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Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance

Part 1

Final Summary Report

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ABSTRACT

This report provides perspectives gained by reviewing 75 Individual Plant Examination (IPE) submittals pertaining to 108 nuclear power plant units. IPEs are probabilistic analyses that estimate the core damage frequency (CDF) and containment performance for accidents initiated by internal events (including internal flooding, but excluding internal fire).

The U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research, reviewed the IPE submittals with the objective of gaining perspectives in three major areas: (1) improvements made to individual plants as a result of their IPEs and the collective results of the IPE program, (2) plant-specific design and operational features and modeling assumptions that significantly affect the estimates of CDF and containment performance, and (3) strengths and weaknesses of the models and methods used in the IPEs. These perspectives are gained by assessing the core damage and containment performance results, including overall CDF, accident sequences, dominant contributions to component failure and human error, and containment failure modes. In particular, these results are assessed in relation to the design and operational characteristics of the various reactor and containment types, and by comparing the IPEs to probabilistic risk assessment characteristics. Methods, data, boundary conditions, and assumptions used in the IPEs are considered in understanding the differences and similarities observed among the various types of plants.

This report is divided into three volumes containing six parts. Part 1 is a summary report of the key perspectives gained in each of the areas identified above, with a discussion of the NRC's overall conclusions and observations (Chapter 8). Part 2 discusses key perspectives regarding the impact of the IPE Program on reactor safety (summarized in Part 1, Chapter 2). Part 3 discusses perspectives gained from the IPE results regarding CDF, containment performance, and human actions (summarized in Part 1, Chapters 3, 4, and 5, respectively). Part 4 discusses perspectives regarding the IPE models and methods (summarized in Part 1, Chapter 6). Part 5 discusses additional IPE perspectives (summarized in Part 1, Chapter 7). Part 6 contains Appendices A, B and C which provide the references of the information from the IPEs, updated PRA results, and public comments on draft NUREG-1560 (including staff responses), respectively.

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EXECUTIVE SUMMARY

Introduction

On August 8, 1985, the U.S. Nuclear Regulatory Commission (NRC) issued its "Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants" (Federal Register, 50FR32138). That policy statement introduced the Commission's plan to address severe accident issues for existing commercial nuclear power plants.

The Commission formulated an approach for systematic safety examination of plants to study particular accident vulnerabilities and desirable, cost-effective changes to ensure that the plants do not pose any undue risk to public health and safety. To implement this approach, the NRC issued Generic Letter (GL) 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 report the results to the Commission."* The purpose and scope of the IPE effort includes examining internal events occurring at full power, including those initiated by internal flooding. In response, the staff received 75 IPE covering regarding 108 nuclear power plant units. The staff then examined the IPE submittals to determine what the collective IPE results imply about the safety of U.S. nuclear power plants and how the IPE program has affected reactor safety. The following sections summarize the results of the IPE Insights Program examination.

Impact of the IPE Program on Reactor Safety

The primary goal of the IPE Program was for licensees to *"identify plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements."* However, GL 88-20 did not specifically define what constitutes a vulnerability; hence, the IPEs exhibit considerable diversity in the criteria used to define a vulnerability. The wording used in some submittals is such that it is not always clear whether a licensee is identifying a finding as a "vulnerability" or as some other issue worthy of attention. Therefore, a problem considered to be a vulnerability at one plant may not have been specifically identified as a vulnerability at another plant. In fact, less than half of the licensees actually identified "vulnerabilities" in their IPE submittals; however, nearly all of the licensees identified other areas warranting investigation for potential improvements. Thus, the IPE program has served as a catalyst for further improving the overall safety of nuclear power plants.

Only four licensees with boiling water reactor (BWR) plants and 15 licensees with pressurized water reactor (PWR) plants explicitly stated that their plants had vulnerabilities. The following vulnerabilities were identified by the four BWR licensees with no common vulnerabilities cited:

- failure of water supplies to isolation condensers
- failure to maintain high-pressure coolant injection and reactor core isolation cooling when residual heat removal has failed
- failure to control low-pressure injection during an anticipated transient without scram (ATWS)
- drywell steel shell melt-through as a Mark I containment issue

Some of the PWR licensees identified common vulnerabilities. Some of the vulnerabilities in the submittals include:

- loss of reactor coolant pump (RCP) seals leading to a loss of coolant accident (LOCA)
- design and maintenance problems that reduce turbine-driven auxiliary feedwater pump reliability

- internal flooding caused by component failures
- failure of the operator to switchover from the coolant injection phase to the recirculation phase
- loss of critical switchgear ventilation equipment leading to loss of emergency buses
- need to enhance operator guidance for depressurization during steam generator tube ruptures
- inadequate surveillance of specific valves leading to interfacing system LOCAs (ISLOCA)
- loss of specific electrical buses
- compressed air system failures
- inability to cross-tie buses during loss of power conditions

In addition, almost all of the licensees identified plant improvements to address perceived weaknesses in plant design or operation. (Over 500 proposed improvements were identified by the plants.) Most of these plant improvements are classified as procedural/ operational changes (approximately 45 percent), design/hardware changes (approximately 40 percent), or both. Few of the improvements involve maintenance-related changes. Typically, the procedural or design changes indicate revised training in order to properly implement the actual change. Approximately 45 percent of the plant improvements are identified by the licensees as implemented, with approximately 25 percent ***implemented and credited*** in the IPEs. Other improvements are either planned or under evaluation. Some improvements are associated with other requirements (primarily the station blackout rule) and utility activities. However, although these improvements were not necessarily identified as a result of the IPE, in some cases, the licensee is using the IPE to prioritize the improvements and to support decisions regarding their implementation. The specific improvements vary from plant to plant. However, numerous improvements that had significant impact on plant safety include changes to AC and DC power, coolant injection systems, decay heat removal systems, heating, ventilating and air conditioning, and PWR RCP seals.

IPE Results Perspectives (Core Damage Frequency)

In many ways, the IPE results are consistent with the results of previous NRC and industry risk studies. The IPE results indicate that the plant core damage frequency (CDF) is often determined by many different sequences (in combination), rather than being dominated by a single sequence or failure mechanism. The largest contributors to plant CDF and the dominant failures contributing to those sequences vary considerably among the plants (e.g., some are dominated by LOCAs, while others are dominated by station blackout). However, for most plants, support systems are important to the results because support system failures can result in failures of multiple front-line systems. Further, the support system designs and dependency of front-line systems on support systems vary considerably among the plants. That variation explains much of the variability observed in the IPE results.

Consistent with previous risk studies, the CDFs reported in the IPE submittals are lower, on average, for BWR plants than for PWR plants, as shown in Figure E.1. Although both BWR and PWR results are strongly affected by the support system considerations discussed above, a few key differences between the two types of plants contribute to this tendency for lower BWR CDFs and cause a difference in the relative contributions of the accident sequences to plant CDF. The most significant difference is that BWRs have more injection systems than PWRs and can depressurize more easily to use low-pressure injection (LPI) systems. This gives BWRs a lower average contribution from LOCAs. However, the results for individual plants can vary from this general trend. As shown in Figure E.1, the CDFs for many BWR plants are actually higher than the CDFs for many PWR plants. The variation in the CDFs is primarily driven by a combination of the following factors:

- plant design differences (primarily in support systems such as cooling water, electrical power, ventilation, and air systems)
- variability in modeling assumptions (including whether the models accounted for alternative accident mitigating systems)
- differences in data values (including human error probabilities) used in quantifying the models.

Table E.1 summarizes the key observations regarding the importance and variability of accident classes commonly modeled and discussed in the IPEs.

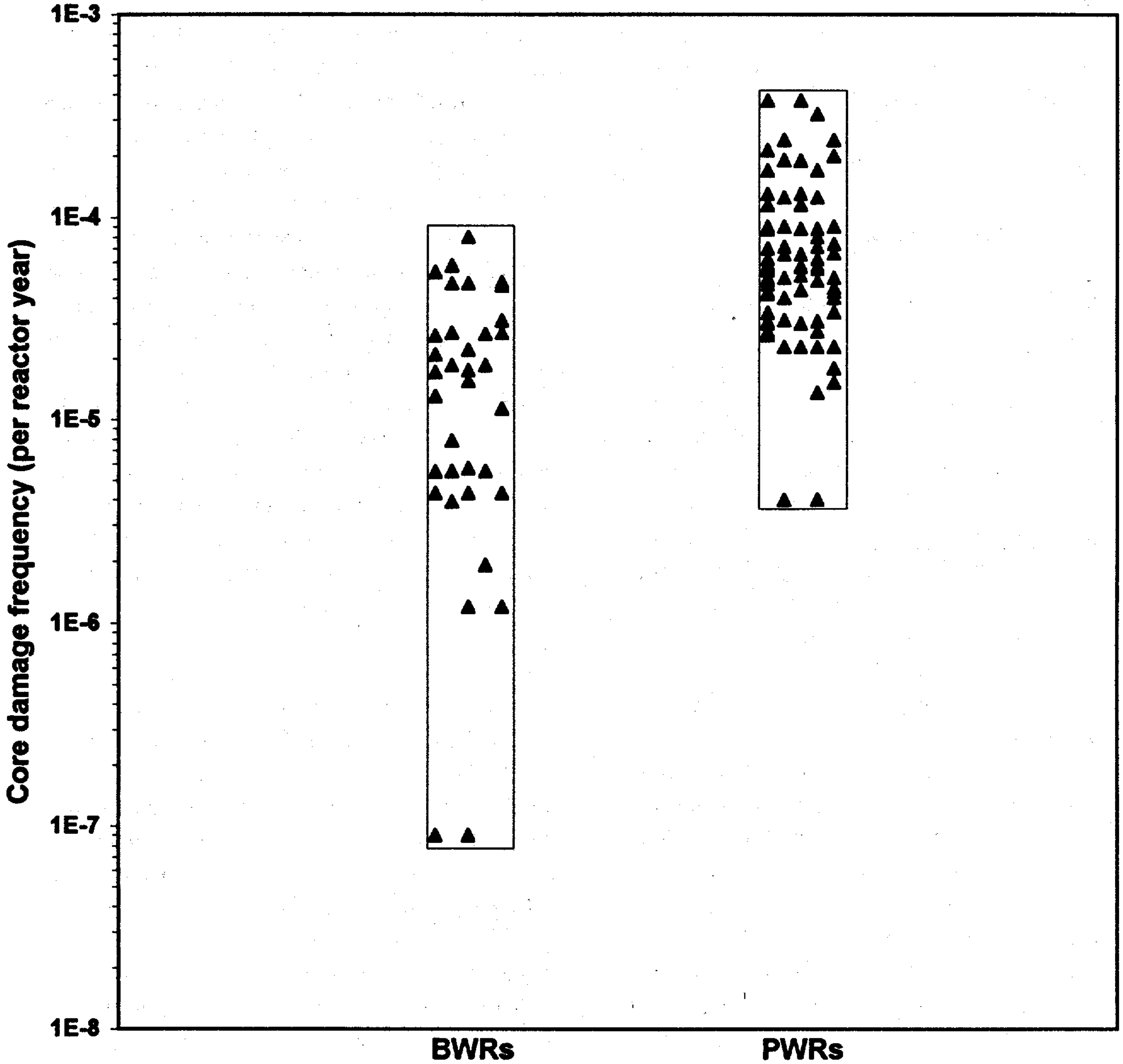

Figure E.1 Summary of BWR and PWR CDFs as reported in the IPEs.

Table E.1 Overview of key CDF observations.

Accident class	Key observations
Transients (other than station blackouts and ATWS)	Important contributor for most plants because of reliance on support systems; failure of such systems can defeat redundancy in front-line systems Both plant-specific design differences and IPE modeling assumptions contribute to variability in results: • use of alternative systems for injection at BWRs • variability in the probability that an operator will fail to depressurize the vessel for LPI in BWRs • availability of an isolation condenser in older BWRs for sequences with loss of decay heat removal (DHR) • susceptibility to harsh environment affecting the availability of coolant injection capability following loss of DHR • capability to use feed-and-bleed cooling for PWRs • susceptibility to RCP seal LOCAs for PWRs • ability to depressurize the reactor coolant system in PWRs affecting the ability to use LPI • ability to cross-tie systems to provide additional redundancy
Station blackouts	Significant contributor for most plants, with variability driven by: • number of redundant and diverse emergency AC power sources • availability of alternative offsite power sources • length of battery life • availability of firewater as a diverse injection system for BWRs • susceptibility to RCP seal LOCAs for PWRs
ATWS	Normally a low contributor to plant CDF because of reliable scram function and successful operator responses BWR variability mostly driven by modeling of human errors and availability of alternative boron injection system PWR variability mostly driven by plant operating characteristics, IPE modeling assumptions, and assessment of the fraction of time the plant has an unfavorable moderator temperature coefficient
Internal floods	Small contributor for most plants because of the separation of systems and compartmentalization in the reactor building, but significant for some because of plant-specific designs Largest contributors involve service water breaks
LOCAs (other than ISLOCAs and steam generator tube ruptures (SGTRs))	Significant contributors for many PWRs with manual switchover to emergency core cooling recirculation mode BWRs generally have lower LOCA CDFs than PWRs for the following reasons: • BWRs have more injection systems • BWRs can more readily depressurize to use low-pressure systems
ISLOCAs	Small contributor to plant CDF for BWRs and PWRs because of the low frequency of initiator Higher relative contribution to early release frequency for PWRs than BWRs because of low early failure frequency from other causes for PWRs
SGTR	Normally a small contributor to CDF for PWRs because of opportunities for the operator to isolate a break and terminate an accident, but important contributor to early release frequency

IPE Results Perspectives (Containment Performance)

For the most part, when the accident progression analyses in the IPEs are viewed globally, they are consistent with typical containment performance analyses. Failure mechanisms identified in the past as being important are also shown to be important in the IPEs. In general, the IPEs confirmed that the large volume PWR containments are more robust than the smaller BWR pressure suppression containments in meeting the challenges of severe accidents; but the IPEs also showed containment bypass was more likely with PWR systems.

Because of the risk importance of early releases, the containment performance analysis descriptions found in the IPE submittals emphasized the phenomena, mechanisms, and accident scenarios that can lead to such releases. These involve early structural failure of the containment, containment bypass, containment isolation failures and, for some BWR plants, deliberate venting of the containment.

As a group, the large dry PWR containments analyzed in the IPEs have significantly smaller conditional probabilities of early structural failure (given core melt) than the BWR pressure suppression containments analyzed. On the other hand, containment bypass and isolation failures are generally more significant for the PWR containments. As seen in Figure E.2, however, these general trends are often not true for individual IPEs because of the considerable range in the results. For instance, conditional probabilities for both early and late containment failure for a number of large dry PWR containments were higher than those reported for some of the BWR pressure suppression containments. Table E.2 summarizes key observations regarding containment performance.

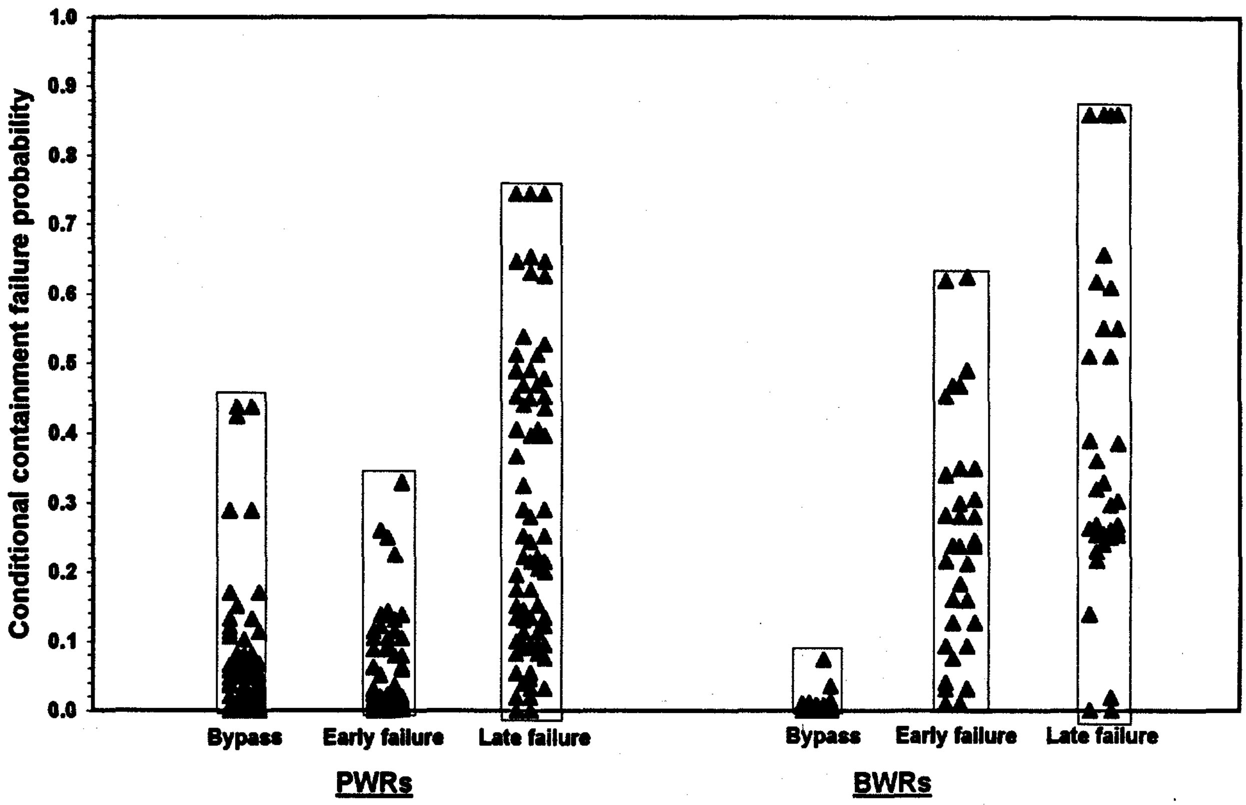

Figure E.2 Summary of conditional containment failure probabilities for BWRs and PWRs as reported in the IPEs.

Table E.2 Key observations regarding containment performance.

Failure mode	Key observations
Early failure	On average, the large volume containments of PWRs are less likely to have early structural failures than the smaller BWR pressure suppression containments Overpressure failures (primarily from ATWS), fuel coolant interaction, and direct impingement of core debris on the containment boundary are important contributors to early failure for BWR containments The higher early structural failures of BWR Mark I containments versus the later BWR containments are driven to a large extent by drywell shell melt-through* In a few BWR analyses, early venting contributes to early releases Phenomena associated with high-pressure melt ejection are the leading causes of early failure for PWR containments* Isolation failures are significant in a number of large, dry and subatmospheric containments The low early failure probabilities for ice condensers relative to the other PWRs appear to be driven by analysis assumptions rather than plant features For both BWR and PWR plants, specific design features lead to a number of unique and significant containment failure modes
Bypass	Probability of bypass is generally higher in PWRs, in part, because of the contribution from SGTRs Bypass, especially SGTR, is an important contributor to early release for PWR containment types Bypass is generally not important for BWRs
Late failure	Overpressurization when containment heat removal is lost is the primary cause of late failure in most PWR and some BWR containments High pressure and temperature loads caused by core-concrete interactions are important for late failure in BWR containments Containment venting is important for avoiding late uncontrolled failure in some Mark I containments The larger volumes of the Mark III containments (relative to Mark I and Mark II containments) are partly responsible for their lower late failure probabilities in comparison to the other BWR containments The likelihood of late failure often depends on the mission times assumed in the analysis

* There has been a significant change in the state-of-knowledge regarding some severe accident phenomena in the time since the IPE analyses were carried out.

The results for BWRs, grouped by containment type, follow expected trends and indicate that, in general, Mark I containments are 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 BWR containment designs and there is significant overlap of the results, given core damage. A large variability also exists in the contributions of the different failure modes for each BWR containment group. However, a significant probability of early or late structural failure, given core damage, was found for plants in all three BWR containment groups. The containment performance results for PWRs indicate that most of the containments have relatively low conditional probabilities of early failure, although a large variability exists in the contributions of the different failure modes for both large dry and ice condenser containments.

The results presented in the IPE submittals are also consistent with previous studies regarding radionuclide release. A significant early release is of particular concern because of the potential for severe consequences as a result of the short time allowed for radioactivity decay and natural deposition, as well as for accident response actions (such as evacuation of the population in the vicinity of the plant). What is considered to be a significant release varies among the licensees. For many, significant release includes instances involving a release fraction of volatile radionuclides equal to or greater than ten percent of core inventory. Using this definition, the reported conditional probability for significant early release varies from less than 0.01 to 0.5 for the BWR IPEs and from less than 0.01 to 0.3 for the PWR IPEs. Frequencies of significant early release are shown in Figure E.3 and vary from negligible to 2E-5 per reactor year (ry) for BWRs and from 1E-8/ry to 2E-5/ry for PWRs. In the BWR IPEs, significant early releases are almost exclusively caused by early containment failure, while containment bypass (especially SGTR), plays an important role in the reported PWR releases.

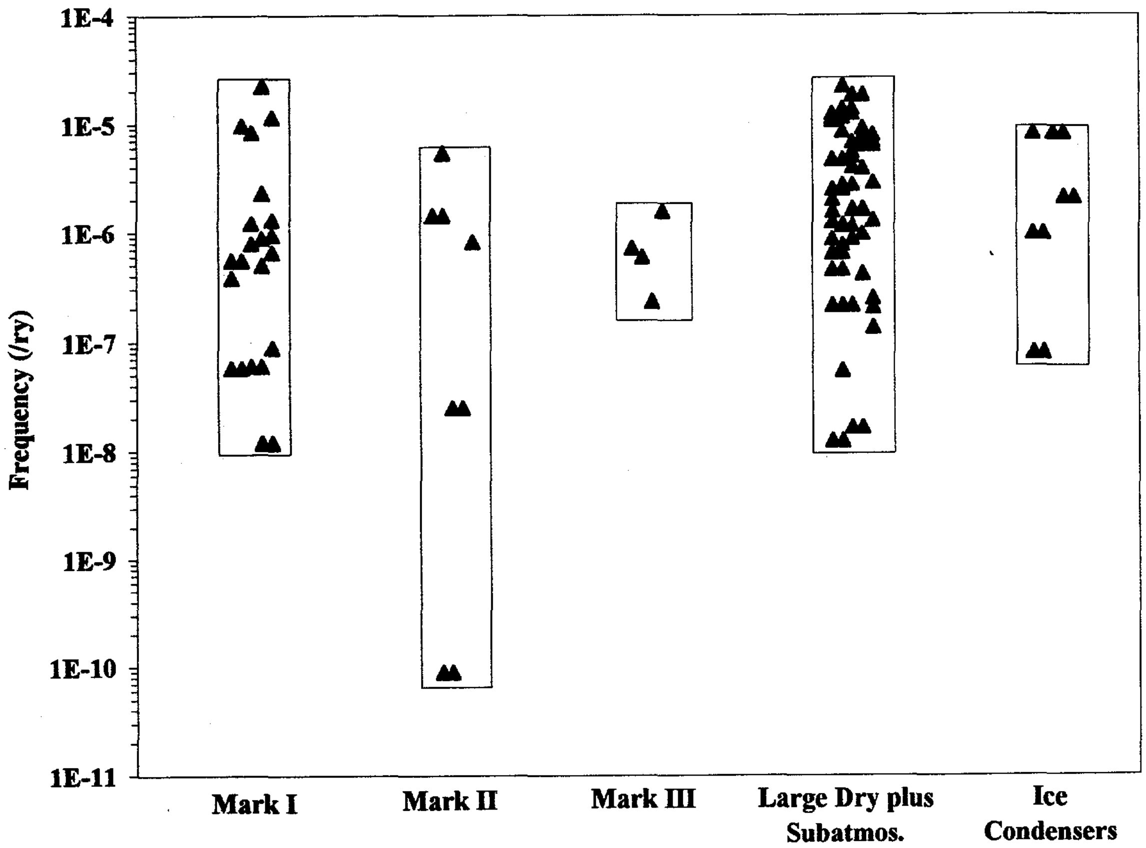

Figure E.3 Frequencies of significant early release (by containment type) as reported in the IPEs.

IPE Results Perspectives (Human Performance)

Only a few specific human actions are consistently important for either BWRs or PWRs as reported in the IPEs. For BWRs, the actions include manual depressurization of the vessel, initiation of standby liquid control during an ATWS, containment venting, and alignment of containment or suppression pool cooling. Manual depressurization of the vessel is more important than expected, because most plant operators are directed by the emergency operating procedures to inhibit the automatic depressurization system (ADS) and, when ADS is inhibited, the operator must manually depressurize the vessel.

Only three human actions are important in more than 50 percent of the IPE submittals for PWRs. These include the switchover to recirculation during LOCAs, initiation of feed-and-bleed, and the actions associated with depressurization and cooldown. Plant-specific features, such as the size of the refueling water storage tank and the degree of automation of the switchover to recirculation, are key in determining the importance of these actions.

While the IPE results indicate that human error can be a significant contributor to CDF, numerous factors can influence the quantification of human error probabilities (HEPs) and introduce significant variability in the resulting HEPs, even for essentially identical actions. General categories of such factors include plant characteristics, modeling details, sequence-specific attributes (e.g., patterns of successes and failures in a given sequence), dependencies, performance shaping factors modeled, application of the human reliability analysis (HRA) method (correctness and thoroughness), and the biases of both the analysts performing the HRA and the plant personnel from whom selected information and judgments are obtained. Although most of these factors introduce appropriate variability in the results (i.e., the derived HEPs reflect "real" differences such as time availability, dependencies and plant-specific factors), several have the potential to cause invalid variability. In order to examine the extent to which variability in the results from the BWRs is caused by valid rather than inappropriate factors, the HEPs from several of the more important human actions appearing in the IPEs were examined across plants.

The results from the examination indicated that much of the observed variability in the HEPs and in the results of the HRAs across IPEs was due to appropriate factors. Thus, the staff concluded that in general the licensees attempted to consider relevant factors in obtaining the HEPs for operator actions and that the results of the HRAs performed by the different licensees were generally consistent, and therefore, useful. On the other hand, it was also concluded that not all of the variability in the examined HEPs was easily explained. That is, after "acceptable" reasons for variation were considered, there still appeared to be some degree of "random" or unexplained variability. Potential reasons for this variability include the lack of precision in existing HRA methods and the fact that some licensees did not perform as thorough HRAs as possible.

While the degree of consistency in HEPs obtained for similar human actions in similar contexts suggests that in general the HRA results from the IPEs were useful in terms of meeting the intent of Generic Letter 88-20, it should be further noted that even when reasonable consistency exists, it is not necessary the case that all the HEPs calculated by a particular plant were realistic and valid for that plant. As noted in Chapters 5 and 13, reasonable consistency can be obtained in HRA without necessarily producing valid HEPs. An HEP is only valid to the extent that a correct and thorough application of HRA principles has occurred.

IPE Models and Methods Perspectives

As a result of the IPE program, licensees elected to perform probabilistic risk assessments (PRAs) for their IPEs. Perspectives on the PRAs used in the IPEs provides information on where the models and methods are sophisticated versus where potential development may be needed. Developing these perspectives requires consideration of several issues:

- What are the characteristics of a PRA?
- How do the IPEs/PRAs compare to these characteristics?
- What can be said about the IPE analyses given the scope of the IPEs (intent of Generic Letter 88-20) and the scope of the staff's IPE reviews?

First, the characteristics of each analytical task of a PRA were defined. The IPE models and methods as described in the individual submittals are compared against the characteristics a PRA. Some general conclusions are then drawn based on the limited reviews that have taken place.

The CDF analyses of the IPEs generally compare well to the characteristics defined for a PRA. The IPEs are generally robust with respect to the identification of dominant accident sequences. This is not to say that particular accident sequence frequencies have been verified, but rather that most of the important accident types have been captured. Although the CDF analyses in the IPEs are generally robust, weaknesses were identified in a number of submittals in the areas of plant-specific data, common cause failure data, and human reliability. An important concern for some IPEs is the HRA, with the most significant concern being the use of invalid HRA assumptions not producing consistently reasonable results.

The IPEs exhibit greater variation in the methods and scope of the accident progression (containment performance) analyses than that found in the CDF analyses. This is commensurate with the guidance of GL 88-20 and NUREG-1335 which allowed significant diversity in the ways licensees could conduct their containment performance analyses. In many of the IPEs, the containment performance analyses and the source term calculations were more simplified than the characteristics identified for a PRA.

Additional IPE Perspectives

The Safety Goal Policy Statement established two qualitative safety goals, which are supported by two quantitative health objectives (QHOs) regarding the risk to the population in the vicinity of a nuclear power plant. Specifically, the safety goals establish that the risk of both prompt fatalities and latent cancer fatalities that might result from reactor accidents should not exceed 0.1 percent of the corresponding risks resulting from all other types of accidents or causes. In responding to GL 88-20, licensees only considered internal events at full power and were not requested to calculate offsite health effects. Therefore, the IPE results cannot be directly compared against the above health objectives. However, it is possible to link the CDF and containment performance results to the safety goals by using surrogate indicators (such as the frequencies of early containment failure and bypass). On the basis of the frequencies of early containment failure and bypass reported in the IPEs, a fraction of the plants have the potential for early fatality risk levels that could approach the QHOs. This subset of plants was further examined using the frequencies of source terms (from the IPEs) with relatively high release factors (>0.03 I, Cs, Te) and adjusted for population. On the basis of this further screening, two BWRs and 12 PWRs remain with the potential for early fatality risk level that could approach the QHOs.

Many of the BWR and PWR plant improvements address station blackout (SBO) concerns and originated as a result of the SBO rule. These improvements had a significant impact in reducing the SBO CDF (an average reduction of approximately 2E-5/ry, as estimated from the CDFs reported by licensees in the IPEs). With the SBO rule implemented, the average SBO CDF is approximately 9E-6/ry, ranging from negligible to approximately 3E-5/ry. Although the majority of the plants that implemented the SBO rule have achieved the goal of limiting the average SBO contribution to core damage to about 1E-5/ry, a few plants are slightly above the goal.

In NUREG-1150, the NRC assessed risk for five nuclear power plants representing both PWR and BWR designs. While these five plants represent only a small sample of designs, it is possible to consider whether the NUREG-1150 results and perspectives are consistent with those found in the IPEs. The average CDFs estimated for both BWRs and PWRs in NUREG-1150 fall within the ranges of the CDFs estimated in the IPEs. The relative contributions of accident sequences in the IPE results is also consistent with the NUREG-1150 results. For PWRs, SBO, transients, and LOCAs are usually the more important contributors; for BWRs, LOCAs and ATWSs are generally less important than SBO and transients.

The conditional probabilities of early containment failure reported in NUREG-1150 (mean values) also fall within the range of the IPE results for each containment type. The IPE results indicate that conditional probabilities for early containment failure are generally higher for BWR containments than for PWR plants. On the basis of absolute frequency, early containment failures for BWRs are similar to those of PWRs because the higher conditional early containment failure probabilities for BWR containments are compensated by the lower values for BWR CDFs.

Overall Conclusions and Observations

In considering the perspectives discussed above, and the results reported in the IPE submittals, certain conclusions and observations can be drawn as summarized below:

- As a result of the IPE program, licensees have generally developed in-house capability with an increased understanding of PRA and severe accidents. Further, the IPE program has served as a catalyst for further improving the overall safety nuclear power plants, and therefore, the generic letter initiative has clearly been a success.

- Areas and issues have been identified where the staff plans to pursue some type of follow-up activity. Areas under consideration are plant improvements, containment performance improvement items either not implemented or not addressed in the IPE submittal, and plants with relatively high CDF or conditional containment failure probability (greater than 1E-4/ry and 0.1, respectively).

- Unresolved safety issue (USI) A-45 ("Shutdown Decay Heat Removal Requirements") and certain other USIs and generic safety issues (GSIs), primarily GSI-23 ("Reactor Coolant Pump Seal Failures"), GSI-105 ("Interfacing System LOCA in Light Water Reactors") and GSI-130 ("Essential Service Water System Failures at Multi-Unit Sites"), were proposed by licensees for resolution on a plant-specific basis. Other safety issues resulting from the IPEs were identified as candidates for further investigation.

- Areas where further research regarding both severe accident behavior and analytical techniques would be useful and should be considered were identified.

- Information from the IPEs/PRAs has the potential to support a diversity of activities such as plant inspection, accident management strategies, maintenance rule implementation, and risk-informed regulation.

- IPE results can be used to indicate areas in PRA where standardization would be useful.

ACKNOWLEDGMENTS

This report provides perspectives on reactor safety and plant performance gained from reviewing the Individual Plant Examination submittals that were performed in response to Generic Letter 88-20.

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ABBREVIATIONS

AC	Alternating Current
ADS	Automatic Depressurization System
ADV	Atmospheric Dump Valve
AFW(S)	Auxiliary Feedwater (System)
AMSAC	ATWS Mitigating System Actuation Circuitry
ANO	Arkansas Nuclear One
ASEP	Accident Sequence Evaluation Program
ATWS	Anticipated Transient Without Scram
B&W	Babcock and Wilcox
BWR	Boiling Water Reactor
BWROG	BWR Owners' Group
BWST	Borated Water Storage Tank
CCI	Core-Concrete Interaction
CCFP	Conditional Containment Failure Probability
CCW	Component Cooling Water
CDF	Core Damage Frequency
CE	Combustion Engineering
CET	Containment Event Tree
CHR	Containment Heat Removal
CPI	Containment Performance Improvement
CRD	Control Rod Drive
CS	Core Spray
Cs	Cesium
CST	Condensate Storage Tank
CVCS	Chemical and Volume Control Tank
CW	Circulating Water
DC	Direct Current
DCH	Direct Containment Heating
DHR	Decay Heat Removal
ECCR	Emergency Core Cooling Recirculation
ECCS	Emergency Core Cooling System
EFW	Emergency Feedwater
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure
EPG	Emergency Procedure Guideline
EPRI	Electric Power Research Institute
ESF	Engineering Safety Feature
ESFA(S)	Engineered Safety Feature Actuation (System)
ESW	Emergency Service Water
FCI	Fuel-Coolant Interaction
FLIM	Failure Likelihood Index Methodology
FSAR	Final Safety Analysis Report
GL	Generic Letter

Abbreviations

GSI	Generic Safety Issue
HCR	Human Cognitive Reliability
HEP	Human Error Probability
HHSI	High Head Safety Injection
HIS	Hydrogen Igniter System
HPCI	High-Pressure Coolant Injection
HPCS	High-Pressure Core Spray
HPI	High-Pressure Injection
HPME	High-Pressure Melt Ejection
HPR	High-Pressure Recirculation
HPSI	High-Pressure Safety Injection
HRA	Human Reliability Analysis
HVAC	Heating, Ventilating, and Air Conditioning
I	Iodine
IA	Instrument Air
IC	Isolation Condenser
IORV	Inadvertently Open Relief Valve
IPE	Individual Plant Examination
IPEP	Individual Plant Examination Partnership
ISGTR	Induced Steam Generator Tube Rupture
ISLOCA	Interfacing System Loss-of-Coolant Accident
kV	Kilovolt
LOCA	Loss-of-Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low-Pressure Coolant Injection
LPCS	Low-Pressure Core Spray
LPI	Low-Pressure Injection
LPR	Low-Pressure Recirculation
LPSI	Low-Pressure Safety Injection
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Program
MFW	Main Feedwater
MOV	Motor-Operated Valve
MSIV	Main Steam Isolation Valve
NMP	Nine Mile Point
NPSH	Net Positive Suction Head
NRC	US Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUMARC	Nuclear Management and Resources Council
ORCA	Operator Reliability Characterization and Assessment
ORE	Operator Reliability Experiments
PCPL	Primary Containment Pressure Limit
PCS	Power Conversion System
PDS	Plant Damage State
PIV	Pressure Isolation Valve
PORV	Power-Operated Relief Valve
PRA	Probabilistic Risk Analysis/Assessment

PSA	Probabilistic Safety Assessment
PSF	Performance Shaping Factor
PWR	Pressurized Water Reactor
QHO	Quantitative Health Objective
RAI	Request for Additional Information
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RES	Office of Nuclear Regulatory Research (NRC)
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWST	Refueling Water Storage Tank
RY	Reactor-Year
SAMG	Severe Accident Management Guidelines
SBO	Station Blackout
SBOR	Station Blackout Rule
SER	Staff Evaluation Report
SG	Steam Generator
SGFP	Steam Generator Feedwater Pump
SGTR	Steam Generator Tube Rupture
SHARP	Systematic Human Action Reliability Procedure
SLC	Standby Liquid Control
SLIM	Success Likelihood Index Methodology
SORV	Stuck Open Relief Valve
SRV	Safety Relief Valve
SSMP	Safe Shutdown Makeup Pump
SSW	Standby Service Water
SW	Service Water
Te	Tellurium
THERP	Technique for Human Error Rate Prediction
TMI	Three Mile Island
TRC	Time Reliability Correlation
TSC	Technical Support Center
TVA	Tennessee Valley Authority
USI	Unresolved Safety Issue
WNP2	Washington (State) Nuclear Power, Unit 2

PART 1

Summary Report

1. INTRODUCTION

1.1 Background

On August 8, 1985, the U.S. Nuclear Regulatory Commission (NRC) issued its "Policy Statement on Accidents Severe Accidents Regarding Future Designs and Existing Plants" (Ref. 1.1). That policy statement introduced the Commission's plan to address severe accident issues for existing commercial nuclear power plants. In the policy statement, the Commission formulated an approach for systematic safety examination of existing plants. The purpose of this examination was to study particular accident vulnerabilities and desirable, cost-effective changes to ensure that the plants do not pose any undue risk to public health and safety. To implement this approach, the NRC issued Generic Letter (GL) 88-20 (Ref. 1.2), requesting that all licensees perform an Individual Plant Examination (IPE) *"to identify plant-specific vulnerabilities to severe accidents and report the results to the Commission"* according to the format and content guidelines outlined in NUREG-1335 (Ref. 1.3).

As stated in GL 88-20, the IPE program has the following purposes:

(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.

(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.

In concert with the objectives of the Commission's Policy Statement, the Deputy Executive Director for Nuclear Reactor Regulation, Regional Operations and Research, issued a memorandum to the NRC's Office of Nuclear Regulatory Research[1.1]. That memorandum explicitly recommended that the NRC *"publish a world-class 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 office of Nuclear Regulatory Research then initiated the IPE Insights Program to document the significant safety insights, based on the IPEs, for the different reactor and containment types and plant designs. This report, which represents the culmination of that program, captures and documents the significant insights and improvements identified from the IPE submittals.

1.2 Objectives of the IPE Insights Program

Several major objectives were pursued in the IPE Insights Program which involved providing perspectives in the following areas:

The impact of the IPE program on reactor safety:

- the number and type of vulnerabilities or other safety issues that have been identified, and the related safety enhancements that have been implemented,

[1.1]Memorandum from James H. Sniezek, Deputy Executive Director for Nuclear Reactor Regulation, Regional Operations and Research, for Eric S. Beckjord, Director, Office of Nuclear Regulatory Research, "IPE Risk Reduction Document," May 12, 1993.

- the impact that the improvements have had on plant safety, and
- whether any of these improvements have "generic" implications for all or a class of plants.

IPE results perspectives:

- important design and operational features that affect core damage frequency (CDF) and containment performance, with regard to the different reactor and containment types,
- the influence of the IPE methodology and assumptions on the results,
- significant plant improvements to reduce CDF and increase containment performance, with regard to the different reactor and containment types,
- operator actions that are consistently important in the IPEs, and
- operator actions that are important because of plant-specific characteristics.

IPE models and methods perspectives:

- models and methods used in the IPE/PRAs (probabilistic risk assessments) that are well established (e.g., consistency in implementation) and robust (sound technical basis with negligible uncertainty),
- models and methods that are weak differentiating between the state-of-the-art versus inappropriate implementation or application of the method or model.

Safety goal implications:

- the implication of the IPE results relative to the current risk level of U.S. plants compared with the Commission's Safety Goals.

Station blackout rule:

- the improvements that have been identified as a result of the station blackout rule and analyzed as part of the IPE, and the impact of these improvements on reducing the likelihood of station blackout.

NUREG-1150 perspectives:

- the results of the IPEs compared with the perspectives gained from NUREG-1150 (Ref. 1.4).

1.3 Scope, Limitations, and General Comments

To accomplish the objectives listed in Section 1.2, the staff received and examined 75 IPE submittals covering 108 nuclear power plant units. The purpose of this examination was to determine what the collective IPE results imply about the safety of U.S. nuclear power plants. Therefore, the perspectives are to be viewed on a "global" basis and not necessarily interpreted or applied on a plant-specific basis. Consequently, the staff studied variations and commonalities among plant results to determine which factors were most influential on the results (CDF and containment performance). In addition, the staff examined the improvements that have been made at the plants, as well as the impact of these improvements on plant CDF and containment performance.

In many cases, licensees submitted a single IPE addressing more than one unit at a given site. In a few of these cases, the multiple units were separately analyzed within the single submittal. In most of these cases, however, a single analysis was performed. For such instances, this report presents the IPE perspectives for each unit individually. For example, a single IPE submittal addresses both Dresden 2 and 3; therefore, information in that submittal was counted twice.

The perspectives provided in this report are based on the original analyses (i.e., PRAs) performed by the licensees for their IPEs (see Appendix A for IPE references). In many cases, licensees have updated these PRAs to reflect plant changes and, in some cases, to incorporate staff concerns, as noted in the staff evaluation report (SER) of the licensee's IPE. For some of these PRAs, the results (e.g., CDF and dominant accident sequences) have changed; however, in most cases, no detailed information was supplied by the licensee. Updated information received by NRC is listed in Appendix B.

In examining the IPEs and developing this report, the staff used the information as reported by the licensees. That is, the staff did not consider the quality (e.g., accuracy) of the analyses when determining the implications of the collective IPE results. Therefore, the staff used information from each IPE, even if a licensee's IPE/PRA was unacceptable (in part or overall), and no adjustment or modification was made. Consequently, it should not be interpreted that the IPE information provided in this report is "approved" by the NRC. Rather, the staff position regarding the acceptability of each IPE submittal is documented in the respective SER.

In responding to GL 88-20, the licensees were not requested to perform a formal uncertainty analysis. Although the licensees estimated mean values for the basic events (e.g., component failure probabilities), most licensees did not propagate the uncertainty of the data for each event in quantifying the CDF and containment failure frequency. Therefore, for most IPEs, the reported CDFs and containment failure frequencies are point estimates and not mean values.

In several areas of a PRA, the state of knowledge has changed since GL 88-20 was issued in 1988. Changes occurred both while the IPEs/PRAs were performed and since the IPEs/PRAs were finalized and submitted to the NRC. Therefore, some issues discussed in this report (including component failure data, common cause data, human reliability analysis, shell melt-through, and direct containment heating, among others) have either been resolved or are being resolved through ongoing advances. The implication of this changing state of knowledge on these and other issues is discussed in this report, where applicable.

This report presents perspectives gained from reviewing the IPE submittals, which address internal initiators (excluding internal fires). This report does not include perspectives based on internal fires and external events (such as seismic events and tornadoes). In addition, risk- and safety-related insights and perspectives discussed in this report are limited to events at full power. Other modes of operation, such as shutdown, are not addressed, and factors such as aging and organizational influence are not explicitly modeled.

The IPE Insights Program is based solely on licensee submittals, which summarize the IPE analyses and do not fully document all design characteristics, analysis assumptions, and results. This limits the ability to fully account for similarities and differences in results. In addition, the IPE Insights Program does not address the correctness of the IPEs; rather, the staff investigated the results to determine whether any safety implications regarding the risk impact of plant and modeling differences can be deduced from the reported information.

Most licensees have indicated that they plan to make improvements to either plant systems or operations. In some cases, licensees have taken credit for these planned changes in their IPEs; in other cases, licensees have not credited improvements that were completed since the "freeze date" for the IPE analysis. This variability introduces nonuniformity when comparing the reported plant results, and the IPE submittals do not provide sufficient information to fully account for these differences (i.e., to "normalize" the results). However, this nonuniformity does not appear to be sufficiently significant (based on a review of the improvements), to mask the safety perspectives that are gained through examination of the IPE results.

1.4 Organization of this Report

The 75 separate IPE submittals covering 108 nuclear units contain a wealth of information. Table 1.1 lists the types of information contained in a single submittal. In examining this information from the IPE submittals relative to the objectives of the program, the report is organized around these objectives and is divided into three volumes. Volume 1, Part 1, is a Summary Report. The key perspectives relative to each of teh objectives discussed in Section 1.2 are summarized in this volume. In addition, the staff's overall conclusions and observations are provided in this volume. Volume 2, Parts 2-5 provide a more detailed discusssiom of the perspectives summarized in Volume 1. Volume 3 contains the appendices to the report. The contents of these volumes and associated chapters and appendices (as shown in Figure 1.1) are described in more detail below.

Table 1.1 Information contained in IPE submittals.

CDF general IPE information	Containment analysis general IPE information
• Plant design and operational information (e.g., system operation, function, dependencies, configuration) • Analysis scope, boundary conditions, data, assumptions, models, methods • Core damage frequency • Initiating events and frequencies • Success criteria • Operator actions and failure probabilities • Equipment failure probabilities • Accident sequence results • CDF and accident sequence dominant contributors • Plant vulnerabilities and improvements	• Plant design and operational information (e.g., cavity geometry, containment strength, spray operation) • Analysis scope, boundary conditions, data, assumptions, models, methods. • Plant damage states and frequencies • Containment event trees • Containment failure frequencies • Containment failure modes and mechanisms • Radionuclide release frequencies • Containment failure contributors • Mitigating systems • Containment performance improvements • Plant vulnerabilities and improvements

Glossary —

The staff anticipates that this report will be used by many different readers, each with different backgrounds. Furthermore, many terms used in this report are dependent on the technical context and, therefore, can vary in definition. A glossary is provided at the end of Volumes 1 and 2 to aid the reader in understanding the specific meaning of each term as used in this report.

Volume 1, Part 1 — Summary Report

Part 1 is a single-volume report divided into eight chapters, as follows:

- Chapter 1 serves as an introduction, providing background information; discussing the objectives of the IPE Insights Program; presenting the scope, limitations, and general comments regarding the program; and serving as a roadmap to the remainder of the document.

- Chapter 2 summarizes the key perspectives on the impact of the IPE program on reactor safety. Chapter 2 is divided into three sections as follows:

  — Section 2.1 discusses the plant vulnerabilities and their impact on reactor safety (as reported by licensees), along with any generic insights.

  — Section 2.2 discusses plant improvements and their impact on reactor safety (as reported by licensees) along with any generic insights.

  — Section 2.3 discusses plant-specific containment performance improvements identified by licensees.

