characteristics, design features and modeling assumptions are discussed below.

- Drywell floor flooded - A flooded drywell floor helps the drywell spray and containment heat removal (CHR) systems to control pressurization and prevent structural failure of the containment. Water can cool the core debris and limit concrete erosion (and hence limit gas generation) so that steam is the main driving force for containment pressurization. The drywell spray and CHR systems are designed to condense steam and remove heat from containment and therefore can control the containment pressure under these circumstances.

- Drywell floor not flooded - If the drywell floor is not flooded (and liner melt-through does not occur) venting may be needed to prevent over pressure failure of the containment. Without water, the hot core debris can cause significant concrete erosion (and hence significant gas release). The heat from this core/concrete interaction can raise the temperature of the drywell to a range where the structural capacity of the steel containment shell is significantly reduced. The quantity of gases released from this interaction also depends on the type of concrete used. For example, limestone concrete releases significantly more gases than basalt concrete. The drywell spray and CHR systems cannot control the pressure in containment if the driving force for pressurization is non condensible gases. Under these circumstance, the only way to control pressure is to relieve gases via venting (preferably from the wetwell in order to benefit from pool scrubbing).

- Containment design configuration - The design of the drywell and pedestal region can limit contact between the water and core debris in some Mark I containments. For example, large sumps in the pedestal region produce deep pools of molten core debris, which are difficult to cool with water. Forming a coolable debris bed is particularly difficult if the water is added after the core debris is in the sumps. Therefore, in some IPEs, core/concrete interactions continued even after water was added to the drywell.

- Core debris characteristics - In the absence of water, the amount of core debris released to the drywell and its temperature determined the extent of core/concrete interactions. If a large amount of core debris at high temperature was assumed released from the reactor vessel then extensive concrete erosion was predicted in the IPEs. Under these circumstances, even if water was added to the core debris , core/concrete interactions were predicted to continue for some Mark I designs. Conversely, if smaller quantities of core debris at lower temperatures were assumed, then much less concrete erosion occurred even without water. Clearly, different modeling assumptions give different results which were considered by utilities when developing strategies to mitigate these failure mechanisms.

Most utilities used a combination of strategies to mitigate gradual pressure build-up caused by core/concrete interactions. The drywell floor flooding strategies designed to prevent liner melt-through if successful will also limit long-term core/concrete interactions and hence limit noncondensible gas generation. If these early flooding strategies were not successful, then most utilities explored other ways of flooding the drywell floor. For instance, the Monticello IPE submittal noted that debris cooling with an alternate injection source, such as fire water, limits the temperature rise in containment and extends the time to containment failure by over-pressurization. In all the IPEs containment sprays were found to be of great benefit for preventing or mitigating late containment failure.

In addition to the advantages mentioned earlier, the cooling provided by the containment sprays will retard the revaporization of fission products deposited on containment surfaces. Sprays can also scrub fission products existing in the containment atmosphere and provide a water source for covering ongoing core/concrete interactions. High temperature effects were also addressed in other ways in some IPEs. Nine Mile Point 1 considered raising the preload on the drywell head bolts as a way of increasing the probability of maintaining containment integrity at elevated temperatures. Finally all utilities have the capability to prevent late structural failure by venting.

Containment venting is an important way of preventing and mitigating core damage in Mark I containments. Venting was used extensively in the IPEs to reduce releases and thus risk, and it was also an important element of the CPI program. Containment venting was used to prevent core damage in accidents involving loss of containment heat removal. It was also used to prevent late structural failure for those accidents in which the core melts through the reactor vessel. However, a few utilities stated in their IPEs that their analyses indicated that the installation of a hardened vent did not significantly impact risk and therefore was only of marginal benefit. In one case the utility stated that they would not install a hardened vent.

In response to the recommendations in Generic Letter 89-16, most utilities with Mark I containments committed to install a hardened wetwell vent system (in some cases a hardened vent was already in place). A hardened vent leading from the wetwell to outside the containment building provides an independent means for containment pressure relief and heat removal while maintaining a habitable environment in the reactor building. The utilities used these venting systems to prevent core damage for some accidents involving loss of containment heat removal. Under these circumstances venting is "clean" because it occurs prior to core damage and involves minimal release of radioactivity.