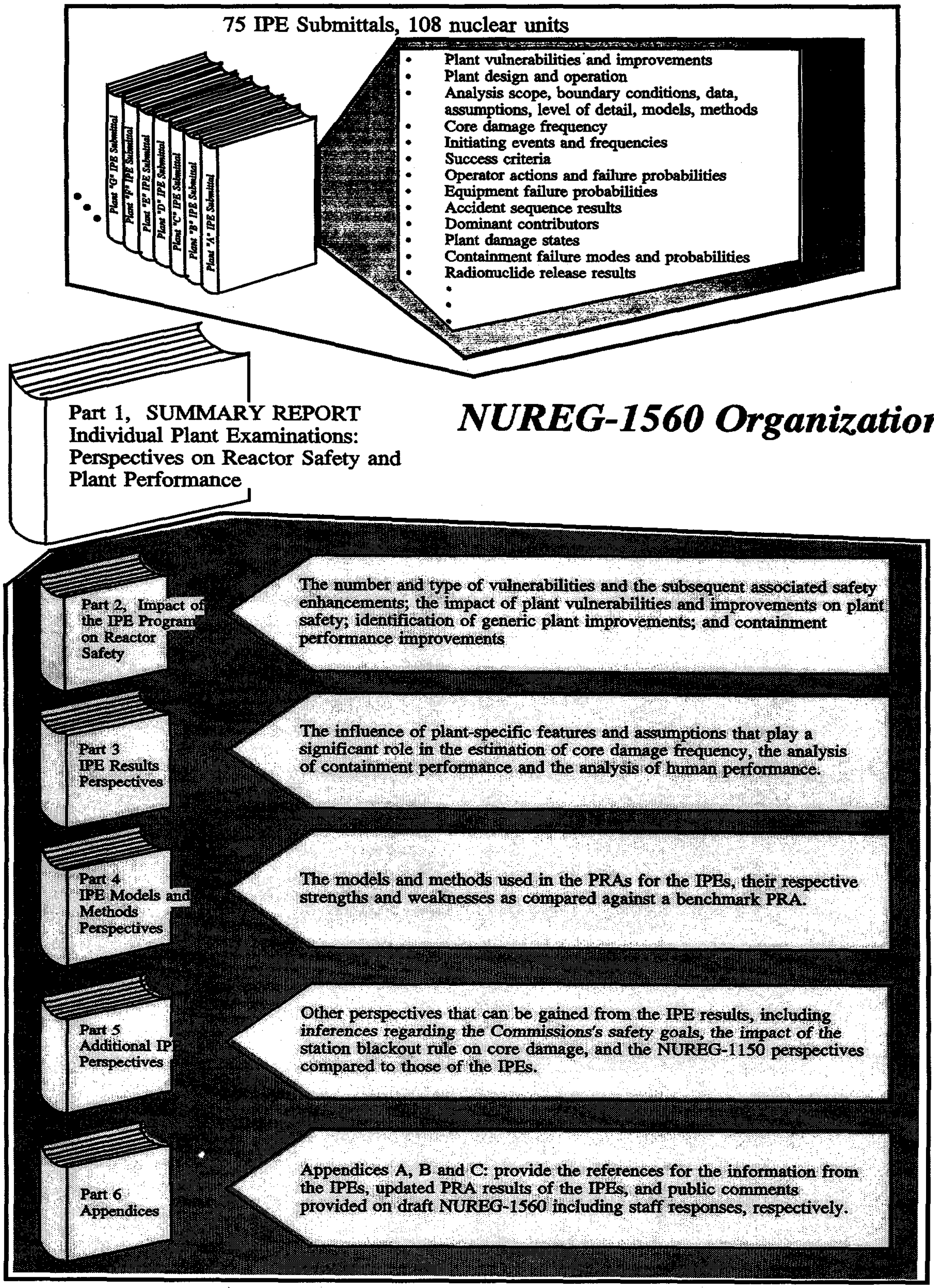


Figure 1.1 IPE NUREG report roadmap.

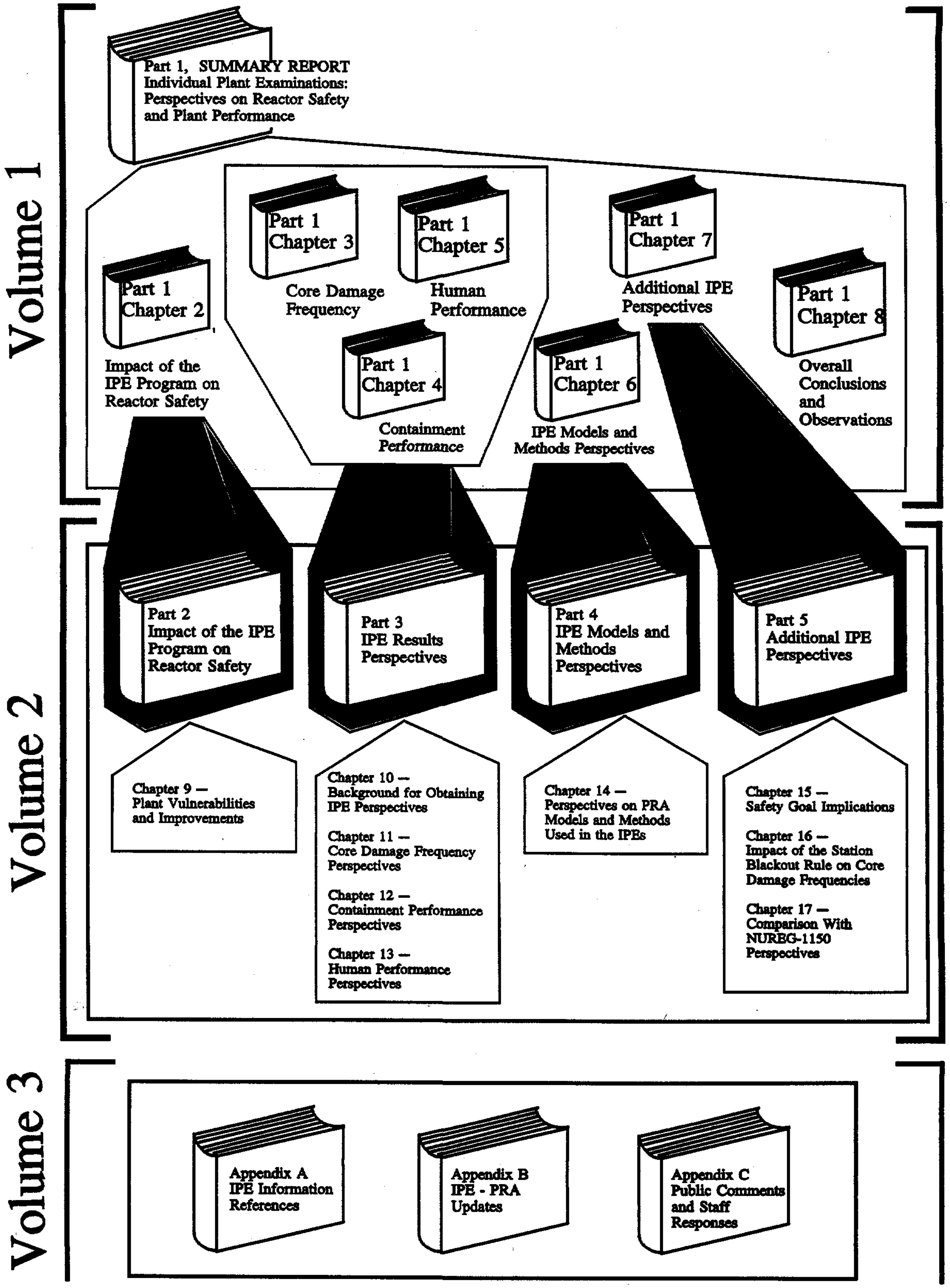


Figure 1.1 IPE NUREG report roadmap (continued).

- Chapter 3 summarizes the key perspectives regarding plant-specific features and assumptions that play a significant role in CDF. For each reactor class, this chapter discusses the key design and operational features that affect CDF, as well as the impact and influence of methods and assumptions on CDF results, and the significant improvements affecting CDF on a core damage accident class basis. The key perspectives discussed include those features, methods, and assumptions that have the greatest impact on causing the variability observed in the results for the given class of plants. Therefore, Chapter 3 is divided into sections aligned with the different classes of boiling water reactor (BWR) and pressurized water reactor (PWR) plants as defined in Tables 1.2 and 1.3, respectively. The perspectives within the different classes are discussed relative to the different accident classes as defined in Table 1.4.

Table 1.2 Summary of BWR plant classes and associated nuclear power plants.

<table>
<tr><th>Class</th><th>IPE submittals</th></tr>
<tr><td rowspan="2">BWR 1/2/3</td><td>• Big Rock Point • Dresden 2&3 • Millstone 1 • Nine Mile Point 1
• Oyster Creek</td></tr>
<tr><td>These plants generally have separate shutdown cooling and containment spray systems and a multi-loop core spray system. With the exception of Big Rock Point, which is housed in a large dry containment, these plants use an isolation condenser.</td></tr>
<tr><td rowspan="2">BWR 3/4</td><td>• Browns Ferry 2 • Brunswick 1&2 • Cooper • Duane Arnold
• Fermi 2 • FitzPatrick • Hatch 1&2 • Hope Creek
• Limerick 1&2 • Monticello • Peach Bottom 2&3 • Pilgrim 1
• Quad Cities 1&2 • Susquehanna 1&2 • Vermont Yankee</td></tr>
<tr><td>These plants are designed with two independent high-pressure injection systems, namely reactor core isolation cooling and high-pressure coolant injection (HPCI). The associated pumps are each powered by a steam-driven turbine. These plants also have a multi-loop core spray system and a multi-mode residual heat removal (RHR) system that can be aligned for low-pressure coolant injection, shutdown cooling, suppression pool cooling, and containment spray functions.</td></tr>
<tr><td rowspan="2">BWR 5/6</td><td>• Clinton • Grand Gulf 1 • LaSalle 1&2 • Nine Mile Point 2
• Perry 1 • River Bend • WNP 2</td></tr>
<tr><td>These plants use a high-pressure core spray (HPCS) system that replaced the HPCI system. The HPCS system consists of a single motor-driven pump train powered by its own electrical division complete with a designated diesel generator. These plants also have a single train low-pressure core spray system, as well as a multi-mode RHR system similar to the system design in the BWR 3/4 group.</td></tr>
</table>

Table 1.3 Summary of PWR plant classes and associated nuclear power plants.

Class	IPE submittals
Babcock & Wilcox (B&W)	• Arkansas Nuclear One,1 • Crystal River 3 • Davis-Besse • Oconee 1,2&3 • Three Mile Island 1
	The B&W plants use once-through steam generators. Primary system feed-and-bleed cooling can be established through the pressurizer power relief valves using the high-pressure injection (HPI) system. The HPI pump shutoff head is greater than the pressurizer safety relief valve setpoint. Emergency core cooling recirculation (ECCR) requires manual alignment to the containment sumps. The reactor coolant pumps (RCPs) are generally a Byron Jackson design.
Combustion Engineering (CE)	• Arkansas Nuclear One,2 • Calvert Cliffs 1&2 • Fort Calhoun 1 • Maine Yankee • Millstone 2 • Palisades • Palo Verde 1,2&3 • San Onofre 2&3 • St. Lucie 1&2 • Waterford 3
	The CE plants use U-tube steam generators with mixed capability to establish feed-and-bleed cooling. Several CE plants are designed without pressurizer power-operated valves. The RCPs are a Byron Jackson design.
Westinghouse 2-loop	• Ginna • Kewaunee • Point Beach 1&2 • Prairie Island 1&2
	These plants use U-tube steam generators and are designed with air-operated pressurizer relief valves. Two independent sources of high-pressure cooling are available to the RCP seals. Decay heat can be removed from the primary system using feed-and-bleed cooling. ECCR requires manual switchover to the containment sumps. The RCPs are a Westinghouse design.
Westinghouse 3-loop	• Beaver Valley 1 • Beaver Valley 2 • Farley 1&2 • North Anna 1&2 • Robinson 2 • Shearon Harris 1 • Summer • Surry 1&2 • Turkey Point 3&4
	This group is similar in design to the Westinghouse 2-loop group. The RCPs are a Westinghouse design.
Westinghouse 4-loop	• Braidwood 1&2 • Byron 1&2 • Callaway • Catawba 1&2 • Comanche Peak 1&2 • DC Cook 1&2 • Diablo Canyon 1&2 • Haddam Neck • Indian Point 2 • Indian Point 3 • McGuire 1&2 • Millstone 3 • Salem 1&2 • Seabrook • Sequoyah 1&2 • South Texas 1&2 • Vogtle 1&2 • Watts Bar 1 • Wolf Creek • Zion 1&2
	The Westinghouse 4-loop group includes nine plants housed within ice condenser containments. Many of these plants have large refueling water storage tanks such that switchover to ECCR either is not needed during the assumed mission time or is significantly delayed. The RCPs are a Westinghouse design.

Table 1.4 Definition of core damage accident classes.

Accident class	Accident class definition
Transients	— *events that disrupt the normal conditions in the plant requiring a reactor trip with the need for core heat removal. Transient initiators include events related to the balance-of-plant (e.g., turbine trip or loss of feedwater) and events associated with plant support systems (e.g., loss of service water or loss of AC bus).*
General Transients	For BWRs and PWRs, transient events followed by failure to successfully remove core heat and bring the reactor to safe shutdown For BWRs, this class is divided into two subclasses: (1) Transients with loss of coolant injection — Events followed by immediate loss of all coolant injection systems resulting in core damage and potentially containment failure (2) Transients with loss of decay heat removal — Events followed by initial success of coolant injection systems and immediate failure of decay heat removal systems. Adverse environments created in the suppression pool and the containment (or the connected building following containment venting or failure) may result in failure of coolant injection systems and subsequent core damage. Containment failure can occur before the initiation of core damage.
Station Blackout	Transient events that strictly involve an initial loss of offsite power followed by a failure of emergency onsite AC power. The failure of AC power results in failure of AC-dependent systems, leaving only the AC-independent system available for core heat removal
Anticipated transient without scram	Transient events followed by a failure to terminate the nuclear chain reaction by failing to insert the control rods
Loss of coolant accidents (LOCAs)	— *events that disrupt the normal conditions in the plant as a result of a breach in the primary coolant causing a loss of core coolant inventory and lead directly to a reactor trip with the need for core heat removal*
General LOCAs	LOCAs that involve primary system pipe breaks of all sizes that occur within the containment, pump seal failures, and inadvertent open relief valve initiating events. (The contribution from transient initiators with a subsequent stuck-open relief valve are included in the transient accident classes.)
Interfacing System LOCAs	LOCAs in systems that interface with the primary system (including the emergency core cooling system) at locations that result in an open path out of the containment
Steam Generator Tube Rupture	LOCAs that involve loss from the primary to the secondary through a ruptured steam generator tube
Internal Flooding	— *events that involve rupture of water lines or operator errors that directly result in failure of required mitigating systems (e.g., through loss of cooling) and/or fail other mitigating systems as a result of submergence or spraying of required components with water.*

- Chapter 4 summarizes the key perspectives on the plant-specific features and assumptions that play a significant role in the containment performance. For each containment class, this chapter discusses the key design and operational features that affect containment performance, as well as the impact and influence of methods and assumptions on containment performance results, and the significant improvements affecting containment performance on a containment failure class basis. The key perspectives discussed include those features, methods, and assumptions that have the greatest impact on causing the variability observed in the results for the given class of plants. Therefore, Chapter 4 is also divided into sections (as seen in Chapter 3) with the perspectives provided for the different BWR and PWR containment classes as defined in Tables 1.5 and 1.6, respectively. The perspectives in Chapter 4 are discussed relative to the different containment failure mode classes (as defined in Table 1.7). In addition, this criteria discusses perspectives on the reported radionuclide releases resulting from containment bypass or early containment failure.

Table 1.5 Summary of BWR containment classes and associated nuclear power plants.

Class	IPE submittals
Mark I	• Browns Ferry 2 • Brunswick 1&2 • Cooper • Dresden 2&3 • Duane Arnold • Fermi 2 • FitzPatrick • Hatch 1&2 • Hope Creek • Millstone 1 • Monticello • Nine Mile Point 1 • Oyster Creek • Peach Bottom 2&3 • Pilgrim 1 • Quad Cities 1&2 • Vermont Yankee
	The Mark I containment consists of two separate structures (volumes) connected by a series of large pipes. One volume, the drywell, houses the reactor vessel and primary system components. The other volume is a torus, called the wetwell, containing a large amount of water used for pressure suppression and as a heat sink. The Brunswick units use a reinforced concrete structure with a steel liner. All other Mark I containments are free-standing steel structures. The Mark I containments are inerted during plant operation to prevent hydrogen combustion.
Mark II	• LaSalle 1&2 • Limerick 1&2 • Nine Mile Point 2 • Susquehanna 1&2 • WNP 2
	The Mark II containment consists of a single structure divided into two volumes by a concrete floor. The drywell volume is situated directly above the wetwell volume and is connected to it with vertical pipes. Most Mark II containments are reinforced or post-tensioned concrete structures with a steel liner, but WNP 2 uses a free-standing steel structure. These containments are also inerted during plant operation to prevent hydrogen combustion.
Mark III	• Clinton • Grand Gulf 1 • Perry 1 • River Bend
	The Mark III containment is significantly larger then Mark I and Mark II containments, but has a lower design pressure. It consists of the drywell volume surrounded by the wetwell volume, with both enclosed by the primary containment shell. The drywell is a reinforced concrete structure in all Mark III containments, but the primary containment is a free-standing steel structure at Perry and River Bend, and a reinforced concrete structure with steel liner at Clinton and Grand Gulf. These containments are not inerted, but rely on igniters to burn off hydrogen and prevent significant accumulation during a severe accident.

Table 1.6 Summary of PWR containment classes and associated nuclear power plants.

<table>
<tr><th>Class</th><th colspan="4">IPE submittals</th></tr>
<tr><td rowspan="2">Large dry and Sub-atmospheric</td><td>• Arkansas Nuclear One,1
• Big Rock Point*
• Calvert Cliffs 1&2
• Diablo Canyon 1&2
• Haddam Neck
• Maine Yankee
• Oconee 1,2&3
• Prairie Island 1&2
• Salem 1&2
• Summer
• Waterford 3</td><td>• Arkansas Nuclear One,2
• Braidwood 1&2
• Comanche Peak 1&2
• Farley 1&2
• Indian Point 2
• Millstone 2
• Palisades
• Robinson 2
• South Texas 1&2
• Surry 1&2
• Wolf Creek</td><td>• Beaver Valley 1
• Byron 1&2
• Crystal River 3
• Fort Calhoun 1
• Indian Point 3
• Millstone 3
• Palo Verde 1,2&3
• Seabrook
• St. Lucie 1&2
• Three Mile Island 1
• Vogtle 1&2</td><td>• Beaver Valley 2
• Callaway
• Davis-Besse
• Ginna
• Kewaunee
• North Anna 1&2
• Point Beach 1&2
• San Onofre 2&3
• Shearon Harris 1
• Turkey Point 3&4
• Zion 1&2</td></tr>
<tr><td colspan="4">The large dry and subatmospheric containment group includes of 65 units, of which 7 have containments kept at subatmospheric pressures. These containments rely on structural strength and large internal volume to maintain integrity during an accident. Most of these containments use a reinforced or post-tensioned concrete design with a steel liner. A few units are of steel construction.</td></tr>
<tr><td rowspan="2">Ice condensers</td><td>• Catawba 1&2
• Watts Bar 1</td><td>• DC Cook 1&2</td><td>• McGuire 1&2</td><td>• Sequoyah 1&2</td></tr>
<tr><td colspan="4">The ice condenser containment is a pressure suppression containment that relies on the capability of the ice condenser system to absorb energy released during an accident. The volumes and strength of these containments are less than those of the large dry containments. Ice condenser containments also rely on igniters to control the accumulation of hydrogen during an accident. Seven of the ice condenser units have a cylindrical steel containment surrounded by a concrete secondary containment. The remaining two units have a concrete containment with a steel liner and lack secondary containments.</td></tr>
<tr><td colspan="5">*Although Big Rock Point has a BWR, it is housed in a large dry containment; therefore, for containment classification purposes, it is considered a PWR containment.</td></tr>
</table>

Table 1.7 Definition of containment failure mode classes.

Failure mode	Containment failure mode definition
Bypass	Failure of the pressure boundary between the high-pressure reactor coolant system and a low-pressure auxiliary system. For PWRs, bypass can also occur because the failure of the steam generator tubes, either as an initiating event or as a result of severe accident conditions. In these scenarios, if core damage occurs, a direct path to the environment can exist.
Early	Structural failure of the containment within a few hours of the start of core damage. Early structural failure can result from a variety of mechanisms such as direct contact of the core debris with the containment, rapid pressure and temperature loads, hydrogen combustion, and fuel-coolant interactions. Failures to isolate containment and vented containments are classified as early containment failures.
Late	Structural failure of the containment several hours after reactor vessel failure. Late structural failure can result from a variety of mechanisms, such as gradual pressure and temperature increases, hydrogen combustion, and basemat melt-through by core debris. Venting containment late in an accident is classified as a late containment failure.

- Chapter 5 summarizes the key perspectives on the importance of the operator's role in CDF estimation and containment performance analysis. The important human actions are discussed for both the BWRs and the PWRs. This discussion includes a description of the human actions generally important for the plants, a summary of the differences between reactor classes, and a discussion of human actions important at only a few plants. In addition, this chapter discusses perspectives on the variability observed in the human actions, with emphasis on one particular operator action (as an example of causes in variability).

- Chapter 6 summarizes the key perspectives on the models and methods used in the IPEs, their perceived strengths and weaknesses.

- Chapter 7 summarizes additional IPE perspectives, and is divided into three sections as follows:

  — Section 7.1 discusses the NUREG-1150 risk results in light of what can be inferred from the IPE results relative to the Commission's Safety Goals.

  — Section 7.2 discusses the plant improvements associated with specific regulations (i.e., station blackout rule) in light of their impact on core damage frequency.

  — Section 7.3 discusses key perspectives identified in NUREG-1150 in light of the results from the IPE analyses.

- Chapter 8 presents overall conclusions and observations considering the various perspectives provided in the previous chapters and the primary purposes of the IPE Insights Program to permit an understanding of how reactor safety has been improved by the IPE initiative. In this regard, Chapter 8 provides perspectives regulatory follow-up activities, potential safety issues, plant inspection, areas for research, safety goal's, PRA and the use of NUREG-1560 as a resource document.

Volume 2, Parts 2 through 5 —

Parts 2 through 5 comprise a single volume divided into ten chapters, as described below.

Part 2 — Impact of the IPE Program on Reactor Safety

Part 2 provides a more in-depth discussion of the information provided in Part 1, Chapter 2, Impact of the IPE Program on Reactor Safety. Part 2 comprises a single chapter, Chapter 9, concerning Plant Vulnerabilities and Improvements (including containment performance improvements). Specifically, Chapter 9 summarizes the criteria used to define vulnerabilities in the IPEs, and discusses specific vulnerabilities identified by the licensees and the actions taken to address those vulnerabilities. This chapter also presents further discussion regarding specific improvements identified by various licensees.

Part 3 — IPE Results Perspectives

Part 3 provides a more in-depth discussion of the information in Part 1, Chapters 3, 4 and 5, regarding CDF and containment performance and human performance perspectives. As such, Part 3 is divided into the following four chapters:

- Chapter 10, Background for Obtaining Reactor and Containment Design Perspectives, explains the approach chosen to obtain the perspectives regarding core damage frequency and containment performance. In addition, this chapter makes the reader aware of the plant and containment characteristics, as well as the boundary conditions, assessments, and assumptions used in IPE modeling that can potentially affect the results reported in the IPEs. This information will help the reader understand the specific perspectives and insights discussed in Chapters 11 and 12.

- Chapter 11, Core Damage Frequency Perspectives, discusses the CDF perspectives relative to reactor design in greater depth than is provided in Chapter 3. This discussion includes the dominant contributors summarized in Chapter 3 for each accident class in each reactor class, along with discussion of other contributors to the accident class CDFs. This chapter also provides quantitative CDF information, indicating the ranges of reported CDFs and averages. In addition, for each reactor class, this chapter discusses the factors causing plants to have the highest and lowest CDFs for each accident class.

- Chapter 12, Containment Performance Perspectives, provides additional details about the perspectives obtained regarding the treatment and results of containment performance in the IPEs, as summarized in Chapter 4. As such, this chapter provides further discussion of the plant-specific features and assumptions that impact the results. In addition, this chapter presents more quantitative information, involving ranges and averages of probabilities and frequencies of containment failure modes and releases, grouped by containment class.

- Chapter 13, Human Performance Perspectives, provides additional perspectives regarding human actions beyond those summarized in Chapter 5. This discussion includes the approach used to obtain the perspectives, as well as additional detail regarding the approaches used to model human actions in the IPEs. In general, this chapter provides more in-depth discussions for the perspectives summarized in Chapter 5 as well as more examples. The discussion regarding the difference in important operator actions relative to reactor class is considerably expanded in Chapter 13, which also provides more examples of the causes of variability in important human actions.

Part 4 — IPE Models and Methods Perspectives

Part 4 provides a more in-depth discussion of the information discussed in Part 1, Chapter 6, IPE Models and Methods Perspectives. Part 4 comprises a single chapter, Chapter 14. It provides a "benchmark" or "characteristics" of a quality PRA and compares these charateristics to the methods and models used in the IPEs to glean perspectives on the strengths and weaknesses of PRA.

Part 5 — Additional IPE Perspectives

Part 5 provides a more in-depth discussion of the additional IPE perspectives discussed in Part 1, Chapter 7, and is divided into the three following chapters:

- Chapter 15 provides a detailed description of how the IPE results were compared to the NRC safety goals and subsidiary objectives. In particular, this chapter provides more detail concerning the approach adopted to infer how the IPE results might be compared to the quantitative health objectives. This comparison was complicated because offsite risk estimates were not reported in most IPEs.

- Chapter 16 provides further information related to the impact of the station blackout rule on CDFs. This information includes details on the approach used to address the impact of the station blackout rule, including a discussion of the type of coping methods used by various plants to comply with the station blackout rule. Chapter 16 also provides further details (beyond those in Section 7.2) on the factors affecting the station blackout CDF for the groups of plants that accounted for implementation of the station blackout rule in the IPEs and those that did not account for implementation of the rule in the IPEs. In addition, this chapter presents results of regression analyses that were performed to determine the key factors affecting the station blackout CDF.

- Chapter 17 provides greater detail regarding a comparison of the IPE results with those reported in NUREG-1150. Specifically, this chapter provides more detail on a numerical comparison of the results and the underlying reasons for the observed differences in the CDF analyses and containment performance assessments. In addition, this chapter contrasts the perspectives derived from the NUREG-1150 study with those drawn from the reported IPE results.

Volume 3, Part 6 — Appendices

APPENDIX A — This appendix provides the reference sources for information from each IPE.

APPENDIX B — This appendix identifies which utility has updated their IPE/PRA.

APPENDIX C — This appendix provides the public comments received on draft NUREG-1560 and the staff responses and disposition of the comments.

REFERENCES FOR CHAPTER 1

1.1 USNRC, "Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants," *Federal Register*, Vol. 50, p. 32138, August 8, 1985.

1.2 USNRC, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f)," Generic Letter 88-20, November 23, 1988.

1.3 USNRC, "Individual Plant Examination: Submittal Guidance, "NUREG-1335, August 1989.

1.4 USNRC, "Severe Accident Risks: An Assessment for Fire U.S. Nuclear Power Plants," NUREG-1150, December 1990.

2. IMPACT OF THE IPE PROGRAM ON REACTOR SAFETY

The primary goal of the Individual Plant Examination (IPE) program was for licensees to identify plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements. While only a small fraction of the licensees actually identified vulnerabilities in their IPE submittals, nearly all of the licensees identified other areas warranting investigation for potential improvements. Thus, the IPE program served as a catalyst for further improving the overall safety of nuclear power plants. This additional effect of the IPE program is a very important element in measuring the success of the IPE program. Meaningful and cost-effective equipment and procedural changes to the plants have benefitted the overall safety of the industry and may not have occurred without implementation of the IPE process, with its inherent, systematic analysis of plant safety.

The IPE submittals do not generally report the quantitative impacts of plant improvements. In the few that do report this information, individual improvements generally reduce core damage frequency (CDF) by less than 50%, but some improvements yield reductions in the total CDF by as high as two orders of magnitude.

No specific definition for what constitutes a "vulnerability" was provided in Generic Letter (GL) 88-20 (Ref. 2.1) or in the subsequent IPE submittal guidance documented by the U.S. Nuclear Regulatory Commission (NRC) in NUREG-1335 (Ref. 2.2). Instead, licensees were asked to decide if a specific vulnerability or weakness exists at their plant and whether some corrective action is needed. Hence, there is considerable variability among the IPE submittals regarding what constitutes a vulnerability. A problem that is explicitly considered to be a vulnerability at one plant may have been identified as a weakness at another plant instead of being called a vulnerability at another plant. Therefore, no connotation of "good" or "bad" should be associated with plants that did or did not identify vulnerabilities. However, all licensees could benefit from reviewing the vulnerabilities identified at other plants and consider if implementation of plant improvements identified by other licensees would substantially improve safety at their plant in a cost-effective manner.

Furthermore, for many plants, the submittal wording is such that it is not always clear whether a licensee is identifying a finding as a "vulnerability" or as some other issue worthy of attention. As a result, this report differentiates between cases in which the licensee appeared to *explicitly* define an issue as a "vulnerability" with proposed resolutions and cases in which the licensee identified plant improvements to address other issues.

The following sections discuss the various definitions of "vulnerability" used by licensees, as well as the identified vulnerabilities. In addition, the discussion presents the plant improvements identified by the licensees to address some of these vulnerabilities and other issues not explicitly identified as vulnerabilities.

The results and perspectives presented below are based on the original IPE submittals forwarded by the licensees to the NRC (refer to Appendix A for sources of IPE information used in this report). Many licensees have updated their IPEs since the orignal submittal and have reported changes in core damage frequency or release frequency; but in most cases, detailed information was not provided. Updated IPE information received by the NRC is listed in Appendix B.

2.1 Plant Vulnerabilities

Many IPE submittals use one of the following sets of quantitative criteria as the basis for defining what constitutes a vulnerability:

- the criteria provided in the "Severe Accident Issue Closure Guidelines," NUMARC Document 91-04 (Ref. 2.3)

- the NRC's policy statement regarding the implementation of safety goals (Ref. 2.4) defining a CDF objective of 1E-4/ry and a large release objective of 1E-6/ry

- importance measures or the results of sensitivity studies to determine which components or systems are the most vital to the plant

The IPE submittals evaluated by the staff identified several variations and combinations of these criteria. No specific definition of vulnerability could be identified for approximately one-third of the plants. However, for a significant number of this group of plants, some sort of criteria were used to identify areas for improving plant safety. Using the various criteria, only about 20 percent of the plants explicitly identified vulnerabilities. The majority of those vulnerabilities related to the CDF analysis portion of the IPE. The vulnerabilities and proposed resolutions are discussed below.

2.1.1 Boiling Water Reactor Vulnerabilities

Only four licensees with boiling water reactors (BWRs) explicitly stated that their plants have vulnerabilities. No common vulnerabilities are identified. However, all BWR licensees could benefit by reviewing the proposed plant improvements to address these vulnerabilities for potential implementation at their plants.

Issues involving failure of the water supplies to the isolation condenser and failure of the operator to initiate the isolation condenser in time to prevent safety relief valves from lifting and subsequently sticking open (effectively failing isolation condenser operation) were defined as vulnerabilities in one BWR 3 IPE. The same licensee also identified drywell steel shell melt-through (a containment performance issue) as a Mark I containment vulnerability. The same licensee identified that operator failure to restore and maintain the reactor pressure vessel (RPV) level as an important operator action that met their criteria for a vulnerability.

Another licensee identified a vulnerability in one BWR 4 IPE involving failure of the heating, ventilating, and air conditioning (HVAC) system in an electrical switchgear room. That failure would result in a delayed loss of power and available heat sinks. Another BWR 4 IPE identified vulnerabilities related to operation of the high-pressure coolant injection (HPCI) system and reactor core isolation cooling (RCIC) system during sequences with loss of containment heat removal. The same licensee also identified vulnerabilities related to automatic injection from the low-pressure coolant injection (LPCI) system that could result in power excursions during an anticipated transient without scram (ATWS).

In another BWR 4 IPE, the licensee identified a unique vulnerability that resulted from a previous plant modification to delete the residual heat removal (RHR) system loop selection logic. That modification realigned RHR-related components to different safety-related buses, resulting in a vulnerability involving the loss of three out of four RHR loops (either directly or through the RHR sevice water system) when either safety-related 4.16 kV bus is lost. The licensee for this plant has implemented procedure modifications and training for using firewater as a backup to the RHR service water (RHRSW) and installation of a cross-tie between RHRSW trains.

No vulnerabilities were defined by licensees with BWR 5/6 plants.

Table 2.1 lists licensee approaches to resolving the vulnerabilities identified in the IPEs. In some cases, the licensee resolved the vulnerability before the IPE was completed and the resolution was reflected in the IPE results. In other cases, no resolution was suggested for the particular vulnerability except to follow research developments concerning the issue and accident management strategies in general.

Table 2.1 Summary of BWR plant vulnerabilities identified by licensees.

Vulnerability description	Licensee approach to resolve vulnerability
BWR 1, 2 & 3 (Isolation Condensers)	
Failure of isolation condenser makeup from city water supply and diesel fire-water pump, resulting in isolation condenser failure.	Procure portable diesel pump; implement procedures to provide isolation condenser shell-side makeup following fire-water system failure.
Operator failure to initiate isolation condenser to prevent safety relief valves from lifting in station blackout.	None identified by licensee.
Operator failure to restore or maintain RPV level following various accident scenarios.	None identified by licensee.
Drywell steel shell melt-through by molten debris following core melt and RPV failure.	Follow research developments in this area and consider strategies as the program develops.
BWR 3 and 4	
Loss of 3/4 RHR loops (directly or through loss of RHRSW) as a result of catastrophic failure of either 4.16 kV Alternating Current (AC) safety bus.	Implemented procedure modification and operator training to allow manual alignment of fire protection system to RHRSW system. Considering installation of an RHRSW header cross-tie; providing a tap for the fire protection system on RHRSW loop B; and providing portable generator to charge safety Direct Current (DC) batteries (to prevent main steam isolation valve closure from battery depletion following loss of a 4.16 kV AC safety bus).
Delayed loss of power caused by loss of switchgear or Class 1E Panel Room HVAC.	Developed recovery procedure to supply alternative ventilation to prioritized rooms.
Upon high suppression pool temperature, procedures require manual operator actions to bypass HPCI suction transfer to suppression pool. Also must bypass high exhaust pressure trips for HPCI and RCIC upon high containment pressure.	Raised HPCI/RCIC back pressure trip setpoints to ensure timely availability and alignment of HPCI and RCIC for high-pressure injection; considering revising HPCI suction transfer control strategy.
Failure of HPCI and condensate during an ATWS is followed by reactor depressurization. Automatic LPCI initiation and injection of full flow for 5 minutes follows. Without immediate flow control by the operator, severe power excursion will occur.	Considering deletion of LPCI control delay or installation of override switch to allow for immediate operator control of LPCI injection.
During loss of off-site power or station blackout, condensate transfer system (CTS) keepfill function is lost; occurrence of waterhammer could cause failure of suppression pool cooling, causing containment failure, unless CTS is available for injection. Failure of the fire main as an injection source during station blackout will also result in vessel and containment failure.	Consider installation of independent, mobile diesel-powered AC generator power supply for CTS pumps.
BWR 5 and 6	
None	----

2.1.2 Pressurized Water Reactor Vulnerabilities

Only 15 licensees with pressurized water reactors (PWRs) explicitly stated that their plants have vulnerabilities. Vulnerabilities were reported in the IPEs for plants with either Combustion Engineering (CE) or Westinghouse reactors. However, none were identified in the IPEs for plants with Babcock and Wilcox (B&W) reactors.

Certain vulnerability issues were common among more than one plant. For instance, concerns related to reactor coolant pump (RCP) seal loss-of-coolant accidents (LOCAs), particularly when induced by loss of seal cooling from the component cooling water (CCW) system, were identified as a vulnerability by six licensees representing both CE and Westinghouse plant designs. This vulnerability can also involve failure to trip the RCPs upon loss of seal cooling, failure of additional seal cooling systems such as the charging pumps, and failure of the high-pressure safety injection (HPSI) system during the recirculation mode because of the loss of the CCW system. For the licensees identifying this vulnerability, resolution of the issue involved implementing or considering alternative RCP seal cooling capabilities, addressing the issue in severe accident management guidelines, or considering new pump seal materials.

Auxiliary feedwater (AFW) system turbine-driven pump reliability is also a common issue defined as a vulnerability by five plants of CE or Westinghouse design, as shown by the following examples:

- At a dual-unit site with CE reactors, the licensee identified a vulnerability associated with a design problem that causes both turbine-driven AFW pumps to be removed from service any time maintenance is required on one of the pump steam admission valves. The licensee implemented a modification in which additional valves have been added on the steam lines to allow continued operation of one AFW pump when the other is out for maintenance.

- One Westinghouse plant licensee identified a vulnerability involving a path that diverts condensate from the condensate storage tank to the main condenser, and thereby reducing the quantity available to the AFW pumps for secondary cooling. The diversion path is created through a valve that fails open upon loss of instrument air or control power. The proposed resolution is to change the valve logic such that it will fail closed upon loss of air or control power. (The licensee was reviewing the basis for the current fail safe position of the valve.)

Common vulnerabilities related to the failure of the operator to switch from the injection phase to the recirculation phase of coolant injection were identified at three Westinghouse plants, as follows:

- At one plant, the switchover is partially automated as the sump recirculation valves are opened upon a low refueling water storage tank (RWST) level; however, the operator must manually close the RWST suction valves. Failure of the RWST low-level signal is defined as a vulnerability, even though manual operator action to open the sump valves per procedures is not credited in the IPE. The resolution proposed in the IPE is to address accidents during the recirculation phase in the accident management guidelines.

- In the IPE for another plant, failure of the operator to transfer suction from the RWST to the containment sump during a large or medium LOCA represents a significant contributor to the CDF. A high probability of operator error is assigned to these scenarios because of the limited time available to establish sump recirculation before the RWST level drains below HPSI pump net positive suction head (NPSH) requirements. Since the RWST inventory is primarily depleted by the operation of the low-pressure safety injection (LPSI) pumps in the injection phase, the licensee is performing an analysis to justify stopping the LPSI pumps at a much earlier step in the sump transfer procedure; this modification would afford the operators more time to perform the switchover procedure.

Another common vulnerability is the loss of critical switchgear HVAC equipment, which results in loss of emergency AC power. This vulnerability was identified at two CE plants and two Westinghouse plants. The Westinghouse plant licensees are reviewing alarm response procedures to determine if they can provide more explicit guidance on how to establish adequate alternative cooling. The licensees for the CE plants have implemented a procedure to use staged portable fans for alternative cooling.

Internal flooding issues are defined as vulnerabilities in five Westinghouse plant IPEs. For example, flooding from failed cooling water components was the most significant vulnerability at one dual-unit site since the source of water is gravity fed and there is little means of isolating the failure. Proposed resolutions to this vulnerability include revising flooding procedures and training, conducting periodic inspections, replacing components identified as potential flood initiators, and improving sump pump protection from flood effects.

Interfacing system LOCAs resulting from multiple valve failures or occurring through normally open valves are defined as vulnerabilities at six Westinghouse plants. The resolutions proposed in these IPEs included procedural improvements that were implemented or under consideration to improve valve testing, LOCA identification and isolation, as well as a modification to close valves that were normally open.

One PWR licensee identified other vulnerabilities involving inadequate surveillance of specific valves, effects of losses of specific electrical buses, compressed air system failures, battery depletion, and the inability to cross-tie buses during loss of power conditions, among other examples.

Table 2.2 lists all of the vulnerabilities identified by licensees in the IPEs for PWR plants, along with the proposed resolutions for each.

Table 2.2 Summary of PWR plant vulnerabilities identified by licensees.

Vulnerability description	Licensee approach to resolve vulnerability
Babcock and Wilcox 2-loop PWR	
None	----
Combustion Engineering 2-loop PWR	
Significant probability of failure of both turbine-driven pumps associated with maintenance or common cause.	Manual isolation valves added upstream and downstream of both turbine-driven pump steam admission valves to allow for maintenance on one pump line at a time (included in requantification).
Loss of two vital AC buses results in 2/4 actuation channels to fail to their actuated state causing inadvertent actuation of engineered safety feature actuation system (ESFAS) (results in bus undervoltage signal and load shedding requiring recovery actions), the reactor protection system (generates an open signal to the power-operated relief valves (PORVs)), or the AFW actuation system (AFWAS) (results in closure of block valves interrupting AFW flow to both steam generators).	Improved awareness through documentation in corrective action system, review of procedures, and additional operator training. Considering redesigning ESFAS to minimize inadvertent undervoltage condition and AWFAS to avoid blocking both steam generators on an inadvertent AFWAS.

Table 2.2 Summary of PWR plant vulnerabilities identified by licensees.

Vulnerability description	Licensee approach to resolve vulnerability
RCP seal and safety injection failure on loss of CCW (a significant contribution to the loss of CCW occurs due to leakage through drain and isolation valves).	Capping downstream piping on all normally isolated drain and vent valves to reduce leakage in CCW system.
Loss of switchgear HVAC resulting in failure of both safety-related 4 kV buses with minimal time for startup of standby unit or compensatory actions on failure of the running unit.	Install pre-staged portable fans and develop procedure for switchgear HVAC recovery actions.
Limited alternatives to depressurization of the reactor coolant system (RCS) during a steam generator tube rupture (SGTR) (primarily operator actions to depressurize the RCS using the main or auxiliary pressure spray). Tripping of RCPs during a SGTR results in loss of main pressurizer spray.	Develop procedure to depressurize the RCS using the pressurizer vent valves.
Minimal surveillance on critical condensate supply manual valves (these valves must operate to align alternative water sources for the AFW pumps when the CST depletes).	Develop surveillance test, preventative maintenance, and performance evaluation procedures to periodically cycle the critical condensate supply manual valves.
On a plant trip, closure of main feedwater regulation valves in conjunction with inability of the steam generator feedwater pump (SGFP) control system to rapidly reduce pump speed, results in tripping the SGFPs on high discharge pressure.	Added digital feedwater system to rapidly reduce pump speed on a reactor trip in order to avoid the high discharge overpressure trip of the SGFPs.
Interfacing system LOCA (ISLOCA) RCP thermal barrier tube rupture.	Install relief valves to limit pressure buildup in the reactor building closed cooling water system.
Westinghouse 2-loop PWR	
In RHR system, normally open motor-operated valves (MOVs) in LPSI lines connected to the reactor coolant system provide an ISLOCA path during normal operations; four pressure isolation valves in the RHR system which are not leak tested are major contributors to an ISLOCA.	Conduct leak testing of four additional valves serving as a boundary between the reactor coolant system and a low-pressure system.
Incomplete procedural guidance for determining where an ISLOCA occurs in the event of failure of RHR pump suction valves.	Modify emergency operating procedure to improve guidance to the operators for identifying and mitigating an ISLOCA.
Internal flooding propagation from turbine building basement to adjoining areas containing safeguards equipment; doors that swing out of the affected room assumed to fail because they cannot withstand the flooding forces.	Modify swing direction of doors separating the turbine building basement from areas containing safeguards equipment to prevent flood propagation into these areas.

Table 2.2 Summary of PWR plant vulnerabilities identified by licensees.

Vulnerability description	Licensee approach to resolve vulnerability
Internal flooding results from failure of the circulating water expansion joint at the main condenser; routine inspections of expansion joints were not conducted.	Improve of inspection method for rubber expansion joints to identify potential flooding problems.
Upon loss of off-site power, 3 of 6 air compressors are unavailable, reducing reliability of instrument air; no procedural steps exist for maintaining a swing bus powering two of the remaining compressors.	Modify emergency operating procedures to ensure that power is supplied to the swing bus upon loss of either safeguard bus in order to have power available to two instrument air compressors.
Makeup valve to condenser fails open on loss of instrument air or control power, diverting condensate from the condensate storage tanks to the main condenser, and reducing the quantity available to the AFW system.	Reviewing design information to determine basis for the current fail safe position of the makeup valve.
Failure of the AFW system contributes approximately 30% to total CDF; reliability of turbine-driven auxiliary feedwater pump directly relates to approximately 20% of CDF.	Improve reliability of turbine-driven auxiliary feedwater pump.
Air compressors are subject to frequent maintenance outages, making the station and instrument air system less reliable.	Modified design to remove the two older air compressors and replace them with air-cooled air compressors.
Charging pump relief valves have opened diverting flow back to the volume control tank, thereby affecting the ability of pumps to provide reactor coolant system makeup and RCP seal cooling.	Investigating actions to prevent diversion of chemical and volume control system water through relief valves.
Westinghouse 3-loop PWR	
Operator failure to promptly provide alternative cooling upon loss of all emergency switchgear ventilation (may lead to loss of all AC power).	Considering change to alarm response procedures to provide more explicit guidance on how to establish sufficient alternative cooling in the event that both emergency switchgear ventilation fan trains fail.
Failure of breakers that transfer 4.16 kV non-emergency buses from unit station service transformers to system station service transformers, can lead to loss of all emergency AC power if the diesel generators also fail.	Develop procedure and training for repair or replacement of failed breakers.
Battery depletion in 3.8 hours following loss of AC power leads to failure of steam generator level instrumentation and subsequent failure of the turbine-driven AFW pump (possibly due to overfilling).	Consider providing more explicit guidance on battery load shedding or providing some means of battery charging during loss of all AC power.

Table 2.2 Summary of PWR plant vulnerabilities identified by licensees.

Vulnerability description	Licensee approach to resolve vulnerability
If reactor trip breakers fail, it is unlikely that operators can remove power to control rods before RCS pressure peaks during ATWS scenarios initiated by loss of main feedwater.	Considering adding the capability for operators to remove power from the bus.
In a station blackout, diesel generators of the other unit cannot be connected to emergency buses of the affected unit, since 4.16 kV emergency AC buses are not cross-tied between units.	Installing cross-tie connecting 4.16 kV normal buses of both units; revise existing procedures and provide training for this cross-tie.
Loss of all RCP seal cooling leads to possible seal failure and LOCA; both thermal barrier cooling and RCP seal injection depend on emergency AC power.	Considering new seal materials and alternative seal cooling systems to address RCP seal integrity for loss of all seal cooling scenarios.
100% load rejection feature not reliable during a loss of off-site power leading to a delayed reactor trip and lifting of the PORVs which can stick open, causing a small LOCA (which shortens time for power recovery).	Considering eliminating the PORV challenge by defeating 100% load rejection capability.
Containment bypass sequences are dominated by SGTR and ISLOCAs.	Change plant procedures and provide training to enhance operator response to SGTRs; improve guidance to operators to identify the key valve to close during an ISLOCA.
Containment overpressurization results from RCS blowdown, early hydrogen burns, and direct containment heating.	Consider extending procedures to all core damage sequences for reducing RCS pressure before vessel breach. (Procedures will provide instruction on aligning recirculation from the containment sump back to the vessel even if core damage has occurred.) Also consider alternative modes to inject water into the reactor cavity and conserve the RWST inventory.
Pressurizer PORV block valves closed during normal operation resulting in insufficient RCS pressure relief upon loss of main feedwater, failure of automatic and manual reactor trip, and failure of ATWS mitigating system actuation circuitry to initiate AFW flow.	Options for reducing the frequency from these ATWS sequences are under review.
Loss of both primary and secondary heat removal in the injection phase primarily caused by failure of the turbine-driven pump during station blackouout (SBO) (unavailability associated with pump testing/ maintenance.	Consider developing severe accident management guidelines to sufficiently to address the vulnerability.
Induced RCP seal LOCAs with loss of primary coolant makeup or adequate heat removal in injection or recirculation phase.	Same as above.

Table 2.2 Summary of PWR plant vulnerabilities identified by licensees.

Vulnerability description	Licensee approach to resolve vulnerability
Small LOCA with loss of primary coolant makeup or adequate heat removal in the injection phase (these sequences deal with failure of emergency feedwater or safety injection).	Same as above.
Small LOCA with loss of primary coolant makeup or adequate heat removal in the recirculation phase (85% of the frequency relates to failure of low-pressure recirculation associated with RWST signal failure following successful high-pressure injection, emergency feedwater actuation, and depressurization).	Same as above.
Medium or large LOCA with loss of primary coolant makeup or adequate heat removal in the injection phase (these sequences deal with failure of safety injection).	Same as above.
Medium or large LOCA with loss of primary coolant makeup or adequate heat removal in the recirculation phase; failure of low-pressure recirculation (associated with failure of RWST signal) following successful high-pressure injection for medium LOCAs.	Same as above.
Failure of reactivity control, primarily caused by reactor trip failure following total loss of instrument air, results in RCS overpressure due to inadequate pressure relief.	Same as above.
SGTRs with loss of effective inventory makeup result in containment bypass.	Plans on replacing the steam generators with a new design that should lower the expected SGTR frequency.
Internal flooding in the turbine building caused by rupture of one of four circulating water (CW) inlet MOVs.	Consider improving flood mitigation procedures and training (including inspection/maintenance improvements, use of submersible MOV operators, improved sump pump capacity/ reliability, and back flow prevention in drain lines).
Internal flooding in the turbine building caused by rupture of one of four service water expansion joints in the valve pits.	Same as above.
Internal flooding in the turbine building caused by severe rupture of one of four service water isolation MOVs in the valve pits.	Same as above.

Table 2.2 Summary of PWR plant vulnerabilities identified by licensees.

Vulnerability description	Licensee approach to resolve vulnerability
Internal flooding in the turbine building caused by rupture of the service water pipe in the valve pit on the CW inlet pipe.	Same as above.
Loss of CCW, combined with "B" charging pump unavailability, leading to an RCP seal LOCA; the high pressure recirculation function is predicted to fail and result in core damage.	Operate charging pump with supplemental cooling; install Service Water hose connections on "A" and "C" pumps (similar to existing hose connection for "B" pump), to allow operation independent of CCW.
Westinghouse 4-loop PWR	
Operator failure to transfer to sump recirculation following a large or medium LOCA, because of the limited time available before the RWST drains below the HPSI pump NPSH requirements.	Conduct analysis to justify stopping LPSI pumps much earlier, giving operators more time to transfer pump suction; increase emphasis in operator training on timely sump transfer emergency operating procedure.
SBO (major contributor to public risk).	Prioritize recovery of off-site power steps in procedure training; develop procedure for severe weather conditions: add air-cooled diesel generator; conduct numerous activities in response to station blackout rule.
ISLOCA (major contributor to public risk).	Add open-valve alarm as part of RHR autoclosure interlock removal; conduct RHR system walkdown in emergency safeguard feature building to determine characteristics of potential releases.
AFW and feed-and-bleed failures are involved in many accident sequences.	Prioritize operator training on the AFW system, recovery of main feedwater, and the primary feed-and-bleed procedure.
Failure of containment sump recirculation is found in dominant sequences.	Implemented design change for the cold leg recirculation array; prioritize training on transfer to sump recirculation steps in emergency operating procedure; prioritize maintenance of service water to containment recirculation cooler MOVs.
RHR valves direct initial leakage to the pressurizer relief tank; operator may transfer to LOCA procedures, never to procedure for LOCA outside of containment (procedures check for LOCAs inside containment before considering the possibility of an ISLOCA).	Consider revising emergency operating procedures related to ISLOCAs.
Internal flooding potential involving a demineralized water line in the relay and switchgear rooms.	Recommended modifying the demineralized water system by inserting an isolation valve and changing procedure for isolation.

2.2 Plant Improvements

As previously discussed, a major goal of the IPE process is to identify plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements. It is clear from the submittals, however, that most licensees went beyond this intent and identified more than 500 other improvements worthy of consideration or implementation, even though no specifically associated vulnerabilities are identified. Many of the plant improvements were identified by multiple licensees. The most often-cited plant improvements for BWRs address station blackout concerns; the PWR improvements address both loss of power and loss of RCP seal cooling. Submittals for both PWR and BWR plants often suggest changes aimed at improving core cooling or injection reliability, particularly for those systems or portions thereof that can operate during loss of AC power. Improvements to address internal flooding and ISLOCAs are identified more often in PWR IPEs than in BWR IPEs. Other less-cited and plant-specific improvements were identified to address a number of other accident class issues at individual plants. Table 2.3 summarizes the commonly identified plant improvements.

Table 2.3 Summary of common plant improvements identified by licensees.

Area of improvement	Applicability		Specific improvement	Status as of submittal
	BWR	PWR		
AC Power	✓	✓	• Add or replace diesel generators	~50% of these improvements had been implemented
	✓	✓	• Add or replace gas turbine generator	
	✓	✓	• Implement redundant off-site power capabilities	
	✓	✓	• Improve bus/unit cross-tie capabilities	
DC Power		✓	• Install new batteries, chargers, or inverters	~50% of these improvements had been implemented
	✓		• Implement alternative battery charging capabilities	
	✓	✓	• Increase bus load shedding	
Coolant Injection Systems	✓		• Replace emergency core cooling system pump motors with air-cooled motors	~30% of these improvements had been implemented
	✓		• Align LPCI or core spray to CST upon loss of suppression pool cooling	
	✓		• Align firewater system for reactor vessel injection	
	✓		• Revise HPCI and RCIC actuation or trip setpoints	
	✓		• Revise procedures to inhibit the automatic depressurization system (ADS) for non-ATWS scenarios	
		✓	• Improve procedures and training regarding switchover to recirculation	
		✓	• Increase training on feed-and-bleed operations	
Decay Heat Removal (DHR) Systems	✓		• Add hard-pipe vent	~70% of these improvements had been implemented
	✓		• Install portable fire pump to provide isolation condenser makeup	
		✓	• Install new AFW pump or improve existing pump reliability	
		✓	• Refill CST when using AFW	
		✓	• Implement a modification to align the firewater pump to the feed steam generator	

Table 2.3 Summary of common plant improvements identified by licensees.

Area of improvement	Applicability: BWR	Applicability: PWR	Specific improvement	Status as of submittal
Support Systems	✓	✓	• Implement procedures and install portable fans for alternative room cooling upon loss of HVAC	~60% of these improvements had been implemented
	✓	✓	• Install temperature alarms in rooms to detect loss of HVAC	
	✓	✓	• Revise procedures and training for loss of support systems	
ATWS	✓		• Revise training on mechanically bound control rods	~25% of these improvements had been implemented
	✓		• Install automatic ADS inhibit for ATWS scenarios	
	✓		• Install alternative boron injection system	
		✓	• Add capability to remove power to the bus upon trip breaker failure	
		✓	• Install Westinghouse ATWS mitigating system	
RCP Seal LOCAs	✓	✓	• Evaluate or replace RCP seal material	~30% of these improvements had been implemented
		✓	• Add independent seal injection or charging pump for SBO	
		✓	• Supply RCP seals with alternative cooling	
		✓	• Conduct operator training on tripping pumps on loss of cooling	
		✓	• Review HPSI dependency on CCW	
SGTRs		✓	• Revise procedure to maintain a higher inventory of water in the borated water storage tank (BWST) or refill BWST	~35% of these improvements had been implemented
		✓	• Implement procedure and training to isolate affected steam generator	
Internal Flooding	✓		• Increase protection of components from flood effects	~60% of these improvements had been implemented
	✓	✓	• Conduct periodic inspections of cooling water piping and components	
		✓	• Revise procedure for inspecting the floor drain and flood barriers	
		✓	• Install water-tight doors	
ISLOCAs	✓	✓	• Review surveillance procedures involving isolation valves	~65% of these improvements had been implemented
		✓	• Modify procedure to depressurize the RCS to reduce leakage	
	✓	✓	• Revise training to deal with ISLOCAs	
Containment Performance	✓		• Provide alternative power source to hydrogen igniters	~10% of these improvements had been implemented
		✓	• Enhance communication between sump and cavity	
		✓	• Inspect piping for cavity flooding systems	
		✓	• Revise procedures to use PORVs to depressurize the vessel following core damage	
Miscellaneous	✓	✓	• Incorporate IPE insights into the operator training program	~50% of these improvements had been implemented

Most of the plant improvements involve procedural/ operational changes (approximately 45%), design/ hardware changes (approximately 40%), or both. Few of the improvements involve maintenance-related changes. Typically, the design or procedural changes indicated revised training in order to properly implement the actual change. Approximately 45% of the plant improvements had been implemented at the time of the IPE submittals, but not all were credited in the IPEs. Approximately 25% were implemented and credited in the IPEs. Other improvements were either planned or under evaluation and may or may not have been implemented as of the date of this report.

Some improvements identified in the IPE submittals are associated with other requirements (primarily the station blackout rule) and other utility activities. However, although these improvements are not necessarily identified as a result of the IPE, in some cases, the licensee is using the IPE to prioritize the improvements and to support decisions regarding their implementation. More details about these plant improvements are provided below.

2.2.1 BWR Plant Improvements

The most often-cited plant improvements for BWRs address station blackout concerns. Many licensees with BWR plants made or are evaluating changes to improve AC power and DC power reliability, increase battery operable life under an extended blackout, and address other system weaknesses. While the area of improvement is the same, the specific improvements vary considerably. For instance, AC power system changes took the form of added or replaced diesel generators, redundant off-site power capabilities, improved AC power recovery potential, and better proceduralized bus cross-tie capabilities.

Unique to the BWR 5/6s, but not explicitly identified for every BWR 5/6, are improvements to increase flexibility in the use of the high-pressure core spray Division III diesel for operating Division I and II loads. A few licensees identified specific diesel generator cooling improvements. DC power system improvements took many different forms, including alternative battery charging capabilities, battery upgrades, additional load shedding, and improved cross-tie capabilities. These changes generally address the ability to maintain at least one train of DC power during loss of AC conditions, which is necessary to operate AC-independent systems and to power instrumentation/indications that provide plant status information to the operator.

The BWR IPE submittals also identified other changes to reduce the vulnerability to station blackout, as well as the potential for core damage from other types of accidents. Generally, these changes provide more reliable core cooling and/or improve the successful use of the AC-independent core cooling systems during prolonged operation. For instance, various isolation condenser improvements were identified for most of the earlier BWR plants. These improvements include better valve reliability, added procedural guidance for use of the isolation condenser during an extended blackout, and prolonged firewater capability for the isolation condensers during an extended station blackout.

Similarly, some licensees for later plants described improvements to ensure better overall reliability and availability of the AC-independent coolant injection systems (i.e., HPCI and RCIC). Among other things, these improvements included ensuring that these systems are indeed AC-independent (e.g., changing a RCIC room exhaust fan from AC-powered to DC-powered). Other licensees are installing a switch to bypass the RCIC high steam tunnel temperature trips. A few licensees are evaluating improvements that would better prepare the operators to replenish the CST (the preferred source of water for HPCI and RCIC). Some licensees also identified improvements regarding the ability to use firewater as a backup to HPCI and RCIC for core injection.

In a few BWR submittals, licensees identified specific improvements to deal with scenarios involving a loss of DHR or ATWS. These improvements involve ensuring the use of the CST for coolant injection pump suction whenever the suppression pool temperature is very high, automatically inhibiting the ADS during an ATWS, and using of an alternative boron injection capability. Such changes clearly target different issues related to either loss of DHR or ATWS accidents. Procedural guidance to use the CST for pump suction improves the long-term operability of coolant injection systems when the suppression pool temperature is above pump design-basis limits. Automatically inhibiting ADS during an ATWS reduces the potential for failing to perform this action when the operator is attempting to respond to a variety of symptoms during this class of accident. Use of an alternative boron injection capability represents an attempt to further increase the probability of successful boron injection when required.

2.2.2 PWR Plant Improvements

Station blackout and related power issues are addressed by improvements in PWRs from all three vendors. Improvements to the AC power system took many forms, indicating that no ***one*** fix is best for all PWRs. These improvements included the addition or replacement of diesels, with emphasis on the addition of swing diesels among multiple units, the addition or upgrading of gas turbines, and improved cross-tie capabilities and bus loading changes. Licensees also appeared to emphasize improvements to AC power equipment room cooling. These improvements included such changes as addition of temperature alarms, increased redundancy in the ventilation systems, and procedural improvements to deal with loss of ventilation. Many PWR licensees also cited DC power system improvements as beneficial to their plants. These improvements took the form of DC bus load shedding, battery upgrades, alternative battery charging features, and improved DC bus cross-tie capabilities. Increased DC power system reliability and prolonged battery life improve the ability to cope with a loss of AC power by providing power for continued operation of AC-independent steam-driven AFW pump trains, as well as power to instrumentation that provides indications of plant status to the operators.

One area of improvement that was addressed in many PWR IPE submittals involves RCP seal LOCAs and related loss of seal cooling issues. These issues are most important for Westinghouse plants. Although the licensees did not identify pump seal LOCAs as important for B&W and CE plants, some licensees with B&W plants did identify improvements to address this issue. These improvements typically involve alternative seal flow capability, sometimes even under loss of power conditions. The addition of high-temperature seals is notably documented in a number of Westinghouse IPE submittals. Hence, dealing with this potential source of primary coolant loss during station blackout as well as during other loss of seal cooling scenarios is generally important to PWR plants.

AFW system improvements were also identified for many PWRs (most commonly for Westinghouse plants). These typically included additional backup water supplies such as the firewater system and redundant pump cooling capability. Other reliability improvements identified for a few of the plants included the ability to operate the AFW manually even under a loss of DC power. These latter improvements address the ability to use the one AC-independent core cooling system in PWRs even during a loss of DC power, thereby increasing the chances of preventing core damage associated with a loss of secondary cooling.

Other PWR improvements include a procedural and some design improvements to deal with internal flooding. Specific improvements varied, indicating the plant-specific nature of dealing with flooding issues. Across all PWRs, licensees identified changes to deal with SGTRs, ISLOCAs, and miscellaneous system weaknesses. In particular, licensees cited enhanced procedural guidance for dealing with SGTRs as well as improved testing and valve status checking to decrease the potential for ISLOCAs.

2.3 Containment Performance Improvements

GL 88-20 and its supplements recommend containment performance improvements (CPIs) that focus on generic containment challenges associated with postulated severe accidents. The CPIs are not intended to be all-inclusive. In GL 88-20, the NRC recognizes that unique plant features warrant additional consideration in the IPE.

Supplement 1 to GL 88-20 (Ref. 2.5) provides CPI recommendations for BWR Mark I containments. These recommendations include installing hardened vents (also identified by separate generic letter (Ref. 2.6)), an alternative water supply for the drywell sprays and RPV injection, enhanced RPV depressurization system reliability, and implementation of the BWR Owners Group (BWROG) Emergency Planning Guidelines (EPGs), Revision 4 (Ref. 2.7). Supplement 3 to GL 88-20 (Ref. 2.8) provides CPI recommendations for Mark II and Mark III containments, PWR ice condenser containments, and PWR dry containments. Facilities with Mark II containments should also consider installing hardened vents for additional containment heat removal. In addition, PWRs should consider hydrogen production and control during severe accidents, particularly the potential for local hydrogen detonation in confined areas during SBO restoration.

The licensees' IPE submittals varied widely in their response to these CPI recommendations. In several cases, the licensees did not address the CPI recommendations. In other instances, the licensees indicated that the CPI recommendations are being considered but were not identified as commitments.

2.3.1 BWR Containment Performance Improvements

The 24 licensees with BWR Mark I containments have installed the hardened vents, but very few of the licensees indicated the modification was performed on the basis of their IPE analyses. Licensees with Mark I containments varied considerably in their assessment of the benefit gained from a hardened vent capability. A number of licensees indicated that the modification yields little or no benefit. Licensees with Mark I containments indicated that they had revised their emergency operating procedures to provide for the use of an alternative RPV water supply and, in some cases, an alternative drywell spray. For example, the Duane Arnold IPE described five separate sources for alternative RPV injection or drywell spray. One source is the fire water system, which is capable of using the independent diesel-driven pump to provide alternative injection and spray during SBO conditions with the RPV and drywell depressurized.

Licensees with Mark I containments did not propose any component modifications to enhance the reliability of RPV depressurization. However, several licenses indicated that they would revise their emergency operating procedures to better address the human reliability aspects of depressurizing the RPV. In each case, where licensees addressed the use of the BWROG EPGs, the staff found that the licensees have adopted Revision 4.

The eight licensees with BWR Mark II containments varied in their responses and commitments to the CPIs. However, the IPEs for the LaSalle and Limerick Units 1&2 specifically discussed venting. At LaSalle, the licensee has installed a hardened vent, provided an alternative RPV injection source using the firewater system, and enhanced the RPV depressurization method. At Limerick Units 1&2, the licensee has not installed a hardened vent pipe, but has committed to a fire hose cross-tie between the fire protection and the RHR systems to facilitate the use of firewater as an alternative water source for RPV injection. The RHR service water system can also be used as an alternative injection source injecting through a hard-pipe cross-tie to the RHR system. The licensee has also implemented procedural enhancements for use of the drywell sprays. At Susquehanna 1&2, modifications to the RHRSW system and procedures for tying the fire water system into RHR/RHRSW for RPV injection and containment injection have been implemented. In addition, all of the IPE submittals that address Revision 4 of the BWROG EPG have implemented those EPGs.

For the BWR Mark III containments, the licensees stated that backup power supplies are not needed for the hydrogen igniters to mitigate local hydrogen detonation in confined areas. However, some of the licensees stated that procedural changes are being considered for containment pressure control.

2.3.2 PWR Containment Performance Improvements

PWR licensees did not identify hydrogen production as a concern that would jeopardize the integrity of large dry and subatmospheric containments. However, some licensees indicated that procedural enhancements are being reviewed to vent the containment to reduce the frequency of containment bypass or overpressurization.

Licensees with ice condenser containments did not identify any specific improvements other than to review containment hydrogen burn strategies. However, one licensee indicated that alternative strategies for igniter operation would be considered for their plant's severe accident management guidelines.

REFERENCES FOR CHAPTER 2

2.1 USNRC, "Individual Plant Examination for Severe Accident Vulnerabilities-10 CFR § 50.54 (f)," Generic Letter 88-20, November 23, 1988.

2.2 USNRC, "Individual Plant Examination: Submittal Guidance," NUREG-1335, August 1989.

2.3 NUMARC, "Severe Accident Issue Closure Guidelines," NUMARC 91-04, January 1992.

2.4 USNRC, "Implementation of Safety Goal Policy," SECY-89-102, March 30, 1989.

2.5 USNRC, "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - CFR § 50.54(f)," Generic Letter 88-20, Supplement 1, August 29, 1989.

2.6 USNRC, "Installation of Hardened Wetwell Vent," Generic Letter 89-16, September 1, 1989.

2.7 General Electric, et al., "BWR Owners Group Emergency Procedures Guidelines, Rev. 4," NEDO-31333, Class 1 Document, March 1987.

2.8 USNRC, "Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities," Generic Letter 88-20, Supplement 3, July 6, 1991.

3. IPE RESULTS PERSPECTIVES: CORE DAMAGE FREQUENCY

This chapter summarizes perspectives gained from the core damage frequency (CDF) evaluations reported in the Individual Plant Examination (IPE) submittals, while Chapter 11 present the results and perspectives in greater detail. The perspectives address the plant-specific features and assumptions that play a significant role in CDF on a reactor class basis. That is, for each reactor class (see Chapter 1 for definition), the key design and operational features that affect CDF, as well as the impact and influence of methods and assumptions on CDF results, and the significant improvements affecting CDF on a core damage accident class basis are discussed. The key perspectives discussed, therefore, include those features, methods, and assumptions that have the greatest impact on causing the variability observed in the results for the given class of plants.

The results and perspective present below are based on the original IPE submittals forwarded by the licensees to the U.S. Nuclear Regulatory Commission (NRC) (refer to Appendix A for sources of IPE information used in this report). Many licensees have updated their IPEs since the original submittals and have reported bottom-line changes in CDF or release frequency; but in most cases, detail information was not provided. Updated IPE information received by the NRC is listed in Appendix B.

3.1 General Core Damage Frequency Perspectives

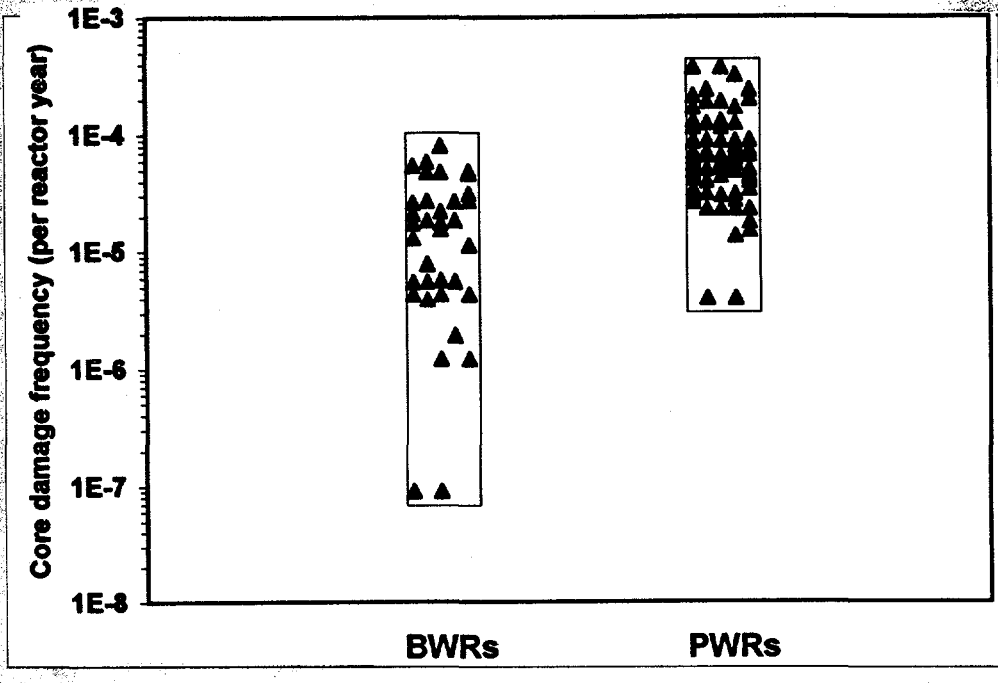
In many ways, the results from the IPEs are consistent with the results of previous risk studies by both the NRC and the industry. The IPEs indicate that the plant CDF[3.1] is determined by many different sequences, rather than being dominated by a single sequence or failure mechanism. The accident class that is the largest contributor to plant CDF varies considerably among the plants, as do the dominant failures contributing to that accident class (e.g., some are dominated by loss of coolant accidents (LOCAs), while others are dominated by station blackout (SBO)). However, for most plants, support systems are important to the results because support system failures can induce failures of multiple front-line systems. (For example, SBO sequences tend to be important contributors for both boiling water reactor (BWR) and pressurized water reactor (PWR) plant groups.) The support system designs and dependencies of front-line systems on support systems vary considerably among the plants, thereby explaining much of the variability observed in the IPE results. This variability was the motivation for the IPE program as noted in the NRC's Severe Accident Policy Statement.[3.2]

Consistent with previous risk studies, the CDFs reported in the original IPE submittals are lower on average for BWR plants than for PWR plants, as shown in Figure 3.1[3.3]. Although both BWR and PWR results are strongly affected by the support system considerations discussed above, there are a few key differences between the plant types. These differences are responsible for the tendency toward lower BWR CDFs as well as the difference observed in the relative contributions of the accident sequences to plant CDF.

[3.1]CDF is the average frequency per reactor-year of accidents potentially involving core damage. A more detailed description of the definition of CDF is provided in Chapter 14.

[3.2]50 *Federal Register* 32138, August 8, 1985.

[3.3]Most of the IPE submittals reported point estimates for the CDFs. In a few cases, uncertainty evaluations were performed, and the mean values were reported in the IPE submittals.



- BWR CDFs are generally lower than PWR CDFs because BWRs have more options for injecting coolant.
- Plant-specific support system design and operation (versus nuclear steam supply system design) drives most of variability.
- Transients and station blackout accidents are important contributors for most plants because of support system dependencies.

Figure 3.1 Reported IPE CDFs and key perspectives for BWRs and PWRs.

The most significant difference is that BWRs have more injection systems and can depressurize more easily than PWRs to use low-pressure injection systems. As a result, BWRs have a lower average contribution from LOCAs.

For transients, most PWRs can remove decay heat either through the steam generators or using the primary system feed-and-bleed function. BWRs remove decay heat directly from the primary system using a shutdown cooling system (or isolation condenser in older plants) or through a process analogous to the feed-and-bleed function, involving coolant injection and subsequent steaming either to the main condenser or the suppression pool. In PWRs, a transient-induced LOCA (e.g., reactor coolant pump (RCP) seal LOCA or stuck-open relief valve (SORV)) will defeat heat removal through the steam generators, and will require injection to maintain the reactor coolant system (RCS) inventory. Transient-induced LOCAs are not a significant problem for most BWRs because most means of decay heat removal require coolant injection and, as noted above, BWRs have more available injection systems than PWRs.

Many BWRs are more susceptible to transients with loss of containment heat removal (CHR) because the sequence results in an adverse environment that causes the failure of emergency core coolant system (ECCS) pumps (generally, due to loss of adequate net positive suction head) and other injection systems. This type of transient sequence is generally not as important for PWRs because of the design of the ECCS pumps.

The results obtained for some plants vary from the general trends noted above. As shown in Figure 3.1, CDF varies considerably within the BWR and PWR plant groups. This variability results in considerable overlap between the CDFs of the two groups.

The specific reasons driving the differences observed in results among the plants (including the significantly lower CDFs for the two outlier plants shown in Figure 3.1) are summarized in Sections 3.2 and 3.3, and discussed in more detail in Chapter 11 in Part 3. The CDF perspectives are based on the original IPE submittals and may not reflect the results of updated assessments performed by the licensees. Sections 3.2 and 3.3 focus on the factors that have the greatest influence on the bulk of the plants, while Part 3 includes a discussion of the plants with the highest and lowest CDFs. The variability is driven by a combination of the following factors:

- plant design differences (primarily in support systems such as cooling water, electrical power, ventilation, and air systems),
- variability in modeling assumptions (including whether the models accounted for alternative accident mitigating systems), and
- differences in data values (including human error probabilities) used in quantifying the models.

Table 3.1 summarizes the key observations regarding the importance and variability of each accident sequence. Section 3.2 and 3.3 provide further details.

Table 3.1 Summary of CDF perspectives for light water reactors.

Accident class	Key observations
Transients	Important contributor for most plants because of reliance on support systems; support system failure can defeat redundancy in front-line systems Both plant-specific design differences and IPE modeling assumptions contribute to variability in results, based on the following factors: • capability to use alternative injection systems (BWRs) • capability to use feed-and-bleed cooling (PWRs) • susceptibility to RCP seal LOCAs (PWRs)

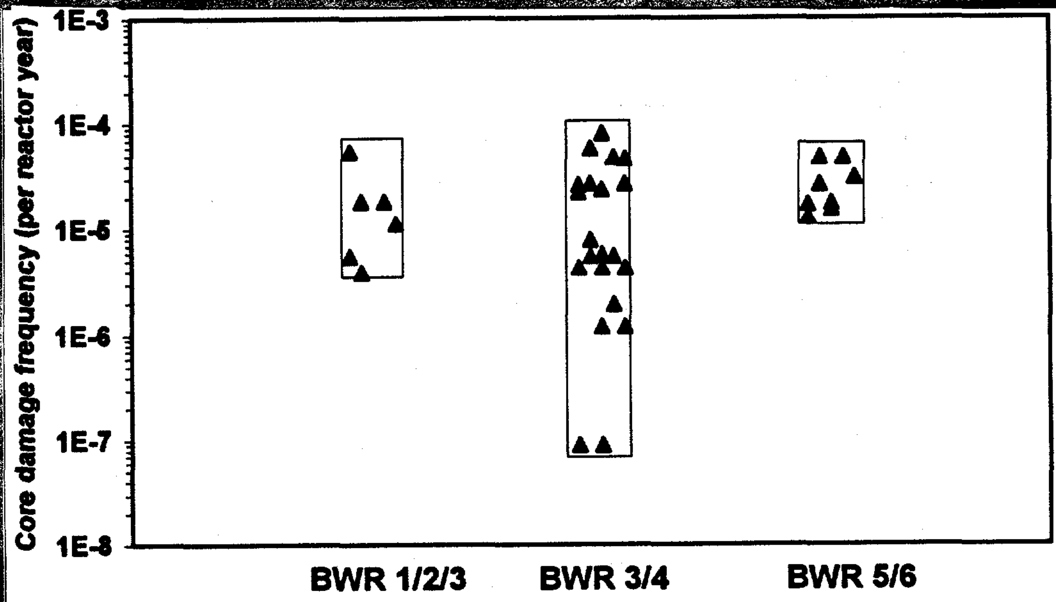
Table 3.1 Summary of CDF perspectives for light water reactors.

Accident class	Key observations
SBOs	Important contributor for most plants, with variability driven by the following factors: • number of emergency AC power sources • alternative off-site power sources • battery life • availability of firewater as an injection source (BWRs) • susceptibility to RCP seal LOCAs (PWRs)
LOCAs	Important contributors for many PWRs BWRs generally have lower LOCA CDFs than PWRs for the following reasons: • BWRs have more injection systems • BWRs can more readily depressurize to use low-pressure injection systems
Internal Floods	Small contributor for most plants, but important for some because of plant-specific designs Largest contributors involve water system breaks that cause the failure of multiple mitigating systems (directly or through flooding effects)
Anticipated transient without scram (ATWS)	Normally a small contributor to plant CDF because of reliable scram function and successful operator responses BWR variability is mostly driven by modeling of human errors but includes some impacts due to plant-specific features; PWR variability is mostly driven by plant operating characteristics and IPE modeling assumptions
Bypass Sequences	Interfacing system LOCAs (ISLOCAs) are a small contributor to plant CDF for BWRs and PWRs because of the low frequency of the initiator Steam generator tube rupture (SGTR) is normally a small contributor to CDF for PWRs because of opportunities for the operator to isolate a break and terminate the accident

3.2 Boiling Water Reactor Perspectives

Figure 3.2 shows the total CDFs for all operating BWRs in each of the BWR plant groups as obtained from the original submittals. With the exception of a few outliers, the total CDF for most BWRs falls within a range spanning two orders of magnitude. Variability in the results is attributed to a combination of factors including plant design differences (especially in support systems such as electrical power, cooling water, ventilation, and instrument air (IA) systems), modeling assumptions, and differences in data values including human error probabilities. The largest variation logically exists in the BWR 3/4 group, which includes the largest number of plants. Among this group, variability in plant design and modeling assumptions resulted in several plants with CDFs below the remaining BWRs, and one plant (two units) considerably below the others. This outlier is discussed in Section 3.2.2.

Table 3.2 summarizes the importance of the various accident classes to the BWR CDFs and the plant-specific features or other factors influencing the results. Significant variability exists for each BWR group with regard to contributions of the different accident classes to total plant CDF. However, licensees in all three BWR groups generally found that three types of accidents are the major contributors to total plant CDF. These three accident categories (i.e., SBOs, transients with loss of coolant injection, and transients with loss of decay heat removal) involve accident initiators and/or subsequent system failures that defeat the redundancy available in systems to mitigate potential accidents.



- CDF does not decrease for newer BWR designs.
- CDF variability primarily results from differences in individual plant design and operation, especially in support systems, modeling assumptions, and data values (particularly human error probabilities).
- Isolation condensers are estimated to be more reliable than RCIC system contributing to a smaller transient with loss of injection contribution for older BWRs.
- Failure to depressurize the reactor, given a transient initiator, for low pressure coolant injection is a dominant contributor to CDF.
- Harsh environments in the containment and reactor building following loss of DHR results in failure of coolant injection systems in many BWRs, but is not as significant for newer BWRs.
- Containment venting helps reduce the CDF for many BWRs.
- ATWS accident sequences are dominated by human errors.

Figure 3.2 Reported IPE CDFs and key perspectives for BWRs.

Table 3.2 Summary of CDF perspectives for BWRs.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
SBO accidents		
Important for most BWRs, regardless of plant group	Availability of AC-independent systems (i.e., high-pressure coolant injection (HPCI) system, diesel-driven firewater system, reactor core isolation cooling (RCIC) interface with suppression pool) Isolation condenser capacity Battery life DC dependency for diesel generator startup Service water (SW) system design and heating, ventilating and conditioning (HVAC) dependency AC power reliability (number of diesel generators, cross-tie capability between buses and units, diverse AC power sources)	Improved operator training Improved DC reliability (cross-tie of buses, portable power supply to charger) Increased DC load shedding Increased AC reliability (alternative AC power source, cross-tie of buses) Increased availability of AC-independent injection systems (diesel-driven firewater, reconfiguring RCIC dependencies)
Transients with loss-of-injection accidents		
Relatively unimportant at BWRs 1/2/3 plants Important for most BWR 3/4 and 5/6 plants	Injection system dependencies on support systems defeats redundancy Availability and redundancy of injection systems (e.g., control rod drive (CRD), motor- driven feedwater pumps, SW cross-tie to residual heat removal (RHR), firewater system) Failure to depressurize influenced by operator direction to inhibit the automatic depressurization system (ADS)	Procedural and hardware enhancements to use alternative systems for injection (e.g., CRD) Increased emphasis in operator training and/or procedure modification on depressurization Improve system reliability by modifying system surveillance procedure to include testing of other system equipment (e.g., pump suction line from suppression pool for the HPCS system), by revising maintenance procedure to reduce common cause failures Enhance procedures to respond to loss of HVAC in ECCS rooms
ATWS accidents		
Relatively unimportant for most BWRs, regardless of plant group	Operator failure to initiate standby liquid control (SLC) in timely manner, to maintain main steam isolation valves (MSIVs) open, to control vessel level, and/or to maintain pressure control Use of alternative means of injecting boron Use of HPCS for mitigation	Improved operator training Installation of automatic inhibit for ADS Installation of an alternative boron injection capability

Table 3.2 Summary of CDF perspectives for BWRs.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
Transients with loss of decay heat removal (DHR) accidents		
Important for most BWRs, regardless of plant group	Limited analysis to support success criteria; no credit for DHR system (e.g., venting) Dependency of support systems for DHR Net positive suction head (NPSH) problems with ECCS on suppression pool Availability of injection system located outside containment and reactor building Capability of ECCS to pump saturated water Use of reactor water cleanup (RWCU) system as alternative DHR system	Improved operator training Increased reliability of equipment (i.e., hardware modifications to replace pump motors with air-cooled motors) Use of alternative systems or alignment for coolant injection (e.g., align low-pressure coolant injection (LPCI) pump to condensate storage tank (CST) Increase availability of injection systems (replenish CST, increase exhaust pressure trip setpoint on RCIC turbine) Revise isolation logic for plant SW and IA Provide control room temperature indicator for rooms containing SW pumps
LOCAs		
Relatively unimportant for all but one BWR plant	High redundancy and diversity in coolant injection systems	Hardware modification for pipe whip constraints (replace torus suction strainers to reduce probability of clogging) Expand environmental qualification program
Interfacing systems LOCAs (ISLOCAs)		
Not important for BWR plants	Compartmentalization and separation of equipment	None identified
Internal flood accidents		
Relatively unimportant at most BWRs, regardless of plant group	Plant layout (separation of mitigating system components and compartmentalization)	Protection of injection system power sources from spray effects Periodic inspection of cooling water pipes Enhance procedures and training to respond to floods, including isolation of the flood source

SBOs involve a loss of both off-site and emergency on-site power sources (primarily diesel generators, but a few plants also have gas turbine generators) that cause the failure of most available mitigating systems except those that do not rely on AC power. The definition of SBO for BWR 5/6s does not include failure of the diesel generator supplying the high pressure core spray (HPCS) system. Most accident sequences contributing to transients with loss of coolant injection involve failure of high-pressure injection systems (such as the feedwater, reactor core isolation cooling, high pressure coolant injection, and HPCS systems) with a subsequent failure to depressurize the plant for injection by low-pressure injection systems. The failure to depressurize effectively defeats a large part of the redundancy in the coolant injection systems. Support system failures (e.g., loss of cooling water systems, AC or DC buses, or IA) impact many of the available accident mitigating systems and contribute to the importance of this accident category, as well as transients with loss of decay heat removal (DHR). In all loss of DHR sequences involving transients or other initiators, redundancy in mitigating systems can be lost as a result of harsh environments in the containment before containment failure or in supporting structures following containment venting or failure.

Lesser contributions from LOCAs, ATWS, and internal flooding are generally reported for all BWRs. However, a few BWRs did report significant contributions from these accident categories. These three accident categories are not important contributors, primarily because they involve low-frequency initiating events. Although ISLOCAs are potentially risk-significant contributors since the containment is bypassed, none of the licensees reported significant CDFs from this accident category, primarily because it involves low-frequency initiating events.

Sections 3.2.1 through 3.2.3 discuss important factors that impact the CDF contributions from these accident categories for each BWR plant group, based on information from the original IPE submittals. Many of these factors are the same for each plant group. However, there are factors worth highlighting that explain some of the differences across the BWR groups. For example, the staff noted that some of the accident class frequencies for the BWR 1/2/3 plant group are generally lower than for the other two BWR plant groups, partially because isolation condensers appear to be more reliable than the RCIC systems that replaced them in later BWR models. RCIC systems have more possible failure modes related to protective trip signals, ventilation failures, and pump operability requirements. Some of these failure modes are only prevalent in the BWR 5/6 IPEs; this partially accounts for the higher SBO CDFs for this group. However, some licensees with isolation condenser plants generally ignored the potential for recirculation pump seal failures, which would effectively defeat the use of the isolation condensers. Finally, the BWR 5/6 plants had lower contributions on average from sequences involving loss of high-pressure injection systems, coupled with failure to depressurize the vessel in order to use low-pressure injection, than did BWR 3/4s. This is partially due to the fact that the HPCS system in the BWR 5/6 plants tends to be more reliable than the HPCI system in the BWR 3/4 plants.

3.2.1 BWR 1/2/3 Perspectives

Six units in the BWR 1/2/3 plant group have isolation condensers. Big Rock Point is the only BWR 1 unit in this group. Two of the units (Oyster Creek and Nine Mile Point 1) are BWR 2s while the other three (Dresden 2&3 and Millstone 1) are BWR 3s. Big Rock Point is also unique in that it is housed in a dry containment; the other five plants in this group are housed within Mark I containments. These plants are listed in Table 3.3.

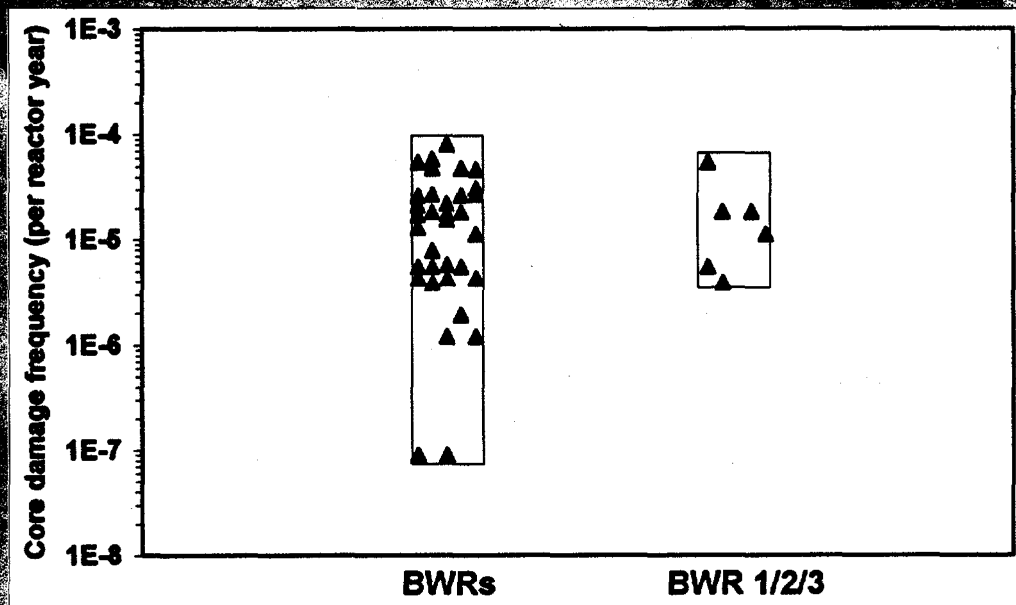
Table 3.3 Plants (per IPE submittal) in the BWR 1/2/3 group.

Big Rock Point	Dresden 2&3	
Millstone 1	Nine Mile Point 1	Oyster Creek

The following accident classes dominate the CDF from the original IPE submittals for at least one of the plants:

- SBO — the loss of all off-site and on-site power
- transients with loss of DHR
- LOCAs

These three accident classes dominate the CDFs at one or more plants because of plant-specific features or modeling assumptions that are unique to those plants. Although some licensees reported significant CDFs from accident classes involving ATWS sequences and transients with loss of coolant injection, these accident classes contribute to a lesser extent to the overall plant CDFs. Only two licensees identified small contributions from internal flood events and ISLOCAs; the remainder of the licensees' findings showed negligible contributions for these two accident classes. As indicated in Figure 3.3, the CDFs for the six BWR 1/2/3 units are within the same range as a majority of the BWRs. The importance of specific accident classes varied significantly from plant to plant, as shown in Figure 3.4.



- The CDF for each plant in this group is dominated by one accident class that varies depending on plant-specific features and modeling assumptions.
- Support system failures are important for this group since they cause the failure of multiple DHR and/or coolant injection systems.
- SORVs are important since they defeat the use of the isolation condenser.
- The availability of motor-driven feedwater pumps and isolation condensers reduces the importance of transients with loss of injection as compared to other BWR groups.
- LOCA is important for Big Rock Point because of its ECCS design similarity to PWRs.

Figure 3.3 Reported IPE CDFs and key perspectives for the BWR 1/2/3 plant group.

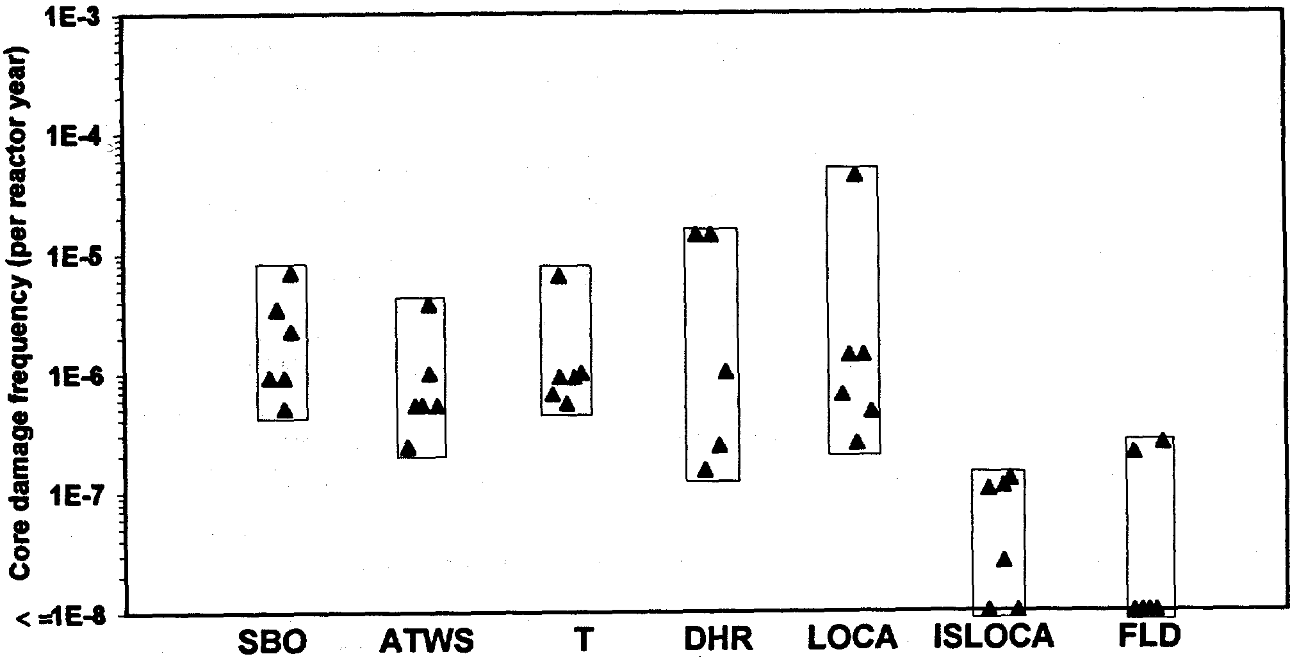


Figure 3.4 Reported IPE accident sequence CDFs for BWR 1/2/3 plants.

The variability in the results for these accident classes is primarily attributable to plant-specific features with some contribution from modeling assumptions and differences in data values (including human error probabilities). Table 3.4 summarizes the key factors influencing the original IPE results and common plant improvements being considered by the licensees to address weaknesses for each important accident class.

Table 3.4 Summary of CDF perspectives for the BWR 1/2/3 plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
SBO accidents		
Dominant contributors for half of the plants in the BWR 1/2/3 group	Availability of AC-independent systems (i.e., HPCI system, diesel-driven firewater system) Isolation condenser capacity Battery life DC dependency for transferring power following scram AC power reliability (number of diesel generators, cross-tie capability between buses and units, diverse AC power sources) Ability to withstand a load rejection and supply house loads from the main generator upon loss of off-site power	Improve isolation condenser Improve operator training Improve DC reliability (cross-tie of buses, portable power supply to charger) Increase DC load shedding Increase AC reliability (alternative AC power source, cross-tie of buses) Increase availability of AC-independent injection systems (diesel-driven firewater, reconfiguring RCIC dependencies) Replace RCP seal material

Table 3.4 Summary of CDF perspectives for the BWR 1/2/3 plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
	Transients with loss of DHR accidents	
Dominant contributors for two plants in the BWR 1/2/3 group	Limited analysis to support success criteria; no credit for DHR system (e.g., drywell sprays) Dependency of DHR systems on support systems Loss of NPSH for ECCS pumps when suppression pool temperatures increase Switchover of ECCS from injection to recirculation mode (applicable only at Big Rock Point)	Install hard-pipe vent Increase reliability of equipment (i.e., isolation condenser modifications; replacement of ECCS pump motors with air-cooled motors) Use alternative systems for DHR (e.g., portable diesel fire pump to supply isolation condenser)
	LOCAs	
Important at Big Rock Point	High redundancy and diversity in coolant injection systems Switchover of ECCS from injection to recirculation mode (applicable only at Big Rock Point)	Implement hardware modification (pipe whip constraints, replacement of torus suction strainers to preclude clogging) Implement environmental qualification program
	ATWS accidents	
Relatively unimportant for all plants in the BWR 1/2/3 group	Operator failure to initiate SLC in timely manner, to maintain MSIVs open, to control vessel level, and/or to maintain pressure control Turbine bypass capacity Boron injection system design	None identified
	Transients with loss of injection accidents	
Relatively unimportant at most BWR 1/2/3 plants	Diversity in available coolant injection systems Injection system dependencies on support systems defeat redundancy Availability of alternative injection systems (e.g., CRD, motor-driven feedwater pumps, firewater system) Failure to depressurize influenced by operator direction to inhibit the ADS	Improve calibration of reactor pressure vessel (RPV) pressure permissive
	Internal flood accidents	
Relatively unimportant at most BWR 1/2/3 plants	Plant layout (separation of mitigating system components and compartmentalization)	None identified
	ISLOCAs	
Not important for the BWR 1/2/3 group	Compartmentalization and separation of equipment	None identified

SBO sequences are the dominant contributors to CDF for three of the six plants in the BWR 1/2/3 group. SBO accidents involve an initial loss of off-site power (LOSP) followed by failure of the emergency on-site AC power sources. Failure of AC power sources results in failure of multiple systems, leaving only the isolation condensers and the steam-driven HPCI system available to mitigate this type of accident (Dresden 2&3 are the only plants in this group that have a HPCI system.) The ability of the isolation condensers to mitigate this type of accident is defeated if a relief valve sticks open (an important SBO scenario for this group).