Venting, after core damage has occurred, as a way of preventing structure failure of the containment was considered to be a last resort by most utilities because it can involve significant fission product release. The advantage of venting from the wetwell (benefit of pool scrubbing) was emphasized in most IPE strategies. The pressure at which venting should be started was also examined in detail by several utilities. The impact of high temperatures on the structural capability of the drywell was also noted. For example, the NMP Unit 1 IPE reported that at 400°F the containment could fail at pressures below the current venting pressure in the EOPs. Further analysis was recommended that could refine the vent actuation pressure.

Containment venting is important to risk in some Mark I containments. If venting occurs shortly after core meltdown and the flow path is directly from the drywell or from the RCS to the environment, then the suppression pool will be bypassed. Under these circumstances, venting would cause a significant release of fission products to the environment. In this context a number of utilities expressed concern about the current BWR Owners Group guidelines for containment flooding (filling the containment solid with water to a level equal with the top of fuel in the RPV) and the venting necessary to carry it out. Since drywell (i.e. unscrubbed) venting is needed to relieve the pressure buildup resulting from the compression of the gas space during containment flooding, there is the potential of an early release of significant magnitude associated with the flooding strategy. A number of utilities speculated that other actions, or even no action, was preferable to carrying out the containment flooding strategy.

Accidents that bypass containment are not important to risk for Mark I containments. If the pressure boundary between the high pressure RCS and a low pressure auxiliary system fails (called an interfacing systems LOCA) then a LOCA outside containment can occur. If water cannot be supplied to the reactor, core damage will occur and a direct path can exists to the environment. Therefore, these accidents can lead to a large early release of fission products. However, interfacing systems LOCA are not risk significant 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. The IPEs reported interfacing systems LOCA frequencies that are about an order of magnitude lower in BWR plants than in PWRs. The lower BWR frequency are in part due to the lower RCS pressures in BWR plants compared with PWRs.

Although interfacing systems LOCA are not important to risk one submittal (the Nine Mile Point Unit 1 IPE) did identify a unique way of bypassing containment. In that IPE, failure of the emergency condenser tubes due to high temperature creep rupture was identified as leading to containment bypass. In a degraded core accident failure between the primary and secondary side of the emergency (or isolation) condenser provides a pathway for release similar to a steam generator tube rupture in PWRs. This failure mode was found to have a relatively low frequency (compared with the frequency of early structural failure) at Nine Mile Point, Unit 1, and was therefore not important to risk. Isolation condensers are found in one other BWR 2 plant and two early BWR 3 plants and presumably this bypass accident is also applicable to these plants. It is therefore necessary to determine that this failure mechanism is also a low risk (low frequency compared with the frequency of early structural failure) event in these other plants.

Accidents that involve failure to isolate containment are not important to risk for Mark I containments. Isolation failures can be preexisting or occur at the time of the initiating event. If the isolation failure is large (i.e., exceeds X volume percent per day) and if core melt occurs, then fission product release can also be large. In addition, because the containment is open at the time of core damage, the offsite site consequences can be significant. These events are not risk significant in BWR Mark I plants because of their relatively low frequencies. Preexisting isolation failures in Mark I plants can be precluded because the containment atmosphere is inerted with nitrogen. Therefore, any loss of containment atmosphere due to preexisting leaks can be easily detected. In addition, failure to isolate containment on demand was found to be a relatively low frequency event compared with the frequencies of other accidents that can cause early structural failure of the containment.

ATWS sequences were risk significant 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 are risk significant 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 indicated that, by proper RPV level control and by opening the maximum number of vent paths, many ATWS scenarios could be controlled. The significance of ATWS events in the different IPEs depended 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 were found to be relatively unimportant to risk in all the IPEs. The ability to vent the containment was a major factor in reducing the importance of this class of accident. Also, the interval between loss of containment heat removal and containment failure is relatively long in these sequences, allowing time for emergency measures on and off site.