Generally, plant design and operational features have a larger impact on the SBO CDF than modeling assumptions. The most influential plant features and modeling assumptions are as follows:

- Differences in AC power sources — All of the plants in the BWR 1/2/3 group have two emergency on-site power sources, and half of the plants are configured with two diesel generators. Millstone 1 has one diesel generator, but also has an air-cooled gas turbine generator that is more unreliable than the plant's diesel generator. The higher turbine generator unreliability contributes to the higher CDF calculated for Millstone 1. Dresden 2&3 each have only one dedicated diesel generator per unit, but share a swing diesel generator. All three diesel generators must fail in order to cause a dual-unit SBO if off-site power is lost at both units. Failure of only two diesel generators (the dedicated diesel and the swing diesel generator) results in a single-unit SBO if off-site power is only lost for one unit. Since Dresden 2&3 also model cross-tieing unit buses for a single-unit LOSP event, the SBO contribution from such events is smaller than for the single-unit plants in the group. The CDF from dual-unit LOSP events is also smaller than the single-unit LOSP CDFs since the dual-unit LOSP frequency is less than the single-unit frequency used for the other plants.

- Availability of a HPCI system — Various SBO sequences (also involving functional failure of the isolation condensers) require that coolant makeup be provided to the reactor vessel. These sequences can involve failures of isolation condenser hardware, failure to provide water to the shell side of the isolation condensers, and loss of primary system coolant through SORVs or pump seal LOCAs (modeled at only one plant in the group). A steam-driven HPCI system (available only at Dresden 2&3) can be used during these types of SBO sequences to inject coolant for a substantial period of time until battery depletion results in loss of system control. With the exception of Big Rock Point, plants without a HPCI system have larger SBO contributions because of their inability to mitigate these sequences.

- Load rejection capability and isolation condenser capacities — Big Rock Point is designed with a 100% turbine bypass capacity that will open upon a LOSP allowing a load rejection without a reactor scram. Continued reactor power operation is thus possible following a LOSP with the main generator operational and providing sufficient power to house loads that include systems required for power operation and power to accident mitigating systems. This feature effectively reduces the frequency of a SBO at the plant. Although Millstone 1 also has a 100% turbine bypass capacity, a load rejection capability similar to that at Big Rock Point could not be identified. Big Rock Point also has a relatively large isolation condenser shell-side capacity that allows heat to be removed for up to 6 hours before makeup coolant is required. The other plants in the group typically require shell-side makeup within 20 to 45 minutes. The larger isolation condenser capacity (partially attributed to the smaller power rating of Big Rock Point) increases the opportunity to recover AC power before all cooling is lost.

- DC power failure — DC power is required during a SBO at plants in this group for operation of the isolation condensers, shell-side coolant makeup for the condensers, HPCI operation, and operation of safety relief valves

(SRVs) to depressurize the vessel for firewater injection (which was modeled for most plants in this group). Battery depletion is identified as an important long-term SBO failure mode at all plants in this group. The battery depletion times vary from 2 to 12 hours, depending on the ability to shed loads, and partially contribute to the spread in the results. At Oyster Creek, DC bus failures were also found to contribute significantly to a SBO since the emergency DC buses are required to transfer power from the auxiliary transformers to the startup transformers following any reactor trip, and also to close breakers for emergency-related components. (The diesel generators can start because their control power is provided by separate batteries.) Thus, a reactor trip concurrent with failure of the two emergency DC buses results in a SBO and the failure of required mitigating systems at Oyster Creek.

Nearly all of the licensees in this group plan or have already implemented plant improvements to address the factors influencing SBO scenarios. These improvements primarily involve hardware modifications directed at increasing AC system reliability or the availability of the isolation condensers. However, several licensees also identified improvements in general operator training to cope with potential accident scenarios. The improvements in AC system reliability identified by the licensees include a reliability analysis to identify potential gas turbine generator improvements, replacing the governor on the gas turbine generator, upgrading or establishing new alternative off-site power sources, and establishing the ability to cross-tie to buses at a sister plant. The licensees also identified DC power improvements, including establishing load shedding procedures and providing portable AC generators to power battery chargers upon loss of normal AC power. Isolation condenser improvements include implementing valve modifications, procuring a portable diesel fire pump and implementing a related procedure for providing makeup coolant to the isolation condenser, and developing a procedure to operate the isolation condenser during an extended SBO. The manual alignment of the firewater system for coolant injection through the core spray (CS) system has been implemented at Oyster Creek and is credited in the IPE. New reactor recirculation pump seal materials have been implemented at Nine Mile Point 1 and are credited in the IPE. In general, the IPEs reflected only a few of the suggested plant improvements, and quantitative evaluations of these improvements were not provided. However, most of the improvements were listed as being implemented.

Transients with loss of DHR are the dominant contributors for two reactors in the BWR 1/2/3 group. Transient sequences with loss of DHR involve accidents in which coolant injection succeeds, but DHR fails. In this situation, direct heat removal from the reactor vessel fails, resulting in relief valve openings that carry steam generated in the vessel to the suppression pool. The CHR systems fail and the suppression pool in BWR 2/3s heats up, leading to containment pressurization and failure if venting is not initiated in time. Coolant injection will also fail either because of high suppression pool temperatures that lead to a loss of NPSH for emergency coolant pumps, or because of adverse environments created in the containment or the reactor building following containment venting or failure. In the Big Rock Point IPE, there is no reported DHR contribution because decay heat is removed through the recirculation of emergency coolant. Thus, loss of DHR sequences are incorporated into loss of injection sequences in the Big Rock Point IPE.

The key factors affecting the CDF from transient sequences with loss of DHR include both plant-specific design and operating features and modeling assumptions:

- Credit for DHR systems — Transients with loss of DHR are the dominant sequences reported in the Dresden 2&3 IPEs. These sequences are important, in part because not all DHR systems are credited or available for preventing core damage in the IPE analysis. The IPE only credits use of the isolation condenser and suppression pool cooling. The licensee did not take credit for use of the shutdown cooling system for direct heat removal from the vessel because it is not a major DHR system during upset conditions. Torus spray has too little flow to be of benefit and drywell spray was assumed to be of little benefit if suppression pool cooling fails. The main condenser is not credited because the licensee states that many unspecified events will result in its unavailability. Containment venting is not credited for preventing core damage since it is not initiated until after adequate ECCS pump NPSH is lost. However, use of containment venting and sprays was credited for preventing containment failure after core damage occurs. Other BWR 2/3 licensees in this group typically credit similar systems in their IPEs and calculate significantly lower contributions from this accident class.

- Support system failures — Systems such as AC and DC power, SW, and IA support the operation of multiple DHR systems in the plants within this group. Initiating events involving the failure of these systems are the dominant contributors to the transient with loss of DHR accident class. Because the support system design is highly plant-specific, the impact of their failure varies from plant to plant. The greatest impact on the plant CDF is reported in the Dresden 2&3 IPEs, where a DC bus initiating event contributes (along with the limited DHR capability discussed above) to making this the dominant accident class.

Most licensees in this group identified potential improvements that would increase the ability to remove decay heat. These include both DHR system modifications and improvements that would help to ensure continued coolant injection system operation under harsh containment environments. The DHR system improvements involve implementing hardware modifications to improve the reliability of the isolation condensers, establishing procedures and installing hardware to perform containment venting (which was credited in most IPEs), replacing drywell spray valve motor operators with environmentally qualified operators, and adding guidance in emergency operating procedures for the use of containment spray. Improvements to coolant injection systems include implementing a procedure to align LPCI or CS pumps to the CST when suppression pool cooling cannot be established, revising surveillance procedures to verify the operability of lubricating oil coolers for LPCI and CS pump motor bearings, and replacing the motors on LPCI and CS pumps with air-cooled motors.

With the exception of the Big Rock Point BWR 1, LOCAs are minor contributors to the CDF in the BWR 1/2/3 group. LOCAs are characterized as important in the Big Rock Point IPE because of the unique design of the CS system and a pessimistic modeling assumption. Following a LOCA, the CS system provides injection into the vessel until the containment water level reaches a prescribed elevation that alerts the operator to switch the suction of the system to the containment sump. In this mode of operation, the water is cooled through heat exchangers before being injected back into the vessel. If recirculation flow cannot be established, emergency operating procedures require continued injection from outside sources until a predetermined level is reached to ensure that the water head will not cause structural failure of the containment. Injection is terminated at that time, leading to a boiloff of the reactor vessel fluid and eventual core damage. Even though this scenario would require days to develop and thus would allow substantial recovery potential, no credit for repair activities is allowed in the analysis. A similar situation occurs for small-break LOCAs with CS failure so that injection must be provided from external sources by the feedwater, condensate, or firewater systems.

The variation in LOCA contributions from the other plants in this group is primarily attributed to differences in available coolant injection and DHR systems. All of the plants in this group have CS systems and motor-driven feedwater systems. Millstone 1 and Nine Mile Point 1 have feedwater coolant injection systems. Three plants (Dresden 2&3 and Millstone 1) have LPCI systems. In addition, some licensees credit the use of alternative injection systems, such as firewater or raw water systems. These variations in coolant injection systems affect the types of sequences that dominate this accident class at each plant, as well as the frequencies of those sequences. In addition, the variation in DHR systems also affect the results. This was most notable at Dresden 2&3, where LOCA sequences involving loss of DHR capability were the most important contributors to this accident class, primarily because of the limited DHR capability modeled in the IPE.

One IPE for this plant group identified specific improvements to address LOCAs. These improvements include installing pipe whip constraints, implementing an environmental qualification program to reduce the failure of required equipment, and replacing the torus suction strainers for the ECCS pumps to preclude clogging. In addition, some of the modifications previously identified for loss of DHR accidents also apply for LOCAs.

ATWS sequences are not generally major contributors to the CDF for the BWR 1/2/3 group. ATWS sequences involve a transient followed by failure to terminate the nuclear chain reaction by inserting the control rods. Power generation continues at levels that can exceed the available coolant injection and DHR system capacities, leading to core damage. An ATWS sequence can be mitigated, however, by injecting boron using the SLC system, controlling coolant injection, and subsequently removing the heat generated at the reduced power level.

Among the BWR 1/2/3 isolation condenser designs, the dominant contributors to the ATWS accident class are quite similar for most plants in this group. The dominant ATWS sequences for these plants involve transient initiators with failure to initiate SLC. The Oyster Creek and Nine Mile Point 1 licensees also identified sequences involving failure to inhibit ADS and to control vessel level as important, while the Dresden 2&3 licensees identified failure to trip the recirculation pumps as a dominant contributor. The important contributors to most of these sequences tend to be human errors involving failure to initiate SLC in a timely manner, failure to maintain MSIVs open, failure to control the vessel water level (including lowering the level to reduce power and avoiding low-pressure system injection) and failure to maintain proper pressure control by inhibiting the ADS. Variability in these human error probabilities results in some variability in the individual ATWS sequence frequencies. Generally, those plants with higher human error probabilities for failure to initiate boron injection have higher CDFs for this accident class. It is also interesting to note that the two plants with 100% turbine bypass capacity have the highest CDFs in this accident class, suggesting that operator error probabilities have more influence on the results than this capability since it is not available for all transients.

The licensees did not identify any specific plant improvements to address ATWS sequences in the IPEs for this plant group. However, improvements for transient sequences with loss of DHR are also generally applicable for ATWS sequences.

Transient sequences are not significant contributors at any plant in the BWR 1/2/3 group. This accident class involves transient sequences with failure of coolant injection. Variations in the CDFs for this accident class and whether the dominant transient sequences involved loss of injection with the vessel at high pressure or low pressure depend somewhat on the injection systems available at each plant. However, because these plants all have motor-driven feedwater pumps, the dominant sequences tend to involve accident initiators that cause feedwater to fail. Examples of these initiators include the loss of feedwater itself or support system initiating events (e.g., loss of AC or DC power, IA, or cooling water systems) that, depending upon the plant design, can cause feedwater and other available coolant injection systems to fail.

The dominant CDF in this accident group, reported for Big Rock Point, relates to the dependence of the mitigating system on the IA system and the use of a pessimistic modeling assumption. Loss of IA results in loss of the main condenser, as well as loss of makeup coolant to the emergency condenser and to the hotwell from the CST (which is required for continued injection from the condensate and feedwater systems). Without IA, makeup coolant for the emergency condenser must be provided by the firewater system. The analysis does not account for the additional use of an alternative shutdown portable pump that can be used to maintain either the emergency condenser shell inventory or firewater makeup to the hotwell.

The licensee did not identify any additional plant improvements to specifically address transients with loss of injection. However, operator training and some injection system improvements previously identified to address other accident classes also apply here.

Internal flooding events are not significant contributors at any of the plants in the BWR 1/2/3 group. Internal flooding events involve rupture of water lines or operator errors that result in a release of water that can directly cause the failure of required mitigating systems (e.g., through loss of cooling) and/or the failure of other mitigating systems as a result of submergence or spraying of required components. The most important factor determining the importance of flooding to the plant CDF is the plant layout. Separation and compartmentalization of mitigating system components reduces the impact of internal flood initiators. Internal flooding events are not dominant contributors at any plant because no internal flood initiator was identified that would cause the complete failure of all systems needed to mitigate a flood-induced transient without additional random failures. However, a few licensees did identify internal flooding sequences that involve feedwater or SW system breaks that directly affect equipment through loss of cooling and through flood impacts on other mitigating equipment.

None of the licensees in this group identified plant improvements related to mitigating an internal flood event.

ISLOCAs are not dominant contributors to CDF for the BWR 1/2/3 plants. The low CDF contribution results from the low frequency of the LOCA initiator combined with its negligible-to-minimal impact on other systems. This low impact results from the compartmentalization and separation of the equipment.

3.2.2 BWR 3/4 Perspectives

The BWR 3/4 group includes 21 units (15 IPE submittals) that have RCIC systems[3.4] as listed in Table 3.5. All of these units are housed in Mark I containments except Limerick 1&2 and Susquehanna 1&2, which are housed in Mark II containments.

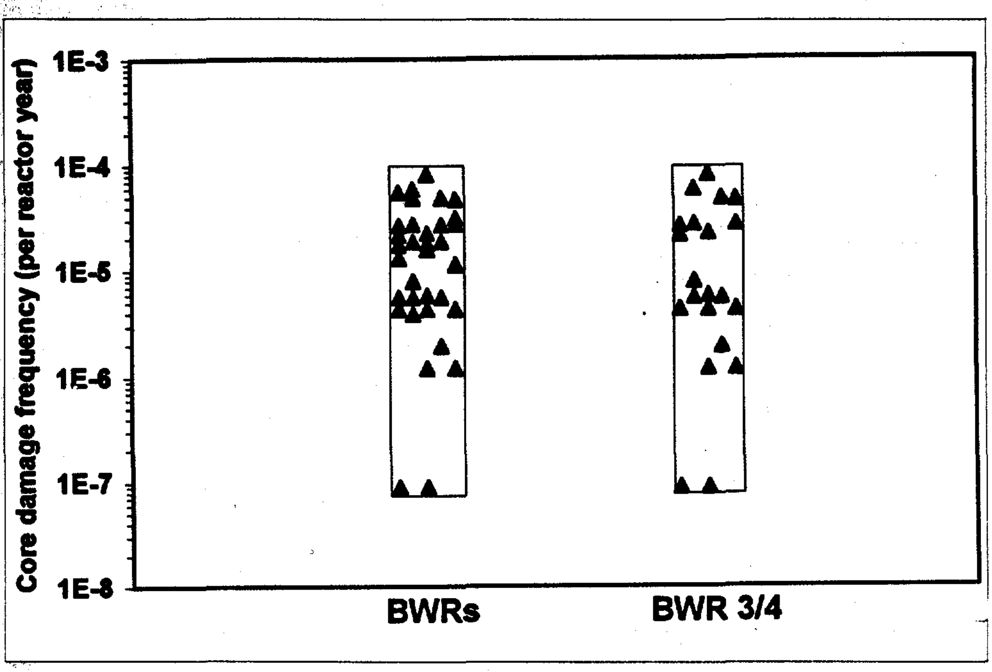
Table 3.5 Plants (per IPE submittal) in the BWR 3/4 group.

Browns Ferry 2	Brunswick 1&2	Cooper	Duane Arnold	Fermi 2
FitzPatrick	Hatch 1&2	Hope Creek	Limerick 1&2	Monticello
Peach Bottom 2&3	Pilgrim 1	Quad Cities 1&2	Susquehanna 1&2	Vermont Yankee

The CDFs reported in the original BWR 3/4 plant IPE submittals have a larger variation than other BWR plant groups, as illustrated in Figure 3.5. Several plants have significantly lower CDFs compared to the majority of the BWR 3/4s, and Susquehanna 1&2 represent outliers[3.5] relative to the other plants.

[3.4]The BWR 3 plants in this group have RCIC systems, while the BWR 3 plants considered in Section 3.2.1 have isolation condensers.

[3.5]Susequehanna 1&2 are outliers primarily due to the approach used by the licensee in performing the IPE. Contributing factors are discussed in the subsequent sections of this report.



- Susquehanna 1&2 have significantly lower CDFs than other plants, in part, because it credits use of additional systems and actions not generally credited in other IPEs and assumes common-cause failures to be improbable.
- Transients with loss of high-pressure injection and failure to depressurize the vessel (for use of low-pressure injection) is the dominant accident class for most plants in this group.
- Human error probabilities for the operator failing to depressurize the reactor varies by four orders of magnitude.
- SBO contribution generally decreases if the plant has more diesel generators, has longer battery depletion times, or credits firewater injection.
- Support system failures are dominant contributors to CDF.

Figure 3.5 Reported IPE CDFs and key perspectives for the BWR 3/4 plant group.

The importance of specific accident classes to the total plant CDFs varies significantly from plant to plant; however, the following accident classes are important for many of these plants (see Figure 3.6):

- SBO — loss of all off-site and on-site AC power
- transients with loss of coolant injection
- transients with loss of DHR

In general, these three accident classes are important contributors to CDF because they involve initiating events and/or subsequent system failures that defeat the redundancy in systems available to mitigate potential accidents. Lesser contributions were identified for the group on average from accident classes involving low-frequency initiating events such as ATWS, LOCAs, and internal floods. However, a few IPEs reported some of these accident classes as being important. Although ISLOCAs are potentially important risk contributors (since the containment is bypassed) none of the licensees report significant CDFs or radionuclide releases from ISLOCAs because this accident class involves low-frequency initiating events.

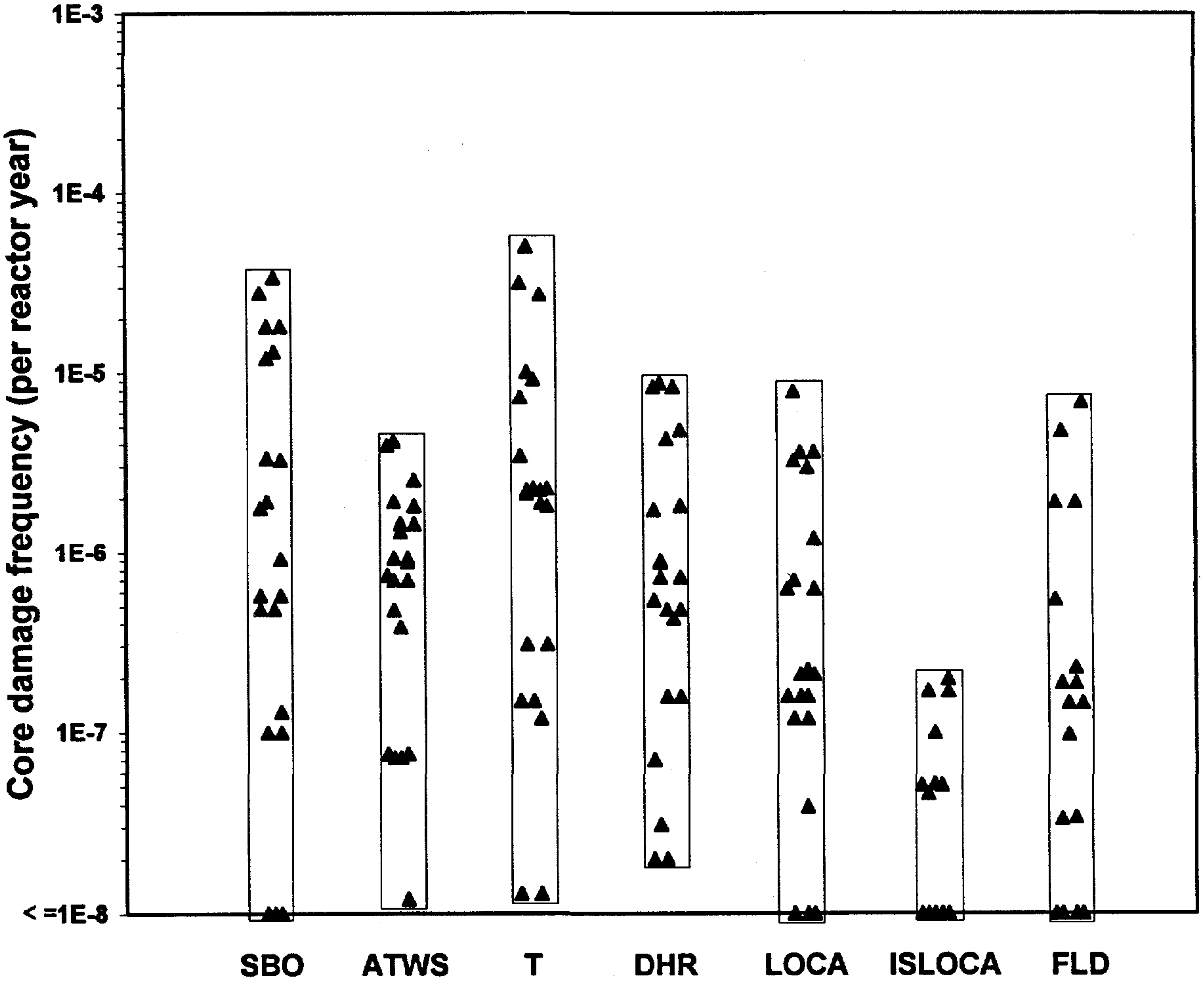


Figure 3.6 Reported IPE accident sequence CDFs for BWR 3/4 plants with RCIC.

The variation in the original IPE results is attributed to many factors including plant-specific design features, modeling assumptions, and variation in data including the probability of operator errors. Table 3.6 summarizes these factors and the improvements being considered by the licensees to address weaknesses, as discussed below for each important accident class. Revised assessments by the licensees may result in different perspectives than those indicated in Table 3.6 and discussed in the remainder of this section.

Table 3.6 Summary of CDF perspectives for the BWR 3/4 plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
	SBO accidents	
Important for most BWRs in this plant group	Availability of AC-independent systems (i.e., HPCI system, diesel-driven firewater system, RCIC interface with suppression pool) Battery life DC dependency for diesel generator startup SW system design and HVAC dependency AC power reliability (number of diesel generators, cross-tie capability between buses and units, diverse AC power sources)	Improve operator training Improve DC reliability (cross-tie of buses, portable power supply to charger) Increase DC load shedding Increase AC reliability (alternative AC power source, cross-tie of buses) Increase availability of AC-independent injection systems (diesel-driven firewater, reconfiguring RCIC dependencies)
	Transients with loss of injection accidents	
Important for most BWR 3/4 plants	Injection system dependencies on support systems defeat redundancy Availability of alternative injection systems (e.g., CRD, motor-driven feedwater pumps, SW cross-tie to RHR, firewater system) Failure to depressurize influenced by operator direction to inhibit the ADS	Implement procedural and hardware enhancements to use alternative systems for injection (e.g., CRD system) Increase emphasis on operator training and/or procedure modification for depressurization
	ATWS accidents	
Significant contributors for a few BWR 3/4 plants	Operator failure to initiate SLC in timely manner, to maintain MSIVs open, to control vessel level, and/or to maintain pressure control Use of alternative means of injecting boron	Revise procedures for responding to mechanically bound control rods Improve operator training
	LOCAs	
Not important for BWR 3/4 plants	High redundancy and diversity in coolant injection systems Depressurization requirement for LPCI during medium LOCAs	Install bypass switch for low-vessel pressure permissive logic required to open LPCI and CS injection valves

Table 3.6 Summary of CDF perspectives for the BWR 3/4 plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
	Transients with loss of DHR accidents	
Important for many plants in this BWR plant group	Limited analysis to support success criteria; no credit for DHR system (e.g., venting) Dependency of DHR systems on support systems Loss of NPSH for ECCS pumps when suppression pool temperatures increase Availability of injection system located outside containment and reactor building and ability to operate following containment failure Use of RWCU system as an alternative DHR system	Improve operator training Install hard-pipe vent Increase availability of injection systems (replenish CST, increase exhaust pressure trip setpoint on RCIC turbine, use of alternative systems)
	Internal flood accidents	
Relatively unimportant at most BWR 3/4 plants	Plant layout (separation and compartmentalization of mitigating system components	Implement protection of injection system power sources from spray effects
	ISLOCAs	
Not important for BWR 3/4 plants	Compartmentalization and separation of equipment	None identified

SBO accidents are important contributors to CDF for most of the plants in this group. SBO accidents involve an initial LOSP followed by failure of the emergency on-site AC power sources. The failure of AC power sources results in failure of multiple mitigating systems, leaving only steam-driven systems (such as RCIC and HPCI) available for coolant injection.

Generally, plant design and operational features have a larger impact on the SBO CDF than do modeling characteristics, but no single factor dominates. Combinations of contributors are usually important, and those combinations vary from plant to plant. With that in mind, the most influential plant features and modeling characteristics are identified as follows:

- Number of emergency AC power sources — The number of emergency diesel generators (usually from two to four per unit) directly affects the reliability of the emergency AC power system. Generally, the higher the number of emergency diesel generators, the lower the SBO contribution. However, diesel generator reliability can also be affected by plant-specific features (such as the lower reliability resulting from the diesel generator cooling water system alignment at Hope Creek) or modeling assumptions (such as the higher diesel generator reliability in the Susquehanna IPE associated with the elimination of common cause failures). The availability of additional and diverse AC power sources (such as the gas turbine generator at Fermi 2) or a separate off-site power source in addition to the normal grid connection (such as exists at Pilgrim and Vermont Yankee) reduces the SBO contributions at those units.

- Battery depletion time — When AC power is lost, the only available injection systems are turbine-driven systems (HPCI and RCIC) or, for some plants, diesel-driven firewater. Battery power is needed to provide control for HPCI and RCIC, or to maintain the ADS valves open so that the low-pressure firewater system can be used. Thus, when the batteries are depleted, all cooling is lost and core damage ensues. Battery depletion times range from 2 hours at Brunswick 1&2 to 14 hours for Pilgrim, with the longer times reflecting plants making extensive use of load shedding. The contribution from SBO accidents is generally lower for units with longer battery depletion times since the probability of recovering AC power and AC-powered mitigating systems increases with time. In fact, units with battery depletion times greater than four hours have significantly lower SBO CDFs than plants with battery depletion times of four hours or less.

- Use of diesel-driven firewater — Some units use diesel-driven firewater systems as a diverse means of supplying coolant injection when HPCI and RCIC have failed. The vessel must be depressurized and maintained at low pressure in order for firewater to be used. Further, this nonstandard use of the firewater system requires the availability of piping connections and power to certain valves, along with appropriate procedures. The ability and time required to inject coolant water using the firewater system thus varies from unit to unit, but for most IPEs, firewater injection is not considered to be feasible for sequences with early failure of RCIC and HPCI. Diesel-driven firewater is most important for sequences with delayed failure of HPCI and RCIC because there is sufficient time to configure the firewater system for injection in those cases.

Nearly all licensees in this group have planned or implemented plant improvements (both procedural improvements and hardware modifications or additions) to address the factors influencing SBO scenarios. These improvements include increasing AC system reliability by establishing dedicated lines to alternative off-site power sources, implementing bus cross-tie procedures, or installing additional diesel generators. DC power improvements (such as establishing load shedding procedures and aligning small diesel generators to supply AC power to the station battery chargers) were also identified. Increasing the reliability of providing coolant injection during a SBO is also a major area for suggested plant improvements, including the addition of procedures and hardware to allow for firewater injection into the vessel, reconfiguring RCIC pump room enclosure fans from AC to DC power sources, and establishing loss-of-ventilation procedures for RCIC and HPCI pump rooms.

The IPE results generally do not reflect these plant improvements, and very few have evaluated the quantitative impact on the CDF. However, a few licensees did perform sensitivity studies regarding some of these improvements. For example, licensees estimated that increasing the battery depletion time from four to six hours would reduce the SBO CDF by 30% at Monticello, and a 40% reduction was obtained for Cooper when the battery depletion time was increased from four to eight hours. The impact of having a gas turbine generator was evaluated in the Fermi 2 IPE. Without the gas turbine generator, the SBO CDF would increase by more than an order of magnitude. In the Monticello IPE, the licensee estimated that addition of an auxiliary diesel generator to supply power to the station battery chargers would reduce the SBO CDF by 70%.

Transients with loss of coolant injection are important contributors to the CDF for most plants in this group. This accident class is dominated by sequences involving loss of the relatively few high-pressure injection systems (typically feedwater, HPCI, and RCIC), and failure to depressurize the vessel so that the multiple low-pressure injection systems can be used. Transients with loss of high-pressure injection, successful vessel depressurization, and failure of low-pressure injection systems are of lesser importance because of the significant redundancy in low-pressure injection systems. Important transient initiating events are those that cause feedwater failure such as LOSP, loss of feedwater, and MSIV closure. Loss of DC buses is also an important initiating event at some plants, because DC power is needed to provide control for HPCI and RCIC, and to maintain the ADS valves open so that the vessel can be depressurized.

For this group of accidents, there are two issues that are critical to the CDF. One issue involves plant-specific design characteristics, while the other issue involves plant operating procedures and training along with modeling assumptions.

- Availability of alternative high-pressure injection systems — The availability of high-pressure injection systems (in addition to feedwater, RCIC, and HPCI) reduces the contribution of this accident class. Several licensees (Brunswick, Cooper, and Susquehanna) use plant-specific calculations to show that the CRD hydraulic system in the enhanced flow mode can provide sufficient coolant injection to maintain core cooling immediately after a reactor scram. The Quad Cities licensees credit a unique safe shutdown injection system that helps to reduce the importance of all loss of injection and loss of DHR sequences. Several licensees with motor-driven feedwater pumps (Monticello, Fermi, and Vermont Yankee) also calculate lower contributions since, unlike steam-driven feedwater pumps, these pumps can continue to operate during transients with MSIV closure.

- Operator failure to depressurize — Current operating procedures at most plants direct the operators to inhibit the ADS for most transients. If high-pressure injection then fails, the operators must recognize this condition and manually depressurize the reactor vessel to allow use of low-pressure injection systems. Operator error probabilities for failure to manually depressurize the vessel vary widely among the IPEs and significantly affect the results. This issue is discussed in Chapter 5.

Plant improvements identified for Vermont Yankee and FitzPatrick that will impact this accident class include CRD hardware and procedural enhancements to allow for sufficient injection to cool the core immediately following a plant scram. Such enhancements improve the reliability of high-pressure injection and reduce the need for manual depressurization. Neither licensee has evaluated the impact of these CRD enhancements. Recognizing the importance of inhibiting ADS, reinforcement in operator training on this issue was recommended in the Monticello IPE submittal. A sensitivity study was performed for the Fermi IPE, in which ADS inhibition was essentially removed from the procedures for non-ATWS scenarios; a 19% reduction in the total CDF was reported in the IPE submittal.

Transients with loss of DHR are important for many BWR 3/4 plants. Loss of DHR transient sequences involve accidents in which coolant injection succeeds, but DHR fails. In this situation, the suppression pool heats up, leading to containment pressurization, and the containment will eventually fail if it is not vented. Coolant injection also eventually fails, as a result of a hot suppression pool, or the adverse conditions created in containment or the reactor building when the containment is vented or fails. These adverse conditions include loss of NPSH for pumps taking suction from the suppression pool or an adverse steam environment in the reactor building.

The key factors affecting the CDF from loss of DHR sequences involve plant-specific design and operating conditions, as well as the assumptions made in the IPEs. The modeling assumptions are important to the results and represent an important area of uncertainty:

- Ability of ECCS pumps to continue operating under harsh containment conditions — ECCS pumps can fail for a variety of reasons during these accidents. NPSH requirements may not be met if the containment fails or is vented. The pumps may fail as a result of high suppression pool temperature or steam in the reactor building. The licensees varied significantly in their assessments of pump operation under these conditions, ranging from the assessment that the harsh environment always causes failure of the ECCS pumps, to the assessment that the pumps are unaffected by the harsh environment. Some of these differences result from actual variations in pump design or venting procedures. Other differences, however, are attributed to the reliance on engineering judgment instead of plant-specific equipment analyses.

- Operability of non-ECCS injection sources following containment failure — All plants have available injection systems (other than ECCS) that can have all required equipment (including associated support equipment) located outside the containment and reactor building; thus, these systems are not subject to the potentially harsh environments noted above. Examples of such systems include the CRD and condensate systems. Plants with such systems that experience no adverse effects following containment failure have lower CDFs.

- Treatment of venting — Most Mark I containments are now equipped with hardened vents to prevent containment failure and harsh environments in the reactor building. Use of these vents can reduce the CDF for this accident class. However, some licensees did not model these vents in their IPEs, and some IPEs accounted for loss of NPSH upon venting. Therefore, there are differences in the results from plant to plant because of differences in the treatment of venting and its effects.

Because adverse containment conditions can impact continued coolant injection during a loss of DHR accident, many licensees suggested plant improvements that will help to ensure continued coolant injection. These suggested plant improvements include implementing procedures to replenish the CST to prevent switching the HPCI suction source to the suppression pool, increasing the RCIC turbine exhaust pressure trip setpoint, modifying the HPCI logic to prevent automatic switchover of suction on high torus level, and implementing procedural changes directing the operator to use alternative systems for injecting water into the vessel and for cooling the RHR system heat exchangers. Most licensees listed installation of a hardened vent as a plant improvement and credited it in the IPE. The Fermi IPE submittal reported that, without containment venting, the total plant CDF will increase by an order of magnitude. The impact of venting on the CDF at other plants varied depending on the available systems for both DHR and coolant injection. For example, venting is not as important at plants that credited high-pressure coolant injection capability from outside the containment.

ATWS sequences are significant contributors for some plants in the BWR 3/4 group. ATWS sequences involve a transient followed by failure to shut down the nuclear chain reaction by inserting the control rods. Power generation continues at levels far in excess of normal decay heat. An ATWS sequence can be mitigated by boron injection using the SLC system, control of coolant injection, and heat removal.

The ATWS results are affected by modeling assumptions and by plant-specific design features. The key factors vary from plant to plant, as follows:

- Failure of boron injection — The SLC system provides boron injection that can shut down the nuclear reaction. An important SLC failure mode reported in the IPEs was the failure to initiate the system. The probabilities for operator failure to initiate SLC vary by orders of magnitude among the IPEs. Some of these variations result from different assumptions about timing, but uncertainties remain. The Monticello and FitzPatrick licensees also credit alternative means of injecting boron using systems such as the CRD system. Generally plants with low operator error probabilities for initiating SLC and with alternative means of injecting boron have lower ATWS contributions from loss of SLC sequences.

- Power reduction using level control — Some licensees (e.g., Hatch 1&2) assume that if boron injection fails, controlling the water level in the core will reduce power to within the turbine bypass capacity of the plant and allow for alternative means of placing the plant in cold shutdown. However, most licensees assume that level control can only be successful in conjunction with the use of SLC. The ATWS contribution from loss of boron injection sequences is reduced for licensees that credit level control by itself as a means of reducing core power to a stable level.

- Operator failure to inhibit the ADS and control coolant injection — If the operator fails to take action to manually inhibit the ADS, the HPCI system will be lost and low-pressure systems will inject at a high flow rate that is not easily controlled. The sequence potentially leads to boron flushing and to a large power surge that repressurizes the system and causes low-pressure injection to cease. Repeated pressurization cycles may eventually lead to failure of either the RCS boundary or the low-pressure injection system. In the IPEs, the probabilities of operator failure to inhibit the ADS varied by several orders of magnitude. This action can be the object of further study, as the reasons for the wide variation are not always obvious from the submittals. Some licensees (including Pilgrim and Brunswick) assumed that failure to inhibit the ADS will not lead to core damage if the operator controls the LPCI flow. These plants generally have lower ATWS contributions from sequences involving failure to inhibit the ADS.

Only a few licensees specifically identified plant improvements to address an ATWS. These improvements generally involve procedural changes and associated training to enhance operator performance in dealing with issues such as level control, inhibiting the ADS, and initiating SLC. These improvements were not credited in the IPEs.

LOCAs are not dominant contributors for most BWR 3/4 plants. Because LOCAs are low-frequency events and BWR 3/4 plants have a variety of diverse injection sources to mitigate LOCAs, such accidents are not usually important to either the CDF or risk for these plants. Differences in the credit given for alternative injection systems is the major parameter accounting for the variability in the LOCA results. For example, the Susquehanna IPE reported a low contribution from small-break LOCAs, partially because of the credit given for enhanced CRD flow for mitigating this size LOCA. Other licensees (such as FitzPatrick) credit limited volume systems (such as condensate) for partially mitigating a large LOCA. Most licensees assume that vessel depressurization is needed for medium LOCAs (including SORVs) in cases involving HPCI failure. This assumption contributes to medium LOCAs being the dominant LOCA core damage contributor at several plants.

The BWR 3/4 IPEs did not identify specific improvements to address LOCAs. However, the improvements identified for transients with loss of injection or loss of DHR also apply for LOCAs. One notable improvement identified by the FitzPatrick licensee involves installing a bypass switch and revising procedures to prevent and mitigate the consequences of miscalibration of the low reactor vessel pressure instruments that provide permissive signals required for opening low-pressure ECCS valves (an important failure mode identified in many IPEs).

Internal flooding is not important for most BWR 3/4 plants. Internal flooding events involve rupture of water lines or operator errors that result in a release of water that can directly cause the failure of required mitigating systems (e.g., through loss of cooling) and/or the failure of other mitigating systems as a result of submergence or spraying of required components. The most important factor in determining the importance of flooding is plant layout. Separation and compartmentalization of mitigating system components reduces the impact of internal flood initiators. Internal flooding events are not dominant at most plants because no internal flood initiator is identified that would cause the complete failure of all systems required to mitigate a flood-induced transient without additional random failures. For a few plants, the licensees identified important internal flooding sequences that generally involve SW system breaks affecting equipment through both loss of cooling and flood-related impacts on other mitigating systems.

Only one plant improvement was identified that specifically relates to internal flooding. That improvement, which is specific to FitzPatrick, involves protection of HPCI- and RCIC-related power sources from spray effects during a specific internal flooding scenario.

ISLOCAs are not dominant contributors to CDF for BWR 3/4 plants. The low CDF contribution results from the low frequency of the LOCA initiator, combined with its negligible-to-minimal impact on other systems. This low impact is attributed to the compartmentalization and separation of the equipment.

3.2.3 BWR 5/6 Perspectives

The BWR 5/6 group (7 IPE submittals) includes eight plants, as listed in Table 3.7.

Table 3.7 Plants (per IPE submittal) in the BWR 5/6 group.

Clinton	Grand Gulf 1
LaSalle 1&2	Nine Mile Point 2
Perry 1	River Bend
Washington Nuclear Power 2	

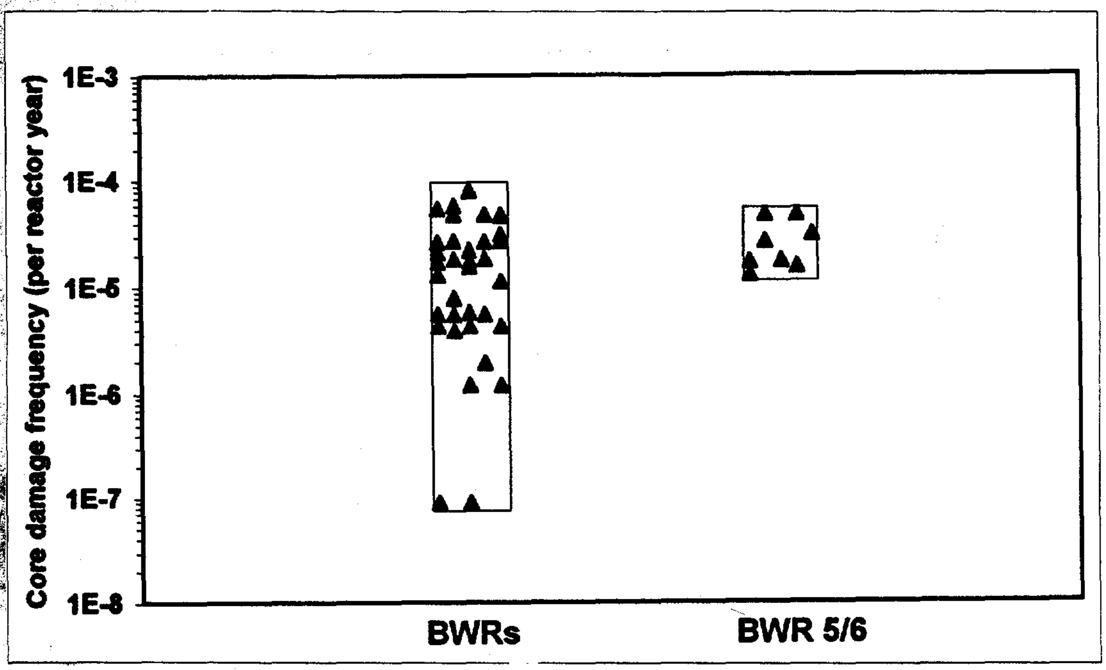
The main feature that differentiates these plants from BWR 3/4 plants is the replacement of the HPCI system containing a steam-driven pump with a HPCS system containing a motor-driven pump. Four of the plants (LaSalle 1&2, Nine Mile Point 2, and Washington Nuclear Power Unit 2) are BWR 5s housed in Mark II containments; the other four (Clinton, Grand Gulf 1, Perry 1, and River Bend) are BWR 6s housed in Mark III containments.

The following accident classes were important contributors at several plants and to the group on average:

- SBO — the loss of all off-site and on-site power
- transients with loss of coolant injection
- transients with loss of DHR

These three accident classes dominate the CDFs at several plants because of plant-specific features or modeling assumptions unique to those plants. Although some licensees reported significant CDFs from ATWS, LOCAs, and internal flooding accidents, these accident classes, on average, have lesser contributions to the overall plant CDF. All of the plants report negligible contributions from ISLOCAs.

The BWR 5/6 plant CDFs obtained from the original IPE submittals exhibit less variability than the other BWR plant groups, but are within the same range as the majority of BWRs, as illustrated in Figure 3.7. However, the CDFs from specific accident classes vary more significantly from plant to plant, as shown in Figure 3.8. The variations in the reported IPE results are attributed to several plant-specific design features and modeling assumptions, none of which totally dominate. These factors and the improvements being considered by the licensees to address identified weaknesses are summarized in Table 3.8 and are discussed below. It should be noted that revised assessments by the licensees may result in different perspectives than those indicated in Table 3.8 and discussed in the remainder of this section.



- The variability in the total plant CDFs is small, but a large diversity exists in the CDF contributors because of a combination of plant-specific design features and modeling assumptions.
- Differences in service water system alignments impact diesel generator availability.
- Variability in modeling RCIC failure modes impacts the CDFs for this group.
- ECCS pumps can pump saturated water during a loss of DHR, but harsh environments in the reactor building following containment failure are identified at some plants as causing the failure of some systems.
- Credit given for use of HPCS as a coolant injection source during ATWS impacts the results.
- Support system failures are important because they cause the failure of multiple systems.

Figure 3.7 Reported IPE CDFs and key perspectives for the BWR 5/6 plant group.

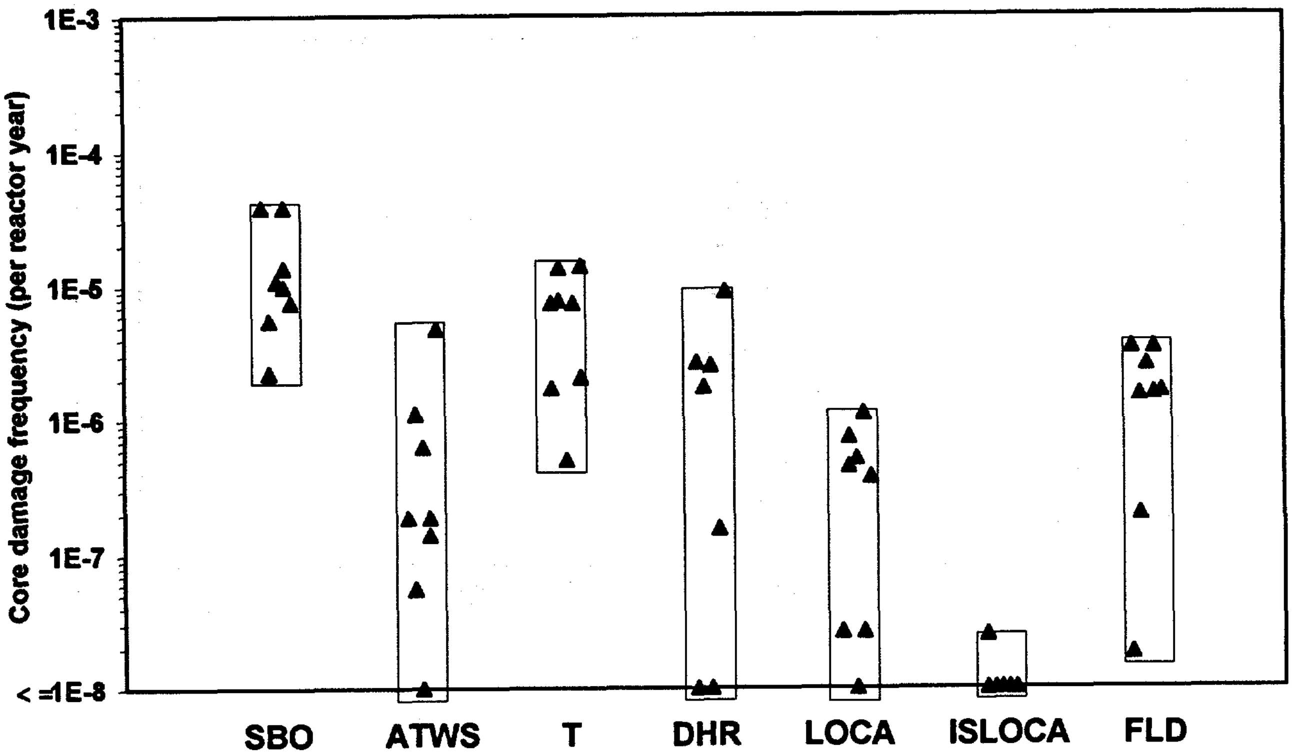


Figure 3.8 Reported IPE accident sequence CDFs for BWR 5/6 plants.

Table 3.8 Summary of CDF perspectives for the BWR 5/6 plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
SBO accidents		
Important for all BWR 5/6 plants	Availability of AC-independent systems (i.e., diesel-driven firewater system, RCIC failure modes) Battery life SW system design and HVAC dependency AC power reliability (number of diesel generators, cross-tie capability between buses, diverse AC power sources)	Implement new procedures Improve operator training Improve DC reliability (cross-tie of buses, portable power supply to charger) Increase DC load shedding Increase AC reliability (alternative AC power source, cross-tie of buses) Increased availability of AC-independent injection systems (diesel-driven firewater, reconfiguring RCIC improvements)

Table 3.8 Summary of CDF perspectives for the BWR 5/6 plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
	Transients with loss of injection accidents	
Important for most BWR 5/6 plants	Injection system dependencies on support systems defeat redundancy Availability of alternative injection systems (e.g., CRD, motor-driven feedwater pumps, service cross-tie to RHR, firewater system) Failure to depressurize is influenced by operator direction to inhibit the ADS	Implement procedural and hardware enhancements to use alternative systems for injection (e.g., firewater system) Increase emphasis on operator training and/or procedure modification for depressurization Improve system reliability by modifying system surveillance procedure to include testing of other system equipment (e.g., pump suction line from suppression pool for the HPCS system), by revising maintenance procedure to reduce common cause failures Enhance procedures to respond to loss of HVAC in ECCS rooms
	Transients with loss of DHR accidents	
Important for many BWR 5/6 plants	Dependency of DHR systems on support systems Availability of injection system located outside containment and reactor building and ability to operate following containment failure Capability of ECCS to pump saturated water	Improve operator training Increase reliability of equipment (i.e., hardware modifications; replacement of pump motors with air-cooled motors) Revise isolation logic for plant SW and IA Provide control room temperature indicator for rooms containing SW pumps
	ATWS accidents	
Relatively unimportant for most BWR 5/6 plants	Operator failure to initiate SLC in a timely manner, to maintain MSIVs open, to control vessel level, and/or to maintain pressure control Use of HPCS for coolant injection	Improve operator training Install automatic inhibit of the ADS Install an alternative boron injection capability
	LOCAs	
Significant for a few BWR 5/6 plants	High redundancy and diversity in coolant injection systems Inclusion/exclusion of inadvertently open relief valve (IORV) in LOCA category	None identified

Table 3.8 Summary of CDF perspectives for the BWR 5/6 plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
	Internal flood accidents	
Significant for a few BWR 5/6 plants	Plant layout: (separation and compartmentalization of mitigating system components)	Conduct periodic inspection of cooling water pipes Enhance procedures and training to respond to floods, including isolation of the flood source
	ISLOCAs	
Not important for BWR 5/6 plants	Compartmentalization and separation of equipment Harsh environments induced by ISLOCA	None identified

SBO accidents are important contributors to the CDFs for all plants in the BWR 5/6 group. SBOs are defined for this group as a LOSP coupled with a loss of both Divisions 1 & 2 of emergency on-site power. The failure of these AC power sources results in loss of multiple mitigating systems, leaving only AC-independent systems (such as the steam-driven RCIC system) and the HPCS system available for coolant injection. The HPCS system is powered by its own AC division of emergency power, complete with a separate diesel generator. Failure of the HPCS-related division of on-site power is not included in the definition of an SBO.

Plant design and operational features have a greater impact on the SBO CDF than do modeling assumptions or differences in data. However, no single factor is responsible for the differences in the reported results. Usually, combinations of factors that vary from plant to plant result in differences in the SBO contributions. Some influential plant features and modeling characteristics are discussed below:

- SW system alignments — A major contributor to the SBO CDF at several plants is the failure of the emergency diesel generators and HPCS pump or diesel generator caused by failures of the SW system. Some plants (such as Grand Gulf and Clinton) have separate standby service water (SSW) systems that only serve emergency loads. Other plants (such as River Bend and Nine Mile Point 2) have SSW systems that share piping with the normal SW system, require isolation of non-safety loads during accidents, and do not have a separate division dedicated to the HPCS system and its diesel generator. Failure of these shared SW systems tends to reduce the on-site AC power reliability more than when there are separate SSW systems. (For example, failure of Divisions 1 and 2 of electrical power at Nine Mile Point 2 results in failure of the HPCS diesel generator through loss of SW cooling.) Generally, plant-specific failure modes of SW systems or their support systems (e.g., the SSW pump room HVAC at Grand Gulf), even at plants with separate SSW systems, contribute to some variation in the SBO CDFs for this group.

- RCIC system failure modes — Variation in the failure modes for the RCIC system include the component failure rates, and the timing of these failures (which affects the potential for off-site power recovery). The bypass of some trip signals modeled in the IPEs accounts for some of the differences in the reported SBO CDFs for this group of plants. How much of this variation results from plant design or operational differences versus modeling assumptions is not clear from information provided in the submittals. However, it is known that there is considerable variation in setpoints and related procedural guidance. RCIC failure modes modeled to various extent in the IPEs include failures caused by high suppression pool temperature, a high turbine exhaust back pressure trip, steam tunnel temperature trips, loss of pump room cooling, and battery depletion varying from one to eight hours. Some licensees took credit for bypassing steam tunnel temperature and high turbine exhaust pressure trips, while others did not. Furthermore, RCIC failure associated with a loss of pump room cooling is not treated as recoverable at some plants when failure is assumed to occur quickly after the loss of HVAC, but is treated as recoverable at other plants. The highest SBO CDF is calculated for LaSalle 1&2, where a sneak circuit failure during a LOSP causes a false high RCIC room temperature signal to isolate the system. This sneak circuit failure and actual trip signals on high pump room temperature contribute significantly to the LaSalle CDF.

- Use of diesel-driven firewater — Some licensees credit the use of diesel-driven firewater as a means of supplying coolant injection during a SBO when HPCS and RCIC fail. The vessel must be depressurized by opening SRVs (which requires DC power) and maintained at low pressure in order for firewater to be used. Not all plants have the capability for firewater injection (i.e., procedures and flow path alignments have not been established). Furthermore, some licensees with the capability to use firewater did not credit its use because of the difficulty of aligning the system and uncertainty in its potential to provide sufficient coolant to the vessel in a timely manner. For those licensees that did model firewater injection, credit was generally taken only in the long-term when sufficient time is available for alignment. Generally, plants that credited firewater injection have lower SBO contributions from long-term sequences (i.e., sequences in which HPCS or RCIC operate for several hours) if battery power is still available to maintain the SRVs open. The lowest SBO contribution was calculated for the Perry IPE, the only plant to indicate that firewater injection can successfully be aligned early during an SBO.

All of the licensees in this group plan or have implemented plant improvements to address factors influencing their SBO sequences. Many of the improvements involve modifying or creating new procedures; however, some involve system hardware modifications. As expected, several of the identified improvements address AC power reliability and include procedures to cross-tie the HPCS diesel generator to one of the other divisions of AC power (identified by three licensees) and to improve the potential for off-site power recovery. DC power improvements include a procedure at one plant for cross-tieing buses and using a portable diesel generator to supply power to a battery charger. Increasing the reliability of coolant injection systems is also a significant area of improvement for this group. Identified improvements include a procedure to address the sneak circuit at LaSalle 1&2 (calculated by the licensee to reduce the CDF from this failure mode by a factor of 20), bypassing RCIC steam tunnel temperature trips (calculated by the licensee to reduce the CDF at one plant by 15%), and aligning the firewater system for injection through other system flow paths (calculated by the licensee to reduce the River Bend CDF by approximately 80% when combined with the use of a portable diesel generator to supply a battery charger).

Transients with loss of coolant injection are important contributors to the CDF for many plants in this group. Sequences involving loss of high-pressure injection systems with failure to depressurize the vessel for low-pressure injection are important for this plant group, as are sequences with successful depressurization and loss of all injection systems. No single factor can be identified to account for the variability in the CDFs calculated for this accident type. Generally, the variability is attributed to multiple factors, including plant design and operational differences, as well as differences in modeling assumptions and data values.

Two factors are worth mentioning for this accident category for BWR 5/6 plants, and both of these factors involve plant-specific design characteristics and (to some extent) modeling assumptions.

- Alternative coolant injection systems — Some variability exists in the available coolant injection systems credited in the IPEs for this group. Most of the plants have motor-driven feedwater pumps that can inject coolant even during transients with MSIV closure. (Steam-driven pumps lose their driving force during this type of accident.) Only two licensees (Grand Gulf and River Bend) credited CRD for injection (based upon a calculation performed in the NUREG/CR-4550 Grand Gulf study) in the short term (i.e., less than three hours following scram). However, both IPEs specify that the vessel must not be depressurized because this would result in CRD pump runout, leading to pump failure. Only two licensees modeled CRD injection in the long-term (more than three hours following scram). Some licensees do not credit the use of the CRD system for unspecified reasons, while others identified plant design features (e.g., isolation signals) that affect its use.

  Most licensees credit the use of the SW system cross-tied to the RHR system and the condensate system as low-pressure injection systems. However, a few licensees identified plant-specific design features cited as limiting the use of these systems (e.g., the ability to provide sufficient makeup to the main condenser). Perry is unique in that the IPE specified that injection can be provided by the condensate transfer system. Several licensees credited the use of the firewater system as an injection source, but only Perry credited it in short-term sequences. This variability in injection systems cannot be directly correlated to the CDF (i.e., the plants that credit more injection systems do not always have a lower CDF for this accident category). However, the types of accident-initiating events dominating this accident category are influenced by the variability in alternative injection systems (see the next item).

- Support system initiating events — Because of the availability of multiple coolant injection systems, including alternative systems credited in some IPEs, the accident class involving transients with loss of coolant injection for this group of plants include a substantial contribution from loss of support systems, especially as initiating events. Loss of an AC bus is particularly important at Grand Gulf and Nine Mile Point 2 because of the isolation of nonsafety-related support systems. These support systems are required for continued operation of coolant injection systems (such as feedwater, condensate, and CRD). LOSP is important at some plants because it causes the failure of the power conversion system (PCS) (including motor-driven feedwater and condensate pumps) and can also cause the failure of the CRD system (depending on whether the CRD system and required support systems are powered by emergency buses). Losses of DC power, instrument air, and cooling water system initiators are also important at some plants because plant-specific alignments of these systems negatively affect coolant injection systems by isolating the systems through loss of control power or essential cooling, or through impacts on other support systems.

Many licensees in this group identified changes to procedures and modifications of system hardware that have the potential to reduce the contribution from this accident class. To improve the reliability of the HPCS system, Clinton is modifying the system surveillance procedure to include testing the pump suction line from the suppression pool (which is calculated by the licensee to reduce the CDF by 13%). Some of the suggested modifications for the RCIC system during a SBO will also increase the system reliability for all loss of injection accidents (e.g., bypassing steam tunnel temperature trips). Recognizing the importance of depressurizing the vessel for low-pressure injection, the licensees suggested several improvements, including revising procedures to not inhibit automatic depressurization for non-ATWS scenarios and emphasizing depressurization more in training. One licensee is revising maintenance procedures to reduce common cause failures of injection system components (e.g., miscalibration of permissive signals for low-pressure injection system valves), and another is enhancing procedures to respond to loss of HVAC in ECCS pump rooms. The Clinton IPE submittal also suggests modifications to the firewater injection path to allow for quicker alignment which reduces the CDF by 13%.

Transients with loss of DHR are important for many BWR 5/6 plants. Loss of DHR transient sequences involve accidents in which coolant injection succeeds, but DHR fails. In this situation, the suppression pool heats up, leading to containment pressurization and failure if it is not vented in time. Coolant injection systems may fail from the harsh environments in the reactor building that are introduced by containment failure. DHR systems available at plants in this group include the PCS, RHR, and venting systems (not credited at several plants).

Two key factors affecting the CDF from loss of DHR transient sequences for this plant group involve plant-specific design and operating features and modeling assumptions:

- Support system failures — Systems such as AC and DC power, cooling water systems, and IA support the operation of multiple DHR systems in the plants within this group. The impact from these support system failures varies because the design of these systems and the dependence of the DHR and other support systems on them is plant-specific. Loss of AC power initiators (primarily loss of AC power to one electrical division) are important contributors at many plants, and the dominant contributors at the plants with the highest CDF from this accident class. Loss of AC power is identified as resulting in cooling water system isolations that cause the PCS and some coolant injection systems to fail. It also results in the loss of venting and loss of a division of the RHR and emergency coolant injection systems.

- Containment failure impacts on coolant injection systems — Unlike most of other BWR plant groups, the continued operation of ECCS pumps in this group is not affected by adverse containment conditions because they are capable of pumping saturated water. However, some licensees with both Mark II & III containments identified the potential for failure of coolant injection systems caused by the harsh environments in the reactor building following containment failure. The potential for such failures is dependent upon the containment design, failure locations, and equipment layout. It is obvious from the IPEs that some licensees assumed that the coolant injection systems will either fail or survive following containment failure, while other licensees provided some basis (e.g., calculations) for survival or failure of the systems. Generally, licensees that identified harsh environment failure modes of the coolant injection systems have higher CDFs for this accident category. One exception is River Bend, which has a lower contribution because it has a unique containment fan system that can prevent overpressurization of the containment.

Only a few licensees identified improvements that can reduce the contribution from loss of DHR sequences for transient (or other) initiators. Perry considered a passive containment vent that the IPE submittal indicated will reduce the plant CDF by 15% (implementation of this passive vent was subsequently rejected by the licensee). River Bend, which has a small vent line that cannot prevent containment overpressure, considered increasing the vent capacity. However, a sensitivity calculation reported in the IPE submittal indicated that this modification will reduce the CDF by only 1% and was thus rejected by the licensee. Grand Gulf is considering plant-specific improvements in several areas that will reduce the contribution from this accident category, and will also affect other accident types. These improvements include revising the isolation logic for plant SW and IA (which currently isolate on loss of Division 1 power) and providing in the control room a temperature indicator for the rooms containing the SSW pumps to help reduce the contribution from HVAC failure in these areas. The IPE submittal indicated that both of these improvements can reduce the plant CDF by approximately 30%.

ATWS sequences are not important contributors for the majority of BWR 5/6 plants. ATWS sequences involve a transient followed by failure to shut down the nuclear chain reaction by inserting control rods. Power generation continues at reduced levels, but far in excess of normal decay heat. An ATWS sequence can be mitigated, however, by injecting boron using the SLC system, by controlling coolant injection, and by subsequently removing the heat generated at the reduced power level.

ATWS contributions to the plant CDFs are affected more by modeling assumptions than by plant-specific design features. Modeling variations are noted in the SLC success criteria (this can be design dependent but modeling differences are also noted), the use of vessel level control by itself as a means of reducing core power to a safe level, and inhibiting ADS and controlling low-pressure injection. Human error probabilities for operator actions to mitigate an ATWS are also variable. Another important factor (unique to this plant group) is the credit taken for use of HPCS as an injection source. The use of HPCS during an ATWS is a concern since spraying cold water into the core will introduce positive reactivity. Some licensees credited the use of HPCS as being successful during an ATWS while others did not due to procedural restrictions. Licensees that credited HPCS have significantly lower ATWS-related CDFs than licensees that did not take credit for HPCS.

The highest ATWS CDF was calculated for Perry, which does not allow, by procedure, the use of HPCS for coolant injection. The Perry licensee also assumed that the operator will always be unable to maintain PCS availability during an ATWS. The latter assumption resulted from the fact that a feedwater runback would occur during an ATWS (not a common feature among BWR 5/6 plants) resulting in a reduction of the vessel coolant level to below the MSIV closure setpoint. The operators can not override the signal and must wait until the valves are fully closed to restore feedwater via the motor-driven feedwater pump.

Two licensees identified improvements aimed at reducing the potential for core damage during an ATWS event. The most significant is the installation of an automatic ADS inhibit during an ATWS, which is being evaluated at Perry. A sensitivity study reported in the IPE submittal indicated that this feature can reduce the plant CDF by 23%. Implementation was subsequently rejected based on cost-effectiveness, the increased complexity of the ADS, and the benefit achieved from other plant and procedural changes in reducing the ATWS CDF. Perry is also evaluating the installation of an alternative boron injection capability. The Clinton IPE submittal is evaluating what changes will be appropriate to their training program with regard to the CRD system and other scram-related hardware.

LOCAs are minor contributors for all plants in the BWR 5/6 group. Because LOCAs are low-frequency events, and because BWR 5/6 plants have a variety of diverse injection sources to mitigate LOCAs, such accidents are not important to either the CDF or the risk for these plants. Plants with higher LOCA CDFs in Figure 3.8 generally have inadvertent open relief valve (IORV) contributions dominating the LOCA CDF. It should be noted that the LOCA contributions at some plants did not include IORV contributions because separation and inclusion of the IORV contribution from the reported results is not always possible. Differences in the credit given for alternative injection systems also affect the results to some extent. For example, a few licensees credited the use of SW cross-tied to the RHR system to provide coolant injection during all LOCAs (if alignment is proceduralized and can be performed from the control room), others only credited the system for mitigating smaller LOCAs resulting either from remote alignment requirements or concerns about an adequate water supply. In addition, the availability of motor-driven feedwater systems at many of the plants resulted in credit taken for this system to mitigate small and medium LOCAs. A common assumption in the IPEs is that vessel depressurization is required for low-pressure injection during a medium LOCA. As a result, many licensees reported medium LOCAs with failure of HPCS and depressurization of the vessel as important LOCA sequences.

The BWR 5/6 IPEs did not identify specific improvements to address LOCAs. However, the improvements identified for transients with loss of injection or loss of DHR will also help reduce the LOCA CDFs.

Internal flooding events are significant contributors to the CDF for several BWR 5/6 plants. Internal flooding events involve rupture of water lines or operator errors that result in a release of water that can directly cause the failure of required mitigating systems (e.g., SW line break will result in loss of cooling to components) and/or the failure of other mitigating systems as a result of submergence or spraying of required components. The most important factors determining the influence of flooding are the plant layout and the dependence of mitigating systems on support systems (particularly cooling water systems). Separation and compartmentalization of mitigating system components reduce the impact of internal flood initiators. No internal flood initiator is identified that will cause the failure of all available mitigating systems. (That is, random failures are required in addition to flood-induced failures to result in core damage.) The dominant flood scenarios identified at the BWR 5/6 plants generally involve ruptures of cooling water system pipes that affect equipment through loss of cooling and through flood impacts on other mitigating systems (generally electrical switchgear).

Two licensees identified procedural improvements that are being considered for internal flooding events. These include periodic inspection of cooling water pipes to reduce the potential for flooding, and enhancement of existing procedures and training to respond to floods (including isolation of the flood source).

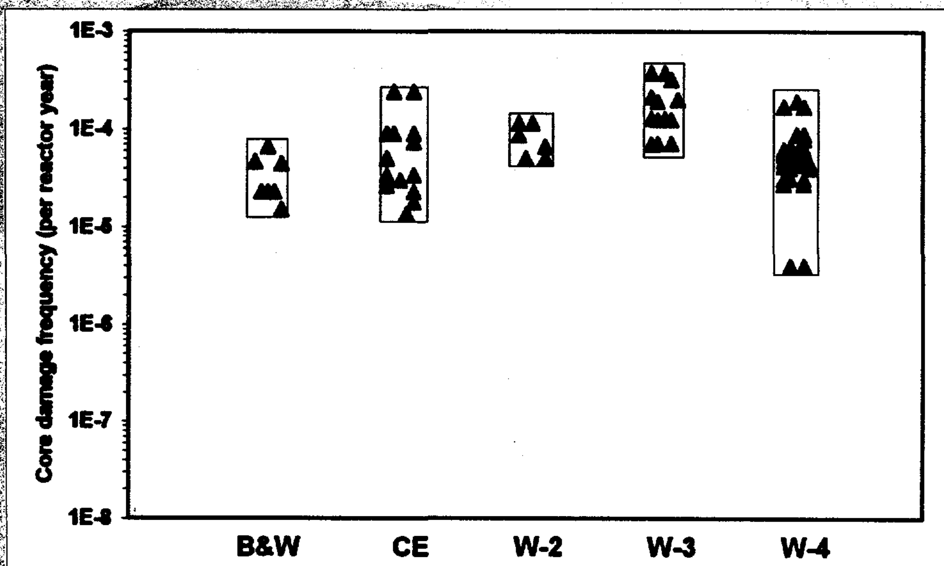
ISLOCAs are not dominant contributors to the CDF for BWR 5/6 plants. The low CDF contribution results from the low frequency of the LOCA initiator, combined with its negligible-to-minimal impact on other systems. This low impact is attributed to the compartmentalization and separation of the equipment.

3.3 Pressurized Water Reactor Perspectives

In general, the plant CDFs reported in the original PWR IPE submittals vary more within individual PWR plant groups than among plant groups. Westinghouse 3-loop plants generally have the highest CDFs, and Babcock and Wilcox (B&W) plants generally have the lowest CDFs, with the CDFs for most B&W plants falling below the CDFs for the Westinghouse 3-loop plants. However, the difference in average CDFs between these two plant groups is about the same as the variability within either of the two plant groups.

The variability in the PWR results is attributed to a combination of factors including plant design differences (especially in support systems such as electrical power, cooling water, ventilation, and IA systems), modeling assumptions, and differences in data values (including human error probabilities). The largest variation exists in the Westinghouse 4-loop plant group, which is the group with the largest number of plants, but the other plant groups also show considerable variability. The Combustion Engineering (CE) plant group contains a 2-unit plant with a CDF well above the other plants in the group, while the Westinghouse 4-loop plant group contains a 2-unit plant with a CDF considerably below the other plants in the Westinghouse 4-loop group. The reasons these plants are "outliers" are discussed in Sections 3.3.2 and 3.3.5, respectively.

Figure 3.9 presents the CDFs for the different PWR plant groups along with key perspectives. Table 3.9 summarizes the importance of the various accident classes to the PWR CDFs and the factors driving variability in the results. For each PWR group, considerable variability exists in the contributions of the different accident classes to the total plant CDF. However, licensees in all five PWR groups generally find that three types of accidents are the major contributors to the total plant CDF. These include transients, LOCAs, and SBO. These three accident classes involve initiators and/or subsequent system failures that defeat the redundancy in systems available to mitigate potential accidents. Lesser contributions are generally reported for ATWSs, SGTRs, ISLOCAs, and internal flooding. However, a few PWRs reported significant contributions from these accident classes, and SGTRs were found to be significant contributors for the Westinghouse 2-loop plants.



- CDF variability is primarily driven by plant-specific support system design and operational differences (rather than differences between vendors).
- Support system failures are generally important for all PWR plant groups.
- Plants with better feed and bleed capability generally have lower CDFs; some plants compensate for feed and bleed limitations by using alternative heat removal systems.
- In the IPE models, B&W and CE plants are generally less susceptible to RCP seal LOCAs than Westinghouse plants;this reduces the transient and SBO CDFs for those plants.

Figure 3.9 Reported IPE CDFs and key perspectives for PWRs.

Table 3.9 Summary of CDF perspectives for PWRs.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
	SBO accidents	
Important for most PWRs, regardless of plant group	Susceptibility to RCP seal LOCAs Redundancy in emergency AC power sources (e.g., number of diesel generators) Battery life Use of plant operating data indicating low frequencies for LOSP and high reliability of emergency diesel generators Implementation of Westinghouse seal LOCA model Backup cooling for RCP seals	Addition of air-cooled diesel generators Increased battery capacity Increased DC load shedding Procedural changes to add redundancy to power supplies Reduced susceptibility to RCP seal LOCAs
	LOCAs	
Important for most PWRs, regardless of plant group	Manual action required to switchover to recirculation Failures of low-pressure injection system from common cause failure of pumps to start, common cause failure of valves, and low-pressure injection pump cooling failures Alternative actions to mitigate LOCA (e.g. depressurizing the RCS using the steam generator atmospheric dump valves (ADVs) when high-pressure injection fails during LOCA) Size of refueling water storage tank (RWST)	Revised training for feed-and-bleed Hardware enhancements, procedural changes and training enhancements to improve reliability of switchover from injection to recirculation
	ATWS accidents	
Relatively unimportant for most PWRs, regardless of plant group	Ability to mitigate by pressure control, boration, and heat removal	Installation of diverse scram systems to comply with the ATWS rule Installation of ATWS mitigating systems actuation circuitry (AMSAC) Addition of alternative scram button
	ISLOCAs	
Relatively unimportant for PWRs, regardless of plant group	Low frequency of rupture Compartmentalization and separation of equipment	Leak testing for isolation valves Procedure modifications for identifying and mitigating ISLOCAs
	SGTR accidents	
Relatively unimportant to CDF for most PWRs	Low frequency of rupture Credit for operator actions and equipment used to mitigate accidents	Procedure modifications for isolating steam generator with ruptured tube Procedure modifications for coping with SGTR

Table 3.9 Summary of CDF perspectives for PWRs.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
Transient accidents		
Important for most PWRs, regardless of plant group	Susceptibility to RCP seal LOCAs Capability for feed-and-bleed cooling Ability to cross-tie between systems and units Dependency of other plant systems on component cooling water (CCW) and/or SW systems Dependency on HVAC and IA Ability to depressurize the steam generators and use condensate for heat removal Use of Westinghouse seal LOCA model Ability to supply long-term water to the suction for auxiliary feedwater (AFW)/emergency feedwater (EFW)	Improved operator training and procedural modifications to reduce frequency of RCP seal LOCA Replacement of manual crossover valves with motor-operated valves Staggering of high-pressure injection pump use during loss of SW events Providing alternative ventilation and improved procedures and training to cope with loss of ventilation to areas such as switchgear rooms Increased reliability of AC power (e.g., additional diesel generators, enhanced procedures for cross-tieing buses) Improved availability of long-term heat removal (e.g., use of firewater for steam generator cooling)
Internal flood accidents		
Important for some PWRs, regardless of plant group	Plant layout (separation and compartmentalization of mitigating system components)	Changes in plant layout Enhanced procedures and training to respond to floods, including isolation of the flood source

Sections 3.3.1 through 3.3.5 discuss the factors that have the greatest influence on the CDF contributions from the most important accident classes for each PWR plant group. Some of these factors reflect concerns that are more prevalent in a particular PWR plant group, but most reflect design differences or modeling assumptions that apply to all PWR plant groups. Design differences among the PWR plant groups are summarized below. It should be noted that revised assessments by the licensees may result in different perspectives than those indicated in Table 3.9 and discussed in these sections.

One of the most important factors affecting PWR CDFs is susceptibility to RCP seal LOCAs for transient and SBO sequences. To prevent core damage in RCP seal LOCA sequences, inventory makeup is required in addition to core heat removal. Both B&W and CE plant groups have less susceptibility to RCP seal LOCAs in the IPE models because most plants in these groups have a seal design that the industry believes to be less prone to seal damage based on some limited experimental data. However, there are several plants in each group that have indicated a significant CDF contribution involving RCP seal LOCAs. This lower susceptibility to RCP seal LOCAs in the B&W and CE IPEs tends to reduce contributions from transient and SBO sequences for B&W and CE plants relative to Westinghouse plants. Because of limited testing of the B&W and CE RCPs, there is considerable uncertainty regarding this issue.

Because the probability of RCP seal LOCAs is generally lower in the B&W and CE IPEs, these plants tend to show more benefit than Westinghouse plants from plant characteristics that improve the reliability of heat removal through the steam generators. (Typical improvements include reliable or redundant feedwater pumps, sustained source of water for feedwater, or longer battery life for control of AFW during SBO.) These factors are less important for many Westinghouse plants because RCP seal LOCAs lead to core damage despite the cooling provided through the steam generators.

Feed-and-bleed cooling is often an important backup for transient sequences with loss of steam generator heat removal. All but one of the B&W plants have high-pressure injection pumps with high shutoff heads that can provide adequate flow for feed-and-bleed cooling even at the SRV setpoint. Some CE plants do not have power-operated relief valves (PORVs) or other means to depressurize. The inability to use feed-and-bleed cooling for these CE plants is generally compensated for by the ability to depressurize the steam generator and use condensate for cooling. Therefore, the lack of PORVs has less influence on the IPE results than might otherwise be expected.

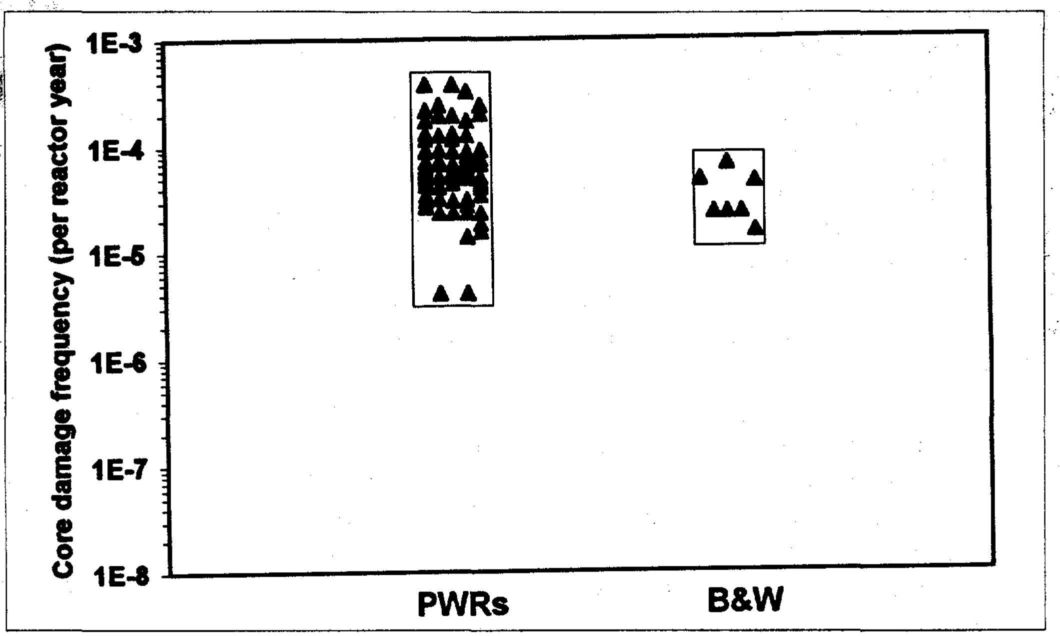
The final factor that tends to show similarities within plant groups is the ECCS recirculation configuration. Plants with a higher degree of automation for performing the switchover and plants that can achieve high-pressure recirculation with fewer components operating tend to have lower failure rates associated with the switchover to recirculation. For plants with manual switchover, variability in the assessment of operator performance of the action is also important. The B&W plants require manual actions for ECCS switchover from injection to recirculation, and the high-pressure pumps must draw suction from the low-pressure pumps to operate in the recirculation mode. The CE plants have automatic switchover, and the high-pressure pumps can draw water directly from the sump rather than drawing suction from the discharge of the low-pressure pumps. The Westinghouse plants are mixed on these factors. Some Westinghouse plants require operator actions to perform the switchover, while other Westinghouse plants have automatic switchover. For some Westinghouse plants, the high-pressure pumps draw directly from the sump during recirculation, while at other Westinghouse plants, the high-pressure pumps must be aligned to draw suction from the low-pressure pumps (which draw from the sump.)

3.3.1 B&W Plant Perspectives

Table 3.10 lists the seven plant units (represented by five IPE submittals) that make up the B&W plant group. In the IPE results, B&W plants as a group have somewhat lower plant CDFs than the bulk of the PWRs, as illustrated in Figure 3.10. The importance of specific accident classes to the plant CDF varies significantly from plant to plant, as shown in Figure 3.11.

Table 3.10 Plants (per IPE submittal) in the B&W group.

Arkansas Nuclear One (ANO) 1	Crystal River 3
Davis-Besse	Oconee 1,2&3
Three Mile Island (TMI) 1	



- Support system failures are important because they defeat redundancy in front-line systems.
- Ability to feed-and-bleed without operator actions reduces CDF for most B&W plants.
- Plants with Byron-Jackson pumps generally have lower contributions from RCP seal LOCAs.
- B&W plants have manual ECCS switchover to recirculation, which increases the contribution from recirculation failures for LOCAs.

Figure 3.10 Reported IPE CDFs and key perspectives for the B&W PWR plant group.

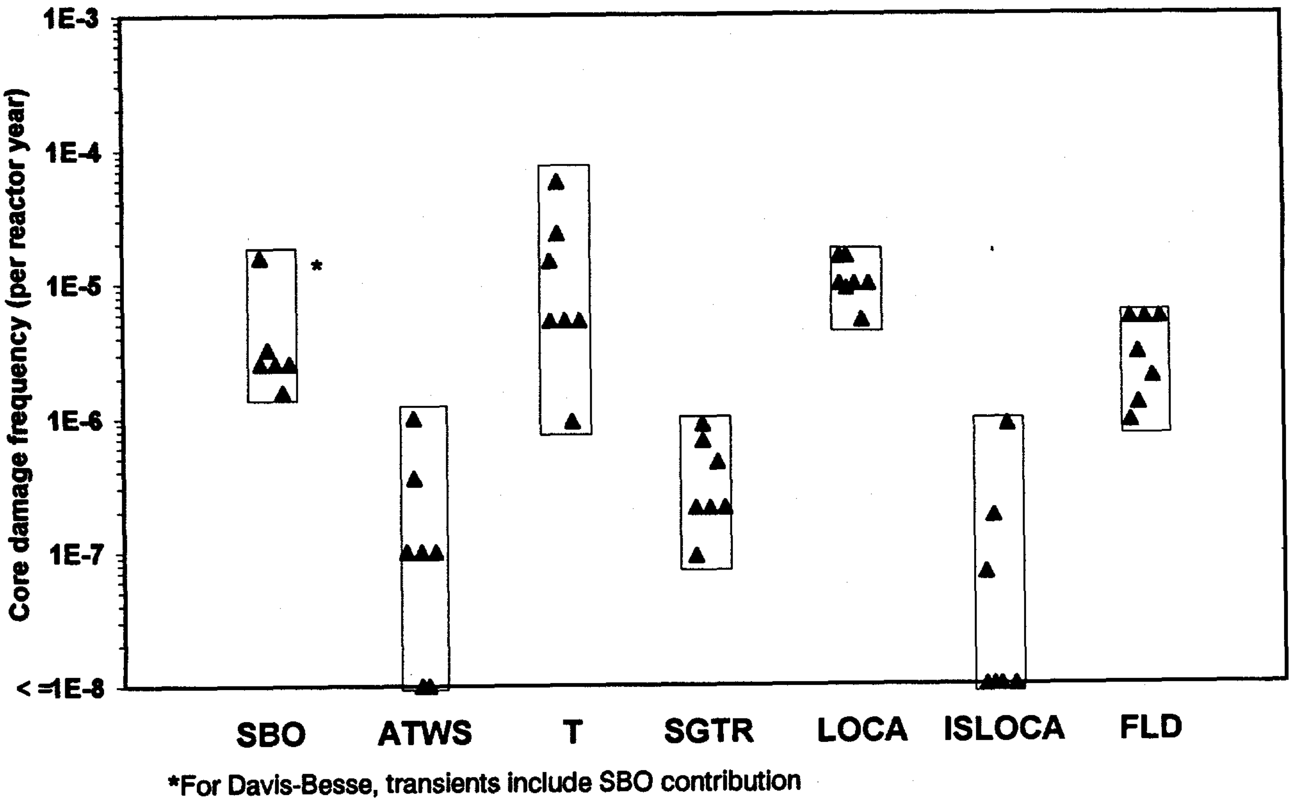


Figure 3.11 Reported IPE accident sequence CDFs for B&W plants.

In general, though, the following accident classes are important for most of these plants:

- transients
- LOCAs
- SBO — the loss of all off-site and on-site AC power

Transients are generally considered to be important contributors to CDF and containment failure frequency because they involve relatively high initiating event frequencies coupled with support system failures that defeat the redundancy in systems available to mitigate potential accidents. LOCAs are more important for these PWR plants than for BWR plants (see Section 3.2) because there are fewer systems available in the PWRs to provide low-pressure injection. SBO accidents are important for some B&W plants because they leave few systems available to prevent core damage.

One licensee (Oconee 1,2&3) identified internal flooding accidents as being important, reflecting a plant-specific weakness to a turbine building flood. None of the licensees found ATWS, SGTRs, or ISLOCAs to be important contributors to CDF. (These accidents are normally low contributors because of the low frequency of the initiating event.) The diverse scram systems installed in all B&W plants for the ATWS rule contribute to the lower ATWS frequency for these plants. Although SGTRs and ISLOCAs are generally found to be low contributors to CDF, they can be important risk contributors since they bypass containment.

The variation in the reported IPE results is attributed to many factors including plant-specific design features, modeling assumptions, and variation in data (including the probability of operator errors). These factors and the improvements being considered by the plants to address weaknesses are summarized in Table 3.11, and are discussed below for each important accident class. It should be noted that revised assessments by the licensees may result in different perspectives than those indicated in Table 3.11 and discussed in the remainder of this section.

Table 3.11 Summary of CDF perspectives for the B&W plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
	Transient accidents	
Important for nearly all B&W plants	Design of RCP seals, which affects susceptibility to RCP seal LOCAs Capability to feed-and-bleed without operator action at most plants SW and CCW design, and dependency of front-line systems on these systems Time window used for plant-specific data Modeling of common cause failures	Training enhancements or procedural modifications to reduce the probability of RCP seal LOCAs Plant-specific operational and design changes: • replacing manual crossover valves between low-pressure injection pumps with motor operated valves (MOVs) that can be remotely operated from the control room • staggering of high-pressure injection pump use during loss of SW events • providing alternative ventilation to maintain SW operation upon loss of HVAC
	LOCAs	
Important for most B&W plants	Factors that affect time to perform the manual switchover to ECCS recirculation: • size of RWST • containment spray setpoint (higher setpoints avoid diverting the ECCS supply to containment sprays) Design of low-pressure injection system and modeling of common cause failures	Procedural changes or training enhancements to improve the reliability of switchover from injection to recirculation Improvements to extend the period of injection from the borated water storage tank (BWST) Improvement to cope with simultaneous high-pressure injection suction valve failures
	SBO accidents	
Important for many B&W plants	Design of RCP seals, which affects susceptibility to RCP seal LOCAs Availability of alternative RCP seal cooling AC power reliability (number of diesel generators, cross-tie capabilities between units, diverse AC power sources) Battery life	Addition of air-cooled diesel generator Additional battery capacity Procedural changes for DC load shedding Procedural changes for replenishing diesel fuel oil for long-term diesel use

Table 3.11 Summary of CDF perspectives for the B&W plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
	Internal Flood Accidents	
Only important for one B&W site	Plant layout (separation and compartmentalization of mitigating system components)	Modified procedures to extend water supplies during internal flooding events Procedural change to minimize potential for flood propagation
	ATWS accidents	
Not important to CDF for any B&W plants	Diverse scram system reduced ATWS contribution for all plants	None identified
	SGTR accidents	
Not dominant for any B&W plants	Modeling of operator actions to isolate the rupture and provide long-term heat removal	Procedural changes to improve the probability of steam generator isolation during SGTR Modified severe accident management guidelines to refill BWST during SGTR
	ISLOCAs	
Not important for any B&W plants	Compartmentalization and separation of equipment	Procedural and training modifications to reduce the likelihood of ISLOCA

Transients are important contributors to CDF for nearly all B&W plants. This accident class involves events that cause the reactor to trip followed by failure to bring the reactor to safe shutdown. The sequences primarily involve failure to remove decay heat by either steam generator cooling or primary system feed-and-bleed cooling. Sequences with failure to replace reactor coolant inventory following an accident-induced LOCA (normally an RCP seal LOCA) are also important at some plants. Transients represent a broad category, covering both general initiators (such as reactor trip or loss of main feedwater) and support-system initiators (such as loss of SW or AC/DC bus).

The failures leading to core damage for transients are found to be quite plant-specific. However, some key factors affecting the CDF from transients are common among many of the submittals. These key factors involve plant-specific design and operating conditions, as well as the modeling of plant systems and operating history in the IPEs, as follows:

- Susceptibility to RCP seal LOCAs — Three B&W sites use Byron Jackson RCPs, which the industry believes are less susceptible to RCP seal LOCAs than other designs. The remaining two sites use either Westinghouse or Bingham International pumps. For the sites with Byron Jackson pumps, the IPEs only considered RCP seal LOCAs to be credible for sequences with both loss of all seal cooling and operator failure to trip the RCPs. The remaining sites include the possibility of RCP seal LOCAs even with the RCPs tripped, which increases the CDF for RCP seal LOCA sequences. The importance of RCP seal LOCAs was also influenced (to a lesser extent) by the availability of backup systems to cool the seals when the normal configuration fails (e.g., safe shutdown facility), and the use of different support systems for cooling the RCP seals than for providing injection (so that loss of a single system does not lead to an unmitigable LOCA).

- Feed-and-bleed cooling — Six B&W units have high-pressure injection pumps with shutoff heads above the SRV setpoint, and one plant (Davis-Besse) has a shutoff head that is near the SRV setpoint. Therefore, operator action is not necessary for feed-and-bleed cooling to succeed for most B&W plants; even if the operator does not open the PORVs, high-pressure injection will automatically actuate, and feed-and-bleed cooling will occur at the SRV setpoint. The Davis-Besse IPE submittal considers the SRV setpoints to be sufficiently uncertain that high-pressure injection might not be able to lift the SRVs; as a result, operator action will be necessary to open the PORV for cooling to succeed.

- SW and CCW dependencies — Most plants have a relatively high dependence of other plant systems on CCW and/or SW systems. Some B&W plants have relatively diverse SW/CCW systems, and have capabilities for cross-tieing systems. These plants generally show less vulnerability to failures in SW and CCW systems. However, loss of either of these support systems is still important to the overall transient CDF for B&W plants.

- Data — Some variability in the results is attributed to the treatment of plant data. Some licensees use data only from the most recent five years, while others included older plant operational data. The more recent data generally indicates fewer plant trips and better equipment reliability; as a result, incorporating older data into the failure frequencies gives higher values than limiting the data to more recent years. A related factor that influences the IPE results is that some licensees tend to use lower values for common cause failures, which reduces the CDF.

All B&W licensees are considering or implementing improvements to address the factors that contribute to the CDF from transients. Most licensees (ANO 1, Davis-Besse, Oconee 1,2&3, and TMI 1) are considering training enhancements or procedural modifications to reduce the probability of RCP seal LOCAs. The remaining improvements discussed in the IPE submittals are plant-specific operational and design changes. Some examples of these changes include replacing manual crossover valves between low-pressure injection pumps with MOVs that can be remotely operated from the control room, staggering high-pressure safety injection (HPSI) pump use during loss of SW events, and providing alternative ventilation to maintain SW operation with loss of HVAC.

LOCAs are important contributors to CDF for most of the plants in this group. The most common LOCA contributors are small LOCAs, but some plants are dominated by medium LOCAs. Small LOCAs are generally the larger contributors to CDF because they have a higher initiator frequency. The dominant contributor to core damage is ECCS failure during recirculation (after the RWST has been drained and the suction is being transferred to the containment sump). Recirculation involves realigning systems, and typically involves more components than required for the injection mode. In addition, operator action is required to perform this switchover at the B&W plants. The reliance on operator action and the complexity in switchover lead to a higher failure probability than for injection failures.

The LOCA CDF results are quite similar for the plants in this group; however, there is variability regarding the most important specific failures:

- Switchover to recirculation — The plants in this group require manual actions to initiate coolant recirculation, and are designed so that high-pressure system pumps draw suction from low-pressure system pumps during recirculation. Some plants have large RWSTs so that the switchover to recirculation is either not necessary or is significantly delayed, giving the operators more time to complete the necessary actions for the switchover to recirculation. The containment sprays also draw from the RWST during the injection mode; as a result, plants with spray actuation occurring at lower containment pressures deplete the RWST more quickly, giving less time for the operators to perform the switchover of ECCS to recirculation. Two submittals have a relatively large contribution from medium LOCAs because of the shorter time available (relative to small LOCAs) for the operators to perform the manual switchover.

- Low-pressure injection system characteristics and modeling — The most important system failures for LOCAs in B&W plants involve failures of the low-pressure injection system. The specific failure mode varies among the plants, but the most common failures involve common cause failure of pumps to start, common cause valve failures, and pump cooling failures.

The IPE submittals list fewer plant improvements related to LOCA concerns than are listed for transients. Davis-Besse and TMI 1 are considering procedural changes or training enhancements to improve the reliability of the switchover from injection to recirculation. In addition, licensees are considering improvements to extend the period of injection from the BWST at Davis-Besse, and to cope with simultaneous high-pressure injection suction valve failures at Oconee 1,2&3.

SBO accidents are important contributors to CDF for many B&W plants. SBO accidents involve an initial LOSP followed by failure of the emergency on-site AC power sources. The failure of AC power sources results in failure of all injection systems and failure of motor-driven AFW. This leaves only turbine-driven AFW available to cool the core and no systems available to provide injection to make up the loss through any RCP seal LOCAs that develop during the transient.

Generally, plant design and operational features have a larger impact on the SBO CDF than do modeling characteristics, but no single factor dominates. Combinations of factors are usually important, and those combinations vary from plant to plant. Overall, the most influential factors are as follows:

- Susceptibility to RCP seal LOCAs — Three B&W sites use Byron-Jackson RCPs, which the industry believes are less susceptible to RCP seal LOCAs than other designs. The remaining two sites use either Westinghouse or Bingham International pumps. As mentioned in the discussion of transients, the IPEs for plants with Byron-Jackson RCPs only consider RCP seal LOCAs to be credible if the RCPs are left running when seal cooling is lost. The RCPs will trip when off-site power is lost, so RCP seal LOCAs are not considered important for SBO sequences at these plants. The remaining sites (those with Bingham International or Westinghouse pumps) include the possibility of RCP seal LOCAs for SBO, which increases the SBO CDF. However, the availability of backup cooling from the safe shutdown facility at one of these plants (Oconee) reduces the impact of RCP seal LOCAs for SBO.

- Number of emergency AC power sources — Most B&W plants have two emergency diesel generators per unit to provide backup power when off-site power is lost. However, Oconee has a unique design in which backup power is provided by lines (hydroelectric and combustion turbine generator) from two other plants. In addition, Oconee has a standby shutdown facility with an independent diesel generator. Three Mile Island Unit 1 has the capability to cross-connect to the diesel generator at Unit 2. The additional redundancy in emergency AC power sources for these plants reduces the SBO CDF.

- Battery depletion time — Battery power is needed to provide AFW control for most plants. (Davis-Besse indicates that AFW can be controlled manually.) Thus, when the batteries are depleted, all cooling is lost and core damage follows. Battery depletion times range from 2 to 6 hours for this group, and the plants with the shortest battery life also have the highest SBO CDFs.

Two licensees indicated that they are considering or implementing plant improvements for SBO-related concerns. ANO 1 added an air-cooled diesel generator and additional battery capacity, while Davis-Besse is evaluating procedural changes to add redundancy to power supplies, provide for DC load shedding, and guide the replenishment of diesel fuel oil for long-term diesel use.

Internal flooding is important for one B&W plant. Internal flooding events involve rupture of water lines that result in a release of water that can directly cause the failure of required mitigating systems and/or other mitigating systems as a result of submergence or spraying of required components. The effects of internal flooding are highly plant-specific, depending on the layout of equipment within the plant and the relative isolation of rooms. Because of this diversity of design and layout, each plant has different vulnerabilities to flooding, and generic conclusions regarding flooding can not be drawn. All of the B&W plants have internal flooding contributions of less than 10% except Oconee 1,2&3. The internal flood contribution for Oconee is dominated by turbine building floods resulting from SW failures that submerge main and EFW pumps because the drains are not large enough to prevent water buildup.