Figure 1 shows the conditional probability for the various Mark I containment failure modes.

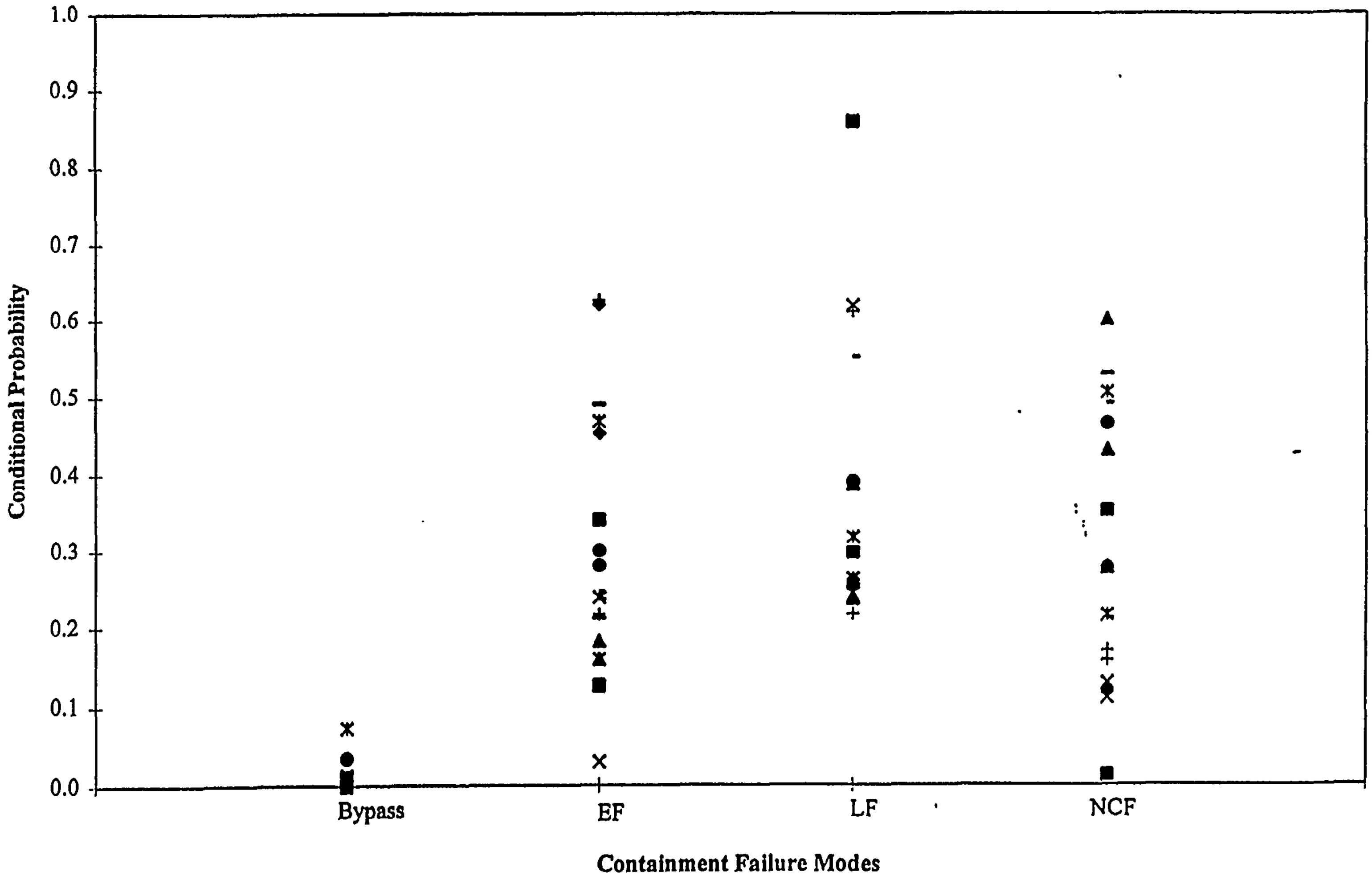

Figure 1 Conditional Containment Failure Probabilities for BWR Mark I Containment