ATWS is not a dominant contributor to CDF for any of the B&W plants. ATWS sequences involve a transient, followed by failure to shut down the nuclear chain reaction by inserting the control rods. Power generation continues at levels far in excess of normal decay heat. An ATWS sequence can be mitigated by pressure control and heat removal. Because of the low frequency of failure to scram, ATWS is a small contributor (less than 3% of CDF) for all the plants in this group. All of the licensees have installed a diverse scram system to comply with the ATWS rule; this system increases the scram probability and decreases the frequency of ATWS sequences.

SGTR is not a dominant contributor to CDF for any of the B&W plants. SGTR sequences involve leakage from the primary to the secondary through a ruptured steam generator tube, followed by either failure to mitigate the leak or failure to establish long-term core heat removal. SGTR accidents are a minor contributor to plant CDF at all B&W plants (less than 5% of CDF) because of the relatively low frequency of the rupture occurring and the relatively high probability of the operators isolating the rupture.

ISLOCAs are not dominant contributors to CDF for B&W plants. ISLOCAs, however, can be significant contributors to risk because the releases bypass containment. The low CDF contribution is attributed to the low frequency of the LOCA initiator combined with its negligible-to-minimal impact on other systems. This low impact results from the compartmentalization and separation of the equipment.

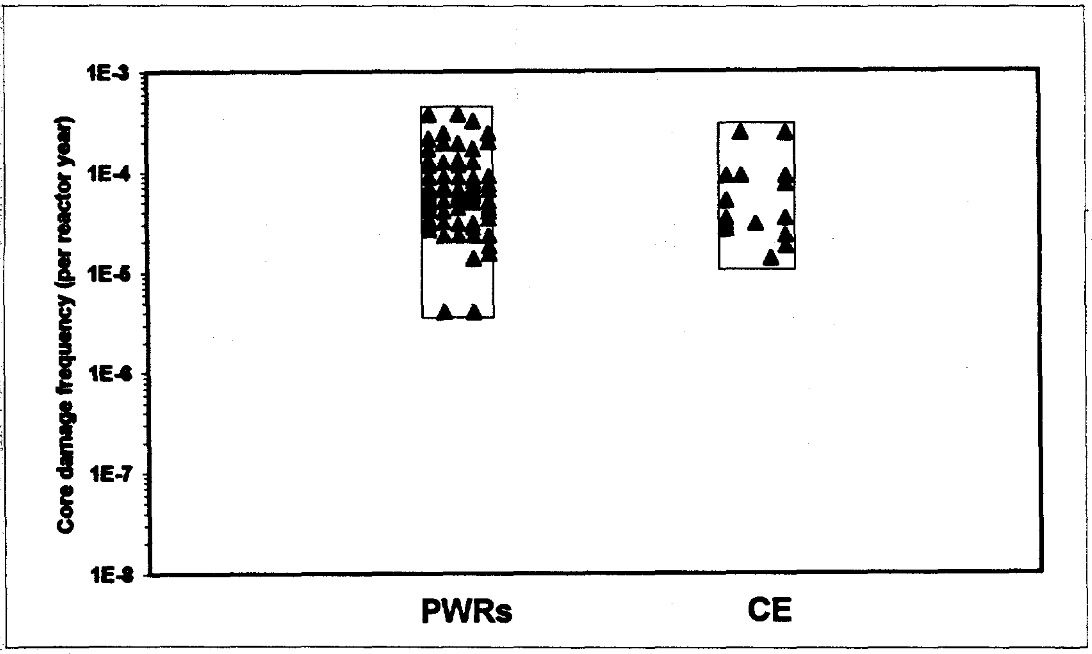
3.3.2 CE Plant Perspectives

Table 3.12 lists the CE plant group which includes 15 plant units (represented by 10 IPE submittals).

Table 3.12 Plants (per IPE submittal) in the CE group.

ANO 2	Calvert Cliffs 1&2
Fort Calhoun 1	Maine Yankee
Millstone 2	Palisades
Palo Verde 1,2&3	San Onofre 2&3
St. Lucie 1&2	Waterford 3

As shown in Figure 3.12, the CDFs for the CE plants as a group are comparable to the CDFs for the bulk of the PWRs. The importance of specific accident classes to the plant CDF varies from one plant to another, as shown in Figure 3.13.



- Support system failures are important because they defeat redundancy in front-line systems.
- Some CE plants without PORVs can feed-and-bleed using other valves.
- Some plants without feed-and-bleed cooling capability can compensate by depressurizing the steam generators to use condensate as an alternative to feedwater.
- LOCAs are important for most plants, but modeling and redundancy in feed and bleed capability reduces the importance for one plant (Fort Calhoun).

Figure 3.12 Reported IPE CDFs and key perspectives for the CE PWR plant group.

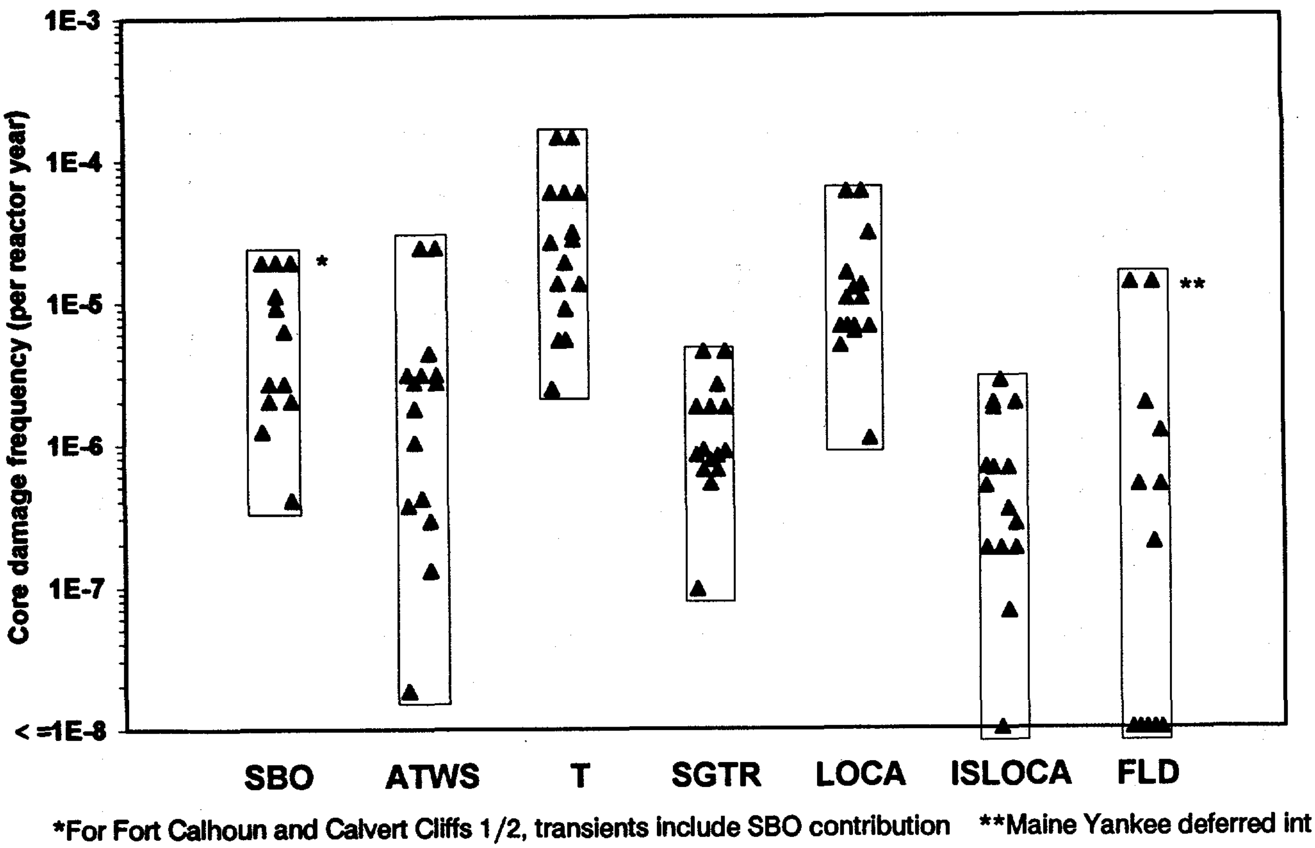


Figure 3.13 Reported IPE accident sequence CDFs for CE plants.

In general, based on the IPE submittals, the following classes of accidents are most important to CE plants:

- transients
- LOCAs
- SBO — the loss of all off-site and on-site AC power

Transients are generally considered to be important contributors to CDF and containment failure frequency because they involve relatively high initiating event frequencies coupled with support system failures that defeat the redundancy in systems available to mitigate potential accidents. LOCAs are important for these plants because there are generally even fewer systems or levels of redundancy available to mitigate such accidents than there are for many transients. SBO accidents are relatively important for CE plants because they leave few systems available to prevent core damage.

For two units (one IPE submittal), ATWS accidents were identified as somewhat important (10% or greater contributor to plant total CDF) mostly because of analysis assumptions. Only one licensee identified internal flooding accidents as being somewhat important (14% contributor to plant total CDF), and only one licensee identified ISLOCAs as contributing 10% or more to the total plant CDF. In these specific cases, a combination of analysis assumptions and plant-specific features caused these accidents to be somewhat important. None of the licensees found SGTR sequences to be important CDF contributors. SGTR and ISLOCAs are normally low contributors, in part because of the relatively low frequency of the initiating event and the relatively high probability of the operators isolating the rupture. Nonetheless, these types of accidents can be important contributors to risk since they bypass containment.

The variation in the reported IPE results is attributed to many factors, including plant-specific design features, modeling assumptions, and variation in data (including the probability of operator errors). These factors and the improvements under consideration by the licensees to address weaknesses are summarized in Table 3.13, and are discussed below for each important accident class. It should be noted that revised assessments by the licensees may result in different perspectives than those indicated in Table 3.13 and discussed in the remainder of this section.

Table 3.13 Summary of CDF perspectives for the CE plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
	Transient accidents	
Important for nearly all CE plants	Degree of high-pressure injection and AFW system dependence on support systems (SW, CCW, AC/DC bus, HVAC) Success criteria for feed-and-bleed (PORV capacity and redundancy, credit for alternative relief valves) Ability to depressurize the steam generators and use condensate for heat removal Lower susceptibility to RCP seal LOCAs because of RCP seal design Long-term makeup to AFW (water supply size, alternative supplies, procedures, operator training)	Improving procedures and training for dealing with loss of ventilation systems, particularly for switchgear, battery, and DC bus and inverter rooms Adding temperature alarms for switchgear, battery, and DC bus and inverter rooms Enhancing procedures and equipment for cross-tieing buses
	LOCAs	
Important for many CE plants	Automatic switchover to ECCS recirculation LOCA initiating event frequency	None identified
	SBO accidents	
Important for many CE plants	AC power reliability (number of diesel generators, cross-tie capabilities between units, diverse AC power sources) Diesel generator cooling dependencies Diesel generator reliability Battery life	Adding additional power sources Extending availability of long-term heat removal: • improving availability of alternative water supplies to AFW • use of firewater for steam generator cooling Adding portable generator for battery charging
	ATWS accidents	
Only important for one CE plant (2 units)	Assessment of the fraction of time that a plant has unfavorable moderator temperature coefficient	None identified

Table 3.13 Summary of CDF perspectives for the CE plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
Internal flood accidents		
Important for two CE sites (three units)	Plant layout (separation and compartmentalization of mitigating system components)	Revising procedure to maintain door open
SGTR accidents		
Not dominant for any CE plants	Modeling of operator actions to isolate the rupture and provide long-term heat removal	None identified
ISLOCAs		
Not important for any CE plants	Compartmentalization and separation of equipment	Adding valve checks or testing

Transients are important contributors to CDF for nearly all CE plants. This accident class involves events that cause the reactor to trip, followed by failure to bring the reactor to safe shutdown (either failure to remove decay heat or failure to replace reactor coolant inventory following an accident-induced LOCA). Transients represent a broad category, covering both general initiators (such as reactor trip or loss of main feedwater) as well as support-system initiators (such as loss of SW or AC/DC bus).

The failures leading to core damage for transients are quite plant-specific. However, some key factors in the IPEs collectively affect the CDF contribution from transients as well as the dominant types of transient scenarios (e.g., core damage during injection and recirculation). These involve plant-specific design and operating conditions, as well as the modeling assumptions made in the IPEs. The modeling issues are important to the results and, in some cases, represent a significant area of uncertainty:

- Degree of support system dependence — Variability in the support systems needed for AFW or EFW and HPSI is important to the overall transient CDF for CE plants. Dependencies on SW, CCW, specific AC/DC buses, and HVAC can be especially significant. For example, only half of the submittals model the need to cool the HPSI pumps for seal/bearing cooling during injection for feed-and-bleed cooling. Omission of this consideration from the other half of the submittals is typically based on a combination of analysis and assumptions. The Calvert Cliffs submittal, which covers two plant units, includes a dependence on HVAC for HPSI. Specific power bus loads and the extent to which these are shared among AFW/EFW, HPSI, and PORVs can also be important, as demonstrated by the significance of specific bus faults to the transient CDF for at least ANO-2, Calvert Cliffs, and Maine Yankee. The latter licensee even models dual bus faults as possible initiating events. The specific configurations vary considerably among the plants, however, and plants with less dependence on these supports and/or greater redundancy in their design tend to be more tolerant of transient initiators.

- Success criteria for feed-and-bleed — Some CE plants do not have PORVs (or alternative relief valves). As a result, they cannot use feed-and-bleed cooling or they do not have redundancy in the features needed for feed-and-bleed cooling, which limits it's effectiveness. For example, the Calvert Cliffs units require both PORVs for successful feed-and-bleed (their PORVs are small). In addition, the licensee for these units did not always credit feed-and-bleed as a possible heat removal system in response to a loss of main feedwater event because the time frame from the initiating event to the prompt for starting feed-and-bleed was believed to be too short (approximately 10 minutes). Maine Yankee, Palisades, and ANO-2 all have less redundant or unusual bleed designs. Plants with lower transient absolute CDF values generally (but not always) either have redundancy in the feed-and-bleed function, and/or take credit for both feed-and-bleed cooling and steam generator depressurization (discussed below).

- Ability to depressurize the steam generators and use condensate for heat removal — Some CE plants have the ability to use steam generator depressurization and heat removal with condensate whenever all feedwater (main and auxiliary) is lost. For example, the Palo Verde and San Onofre units do not have PORVs and so cannot use feed-and-bleed cooling. However, these units credit steam generator depressurization and use of condensate as a successful means of heat removal. Calvert Cliffs does not credit steam generator depressurization and use of condensate for heat removal because of the small single ADV for each steam generator. The ability to use steam generator depressurization and condensate for heat removal is plant-specific, and reduces the transient CDF for those IPEs that include this strategy.

- Susceptibility to RCP seal LOCAs — The industry generally considers susceptibility to RCP seal LOCAs to be lower for CE plants than for many other PWRs particulary when the RCPs are stopped. This assumption is based on the typical Byron- Jackson 4-stage seal design (or an equivalent) used in CE plants, and the analyses and actual tests performed by the CE Owner's Group regarding this issue. Because of these analyses and tests, all CE submittals use models that are considered realistic by the licensees for the RCP seal failure potential in CE plants. Nevertheless, some variability exists among the CE plant IPE submittals in the associated flow rate and probability of these LOCAs. In some cases, a small but possible chance for RCP seal LOCAs is considered in the IPE and is generally associated with the failure of the operator to trip the RCPs following loss of seal cooling.. In other cases, the IPE model does not consider the potential for RCP seal LOCAs. Since such events increase the demand for coolant injection during transient scenarios, the specific models used and their bases are a factor in the relative importance of transient scenarios.

- Long-term makeup to the AFW/EFW — The long-term ability to supply water to the suction for AFW/EFW and the associated actions are particularly significant for transients with no induced LOCA conditions. This source of water is required to ensure long-term heat removal until shutdown cooling can be initiated. Variability in water supply size, alternative supplies, procedures, and operator training all influence the relative importance of this characteristic.

Most licensees are considering or have already implemented improvements to address the factors that impact the CDF from transients. The most common plant improvements are aimed at increasing power reliability during LOSP scenarios. Many of these improvements can also benefit other types of accidents, especially those involving SBO. These improvements include adding additional power sources, improving procedures and training for dealing with the loss of ventilation systems (particularly for switchgear, battery, and DC bus and inverter rooms) adding temperature alarms for these rooms, enhancing procedures and equipment for cross-tieing buses, and extending the availability of long-term heat removal (such as improving availability of alternative water supplies to AFW or considering the use of firewater for steam generator cooling). In a few cases, these modifications have already been completed and are credited in the IPEs; at most plants, however, the modifications are either being installed or are still under consideration.

LOCAs are important contributors to CDF for many plants in this group. The most common LOCA contributors are small LOCAs, but all CE plant IPE submittals conclude that large or medium LOCAs have a relatively significant contribution. Small LOCAs are generally the larger contributors to CDF because they have a higher frequency and, if HPSI should fail, they cannot be directly mitigated using core flood tanks and low-pressure injection without manual depressurization. LOCAs are dominated by failures during either injection or recirculation, depending on the specific plant. The relative contribution from injection failures is generally higher for CE plants than for other PWR plant groups.

The plant features and assumptions mentioned above for transients also influence the specific contributions to LOCA core damage scenarios for CE plants and affect the relative importance of injection and recirculation accidents. Nevertheless, the absolute CDFs from LOCAs for the CE plants all fall within approximately one order of magnitude with the exception of one unit, Ft. Calhoun, which has the lowest LOCA-related CDF. Ft. Calhoun has redundancy in both its feed-and-bleed capabilities (even for very small LOCAs), and HPSI is not particularly susceptible to support system faults. However, perhaps the most noteworthy difference between Ft. Calhoun and the other CE plants is the relatively low LOCA initiating event frequencies reported in the Ft. Calhoun IPE submittal. In all of the LOCA categories (small, medium and large), these frequencies are generally 4 to 20 times lower than the frequencies reported in other CE plant IPEs. However, this factor alone can account for the differences in the LOCA CDF results.

None of the licensees indicated that plant improvements are being made specifically to address LOCA-related issues. However, the improvements for transients indicated above will somewhat affect the LOCA CDF to the extent that the overall reliability of heat removal is increased.

SBO accidents are important contributors to CDF for some of the CE plants. SBO accidents involve an initial LOSP followed by failure of the emergency on-site AC power sources. The failure of AC power results in failure of all injection systems and motor-driven AFW or EFW. This leaves only turbine-driven AFW/EFW available to cool the core and no systems available to provide injection to make up the loss through any RCP seal LOCAs or stuck-open primary relief valves that may develop during the transient.

Generally, plant design and operational features have a larger impact on the SBO CDF than do modeling characteristics, but no single factor dominates. Combinations of contributors are usually important, and those combinations vary from plant to plant. Overall, the two most influential factors appear to be as follows:

- Number of emergency AC power sources and their dependencies — This factor has a direct influence on the probability of a SBO, and thus is an important factor in the significance of core damage induced by SBOs. The number of emergency diesel generators (usually two per unit) directly affects the reliability of the emergency AC power system. Further, a few plants have a diverse AC power source, such as a gas turbine generator or swing or other diesels, and so are less susceptible to common cause failures of diesel generators. Configuration differences range from units with just one diesel per unit with a swing diesel between units (such as at Calvert Cliffs) to two diesels per unit with additional backup capability existing or being added (such as at Palo Verde). Calvert Cliffs is in the process of adding diesels to improve their on-site power reliability (not credited in the IPE). Additionally, the staff noted that the nature of diesel dependencies can somewhat influence the overall on-site reliability. For example, Ft. Calhoun and San Onofre use self-cooled diesels with a radiator design, making them less susceptible to cooling dependency faults. Maine Yankee has a backup diesel that uses firewater as the cooling source. Plant-specific operating history can also be important, accounting for the order-of-magnitude difference in diesel generator failure-to-start probabilities identified in the IPEs.

- Battery depletion time — Plants need battery power to provide plant status indication and control for the steam-driven AFW/EFW, including assurance of a long-term water supply for heat removal. The longer the battery life, the more likely it is to successfully provide heat removal using AFW/EFW while simultaneously increasing the chances of recovering AC power to eventually achieve a safe and stable plant state. Thus, for all plants in this group, when the batteries are depleted, all cooling is lost and core damage ensues. One notable exception involves the Palisades plant, which credits AFW operation for six hours beyond battery depletion on the basis of manual control and the nitrogen station capacity for keeping the AFW valves open. Battery depletion times range from two to eight hours for this group, with the longer times reflecting plants using load shedding. Plants with the longer credited battery depletion times tend to exhibit a lower SBO CDF.

The staff noted that the RCP seal failure models used in the majority of the CE plant IPEs do not assume seal leakage if the RCPs are tripped as would occur during an SBO. Thus, no RCP seal LOCA contributions to the SBO-related CDF were calculated for these plants (the exceptions being Calvert Cliffs 1&2 and possibly Palo Verde 1,2&3). The lack of an RCP seal LOCA minimizes the need for high-pressure injection during a SBO, increases the significance of the continued operability of just the steam-driven AFW/EFW, and increases the significance of the battery depletion time.

Other features that might at first be thought to be important to SBO are in fact not so important. For instance, a low or high initiator frequency for off-site power does not necessarily correspond to an associated low or high SBO CDF. In fact, in some cases, the opposite result occurs. Since all the plants in this group rely on a steam-driven AFW/EFW train to provide heat removal during a SBO, variability in the availability and reliability of this equipment among the plants will be expected to be important to plant CDF. However, differences in the failure data for this equipment are not very large, making this variability a relatively small factor compared with the others discussed above.

The plant improvements related to increasing the reliability of AC power that are discussed above for transients are also beneficial for SBO accidents. In addition, one licensee (Waterford), indicated that they would assess the benefit of adding a portable generator for battery charging.

ATWS is not a dominant contributor for CE plants. ATWS sequences involve a transient, followed by failure to shut down the nuclear chain reaction by inserting the control rods. Power generation continues at levels far in excess of normal decay heat. An ATWS sequence can be mitigated by pressure control, boration, and heat removal. Because of the low frequency of failure to scram and the credit taken for the above mitigating features, ATWS is a relatively low contributor to CDF (less than 5E-6/ry) for all but one plant in this group (Calvert Cliffs). In that single exception, the higher ATWS contribution reflects the analysis assumptions and the assessment that the plant has an unfavorable moderator temperature coefficient for a large fraction of the time (40%). The result is an analysis that does not credit the mitigating features mentioned above, assuming instead that an ATWS leads directly to core damage. This pessimistic stance leads to an ATWS CDF that is a factor of six greater than the next highest plant ATWS CDF among the CE plants.

Internal flooding is generally not important for CE plants. Internal flooding events involve ruptures of water or steam lines resulting in water or steam releases that can directly cause the failure of required mitigating systems or cause their failure as a result of submergence or spraying of required components. The effects of internal flooding are highly plant-specific, depending on the layout of equipment within the plant and the relative isolation of rooms. Because of this diversity in design and layout, each plant has different vulnerabilities to internal flooding events. Nevertheless, based on screening methods, all plants in this group report CDFs from internal flooding in the range of 1E-6/ry to 1E-7/ry (or even lower) except for Calvert Cliffs and Ft. Calhoun. Calvert

Cliffs performed a more rigorous analysis that accumulated the combined effects of nearly a dozen flood scenarios (rather than using a series of individual screening arguments) that collectively contribute to an internal flood CDF of about 1.5E-5/ry (a 6% contribution to total CDF). It is unclear whether the flooding contribution will be significantly higher for other plants if they use the screening approach employed by Calvert Cliffs. For Ft. Calhoun, the flood-related CDF (about 2E-6/ry) represents about 14% of the total plant CDF. As a result, the licensee is implementing plant improvements to address this issue.

SGTR is not a dominant contributor to CDF for CE plants. SGTR sequences involve leakage from the primary to the secondary through a ruptured steam generator tube. Core damage results either from failure to mitigate the leak or failure to establish coolant makeup and long-term core heat removal. Relative to other contributors, SGTR sequences are minor contributors to plant CDFs at CE plants (less than 5E-6/ry) because of the low frequency of the rupture occurring, coupled with credit taken for operator actions and equipment used to mitigate the event. Although minor, the contribution to CDF is highly dependent on the assessed failure probabilities associated with human actions designed to mitigate such events. Although this is an area of higher uncertainties, it does not have a significant impact on the overall plant CDF. However, SGTR can contribute significantly to risk because the releases bypass containment.

ISLOCAs are not dominant contributors to CDF for CE plants. ISLOCAs, however, can be significant contributors to risk because the releases bypass containment. The low CDF contribution results from the low frequency of the LOCA initiator, combined with its negligible-to-minimal impact on other systems. This low impact is attributed to the compartmentalization and separation of the equipment.

3.3.3 Westinghouse 2-Loop Perspectives

Table 3.14 lists the Westinghouse 2-loop plant group, which includes 6 plant units (represented by 4 IPE submittals).

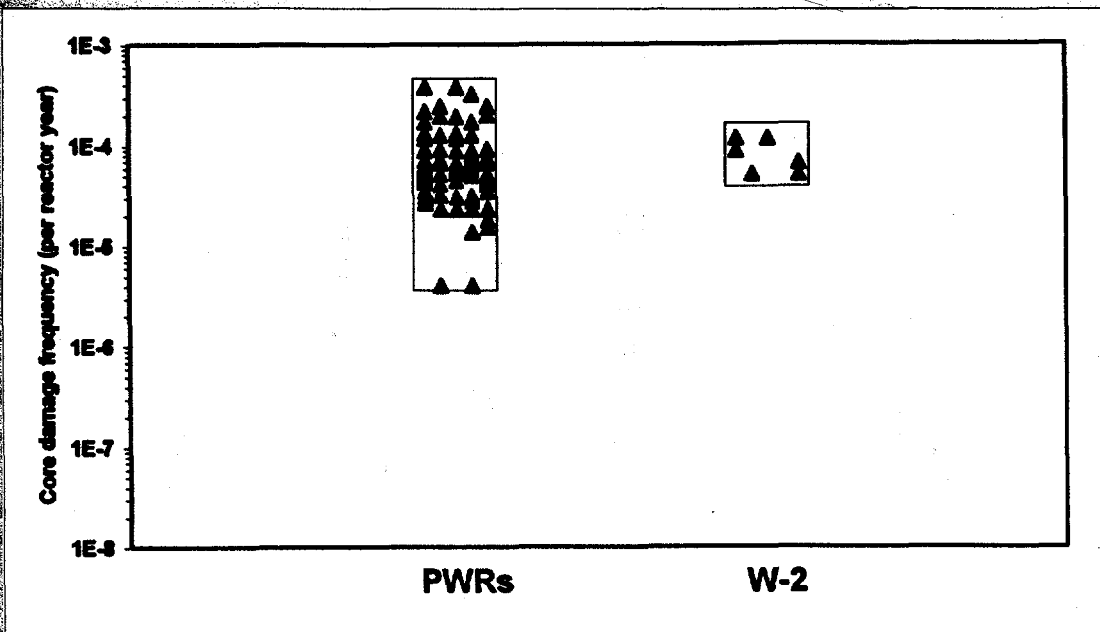
Table 3.14 Plants (per IPE submittal) in the Westinghouse 2-loop group.

Ginna	Kewaunee
Point Beach 1&2	Prairie Island 1&2

As a group, the Westinghouse 2-loop plants have CDFs in the middle of the range spanned by all PWRs, as illustrated in Figure 3.14. The importance of specific accident classes to CDF varies significantly from plant to plant, as shown in Figure 3.15. However, the following accident classes are most important for the Westinghouse 2-loop plants:

- transients
- LOCAs
- SBO — the loss of all off-site and on-site AC power
- SGTRs

In general, transients are important contributors to CDF and containment failure frequency because they involve relatively high initiating event frequencies coupled with support system failures that defeat the redundancy in systems available to mitigate potential accidents. LOCAs are more important for these PWR plants than for BWR plants (see Section 3.2) because fewer systems are available in the PWRs to provide low-pressure injection. SBO accidents are important for some Westinghouse 2-loop plants because they leave few systems available to prevent core damage. Half of the Westinghouse 2-loop IPE submittals had contributions from SGTR sequences of less than 10%, while the other half had contributions greater than 10%.



- Support system failures are important because they defeat redundancy in front-line systems.
- Relatively high dependence of front-line systems on SW and CCW increases CDF for transients.
- A substantial contribution to transients arises from instrument air failures that fail PORVs.
- LOCA CDF is increased by the requirement for manual actions to switch to ECCS recirculation.

Figure 3.14 Reported IPE CDFs and key perspectives for the Westinghouse 2-loop PWR plant group.

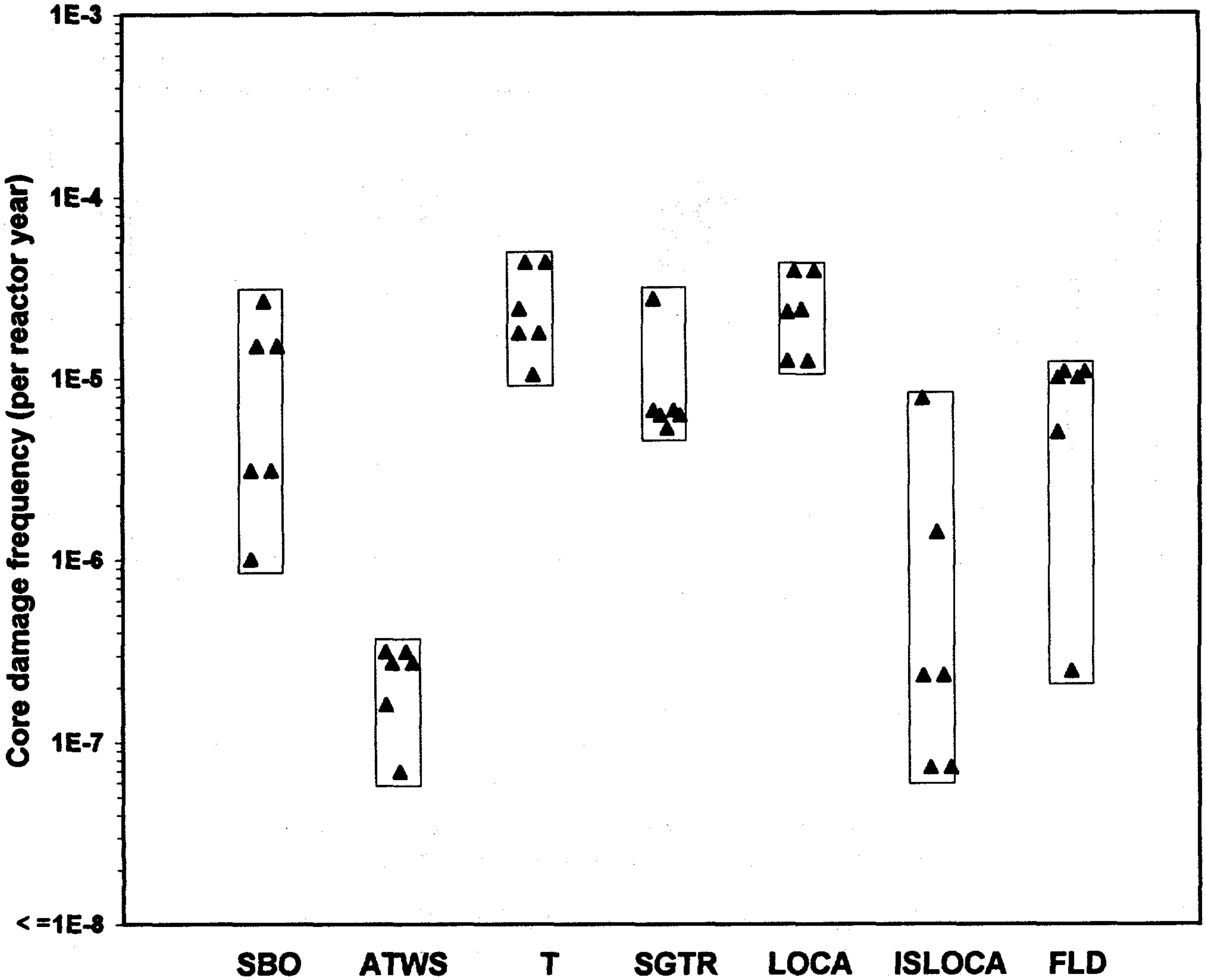


Figure 3.15 Reported IPE accident sequence CDFs for Westinghouse 2-loop plants.

One licensee identified internal flooding accidents as being important. All of the plants had ATWS contributions of less than 1%, and all of the plants had ISLOCA contributions of less than 10%. These accidents are normally low contributors because of the low frequency of the initiating event and the failure to scram. Although ISLOCAs are generally found to be low contributors to CDF, they can be important risk contributors since the releases bypass containment.

The variation in the reported IPE results is attributed to many factors including plant-specific design features, modeling assumptions, and use of data (including the probability of operator errors). These factors and the improvements being considered by the plants to address weaknesses are summarized in Table 3.15, and are discussed below for each important accident class. It should be noted that revised assessments by the licensees may result in different perspectives than those indicated in Table 3.15 and discussed in the remainder of this section.

Table 3.15 Summary of CDF perspectives for the Westinghouse 2-loop plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
	Transient accidents	
Important for all Westinghouse 2-loop plants	Degree of high-pressure injection and AFW system dependence on SW and CCW RCP seal cooling is not dependent on SW or CCW, except in one plant Modeling of RCP seal LOCA probability and size Dependence of PORVs on IA	Upgrades related to IA: • adding cross-ties • reducing IA dependencies on other support systems • changing valve failure modes to avoid water diversion upon loss of IA
	LOCAs	
Important for all Westinghouse 2-loop plants	Manual actions needed for switchover to ECCS recirculation Size of RWST (smaller tanks give less time for operators to perform the switchover) Whether containment sprays will be actuated (causing more rapid depletion of the RWST) Assessment of human error probability for performing switchover to recirculation	Revised hardware and procedures needed for recirculation Revised training programs for feed-and-bleed procedures Revised technical specifications to align ECCS pumps to the RWST rather than boric acid storage tank while in standby
	SBO accidents	
Important for some Westinghouse 2-loop plants	AC power reliability (number of diesel generators, cross-tie capabilities between units, diverse AC power sources) Diesel generator reliability Frequency of LOSP Modeling of RCP seal LOCA probability and size Backup cooling for RCP seals	Added or upgraded diesel generators or gas turbine generators Upgraded power distribution Added batteries Improved reliability of turbine-driven AFW: • connection to use firewater for cooling AFW pumps • equipment modifications to improve reliability of AFW pumps Provided alternative RCP seal cooling from technical support center diesel generator
	SGTR accidents	
Important for one Westinghouse 2-loop plant	Modeling of operator actions to isolate the rupture and provide long-term heat removal	Modified procedures for coping with SGTRs

Table 3.15 Summary of CDF perspectives for the Westinghouse 2-loop plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
	Internal flood accidents	
Important for some Westinghouse 2-loop plants	Plant layout (separation and compartmentalization of mitigating system components)	Changes to procedures for identifying and coping with floods (e.g., maintaining doors closed, or revising procedures to isolate leaks) Changes to plant layout (e.g., changing swing direction of doors)
	ATWS accidents	
Not dominant for Westinghouse 2-loop plants	Assessment of the fraction of time that the plant has an unfavorable moderator temperature coefficient	None identified
	ISLOCAs	
Not important for Westinghouse 2-loop plants	Compartmentalization and separation of equipment	Leak testing for isolation valves Procedure modifications for identifying and mitigating ISLOCAs

Transients are important contributors to CDF for all Westinghouse 2-loop plants. This accident class involves events that cause the reactor to trip followed by failure to bring the reactor to safe shutdown (either failure to remove decay heat or failure to replace the reactor coolant inventory following a transient-induced LOCA, normally an RCP seal LOCA). Transients represent a broad category, covering both general initiators (such as reactor trip or loss of main feedwater) as well as support-system initiators (such as loss of SW or AC/DC bus).

The failures leading to core damage for transients are quite plant-specific. However, some factors that affect the CDF from transients are common among many of the submittals. The key factors involve plant-specific design and operating conditions as well as the assumptions made in the IPEs. The modeling assumptions regarding RCP seal LOCAs affect the results and represent an important area of uncertainty. These assumptions and other important factors are as follows:

- SW and CCW dependencies — The Westinghouse 2-loop plants have a relatively high dependence of other plant systems on CCW and/or SW. The configurations vary considerably among the plants, however. For example, the motor-driven AFW pumps depend on SW for cooling at Point Beach and Ginna, but are independent of cooling water (except as an alternative water source) for Prairie Island. An important cooling water dependency for the Westinghouse 2-loop plants is the configuration for cooling the RCP seals and the charging and HPSI pumps, which is discussed below. Because of the relatively high dependence on CCW and/or SW, loss of either of these support systems is important to the overall transient CDF, but the contribution is generally lower than for the other Westinghouse plant groups.

- RCP seal cooling — RCP seal LOCAs do not contribute as much to the CDF for Westinghouse 2-loop plants as for other Westinghouse plant groups because of plant design characteristics that reduce the associated threat and because of modeling assumptions used in the IPEs (see the discussion below). However, the contribution from RCP seal LOCAs is still significant. Charging pumps, which are separate from HPSI pumps, can cool the RCP seals for all Westinghouse 2-loop plants. These charging pumps are air-cooled at all Westinghouse 2-loop plants, but the Prairie Island charging pumps indirectly depend on SW because SW failure results in a delayed failure of AC to the charging pumps. On average, this group of Westinghouse plants is less dependent on CCW and SW of RCP seal cooling, thereby reducing the contribution from transients with RCP seal LOCAs relative to other Westinghouse plant groups. Other design factors that affect the frequency of RCP seal LOCAs include the availability of backup systems (e.g., technical support center diesel generator for Kewaunee and Ginna, or CCW cross-ties at Prairie Island) to cool the seals when the normal configuration fails.

- RCP seal LOCA model — Most of the Westinghouse 2-loop licensees used the Westinghouse seal LOCA model, which tends to give lower probabilities of seal failures and lower leak rates than other models used in IPEs for Westinghouse plants. However, implementation of the Westinghouse model varied somewhat among the Westinghouse 2-loop plants, and this variation had a significant impact on the results. For example, Prairie Island used a relatively low leak rate from the RCPs for cases with seal LOCAs, giving considerable time for recovery of failed equipment.

- IA dependency — Although not generally the dominant contributor, IA failures are substantial contributors to transients for the Westinghouse 2-loop plants. The pressurizer PORVs rely on IA for all plants in this group. As a result, loss of IA removes the capability to use feed-and-bleed cooling. In addition, the unavailability of the PORVs to relieve RCS pressurization imposes more demands on the SRVs, thereby increasing the probability of stuck-open relief valves. The Ginna IPE shows the greatest dependence on IA, with about one-quarter of the plant CDF initiating with IA failure.

All of the licensees are considering or have implemented improvements to address the factors that increase the transient-related CDF. Most of the IPEs addressed improvements aimed at increasing the reliability of AC power during LOSP scenarios (e.g., adding or upgrading diesel generators or gas turbine generator, upgrading power distribution, and adding batteries). About half of the plants addressed upgrades related to IA (e.g., adding cross-ties, reducing IA dependencies on other support systems, or changing valve failure modes to avoid water diversion upon loss of IA). All of the licensees also considered improvements to address plant-specific issues. At some plants, these modifications have been completed and were credited in the IPEs; at other plants, the modifications are still under consideration.

LOCAs are important contributors to CDF for all plants in this group. The LOCA contributions for the Westinghouse 2-loop plants are generally dominated by small LOCAs, because of the higher small LOCA initiator frequency. The dominant contributor to core damage for LOCAs is ECCS failure during recirculation. Recirculation involves realigning systems, and typically involves more components than required for the injection mode; this complexity leads to a higher failure probability. In addition, the Westinghouse 2-loop plants require operator action to align the systems for recirculation; this also increases the probability of failure.

The IPE submittals for the Westinghouse 2-loop plants did not show a large variability in the contribution from LOCAs. The relatively small variability observed among the plants, as well as the difference in contribution relative to other Westinghouse plants are primarily related to the switchover of ECCS from injection to recirculation. Generally, plant design and operational features had about the same impact on the LOCA CDF as did modeling characteristics. Overall, the most influential factors are as follows:

- Switchover to recirculation — All of the Westinghouse 2-loop plants require manual actions to initiate recirculation. This need for manual action increases the CDFs, particularly for large and medium LOCAs in which there is less time available for the operators to perform the actions. For small LOCAs that remain at RCS pressures above the shutoff head of the low-pressure injection pumps, high-pressure injection and recirculation is needed. The Westinghouse 2-loop plants achieve high-pressure recirculation by aligning the high-pressure pumps to draw suction from low-pressure systems during recirculation. The time available for performing the realignment varies among the plants because of factors such as the size of the RWST, and whether the containment sprays will be actuated for the LOCAs (causing more rapid depletion of the RWST). These factors, as well as the assessed probabilities of the operator successfully performing the alignment affect the reliability of the switchover from injection to recirculation. For example, Point Beach had the greatest large LOCA contribution to CDF for the Westinghouse 2-loop plants, primarily because the Point Beach licensee based the IPE on a pessimistic value for operator failure to perform the switchover.

Most licensees indicated that plant improvements are being made to address LOCA issues. These improvements involve hardware and procedures needed for recirculation, revised training programs for feed-and-bleed procedures, and revised technical specifications to align ECCS pumps to the RWST (rather than the boric acid storage tank) while in standby.

SBO accidents are important contributors to CDF for some Westinghouse 2-loop plants. SBO accidents involve an initial LOSP followed by failure of the emergency on-site AC power sources. This failure of AC power sources results in failure of all injection systems and failure of the motor-driven AFW. This leaves only turbine-driven AFW available to cool the core and no systems available to provide injection to make up the loss through any RCP seal LOCAs that develop during the transient.

Generally, plant design and operational features have a greater impact on the SBO CDF than do modeling characteristics, but no single factor dominates. Combinations of contributors are usually important, and those combinations vary from plant to plant. Overall, the most influential factors are those discussed below:

- Number of emergency AC power sources — The number of emergency diesel generators directly affects the reliability of the emergency AC power system, and varies among the four Westinghouse 2-loop sites. The two single-unit sites each have two emergency diesel generators. Of the dual-unit sites, one has two emergency diesel generators both of which are shared between units, and the other has two emergency diesel generators per unit which can be cross-tied. In addition, one dual-unit site (Point Beach) has a gas turbine generator available to provide emergency power. Technical support center diesel generators are available at the two single-unit sites, and these diesel generators can power the charging pumps to cool the RCP seals. This relatively high availability of emergency power at all of the Westinghouse 2-loop plants contributes to the lower contribution from SBO relative to the other Westinghouse plants.

- Plant operating data — The plant data for Ginna and Prairie Island indicate low LOSP frequencies and high reliability of the emergency diesel generators, respectively. This is a key reason that these two plants have the lowest SBO contributions for the Westinghouse 2-loop plant group.

- Modeling of RCP seal LOCAs — Because seal cooling is lost during a SBO, RCP seal LOCAs are important for most Westinghouse plants. As noted in the discussion of transients above, most of the Westinghouse 2-loop IPEs used the Westinghouse model for characterizing RCP seal LOCA frequency and leak rate. As a result, the Westinghouse 2-loop plants have a lower overall contribution from RCP seal LOCAs than other Westinghouse plant groups. However, the implementation of the seal LOCA model varied somewhat among the submittals, introducing some variability into the results.

- Backup cooling for RCP seals — Some plants have backup capabilities for seal cooling (e.g., use of the technical support center diesel generator to power the charging pumps at Kewaunee and Ginna). This capability reduces the frequency of SBO sequences with RCP seal LOCAs.

The plant improvements related to increasing the reliability of AC power that are discussed above for transients are also beneficial for SBO accidents. In addition, some licensees indicated that plant improvements are under consideration or have been implemented to enhance the reliability of turbine-driven AFW (e.g., providing connection to use firewater for cooling the turbine-driven AFW pumps, or equipment modifications to improve the reliability of the pumps), or to provide alternative RCP seal cooling during SBO accidents from the technical support center diesel generator.

SGTR is an important contributor to CDF for one of the Westinghouse 2-loop plants. SGTR sequences involve leakage from the primary to the secondary through a ruptured steam generator tube, followed by failure to either mitigate the leak or establish long-term core heat removal. The SGTR accident class is an unusually high contributor to plant CDF for one Westinghouse 2-loop plant (Ginna). Although lower than Ginna, the other Westinghouse 2-loop plants have SGTR CDFs that are generally higher than for the other PWRs. Because the releases bypass containment, the impact on risk results is also important for these plants.

The SGTR contribution to CDF is primarily driven by the treatment of operator actions. These accidents require considerable operator involvement, with procedures that vary among the plants. In addition, modeling of operator errors and strategies to respond to the SGTR varies considerably among the IPEs. The high SGTR contribution at Ginna appears to be primarily an artifact of the stringent success criteria applied for inventory control. However, the licensee indicated that, as a result of the IPE, the procedures for coping with SGTRs are being evaluated for possible improvements.

Internal flooding is important for some Westinghouse 2-loop plants. Internal flooding events involve rupture of water lines that result in a release of water that can directly cause the failure of required mitigating systems and/or other mitigating systems because of submergence or spraying of required components. The effects of internal flooding are highly plant-specific, depending on the layout of equipment within the plant and the relative isolation of rooms. Because of this design and layout diversity, each plant has different vulnerabilities to flooding, and generic conclusions regarding flooding can not be drawn. The plants with the largest flood contributions typically were dominated by floods that affected support systems such as electric power and SW, which have plant-specific designs.

Most licensees indicated that plant improvements are under consideration or have been implemented to reduce the potential for core damage from flood scenarios. These improvements include changes to procedures for identifying and coping with floods (e.g., maintaining doors closed, or revising procedures to isolate leaks), as well as changes to plant layout (e.g., changing the swing direction of doors).

ATWS is not a dominant contributor for any of the Westinghouse 2-loop plants. ATWS sequences involve a transient, followed by failure to shut down the nuclear chain reaction by inserting the control rods. Power generation continues at levels far in excess of normal decay heat. An ATWS sequence can be mitigated by pressure control and heat removal. Because of the low frequency of failure to scram, and because none of the plants in this group operate with PORV block valves closed, ATWS is a low contributor (less than 1% of CDF) for all of these plants.

ISLOCAs are not dominant contributors to CDF for the Westinghouse 2-loop plants. ISLOCAs, however, can be significant contributors to risk because the releases bypass containment. The low CDF contribution results from the low frequency of the LOCA initiator, combined with its negligible-to-minimal impact on other systems. This low impact is attributed to the compartmentalization and separation of the equipment.

3.3.4 Westinghouse 3-loop Perspectives

Table 3.16 lists the Westinghouse 3-loop plant group, which includes thirteen plant units (represented by 9 IPE submittals).

Table 3.16 Plants (per IPE submittal) in the Westinghouse 3-loop group.

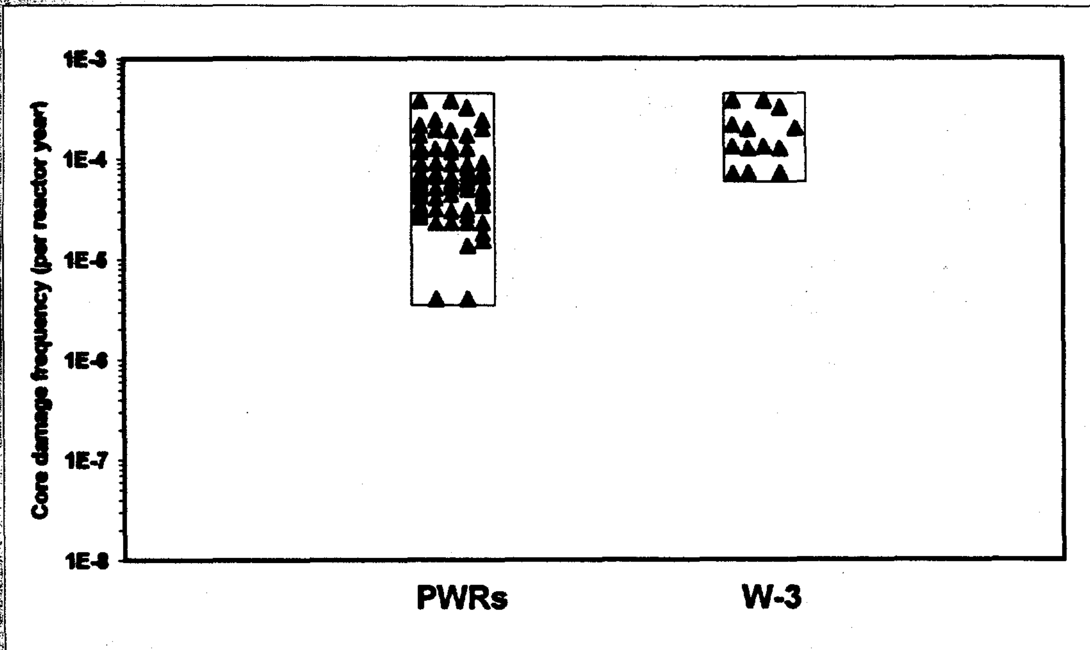
Beaver Valley 1	Beaver Valley 2
Farley 1&2	North Anna 1&2
Robinson 2	Shearon Harris 1
Summer	Surry 1&2
Turkey Point 3&4	

As a group, the Westinghouse 3-loop plants tend to have the highest plant CDFs of the PWRs, as illustrated in Figure 3.16. The importance of specific accident classes to CDF varies significantly from plant to plant, as shown in Figure 3.17. However, the following accident classes are important for many of these plants:

- transients
- LOCAs
- SBO — the loss of all off-site and on-site AC power

In general, transients are important contributors to CDF and containment failure frequency because they involve relatively high initiating event frequencies coupled with support system failures that defeat the redundancy in systems available to mitigate potential accidents. LOCAs are more important for these and most other PWR plants than for BWR plants (see Section 3.2) because fewer systems are available in the PWRs to provide low-pressure injection. SBO accidents are relatively important for the Westinghouse 3-loop plants because they leave few systems available to prevent core damage.

A few licensees identified internal flooding accidents as being important. All but one of the plants had ATWS contributions of less than 10%, and all of the plants have SGTR sequences of 10% or less. All of the licensees reported CDFs from ISLOCAs that are less than 3% of the plant CDF. These accidents are normally low contributors because of the low frequency of the initiating event and the relatively high probability of the operators isolating a steam generator with ruptured tubes. Nonetheless, SGTR and ISLOCAs can be important risk contributors since the releases bypass containment. The variation in the reported IPE results is attributed to many factors, including plant-specific design features, modeling assumptions, and use of data (including the probability of operator errors). These factors and the improvements being considered by the plants to address weaknesses are summarized in Table 3.17, and are discussed below for each important accident class. It should be noted that revised assessments by the licensees may result in different perspectives than those indicated in Table 3.17 and discussed in the remainder of this section.



- Support system failures are important because they defeat redundancy in front-line systems.
- Relatively high dependence of front-line systems on SW, CCW, and HVAC increases CDF for transients.
- RCP seal LOCAs are higher contributors to transients for Westinghouse three-loop plants than for other Westinghouse plant groups because of generally higher dependence on CCW and modeling assumptions.
- LOCA CDF is increased for most plants by the requirement for manual actions to switch to ECCS recirculation at those plants.
- Variability in SBO is primarily driven by redundancy of emergency AC power sources and modeling of RCP seal LOCAs.

Figure 3.16 Reported IPE CDFs and key perspectives for the Westinghouse 3-loop PWR plant group.

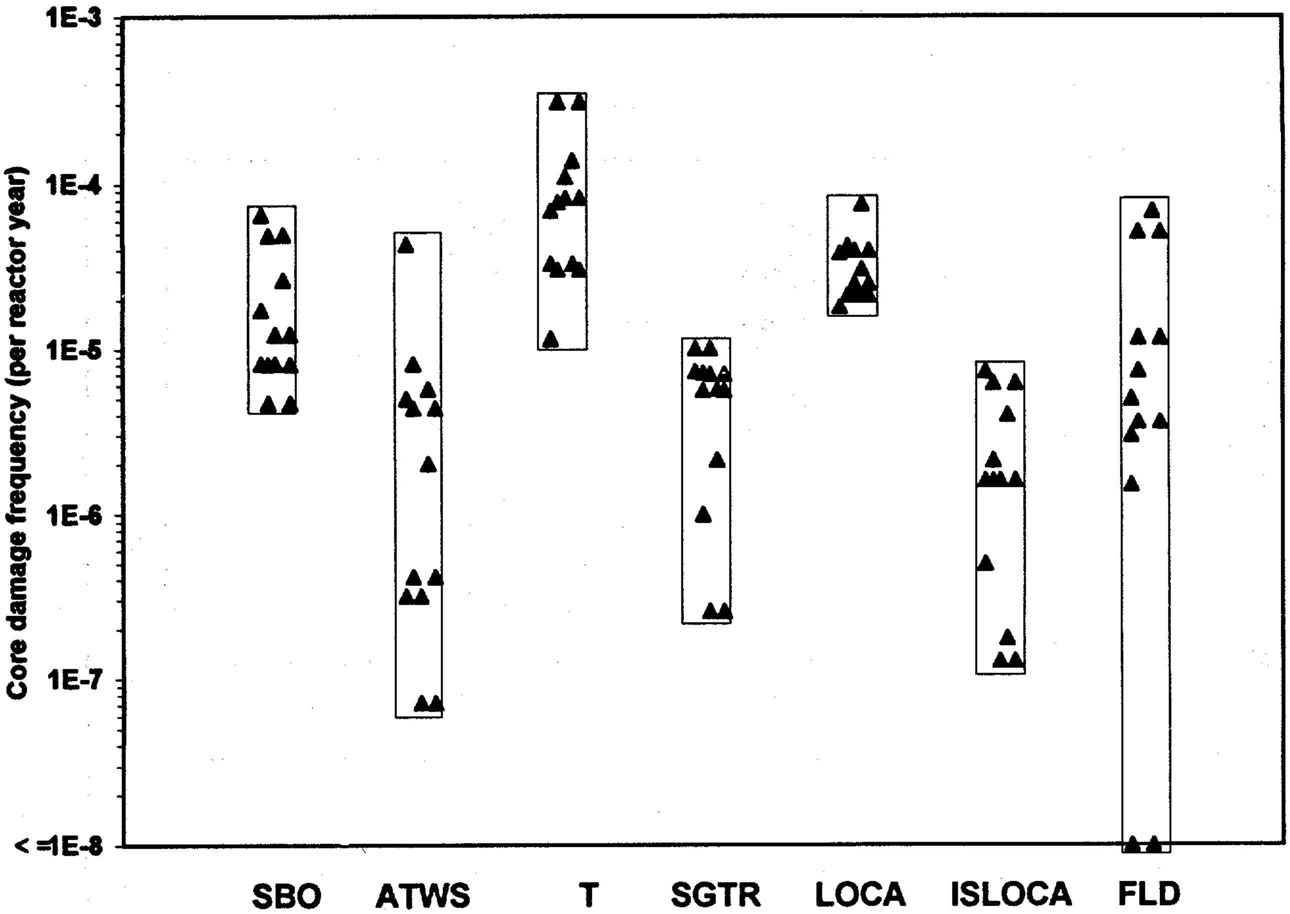


Figure 3.17 Reported IPE accident sequences CDFs for Westinghouse 3-loop plants.

Table 3.17 Summary of CDF perspectives for the Westinghouse 3-loop plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
	Transient Accidents	
Important for all Westinghouse 3-loop plants	Degree of high-pressure injection and AFW system dependence on SW and CCW Dependence of RCP seal cooling on SW or CCW Modeling of RCP seal LOCA probability and size Dependence on HVAC (particularly switchgear)	Providing backup cooling for charging pumps so that RCP seal cooling can be maintained Replacing RCP seals with the newer temperature-resistant design

Table 3.17 Summary of CDF perspectives for the Westinghouse 3-loop plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
	LOCAs	
Important for most Westinghouse 3-loop plants	Most plants have automatic switchover for low-pressure recirculation, but require manual actions for switchover to high-pressure recirculation Ability to depressurize the RCS by aggressively cooling down using steam generator ADVs so low-pressure injection can be used when high-pressure injection fails Ability to refill the RWST	Revising hardware and procedures needed for recirculation Changing feed-and-bleed procedures
	SBO accidents	
Important for many Westinghouse 3-loop plants	AC power reliability (number of diesel generators, cross-tie capabilities between units, diverse AC power sources) Battery life Modeling of RCP seal LOCA probability and size Backup cooling for RCP seals Dependence of switchgear on HVAC	Adding diesel generators Improving diesel generator reliability and maintenance programs Adding procedures for cross-tieing buses Improving load shedding procedures Providing independent battery charging capability Providing alternative RCP seal cooling during SBO
	ATWS accidents	
Not important for most Westinghouse 3-loop plants	Plant operation with PORV block valves closed	Implementing modifications to remove power to the bus if the reactor trip breaker fails
	Internal flood accidents	
Important for some Westinghouse 3-loop sites	Plant layout (separation and compartmentalization of mitigating system components)	Changing procedures for identifying and coping with floods Changing plant layout (e.g., adding dikes, water-tight doors, or backflow prevention devices in drains)

Table 3.17 Summary of CDF perspectives for the Westinghouse 3-loop plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
	SGTR accidents	
Not important for Westinghouse 3-loop plants	Modeling of operator actions to isolate the rupture and provide long-term heat removal	Implementing procedures to isolate steam generator with ruptured tube Implementing procedures to depressurize steam generators when high-pressure injection fails
	ISLOCAs	
Not important for any Westinghouse 3-loop plant	Compartmentalization and separation of equipment	Implementing procedures and training to enhance operator response to ISLOCAs

Transients are important contributors to CDF for all Westinghouse 3-loop plants. This accident class involves events that cause the reactor to trip followed by failure to bring the reactor to safe shutdown. This may involve either failure to remove decay heat or failure to replace reactor coolant inventory following an accident-induced LOCA, normally an RCP seal LOCA. Transients represent a broad category, covering both general initiators (such as reactor trip or loss of main feedwater) and support-system initiators (such as loss of SW or AC/DC buses).

The failures leading to transient-related core damage are quite plant-specific. However, some key factors affecting the transient-related CDF are common among many of the submittals. These key factors involve plant-specific design and operating conditions, as well as the assumptions made in the IPEs. The modeling assumptions regarding RCP seal LOCAs affect the results and represent an important area of uncertainty. The key factors are identified and discussed below:

- SW and CCW dependencies — At most of the plants, there is a relatively high dependence of other plant systems on CCW and/or SW. The configurations vary considerably among the plants, however. For example, motor-driven AFW requires SW cooling at H.B. Robinson, while the Shearon Harris pumps are self-cooled. Particularly important for the Westinghouse 3-loop plants is the configuration for cooling the RCP seals and charging/HPSI pumps, which is further discussed below. Because of the relatively high dependence on CCW and/or SW, loss of either of these support systems is important to the overall transient-related CDF.

- Susceptibility to RCP seal LOCAs — For some plants, the importance of RCP seal LOCAs is reduced because of plant design characteristics that reduce the associated threat. These include the availability of backup systems (e.g., firewater or cross-ties to a second unit at the site) to cool the seals when the normal configuration fails. They also include the use of different support systems to cool the RCP seals so that loss of a single system does not lead to an unmitigable LOCA. The modeling of RCP seal LOCAs varied considerably among the IPEs, with some using low leakage probabilities and leak rates, while others used much higher values for both parameters. This variability has a significant impact on the results. For example, the North

Anna IPE indicated a low contribution from sequences involving RCP seal LOCAs, primarily because seal leakage was assumed to be minimal if the operators are able to successfully depressurize the primary system. At the other extreme, the dominant transient sequence at Turkey Point involved loss of CCW that induced the failure of both high-pressure injection pumps and cooling to the RCP seals, leading to an RCP seal LOCA that cannot be mitigated.

- HVAC dependency — The dependency of systems on HVAC for the Westinghouse 3-loop plants varied between the extremes of HVAC not being needed to HVAC being required for key systems such as electrical switchgear. This variation affects the transient-related CDF for this group, with dependency on HVAC increasing the CDFs for the affected plants. For example, Surry, H.B. Robinson, and Beaver Valley have significant contributions to CDF from transients with failure of HVAC to electrical switchgear.

All of the licensees are considering or have implemented improvements to address the factors that increase the transient-related CDF. The most common plant improvements are aimed at reducing the potential for RCP seal LOCAs (e.g., providing backup cooling for charging to maintain RCP seal cooling, or replacing seals with the newer temperature-resistant design). Other improvements increase the reliability of AC power during LOSP scenarios (e.g., adding diesel generators, improving diesel reliability and maintenance programs, adding procedures for cross-tieing buses). All of the licensees also are considering improvements to address plant-specific issues. At some plants, these modifications have already been completed and were credited in the IPEs; at other plants, the modifications are still under consideration.

LOCAs are important contributors to CDF for most plants in this group. The LOCA contributions for all Westinghouse 3-loop plants except H.B. Robinson are dominated by small LOCAs, because of the higher small LOCA initiator frequency. The dominant contributor to core damage is failure of ECCS during recirculation, but a few plants are dominated by failures during injection. Recirculation involves realigning systems, and typically involves more components than required for the injection mode; this complexity leads to a higher failure probability. In addition, some plants require operator action to align the systems for recirculation; this generally increases the probability of failure.

Generally, plant design and operational features had about the same impact on the LOCA CDF as did modeling characteristics. Overall, the most influential factors are those discussed below:

- Switchover to recirculation — At most of the plants some manual actions are required to initiate coolant recirculation because high-pressure system pumps draw suction from low-pressure systems. However, the switchover of the low-pressure systems from the RWST to the containment sump is automatic at most of the plants; this simplifies the actions required of the operators, thereby increasing the probability of successfully completing the action. Plants without this automation tend to have higher LOCA CDFs, particularly for large and medium LOCAs, in which there is less time available for the operators to perform the actions. At H.B. Robinson, for example, the switchover is manual, and this plant has the highest LOCA CDF for the Westinghouse 3-loop plants.

- Alternative actions to mitigate a LOCA — Some licensees credit alternative actions such as depressurizing the RCS using the steam generator ADVs when high-pressure injection fails during a LOCA or refilling the RWST if recirculation fails. The ability of these strategies to succeed is plant-specific and it is not clear whether other licensees can use the same strategies. However, the strategies are found to be important for those submittals that credited the actions.

Very few licensees indicated that plant improvements are being made to address LOCA issues. The indicated improvements involve hardware and procedures needed for recirculation, and changes to feed-and-bleed procedures.

SBO accidents are important contributors to CDF for many of the Westinghouse 3-loop plants. SBO accidents involve an initial LOSP followed by failure of the emergency on-site AC power sources. The failure of AC power sources results in failure of all injection systems and failure of motor-driven AFW. This leaves only turbine-driven AFW available to cool the core and no systems available to provide injection to make up the loss through any RCP seal LOCAs that develop during the transient.

Generally, plant design and operational features have a larger impact on the SBO CDF than do modeling characteristics, but no single factor dominates. Combinations of contributors are usually important, and those combinations vary from plant to plant. Overall, the most influential factors are those discussed below:

- Modeling of RCP seal LOCAs and seal design — Because seal cooling is lost during a SBO, RCP seal LOCAs are important for most Westinghouse 3-loop plants. As noted in the discussion of transients above, RCP seal LOCA modeling varies among the submittals, and has a significant impact on the results. Backup capabilities for seal cooling (e.g., cross-ties between units for North Anna and Surry) are also important.

- Number of emergency AC power sources — The number of emergency diesel generators (usually two or three per unit) directly affects the reliability of the emergency AC power system. Further, a few plants have the ability to cross-tie between units or have extra diesel generators available. This reduces the SBO CDF, particularly if the diesel generators are of diverse design, thereby reducing the potential for common cause failure. For example, Turkey Point has four emergency diesel generators and five black-start diesel generators at the site (two nuclear units). This significantly reduces the SBO CDF for the plant. Modeling of common cause failures can also be important; for example, use of low common cause failure probabilities significantly lowers the SBO CDF for Shearon Harris and Farley.

- Battery depletion time — Although some licensees indicated that they have the capability to manually control AFW when DC power is lost, most licensees indicated that they need battery power to provide AFW control. Thus, for most plants, when the batteries are depleted, all cooling is lost and core damage ensues. Battery depletion times range from two to eight hours for this group, with the longer times tending to reduce the SBO CDF.

The plant improvements related to increasing the reliability of AC power that were discussed above for transients are also beneficial for SBO accidents. In addition, some licensees indicated that plant improvements are under consideration or have been implemented to extend the availability of DC power (e.g., improved load shedding procedures or providing independent battery charging capability). Other improvements will reduce the susceptibility to RCP seal LOCAs (e.g., replacing RCP seals with high-temperature o-rings or providing alternative RCP seal cooling during SBO accidents).

Internal flooding is important for some Westinghouse 3-loop plants. Internal flooding events involve rupture of water lines that result in a release of water that can directly cause the failure of required mitigating systems and/or cause system failure because of submergence or spraying of required components. The effects of internal flooding are highly plant-specific, depending on the layout of equipment within the plant and the relative isolation of rooms. Because of this design and layout diversity, each plant has different vulnerabilities to flooding, and generic conclusions regarding flooding can not be drawn. The plants with the largest flood contributions are typically dominated by floods that affect support systems (such as electric power and SW), which have plant-specific designs.

The plants with the highest CDF contributions from internal floods indicated that plant improvements are being considered or have been implemented to reduce the potential for core damage from flood scenarios. These improvements include changes to procedures for identifying and coping with floods, as well as changes to plant layout (e.g., adding dikes, water-tight doors, or backflow prevention devices in drains).

ATWS is not an important contributor for most Westinghouse 3-loop plants. ATWS sequences involve a transient, followed by a failure to shut down the nuclear chain reaction by inserting the control rods. Power generation continues at levels far in excess of normal decay heat. An ATWS sequence can also be mitigated by pressure control and heat removal. Because of the low frequency of failure to scram, ATWS is a relatively low contributor (less than 10% of CDF) for nearly all plants in this group. The single plant with a significant ATWS contribution (Beaver Valley 1) operates with two of the three PORV block valves closed, thereby reducing the relief capacity of the primary system during the early phase of an ATWS.

SGTR is not a dominant contributor to CDF for Westinghouse 3-loop plants. SGTR sequences involve leakage from the primary to the secondary through a ruptured steam generator tube, followed by either failure to mitigate the leak or failure to establish long-term core heat removal. SGTR accidents are a minor contributor to plant CDF (less than 10%) at the Westinghouse 3-loop plants because of the low frequency of the rupture occurring and the relatively high probability of the operators isolating the rupture. However, the contribution to risk is more significant at many plants because the releases bypass containment.

Although minor, the SGTR contribution to CDF is primarily driven by the treatment of operator actions in the IPEs. SGTR accidents require considerable operator involvement, with procedures that varied among the plants. In addition, modeling of operator errors and strategies varied considerably among the IPEs. Although this area contains large uncertainties, it does not have a large impact on overall plant CDF.

ISLOCAs are not dominant contributors to CDF for Westinghouse 3-loop plants. ISLOCAs, however, can be significant contributors to risk because the releases bypass containment. The low CDF contribution results from the low frequency of the LOCA initiator, combined with its negligible-to-minimal impact on other systems. This low impact is attributed to the compartmentalization and separation of the equipment.

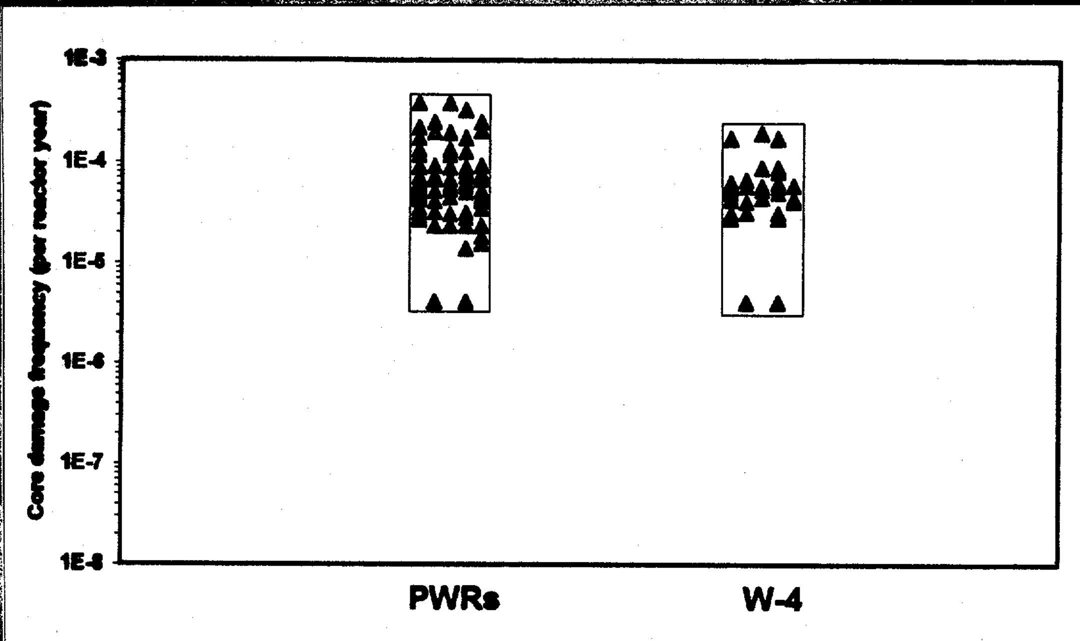
3.3.5 Westinghouse 4-loop Perspectives

Table 3.18 lists 32 plant units (represented by 20 IPE submittals) of the Westinghouse 4-Loop plant group.

Table 3.18 Plants (per IPE submittal) in the Westinghouse 4-loop group.

Braidwood 1&2	Byron 1&2	Callaway	Catawba 1&2
Comanche Peak 1&2	DC Cook 1&2	Diablo Canyon 1&2	Haddam Neck
Indian Point 2	Indian Point 3	McGuire 1&2	Millstone 3
Salem 1&2	Seabrook	Sequoyah 1&2	South Texas 1&2
Vogtle 1&2	Watts Bar 1	Wolf Creek	Zion 1&2

The Westinghouse 4-loop plants span nearly the entire range of CDFs reported for PWRs, as illustrated in Figure 3.18. One dual-unit plant (Zion 1&2) has a CDF considerably below the other PWRs. This outlier primarily reflects more optimistic modeling (relative to the other PWRs) of success criteria, operator actions, and common cause failures. The importance of specific accident classes to CDF varied significantly from plant to plant, as shown in Figure 3.19.



- Support system failures are important because they defeat redundancy in front-line systems.
- Relatively high dependence of front-line systems on SW and CCW increases CDF for transients.
- Variability in contribution from RCP seal LOCAs reflects differences in both plant designs and modeling assumptions.
- Plants requiring manual actions for ECCS switchover to recirculation tend to have higher LOCA CDFs.

Figure 3.18 Reported IPE CDFs and key perspectives for the Westinghouse 4-loop PWR plant group.

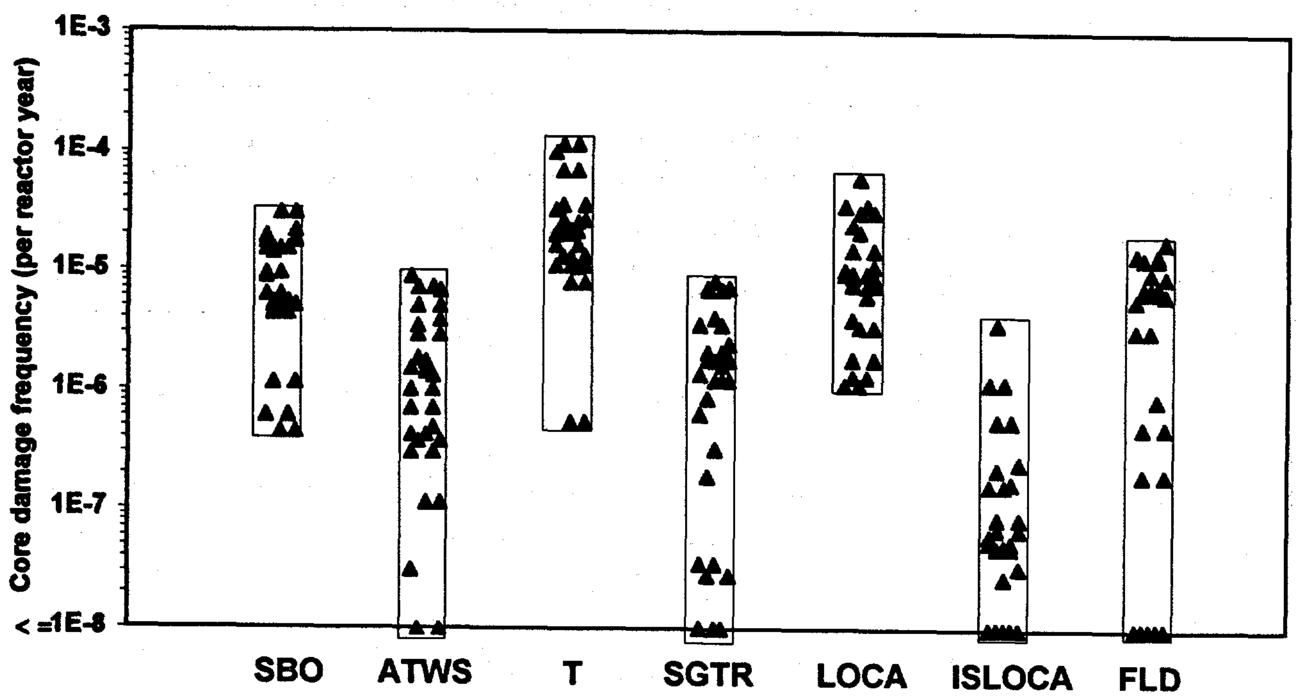


Figure 3.19 Reported IPE accident sequences CDFs for Westinghouse 4-loop plants.

The following accident classes, however, were important for many of these plants:

- transients
- LOCAs
- SBO — the loss of all off-site and on-site AC power

In general, transients are important contributors to CDF and containment failure frequency because they involve relatively high initiating event frequencies coupled with support system failures that defeat the redundancy in systems available to mitigate potential accidents. LOCAs are more important for these PWR plants than for BWR plants (see Section 3.2) because fewer systems are available in the PWRs to provide low-pressure injection. SBO accidents are relatively important for the Westinghouse 4-loop plants because they leave few systems available to prevent core damage.

A few licensees identified internal flooding accidents as being important, generally reflecting plant-specific weaknesses. With the exception of a single outlier plant in each category, none of the licensees found ATWS or SGTR sequences to be important contributors, and none of the licensees reported significant CDFs from ISLOCAs. These accidents are normally low contributors because of the low frequency of the initiating event and the relatively high probability of the operators isolating ruptures in steam generator tubes. Although SGTR and ISLOCAs were generally found to be low contributors to CDF, the releases can be important risk contributors since they bypass containment.

The variation in the reported IPE results is attributed to many factors including plant-specific design features, modeling assumptions, and use of data (including the probability of operator errors). These factors and the improvements under consideration by the licensees to address weaknesses are summarized in Table 3.19, and are discussed below for each important accident class.

Table 3.19 Summary of CDF perspectives for the Westinghouse 4-loop plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
Transient accidents		
Important for nearly all Westinghouse 4-loop plants	Degree of high-pressure injection and AFW system dependence on SW and CCW Dependence of RCP seal cooling on SW or CCW Modeling of RCP seal LOCA probability and size	Adding a new CST Using firewater or backup city water to extend the availability of AFW Replacing RCP seals with the newer temperature-resistant design Providing backup cooling for RCP seals
LOCAs		
Important for many Westinghouse 4-loop plants	Degree of automation of switchover of ECCS to recirculation Design of high-pressure recirculation systems (drawn directly from sump instead of drawing suction from low-pressure recirculation systems) Size of the RWST Ability to depressurize the RCS by aggressively cooling down using steam generator ADVs so low-pressure injection can be used when high-pressure injection fails Ability to refill the RWST	Revising hardware and procedures needed for recirculation Upgrading feed-and-bleed training
SBO accidents		
Important for many Westinghouse 4-loop plants	AC power reliability (number of diesel generators, cross-tie capabilities between units, diverse AC power sources) RCP seal material Backup cooling for RCP seals Modeling of RCP seal LOCA probability and size Battery life Diesel generator reliability	Adding or upgrading diesel generators or gas turbines Improving diesel generator reliability and maintenance programs Adding procedures for cross-tieing buses Improving load shedding procedures Improving battery capacity Providing independent battery charging capability Implementing procedures for manual operation of AFW when control power (DC) is lost

Table 3.19 Summary of CDF perspectives for the Westinghouse 4-loop plant group.

Accident importance	Important design features, operator actions, and model assumptions	Example plant improvements
Internal flood accidents		
Important for some Westinghouse 4-loop sites	Plant layout (separation and compartmentalization of mitigating system components)	Modifying demineralized water system
ATWS accidents		
Not dominant for most Westinghouse 4-loop plants	Plant operation with PORV block valves closed	Installing AMSAC Adding alternative scram button
SGTR accidents		
Not important for most Westinghouse 4-loop plants	Modeling of operator actions to isolate the rupture and provide long-term heat removal	Implementing procedures for coping with SGTR
ISLOCAs		
Not important for any Westinghouse 4-loop plant	Compartmentalization and separation of equipment	Implementing procedure modifications Providing RHR isolation valve leakage monitoring system

Transients are important contributors to CDF for nearly all Westinghouse 4-Loop plants. This accident class involves events that cause the reactor to trip, followed by failure to bring the reactor to safe shutdown (either failure to remove decay heat or failure to replace the reactor coolant inventory following an accident-induced LOCA, normally an RCP seal LOCA). Transients represent a broad category, covering general initiators (such as reactor trip or loss of main feedwater), as well as support-system initiators (such as loss of SW or AC/DC bus).

The failures leading to core damage for transients are quite plant-specific. However, some key factors affecting the transient-related CDF are common among many of the submittals. These key factors involve plant-specific design and operating conditions, as well as the assumptions made in the IPEs. The modeling of RCP seal LOCAs affects the results and represents an important area of uncertainty. The key factors are identified and discussed below:

- SW and CCW dependencies — Most of the plants exhibit a relatively high dependence of other plant systems on CCW and/or SW. Thus, loss of either of these support systems is important to the overall transient CDF. For example, the dominant transient sequence at D.C. Cook is a loss of CCW leading to a RCP seal LOCA that cannot be mitigated. The configurations vary considerably among the plants, however, and plants with the ability to use alternative cooling configurations (when the primary cooling system is lost) generally have lower transient-related CDFs.

- Susceptibility to RCP seal LOCAs — For some plants, the importance of RCP seal LOCAs is reduced because of plant design characteristics that reduce the associated threat. These include the use of the newer seal material (which is less susceptible to leakage than the older material), availability of backup systems (e.g., safe shutdown facility) to cool the seals or charging pumps when the normal configuration fails, and use of a different support system to cool the RCP seals than is used to cool the charging and HPSI pumps (so that loss of a single system does not lead to an unmitigable LOCA). The modeling of RCP seal LOCAs varies considerably among the IPEs, with some using low leakage probabilities and leak rates, while others used much higher values for both parameters. This variability had a significant impact on the results.

Most licensees are considering or have implemented improvements to address the factors that increase the transient-related CDF. The most common plant improvements are aimed at increasing the reliability of AC power during LOSP scenarios (e.g., adding or upgrading diesel or gas turbines, improving diesel reliability and maintenance programs, adding procedures for cross-tieing buses). Other improvements include extending AFW availability (e.g., adding a new CST, using firewater, using backup city water), and reducing the potential for RCP seal LOCAs (e.g., replacing seals with the newer temperature-resistant design, or providing backup cooling for RCP seals). At some plants, these modifications have already been completed and were credited in the IPEs; at other plants, the modifications are still under consideration.

LOCAs are important contributors to CDF for many plants in this group. The most important LOCA contributors at most plants are small LOCAs, but some plants are dominated by large or medium LOCAs. The small LOCAs are generally the larger contributors to CDF because they have a higher frequency. The dominant contributor to core damage is ECCS failure during recirculation, but a few plants are dominated by failures during injection. Recirculation involves realigning systems, and typically involves more components than required for the injection mode; this complexity leads to a higher failure probability.

Generally, plant design and operational features had a larger impact on the LOCA CDF than did modeling characteristics. Overall, the most influential factors are those discussed below:

- Switchover to recirculation — Most plants require some manual actions to initiate coolant recirculation because high-pressure recirculation system pumps draw suction from low-pressure systems. However, the switchover of the low-pressure systems from the RWST to the containment sump is automatic at many plants. This simplifies the actions required of the operators, and increases the probability of successfully completing the action. At Haddam Neck, for example, the switchover is manual, and this plant has the highest LOCA CDF for the Westinghouse 4-loop plants. For ice-condenser plants in particular, the lower containment design pressure results in earlier actuation of containment sprays. As a result, so there is less time to perform the switchover. Therefore, the degree of automation is particularly important for ice-condenser plants.

- Size of RWST — Some plants have large RWSTs so that the switchover to recirculation (and the associated complications discussed above) is either not necessary for small LOCAs within the modeled mission time or is significantly delayed, giving the operators more time to complete the necessary actions for the switchover to recirculation. Larger RWSTs were found to have an important effect on the LOCA-related CDF. For example, the two plants with the lowest LOCA CDFs (Braidwood and Byron) have relatively large RWSTs, and do not require switchover to recirculation for small LOCAs.

- Alternative actions to mitigate a LOCA — Some licensees credit alternative actions such as depressurizing the RCS using the steam generator ADVs when high-pressure injection fails during a LOCA, or refilling the RWST if recirculation fails. The ability of these strategies to succeed is plant-specific, and the strategies may not be feasible at other plants. However, the strategies were found to be important for those submittals that credited the actions.

Very few of the licensees indicated that plant improvements are being made to address LOCA issues. The indicated improvements primarily involve hardware and procedures needed for recirculation, and feed-and-bleed training upgrades.

SBO accidents are important contributors to CDF for many Westinghouse 4-Loop plants. SBO accidents involve an initial LOSP followed by failure of the emergency on-site AC power sources. The failure of AC power sources results in failure of all injection systems and failure of motor-driven AFW. This leaves only turbine-driven AFW available to cool the core and no systems available to provide injection to make up the loss through any RCP seal LOCAs that develop during the transient.

Generally, plant design and operational features have a larger impact on the SBO CDF than do modeling characteristics, but no single factor dominates. Combinations of contributors are usually important, and those combinations vary from plant to plant. Overall, the most influential factors are those discussed below:

- Modeling of RCP seal LOCAs and seal design — Because seal cooling is lost during a SBO, RCP seal LOCAs are important for the Westinghouse 4-Loop plants. As noted in the discussion of transients above, RCP seal LOCA modeling varies among the submittals, and has a significant impact on the results. Another important factor is whether the plants have replaced the RCP seals with the newer temperature-resistant design. Callaway for example, has installed the new seals, and as a result has a relatively low contribution from SBO accidents that involve RCP seal LOCAs.

- Number of emergency AC power sources — The number of emergency diesel generators (usually two or three per unit) directly affects the reliability of the emergency AC power system. Further, a few plants have a diverse AC power source, such as a gas turbine generator or an independent safe shutdown facility, and are less susceptible to common cause failures of diesel generators. Plant-specific operating history can also be important.

- Battery depletion time — Although some licensees indicated that they have the capability to manually control AFW when DC power is lost, most licensees indicated that they need battery power to provide AFW control. Thus, for most plants, when the batteries are depleted, all cooling is lost and core damage ensues. Battery depletion times range from 1 to 12 hours for this group, with the longer times reflecting plants making extensive use of load shedding.

Plant improvements related to increasing the reliability of AC power that are discussed above for transients are also beneficial for SBO accidents. In addition, many licensees indicated that plant improvements are under consideration or have been implemented to extend the availability of DC power (e.g., improvements to battery capacity, improved load shedding procedures, providing independent battery charging capability). In addition, a few licensees implemented procedures for manual operation of AFW when control power (DC) is lost.

Internal flooding is important for some Westinghouse 4-Loop plants. Internal flooding events involve rupture of water lines that result in a release of water that can directly cause the failure of required mitigating systems and/or cause system failure because of submergence or spraying of required components. The effects of internal flooding are highly plant-specific, depending on the layout of equipment within the plant and the relative isolation of rooms. Because of this design and layout diversity, each plant has different vulnerabilities to flooding, and generic conclusions regarding flooding can not be drawn. The plants with the largest flood contributions are typically dominated by floods that affect support systems such as electric power and SW, which have plant-specific designs.

ATWS is not an important contributor for most Westinghouse 4-Loop plants. ATWS sequences involve a transient, followed by failure to shut down the nuclear chain reaction by inserting the control rods. Power generation continues at levels far in excess of normal decay heat. An ATWS sequence can be mitigated by pressure control and heat removal. Because of the low frequency of failure to scram, ATWS is a relatively low contributor (less than 10% of CDF) for nearly all plants in this group. The single plant with a significant ATWS contribution (Indian Point 3) operates with the PORV block valves closed, thereby reducing the relief capacity of the primary system during the early phase of an ATWS.

SGTR is not an important contributor to CDF for most Westinghouse 4-Loop plants. SGTR sequences involve leakage from the primary to the secondary through a ruptured steam generator tube, followed by either failure to mitigate the leak or failure to establish long-term core heat removal. SGTR accidents are a minor contributor to plant CDF at most Westinghouse 4-loop plants because of the low frequency of the rupture occurring and the relatively high probability of the operators isolating the rupture. However, the contribution to risk is more significant at many plants because the releases bypass containment.

Although nearly always minor, the SGTR contribution to CDF is primarily driven by the treatment of operator actions in the IPEs. These SGTR accidents require considerable involvement from operators, with procedures that vary among the plants. In addition, modeling of operator errors and strategies varied considerably among the IPEs. Although this area contains large uncertainties, it does not have a significant impact on the overall plant CDF.

ISLOCAs are not dominant contributors to CDF for the Westinghouse 4-loop plants. ISLOCAs, however, can be significant contributors to risk because the releases bypass containment. The low CDF contribution results from the low frequency of the LOCA initiator, combined with its negligible-to-minimal impact on other systems. This low impact is attributed to the compartmentalization and separation of the equipment.

4. IPE RESULTS PERSPECTIVES: CONTAINMENT PERFORMANCE

Consistent with the U.S. Nuclear Regulatory Commission's (NRC's) "defense-in-depth" principle, the importance of the containment as a vital barrier to the release of fission products to the environment was recognized in Generic Letter (GL) 88-20[4.1], where evaluation of containment performance was identified as an important aspect of the Individual Plant Examination (IPE) program. Appendix 1 of GL 88-20 was dedicated to providing guidance on the examination of containment system performance. In summarizing the guidance on the approach to the Level 2 analysis consistent with the IPE objectives, the Appendix states that the approach should focus on containment failure mechanisms and timing. Further guidance on evaluation of containment response is provided in NUREG-1335, "Individual Plant Examination: Submittal Guidance."[4.2] NUREG-1335 states: "The primary objective is to provide the utility with a framework for obtaining an understanding of and appreciation for containment failure modes, the impact of phenomena and plant features, and the impact of operator actions."

This chapter summarizes perspectives on the containment performance evaluations reported in the IPE submittals, while Chapter 12 presents the results and perspectives in greater detail. The impact of nuclear plant operation on public safety is ultimately related to the frequency of releases from the containment and the IPE reported containment failure and release frequencies are described in Chapter 12. However, the emphasis in both chapters is on conditional containment failure probability (CCFP)[4.3] since a main objective of these chapters is to obtain perspectives on the performance of the various containment types independent from other plant features. For this purpose, the CCFP is a useful parameter since it decouples containment failure from core damage frequency. The value of CCFP was also recognized by the majority of licensees since CCFPs are reported directly in most of the IPE submittals. The results and perspectives presented below are based on the original IPE submittals forwarded by the licensees to the NRC.[4.4] Many licensees have updated their IPEs since the original submittal and have reported bottom-line changes in core damage frequency or release frequency (but in most cases detailed information was not provided). Updated IPE information received by NRC is listed in Appendix B.

4.1 General Containment Performance Perspectives

When the accident progression analyses in the IPEs are viewed globally, they are, for the most part, consistent with probabilistic risk analysis (PRA) containment performance analyses performed previously. Failure mechanisms identified as important in the past were shown to be important in the IPEs as well.

[4.1]USNRC, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," Generic Letter 88-20, November 23, 1988.

[4.2]US NRC, "Individual Plant Examination: Submittal Guidance," NUREG-1335, August 1989.

[4.3]Conditional containment failure probability is defined as the probability of containment failure conditional on core damage having occurred. Chapter 14 provides a more detailed discussion on the definition and estimation of conditional failure probability.

[4.4]Refer to Appendix A for sources of IPE information used in this chapter.

The significance of individual containment failure mechanisms is often determined by particular features of a containment class. In general, the IPEs confirmed that large volume containments are less likely to experience early structural failures than the smaller boiling water reactor (BWR) pressure suppression containments. However, as indicated in Figure 4.1, considerable variability exists in the CCFPs reported within each containment class. This variability exists in the reported frequencies of containment failure as well.

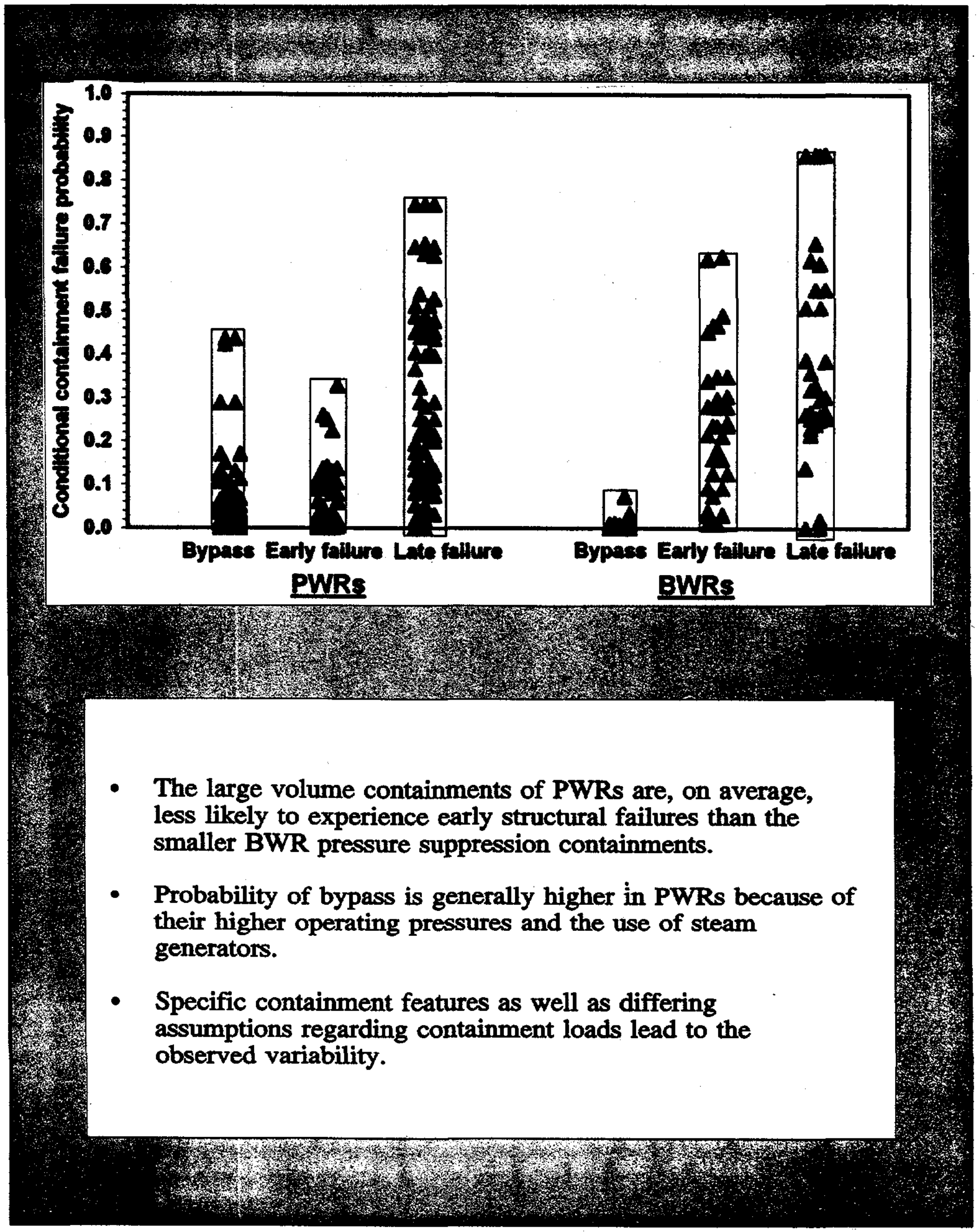


Figure 4.1 Reported IPE CCFPs (given core melt) and key perspectives on containment performance for all plants.

The importance of early radionuclide releases to all risk measures (i.e., acute and latent health effects including land contamination), has been established in previous PRAs that included consequence calculations. In keeping with the significance of such early releases, the containment performance analyses described in the IPE submittals emphasized the phenomena, mechanisms, and accident scenarios that could lead to early releases. These involve early structural failure of the containment, containment bypass, containment isolation failures, and for some BWR plants, deliberate venting of the containment.

As a group, the pressurized water reactor (PWR) large dry containments analyzed in the IPEs had significantly smaller conditional probabilities of early structural failure than the BWR pressure suppression containments analyzed. On the other hand, containment bypass and isolation failures are, in general, more significant for the PWR containments. As shown in Figure 4.1, however, these general trends often do not hold true for individual IPEs because of the considerable range in the results. For instance, CCFPs for both early and late failure for a number of PWR large, dry containments are higher than those reported for some of the pressure suppression containments.

Differences in containment designs account for much of the observed differences in failure probabilities indicated in Figure 4.1. This is also true for the variations between containment classes, as well as differences between individual plants in the same containment class. In a significant number of cases, unique plant-specific containment features are identified in the analyses as leading to important failure mechanisms.

Differing assumptions, however, in the accident progression modeling also play a major role in explaining the significant variability in the results. Differences in modeling assumptions are not surprising, since considerable uncertainty still exists regarding the loads imposed on containments by the phenomena postulated in an accident progression analysis.

Table 4.1 summarizes key observations regarding containment performance. The subsequent sections of this chapter discuss the containment performance perspectives in some detail. Chapter 12 of Part 3 of this report provides additional in-depth discussion of the containment performance results reported in the IPEs and the perspectives drawn from them.

Table 4.1 Summary of key containment performance perspectives for LWR containments.

Failure mode	Key observations
Early failure	The large-volume containments of PWRs are, on average, less likely to experience early structural failures than the smaller BWR pressure suppression containments
	Overpressure failures, primarily from anticipated transients without scram (ATWS), fuel-coolant interaction (FCI), and failures due to direct impingement of core debris are found to be important contributors to early failure for most BWR containments; hydrogen burns are found important in some Mark III containments
	The higher probability of early structural failures of BWR Mark I plants, compared to the later BWR containments, is driven to a large extent by drywell shell melt-through*
	Phenomena associated with high-pressure melt ejection (HPME) are the leading causes of early failure for PWR containments*
	Isolation failures are found to be significant in a number of large dry and subatmospheric containments
	The low probability of early failures for ice condensers (relative to the other PWRs) appear to be driven by analysis assumptions rather than plant features
	For both BWR and PWR plants, specific design features lead to a number of unique and significant containment failure modes

Table 4.1 Summary of key containment performance perspectives for LWR containments.

Failure mode	Key observations
Bypass	Probability of bypass is generally higher in PWRs, in part, because of the contribution from steam generator tube ruptures (SGTRs) Bypass, mostly from SGTR, has probabilities comparable to early structural failure for both PWR containment types Bypass is generally not important for BWRs
Late failure	Overpressurization when containment heat removal (CHR) is lost is the primary cause of late failure in most PWR and some BWR containments High-pressure and temperature loads caused by core-concrete interactions (CCIs) are important for late failure in BWR containments Containment venting is found to be important for avoiding late uncontrolled failure in some Mark I IPEs The larger volumes of the Mark III containments are partly responsible for their lower late failure probabilities (in comparison to the other BWR containments) The likelihood of late failures often depends on the mission times assumed in the analysis

* As noted in Chapter 8 there has been a considerable change in the state-of-knowledge regarding some severe accident phenomena in the time since the IPE analyses were carried out.

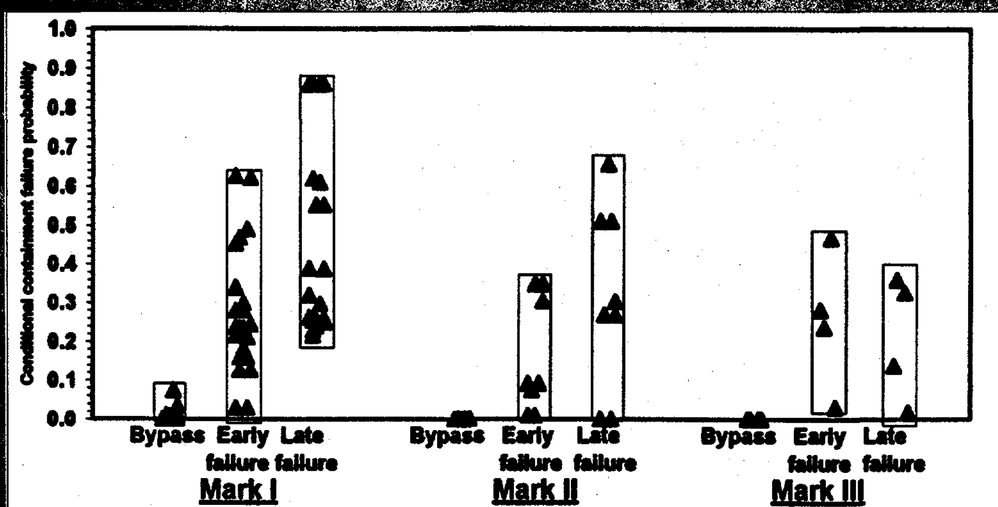
4.2 BWR Containment Performance Perspectives

As noted in Section 1.4, BWR plants are separated into three groups (Mark I, II and III), according to the type of pressure suppression containment used, for the purpose of identifying containment performance perspectives from the submitted IPEs. One early BWR, Big Rock Point, is housed in a large dry containment (similar to that used for PWRs) and therefore is discussed in Section 4.3.1.

The results follow expected trends and indicate that the early Mark I containments are generally more likely than the later Mark II and Mark III designs to fail during a severe accident. 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.

Significant variability exists with regard to the contributions of the different failure modes for each containment group. 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.

Sections 4.2.1 through 4.2.3 discuss important factors that impact the probabilities of the failure modes shown in Figure 4.2 for each containment group. In general, each containment group has different factors that influence the failure modes. This often occurs because of differences in containment design between the three groups. For example, shell melt-through (caused by contact with core debris) is found to be the most important contributor to early containment failure for Mark I containments. This failure mechanism is possible for Mark I containments because the pedestal and drywell floor are at the same level, and core debris can reach the containment wall (which is usually steel). By contrast, the core debris cannot easily reach the containment wall in Mark II and Mark III containments; therefore, other failure mechanisms are found to be important for these designs. Specifically, accidents in which CHR is lost or inadequate are found to be important contributors to early failure in Mark II plants. In Mark III plants, early failure is primarily caused by energetic events, such as FCIs and hydrogen combustion events. Hydrogen combustion is unlikely in Mark I and II plants because their atmospheres are inerted during operation.



- **The higher early structural failures of Mark I plants (versus the later BWR containments) are driven to a large extent by drywell shell melt-through.**
- **Bypass and containment isolation failures are generally not important for BWRs.**
- **The large volumes of Mark III plants are partly responsible for their lower late failure probabilities (in comparison to other BWR containments).**

Figure 4.2 Reported IPE CCFPs (given core melt) and key perspectives on containment performance for BWR plants.

Late containment failure can be caused by gradual pressure buildup associated with non-condensible gas release, basemat melt-through, or hydrogen combustion events. Gradual pressure buildup caused by CCI is found to be an important contributor to late failure in Mark I and II containments. However, hydrogen combustion is found to be important to the probability of late failure in some Mark III containments.

Finally, venting can be important to the probability of containment integrity loss (early and late in an accident sequence) for some BWR plants. Differences in the modeling of venting contribute to the variability of the results.

The containment performance results for all operating BWRs using pressure suppression containments are shown in Figure 4.2 and perspectives are summarized in Table 4.2.

Table 4.2 Summary of containment performance perspectives for BWRs.

Failure mode	Failure mode importance	Important design features, operator actions, and model assumptions	Important plant improvements
Early failure	Significant probability for most BWRs, regardless of containment type Isolation failures not important for BWRs	High pressure loads at the time the core debris melts through the reactor vessel, FCI, and direct impingement of core debris are identified as contributors to early failure in BWRs Shell melt-through is found to be the most important contributor to early failure for Mark I plants. Specific design features, and assumptions regarding core debris characteristics and the absence or presence of water in the drywell, determine the importance of shell melt-through for individual plants Hydrogen burns are found to be important in some Mark III IPEs ATWS sequences are found to be important contributors in some BWR IPEs Specific design features play an important role in many analyses	Alternative water sources for flooding of the drywell floor Less restrictive drywell spray initiation criteria Operator training on depressurization
Bypass	Not important for BWRs as a group	Bypass via emergency condenser is identified by one IPE	None identified
Late failure	Significant probability for most BWRs, somewhat less important for Mark III containments	High-pressure and temperature loads caused by CCI are an important failure mode. Specific design features (size of sumps), as well as assumptions regarding core debris characteristics and the absence or presence of water in the drywell, determine the importance for individual plants Excessive safety relief valve (SRV) discharge into a hot suppression pool is also found to lead to late failure in some cases Late combustible gas burns are important in some Mark IIIs plants Containment venting is found to be an important way of avoiding uncontrolled containment failure for some Mark I and III containments	Ensuring that the drywell floor is flooded Altering venting criteria to account for temperature effects

4.2.1 BWR Mark I Perspectives

Table 4.3 lists the Mark I containment group, which includes 22 BWR units (represented by 17 IPE submittals). 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. Figure 4.3 shows the Mark I CCFPs of the various containment failure modes reported in the IPEs, and Table 4.4 summarizes the IPE results.

Table 4.3 Plants (per IPE submittal) in Mark I containment group.

Browns Ferry 2	Brunswick 1&2	Cooper
Dresden 2&3	Duane Arnold	Fermi 2
Fitzpatrick	Hatch 1&2	Hope Creek
Millstone 1	Monticello	Nine Mile Point 1
Oyster Creek	Peach Bottom 2&3	Pilgrim 1
Quad Cities 1&2	Vermont Yankee	

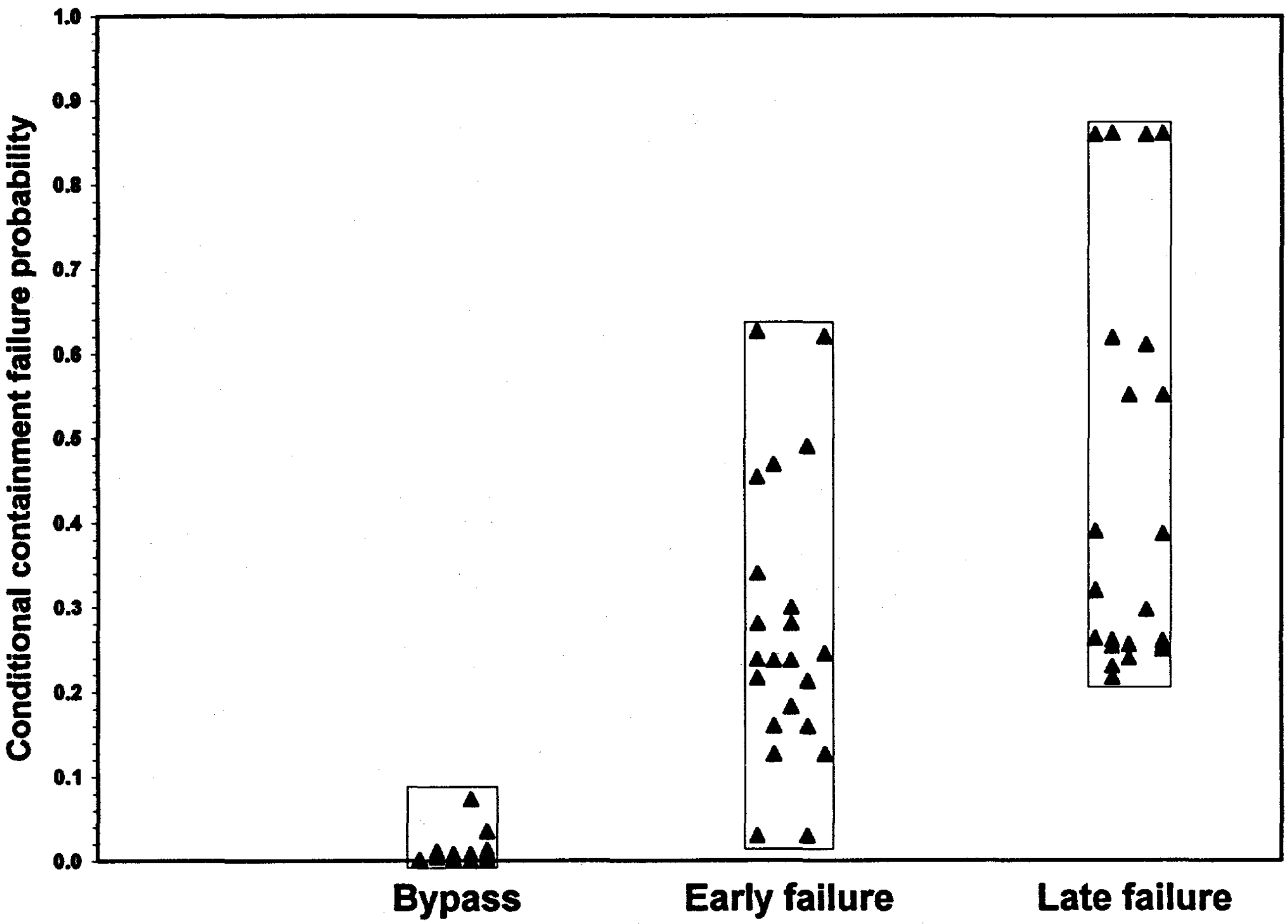


Figure 4.3 Reported IPE CCFPs for BWR Mark I containments.

Table 4.4 Summary of Mark I containment performance perspectives.

Failure mode	Failure mode importance	Important design features, operator actions, and model assumptions	Suggested plant improvements
Early failure	High probability for most plants in this group	Shell melt-through is found to be the most important contributor to early failure of Mark I plants, which are susceptible to direct impingement by core debris as a result of their steel containment and reactor pedestal-to-drywell communication Specific design features, as well as assumptions regarding core debris characteristics and the absence or presence of water in the drywell, determine the importance of shell melt-through for individual plants High pressure loads at the time the core debris melts through the reactor vessel is also a significant contributor to early failure ATWS sequences are found to be important contributors in some IPEs Isolation failures are not important	Alternative water sources for flooding of the drywell floor Less restrictive drywell spray initiation criteria Operator training on depressurization
Bypass	Not important for this plant group	Bypass via emergency condenser is identified by one IPE	None identified
Late failure	Significant probability for most plants in this group	High-pressure and temperature loads caused by core/concrete interactions are an important failure mode Specific design features (size of sumps), as well as assumptions regarding core debris characteristics and the absence or presence of water in the drywell, determine the importance for individual plants Containment venting is found to be an important way of avoiding uncontrolled containment failure in some IPEs	Ensuring that the drywell floor is flooded Altering venting criteria to account for temperature effects

The IPE results indicated a significant probability of early and/or late containment failure for most Mark I containments. Accidents that cause structural failure of the drywell shortly after the core debris melts through the reactor vessel have been found to be dominant contributors to risk in past PRAs. The importance of individual failure mechanisms depends on plant-specific features and, in some cases, on modeling assumptions; however, the following mechanisms were found to be important causes of early structural failure for many Mark I containments:

- drywell shell melt-through caused by direct contact with the core debris (i.e., shell melt-through)
- drywell failure caused by rapid pressure and temperature pulses at the time of reactor vessel melt-through

In general, these failure mechanisms can be important to risk because of the relatively short time available for radioactivity decay, natural deposition processes, and accident response actions. In addition, drywell failure implies that radionuclides released from the damaged core bypass the suppression pool. (Significant retention can occur if aerosol radionuclides pass through a suppression pool.) Because of the relatively short time to radionuclide release and the magnitude of the release, these failure mechanisms have been found to be important to all risk measures (i.e., acute and latent health effects including land contamination) in past studies that included estimates of offsite consequences. These failure mechanisms can also occur for any accident class that involves release of a significant amount of core debris from the reactor vessel.

Other failure mechanisms are identified as being important in a few IPEs. For example, drywell failure caused by gradual pressure and temperature buildup associated with gases and steam released during CCI was important in some IPEs. In other IPEs, venting was found to be an important contributor to the release of radionuclides. However, accidents that bypass containment (such as interfacing systems loss of coolant accident (ISLOCAs) or involve containment isolation failure were not important contributors to the core damage frequency (CDF) in any of the IPEs for Mark I plants. These accidents were also not important to the likelihood of either early or late containment failure 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.

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 in Mark I containments. This is because, for most Mark I containments the reactor pedestal and drywell floor are at the same level and openings exist between the pedestal region and the floor. This design allows core debris to flow across the drywell floor and induce the failure of the steel drywell shell either by direct melt-through or by 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 shell 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 mitigates shell melt-through in all of the submittals. The benefit of water on the drywell floor before vessel failure (at low pressure) as a mitigating mechanism for shell melt-through is significant (NUREG/CR-5423[4.5]) and utilities with Mark I containments may wish to consider this benefit when developing their accident management plans.
- Containment design configuration — 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 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

[4.5]Theofanous, T.G., et al., "Probability of Linear Failure in a Mark I Containment," NUREG/CR-5423, University of California, Santa Barbara, August 1991.

concrete curb prevents or limits core debris from reaching the containment shell. Also, the Brunswick containment is unique among Mark I designs because it is constructed of concrete (with a steel liner) rather than steel. 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 determine whether or not shell melt-through occurs. Shell melt-through is found to be an important contributor to the likelihood of early containment failure if a large amount of high-temperature core debris is assumed to be released to the drywell. Under these circumstances, the core debris can flow across the floor and melt through the shell. Shell melt-through is not an important mechanism for causing containment failure if smaller quantities of core debris at lower temperatures (less able to flow across the floor) are assumed to be released into the drywell. As different modeling assumptions can produce such significantly different results (i.e., containment failure versus no failure), any actions taken by the utilities to mitigate this failure mechanism should reflect this uncertainty. Therefore, as water can effectively mitigate shell melt-through, utilities with Mark I containments may wish to ensure a flooded drywell floor in order to eliminate the uncertainty regarding containment failure caused by this mechanism.

A number of utilities are being proactive in identifying minor hardware modifications and procedural changes to ensure a flooded drywell floor before reactor vessel melt-through. The availability of alternative water sources to the drywell spray header, such as water from a diesel-driven fire water pump during a station blackout (SBO) has shown to significantly reduce the likelihood of early failure in the Browns Ferry IPE. Another example is the Monticello plant, where connections are available to enable the operators to use residual heat removal 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. At Peach Bottom, the capability exists to supply the sprays with water from an external pond or the emergency cooling tower. Several IPEs, such as those for Duane Arnold and Monticello, also discuss the possibility of relaxing restrictions on drywell spray initiation in the current emergency operating procedures (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 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 reactor coolant system (RCS) depressurizes as the core debris melts through the reactor vessel. Hydrogen (from clad oxidation) and steam are the driving forces for pressurization. If the pressure pulse exceeds the ultimate pressure capability of the containment, failure will occur at the weakest location in either 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 the amount of core debris dispersed determine whether this failure mechanism is a significant contributor to early containment failure for individual Mark I containments. The most important accident characteristic, design features, and modeling assumptions are discussed below:

- RCS pressure at the 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 risk significance of this failure mechanism therefore depends on the importance of accident classes in which the RCS is at high-pressure (such as transient events with failure of the automatic depressurization system). A number of utilities explored the possibility of enhancing the depressurization capability of the RCS, but they identified adverse effects that require careful consideration.

- Containment failure location — The containment failure location can significantly influence the importance of this failure mechanism. If failure occurs in the wetwell, significant retention of the aerosol radionuclides occurs in the suppression pool, making it less likely that this failure mechanism will lead to the release of significant quantities of radionuclides. Conversely, if failure occurs in the drywell, the radionuclides are released without the benefit of pool scrubbing and the release can be much higher.

- RCS depressurization characteristics — The rate of RCS depressurization (at vessel breach), steam generation, and characteristics of core debris dispersal determine the importance of this failure mechanism. If rapid depressurization is assumed (as a result of a large opening in the reactor vessel) high pressure pulses can occur. These pulses have a high likelihood of causing containment failure. In addition, if a large amount of high-temperature core debris is assumed to be released and dispersed into the containment atmosphere, it can directly heat the atmosphere 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.

Ways of preventing or mitigating the pressure and temperature loads at vessel melt-through include 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. Similarly, spray operation cannot effectively mitigate all pressure loads associated with RCS depressurization during severe accidents.

In the IPEs, a number of utilities explored controlled depressurization of the RCS before melt-through of the reactor vessel as a strategy for mitigating rapid overpressure failure of Mark I containments. Enhancement of the emergency depressurization capability was also an issue raised as part of the NRC's Containment Performance Improvement (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 (LPI) systems are not available, depressurization causes a loss of coolant inventory that 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 potentially adverse effects, some utilities recommend further study before 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 non-condensible gases and steam released from the drywell floor during core-concrete interactions can induce the failure of 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 has been found to be less risk significant than the two early failure mechanisms discussed above. This is because of the longer time available for radioactive decay, natural deposition processes, and accident response. However, even for late failures, if the failure location is in the drywell, significant radionuclide release can still occur. As a result, this failure mechanism has been found to be important to longer-term risk measures (i.e., latent health effects and land contamination) in several previous PRAs.

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 characteristic, design features, and modeling assumptions are discussed below:

- Drywell floor flooded — With a flooded drywell floor, it is more likely that the drywell spray and CHR systems can 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. Therefore, these systems can control the containment pressure under these circumstances.

- Drywell floor not flooded — If the drywell floor is not flooded (and shell melt-through does not occur), venting may be needed to prevent overpressure 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 basaltic 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 circumstances, 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 continue even after water is added to the drywell.

- Core debris characteristics — In the absence of water, the amount and temperature of core debris released to the drywell determine the extent of core-concrete interactions. If a large amount of high temperature core debris is assumed to be released from the reactor vessel, the IPEs predict extensive concrete erosion. Under these circumstances, even if water is added to the core debris, core-concrete interactions are predicted to continue for some Mark I designs. Conversely, if smaller quantities of core debris at lower temperatures are assumed, much less concrete erosion occurs even without water. Clearly, different modeling assumptions give different results that utilities should consider when developing strategies to mitigate these failure mechanisms.

Most utilities use a combination of strategies to mitigate gradual pressure buildup caused by core-concrete interactions. The drywell floor flooding strategies designed to prevent shell melt-through (if successful) will also limit long-term core-concrete interactions and hence noncondensible gas generation. If these early flooding strategies are not successful, most utilities will explore other ways of flooding the drywell floor. For instance, the Monticello IPE submittal noted that debris cooling with an alternative injection source (such as fire water) limits the temperature increase in containment and extends the time to containment failure associated with over-pressurization.

In all of 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 radionuclides deposited on containment surfaces. Sprays can also scrub radionuclides existing in the containment atmosphere and provide a water source for covering ongoing core-concrete interactions.

High-temperature effects are also addressed in other ways in some IPEs. For example, Nine Mile Point 1 considers 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 is used extensively in the IPE analyses to reduce releases and the associated risk. It is also an important element of the CPI Program. In addition, containment venting is sometimes credited with preventing core damage in accidents involving a loss of CHR. It is also used to prevent late structural failure for 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 does not significantly reduce risk and, therefore, is only of marginal benefit. In one case, the utility stated that the installation of a hardened vent lead to less than 1% reduction in CDF.

In response to the recommendations in Generic Letter 89-16[4.6], 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. Chapter 3 discusses how the utilities use these venting systems to prevent core damage for some accidents involving loss of CHR. Under these circumstances, venting is "clean" because it occurs before core damage and involves minimal release of radioactivity.

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

If venting occurs shortly after core meltdown, and the flow path is directly from the drywell or from the RCS to the environment, the suppression pool will be bypassed. Under these circumstances, venting would cause a significant release of radionuclides 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 to the top of fuel in the reactor pressure vessel (RPV)) and the venting necessary to carry it out. Since drywell venting (i.e., unscrubbed) is needed to relieve the pressure buildup resulting from the compression of the gas space during containment flooding, there is the potential for an early release of significant magnitude associated with the flooding strategy. A number of utilities speculated that other actions, or even no action, would be preferable to carrying out the containment flooding strategy.

Accidents that bypass containment are not important for Mark I containments. A loss of coolant accident (LOCA) can occur outside containment if the pressure boundary fails between the high pressure RCS and a low pressure auxiliary system (called an interfacing systems LOCA or ISLOCA). If water cannot be supplied to the reactor, core damage will occur and a direct path to the environment can exist. Therefore, these accidents can lead to a large early

[4.6]USNRC, "Installation of Hardened Wetwell Vent," Generic Letter 89-16, September 1, 1989.

release of radionuclides. However, ISLOCAs are found to be unimportant for BWR Mark I containments because of their relatively low frequency (compared with the frequency of accidents that dominate the CDF and can lead to early structural failure). The IPEs reported bypass CDF frequencies that are about an order of magnitude lower in BWR plants than in PWRs. The higher PWR frequencies result in part from the SGTR contribution in the PWR systems.

Although ISLOCAs are not important for BWRs, the Nine Mile Point 1 IPE identified a unique way of bypassing containment. In that IPE, failure of the emergency condenser tubes caused by 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 an SGTR in PWRs. This failure mode is found to have a relatively low frequency (compared with the frequency of early structural failure) at Nine Mile Point 1, and is therefore not important. Isolation condensers are found in one other BWR 2 plant and two early BWR 3 plants; presumably, this bypass accident also applies to these plants, but was not discussed in the IPE submittals.

Accidents that involve failure to isolate containment are not important for Mark I containments. Isolation failures can be preexisting or can occur at the time of the initiating event. If the isolation failure is large, and if core melt occurs, radionuclide release can also be large. In addition, because the containment is open at the time of core damage, the offsite consequences can be significant. These events are not important 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 associated with preexisting leaks can easily be detected. In addition, failure to isolate the containment on demand is found to be a relatively low-frequency event compared with the frequencies of other accidents that can cause early structural failure of the containment.

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 CHR and containment venting are inadequate. In ATWS events, the energy deposited to the containment can overwhelm the normal CHR 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 a loss of emergency core cooling systems (as a result of the 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 to be risk significant in past PRAs since core damage, vessel failure, and containment failure can occur within a short time, 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 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 CHR were found to be relatively unimportant in all IPEs. The ability to vent the containment is a major factor in reducing the importance of this class of accident. Also, the interval between loss of CHR and containment failure is relatively long in these sequences, allowing time for emergency measures on- and off-site.

4.2.2 BWR Mark II Perspectives

Table 4.5 lists eight BWR units (represented by five IPE submittals) in the Mark II containment group. Four units (Limerick 1&2 and Susquehanna 1&2) are of the BWR 4 type, while the other four units (LaSalle 1&2, Nine Mile Point 2, and Washington Nuclear Power 2 (WNP2)) are BWR 5 designs.

Mark II containments retain many features of the older Mark I containments from which they evolved. They also are characterized by relatively high strength but small volume. 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 WNP2, where the containment consists of a steel shell. The IPE results for this group are shown in Figure 4.4 and summarized in Table 4.6.

Table 4.5 Plants (per IPE submittal) in Mark II containment group.

LaSalle 1&2	Limerick 1&2	Nine Mile Point 2
Susquehanna 1&2	Washington Nuclear Power 2	

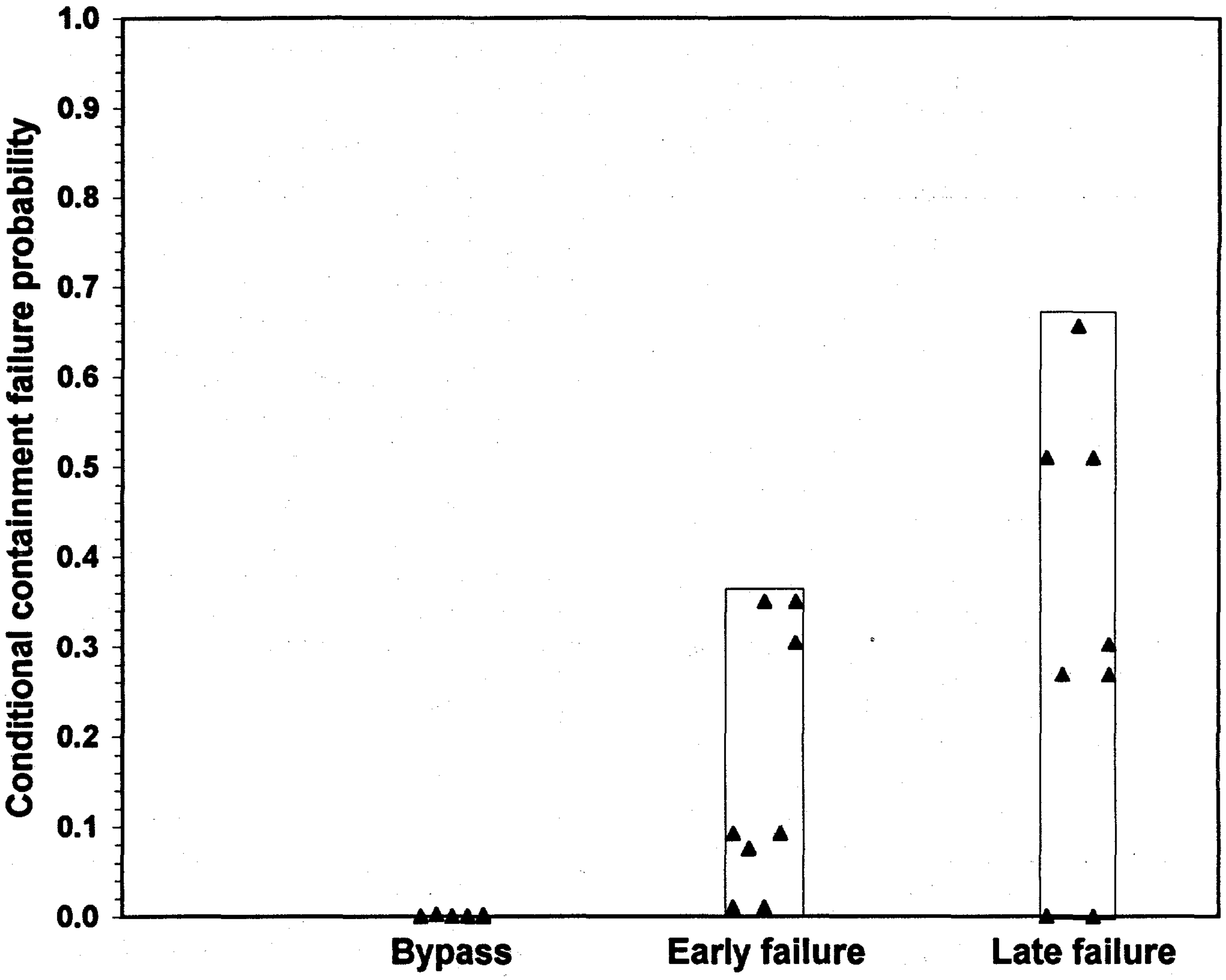


Figure 4.4 Reported IPE CCFPs for BWR Mark II containments.

Table 4.6 Summary of Mark II containment performance perspectives.

Failure mode	Failure mode importance	Important design features, operator actions, and model assumptions	Suggested plant improvements
Early failure	Quite high for several plants in this group Isolation failures not important	Overpressure failures, primarily from ATWS, are found to be important FCI and direct impingement of core debris are important in some of the analyses Specific design features, especially reactor pedestal design, play an important role in many analyses Assumptions regarding the likelihood and magnitude of some severe accident phenomena vary considerably in the analyses	Relaxation of drywell spray initiation criteria Further study and training on reactor depressurization
Bypass	Not important for this plant group	Low frequency is reported in CDF analysis	None identified
Late failure	Probability quite high for this plant group	High-pressure and temperature loads caused by core-concrete interactions are an important failure mode Excessive SRV discharge into a hot suppression pool is also found to lead to late failure in some cases Venting is not important for most plants in this group	Reconsideration of current containment flooding guidelines Modification of current venting procedures

Accidents progressing to structural failure of the containment, particularly in the drywell, before or shortly after reactor vessel failure lead to the most significant radionuclide releases. As indicated in Figure 4.4, the conditional probability of these 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, but modeling assumptions play a role as well. The following failure mechanisms have been found to lead to early failure of Mark II containments:

- containment overpressure failure caused by a loss of CHR or (inadequate CHR) is important in most Mark II IPE analyses.

- FCI and direct impingement of core debris on the containment boundary play a significant role in some of the analyses.

- rapid pressure and temperature increases at the time of reactor vessel failure are significant in only a few Mark II IPE analyses.

As noted for Mark I containments, these failure mechanisms are important to risk because of the relatively short time available for radioactivity decay, natural deposition processes, and accident response actions. In addition, drywell failure implies that radionuclides released from the damaged core bypass the suppression pool. (Significant retention can occur if aerosol radionuclides pass through a suppression pool.) Because of the relatively short time to radionuclide release and the magnitude of the release, these failure mechanisms have been found to be important to all risk measures (i.e., acute and latent health effects including land contamination) in previous PRAs.

Containment venting does not play a significant role in accident progression in Mark II plants, except in the LaSalle 1&2 analysis. Accidents that bypass containment (such as ISLOCAs) or involve containment isolation failure were not important contributors to the CDF in any of the IPEs for Mark II plants. These accidents are also not important to the likelihood of containment failure because their frequencies of occurrence are so much lower than the frequencies of early structural failure caused by other accidents that dominate the CDF.

Specific plant features play an important role in accident progression in Mark II containments. While the designs of the primary containment for all Mark II plants are similar, and all of them are inerted with nitrogen, there are differences that may affect accident progression and the resultant containment failure modes. The most important design differences are as follows:

- Containment Construction — The primary containment of most Mark II plants is of concrete construction; however, WNP2 uses a free-standing steel vessel as its primary containment. In the WNP-2 IPE, the licensee assumed that if the reactor vessel fails at elevated pressure and an HPME results, sufficient melt could escape the pedestal cavity and damage the containment shell in a manner similar to the shell melt-through failure postulated for Mark I containments.

- Reactor Pedestal Floor Elevation — Among Mark II plants, Limerick 1&2 and Susquehanna 1&2 are BWR 4 reactors while the others are BWR 5 reactors. This difference in BWR type does not itself have a significant effect on accident progression within the containment. However, the reactor pedestal cavity design may. In general, BWR 5 plants have a recessed in-pedestal region (reactor cavity) BWR 4 plants have a flat in-pedestal floor at approximately the same elevation as the ex-pedestal drywell floor. After vessel failure and the discharge of core debris a cavity with its floor at the same level as the rest of the drywell floor is more likely to allow the corium to spread out through the doorway onto the drywell floor, with the possibility of contacting and eroding the downcomer pipes thus creating a suppression pool bypass condition.

- Presence of Downcomers in the Reactor Pedestal Region — Except for Nine Mile Point 2, there are no downcomer pipes in the cavity region of Mark II containments. For Nine Mile Point 2, with downcomers inside the pedestal region, corium released from the reactor vessel may more easily enter the suppression pool, thus increasing the potential for a severe FCI in the suppression pool. Among the plants without downcomers in the cavity region, LaSalle has no water in the wetwell directly below the cavity area.

- Presence of Drain Tubes — In all Mark II plants considered except Susquehanna 1&2, there are drain tubes located in the drywell floor. These drain tubes may fail as a result of direct corium attack or FCI after vessel breach. The failure of the drain tubes will result either in direct containment failure or in a suppression pool bypass.

- Size of the Reactor Pedestal (Cavity) Area — In LaSalle 1&2, the reactor cavity is large enough to contain all of the core debris postulated to be released at vessel breach. This means that failure of the downcomers in the drywell floor (and consequent suppression pool bypass) as a result of direct contact with core debris is less likely than if the core debris could flow out of the cavity and across the drywell floor. On the other hand, the large amount of water that could accumulate in this cavity could lead to significant fuel/coolant interactions when the vessel fails.

The variation in predicted Mark II containment performance can be attributed in part to design differences, however differences in modeling assumptions also play a role. For instance, the small early failure probability for Nine Mile Point 2 may be partially explained by the significantly higher containment free volume-to-thermal power ratio of this plant compared to other Mark II plants. WNP2 has the smallest containment free volume to thermal power ratio. In addition WNP2 uses a steel containment and its higher failure probability may also be partially attributed to the consideration given in the IPE to containment failure induced by the impact of a hot debris jet on the containment shell during HPME.

The higher failure probability for LaSalle 1&2 may be partially attributed to the assumption that the cavity drain pipe failures caused by FCI lead to containment failure because a valve in the line fails outside of containment. In WNP2 there is also a high probability of drain line failure from core debris attack or steam explosion, but this is modeled as a suppression pool bypass (not as a direct containment failure). However, ex-vessel steam explosions may cause the WNP2 containment to fail in other locations, but with a smaller probability. Drain line failure in Limerick 1&2 is also modeled as a suppression pool bypass scenario. In WNP2 and Limerick 1&2, drain line failure leads to containment failure only if suppression pool cooling is lost. All of the reasons for the observed variation in the conditional probabilities of early failure for the Mark II containments were not apparent, since the contributions from the different containment failure modes were not discussed in detail in the IPE submittals.

The approach used in the Susquehanna 1&2 IPE differs in many respects from that used in other IPEs. As a result, the Susquehanna 1&2 CDF (9E-8/ry) is orders of magnitude lower than that of other plants, and the conditional reactor vessel and containment failure probabilities are also very low. Of the total CDF, only about 5% involves vessel breach, 1% involves containment failure, and less than 0.1% involves both containment failure and vessel breach. More than 80% of the CDF is attributed to ATWS events, as is more than 95% of containment failure.

Although many improvements in equipment, procedures, and training have been credited in the Susquehanna 1&2 IPE analysis, several assumptions used clearly are important contributors to the significantly lower CDF and containment failure probabilities. One of the most important involves emergency procedures and operator actions. In the Susquehanna 1&2 IPE, the licensee assumed that the EOPs cover all credible plant conditions without ambiguities, inadequacies, or improper actions; that the operator will not make any procedural errors; and that the operator execution error is comparable to the unavailability rate of the equipment required in the operator action. In addition to the high recovery probabilities used in the IPE, the licensee also assumed that there will be a gradual release of core debris from the vessel after breach, and the IPE did not consider most of the loading conditions associated with vessel breach, that may cause an early containment failure. The licensee further assumed that the corium is quenched, and CCI is prevented, if a water pool is present and continuously resupplied when the debris pour commences; that downcomer and drywell shell melt-through are prevented if there is an overlying water pool for the corium; and that containment venting or failure will not result in the loss of core injection as a result of adverse operating conditions. In general, accident progression issues that have significant uncertainties were either not treated, or were included in the IPE with simplifying and optimistic assumptions.

Accidents in which CHR is lost or inadequate are important contributors to early containment failure in the IPEs for Mark II plants. In these accidents, the containment is pressurized by steam generation from an overheated suppression pool, and containment failure often occurs before reactor vessel failure. ATWS sequences in which CHR systems are inadequate are primary contributors to these accident types in most of the IPEs. The LaSalle 1&2 IPE reports a conditional containment failure probability before vessel breach of about 0.1. For Limerick 1&2, the major contributor to early failure is ATWS with a probability also of about 0.1. In Nine Mile Point 2, the leading contributor to early failure is also ATWS, with a probability about an order of magnitude lower. In the WNP2 analysis, early containment failure also seems to result primarily from overpressure failures typified by sequences in which injection is successful, but all viable means of CHR are lost. When the containment fails in these accidents, the energetic release of steam into the reactor building leads to the failure of all injection systems and therefore to core melt.

High-pressure and temperature loads at the time the reactor vessel fails are not significant contributors to early containment failure for most Mark II containments. As in Mark I plants, this failure mechanism could occur in Mark II 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 how significant this failure mechanism is to early containment failure for individual Mark II containments. However, this failure mechanism only appears to contribute significantly to early failure in the Nine Mile Point 2 IPE, and the early failure probability for this plant is small (on the order of 0.05).

Ways of preventing or mitigating the pressure and temperature loads at vessel melt-through include enhanced RCS depressurization capability, containment venting, and spray operation. Of these possible actions, RCS depressurization is potentially the most effective.

Enhancing the capability for emergency depressurization was recommended for Mark II containments as part of the NRC's CPI program. However, the potentially adverse effects of depressurizing too soon, noted by some utilities with Mark I plants apply to Mark II plants as well. For instance, in the WNP2 IPE, the licensee performed a sensitivity study for depressurization for short-term SBO sequences. According to the IPE, without a source of coolant makeup, the inventory lost during depressurization will result in the core melting about one hour sooner than if the system were left at high-pressure. The initial conclusion from the sensitivity analysis is that depressurization of the primary system should be delayed as long as possible; however, once core melt begins it seems prudent to depressurize as quickly as possible. This has two benefits. Namely, if power is restored, LPI systems become available as sources of core cooling, and if depressurization is at least partially successful, the chances are enhanced for delaying containment failure or maintaining the integrity of the containment.

Again, similar to some Mark I IPEs, a number of Mark II IPEs (Limerick 1&2 and Nine Mile Point 2) mention the possible benefit of relaxing the restrictions on the use of drywell sprays for accident management. This would help to control drywell temperature and provide some radionuclide release control.

High-pressure and temperature loads caused by core-concrete interactions are significant contributors to late containment failure for Mark II containments. Gradual pressurization at high temperatures caused by noncondensible gases and steam released from the drywell floor and the reactor pedestal area during core-concrete interactions can induce the failure of Mark II containments several hours after vessel failure. This failure mechanism occurs because of the relatively small volume of Mark II containments. Failure can occur either in the wetwell or in the drywell. Generally, this failure mechanism is less significant than the early failure mechanisms discussed above because of the longer time available for radioactive decay, natural deposition processes, and accident response. However, even for late failures, if the failure location is in the drywell, significant radionuclide release can occur, making this failure mechanism more important. Containment strength as a function of temperature is also an important issue for Mark II containments because drywell temperature can be very high (up to 1000°F) during CCI. The containment pressure capability is weakened at high temperature and the drywell seals (e.g., drywell head seal) may fail at high temperature. Core-concrete interactions in Mark II containments can also lead to reactor pedestal failure with subsequent reactor vessel structural support failure. In some Mark II containments (i.e., those with downcomers or drain lines in the pedestal floor) suppression pool bypass can occur after vessel failure as a result of downcomer failure or drywell floor failure with subsequent containment overpressure from steam.

The significance of these failure mechanisms to late containment failure is determined by the design configuration of the drywell and reactor pedestal area, the availability of sprays or venting, and modeling assumptions regarding the quantity and temperature of core debris dispersed across the drywell floor.

In some IPEs, late containment failure also results when significant discharge occurs from SRVs into a hot suppression pool. Containment failure is assumed to occur in the Limerick 1&2 and Nine Mile Point 2 IPEs when substantial power is produced in the core and discharged through the SRVs to a suppression pool already at high temperature. 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.

With the exception of one plant, containment venting does not play a significant role in accident progression for plants with Mark II containments. Containment venting can be used under a variety of situations. Venting can be used to prevent containment failure by providing a controlled release of the containment atmosphere if the containment pressure approaches a predetermined primary containment pressure limit, (PCPL). Venting would usually be used in situations where there is a gradual increase of containment pressure. Under these conditions, venting would usually be grouped among the late release categories. Wetwell venting is the preferred venting path because of the radionuclide scrubbing capability provided by the suppression pool for wetwell venting. Containment venting for containment pressure control is also used during containment flooding. Drywell venting is required after the wetwell venting path is submerged during the containment flooding process. Since drywell venting does not have the benefit of suppression pool scrubbing, the associated releases may be quite severe. In addition to containment pressure control, containment venting could also be used for combustible gas control, should the containment become deinerted.

Excluding LaSalle 1&2, the conditional probability of containment venting is 0.1 or less in the IPEs for plants with Mark II containments and any releases are grouped with the late failure category. LaSalle 1&2 is found to have a total containment venting probability of about 0.5, (i.e., about 0.4 from late venting and the remainder from early venting). Late venting is frequently used in the LaSalle 1&2 IPE for transient events. About one-fifth of the total venting probability occurs for sequences without vessel breach. Operators at LaSalle 1&2 are trained to vent the containment to stay below the PCPL curve. According to the licensee, intermittent venting in this manner will not create severe environments and comprise equipment.

Although the probability values of the various venting modes are usually not presented in the IPE submittals, some information can be inferred. For example, the Limerick 1&2 IPE considers all three venting modes (i.e., for combustible gas control, containment pressure control, and containment flooding), and the total venting probability is split approximately equally between drywell venting and wetwell venting. In the Nine Mile Point 2 IPE, nearly all of the venting probability determined in the Level 2 analysis is associated with drywell venting, probably used as part of the containment flooding procedure. In the Susquehanna 1&2 IPE, venting is only used when there is no core damage. In the WNP2 IPE, according to the containment event trees, venting is used only when CHR is lost. The window of opportunity for containment venting for WNP2 is rather small since the emergency operating procedures call for venting at about 40 psig; however, if the differential pressure across the valve disk in the preferred vent path exceeds 50 psig, the operators cannot generate sufficient force to open it. This limited opportunity for venting in WNP2 may be part of the reason why venting is not of greater benefit for sequences where CHR is lost.

Several of the Mark II IPEs discuss modifications to current venting procedures. The Nine Mile Point 2 analysis, for example, shows that containment failure is predicted to occur below the currently recommended Nine Mile Point 2 venting pressure when the containment temperature is greater than 650°F. However, this IPE also noted that, at lower containment temperatures, venting at the recommended pressure of 45 psig results in a radionuclide release substantially greater than if venting were not called for until a higher pressure is reached. Therefore, the IPE suggests recalculating the venting initiation criteria and inclusion of a temperature-dependent venting pressure.

The Nine Mile Point 2 IPE also questioned the venting called for in the BWR Emergency Procedure Guidelines (EPGs) when implementing the containment flooding contingency (i.e., drywell venting or venting of the reactor vessel through the main steam isolation valve (MSIVs)). As an alternative, the IPE advocates possible improved response for containment flooding that does not require opening the RPV vent and avoids using the drywell vent unless no other alternative exists. According to the IPE, alternative actions have been shown to produce substantially lower releases and much longer times to failure. The IPE submittal stated that even no action is likely to be preferable to the action directed by the EPGs. It should be noted that containment venting may be used in sequences where the containment fails as a result of other causes. In the IPEs, these cases are usually grouped into the containment failure category that has a more severe release.

Accidents that involve failure to isolate containment or that bypass containment have low frequencies for plants with Mark II containments. Isolation failures can be preexisting or can occur at the time of the initiating event. If the isolation failure is large and core melt occurs, radionuclide release can also be large. In addition, because the containment is open at the time of core damage, the offsite consequences can be significant. These events are not significant in BWR Mark II plants because of their relatively low frequencies. Preexisting isolation failures in Mark II plants can be precluded because the containment atmosphere is inerted with nitrogen. Therefore, any loss of containment atmosphere associated with preexisting leaks can easily be detected. In addition, failure to isolate containment on demand is found to be a relatively low-frequency event compared with the frequencies of other accidents that can cause early structural failure of the containment. In the Nine Mile Point 2 analysis, for instance, isolation failure occurs for only 1% of the total CDF despite the fact that Nine Mile Point 2 uses motor operated valves at some of the isolation points that would not automatically close under SBO conditions. The IPE indicated that it could be useful in future operator training to emphasize the need to locally close these valves to provide isolation.

If the pressure boundary between the high pressure RCS and a low pressure auxiliary system fails (called an ISLOCA) a LOCA outside containment can occur. If water cannot be supplied to the reactor, core damage will occur and a direct path can exist to the environment. Therefore, these accidents can lead to a large early release of radionuclides. However, ISLOCAs are not significant contributors to early containment failure for BWR Mark II containments because of their relatively low frequency compared with the frequency of accidents that dominate the CDF and can lead to early structural failure. The IPEs reported ISLOCA frequencies that are about an order of magnitude lower in BWR plants than in PWRs. The higher PWR frequencies are, in part, attributed to the contribution from SGTRs in PWRs.

4.2.3 BWR Mark III Perspectives

Table 4.7 lists the Mark III containments (4 single-unit BWRs) described in four separate IPE submittals.

All four plants use 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 a Mark I or Mark II containment while the containment design pressure and estimated failure pressure are significantly lower than those of Mark I and Mark II containments. Because of their relatively larger volume, Mark III containments are not inerted. Instead, they rely on glow plug igniters to burn off accumulating hydrogen during a severe accident and prevent energetic hydrogen events.

The IPE results for this group are shown in Figure 4.5 and summarized in Table 4.8.

Table 4.7 Plants (per IPE submittals) in Mark III containment group.

Clinton	Grand Gulf 1	Perry 1	River Bend

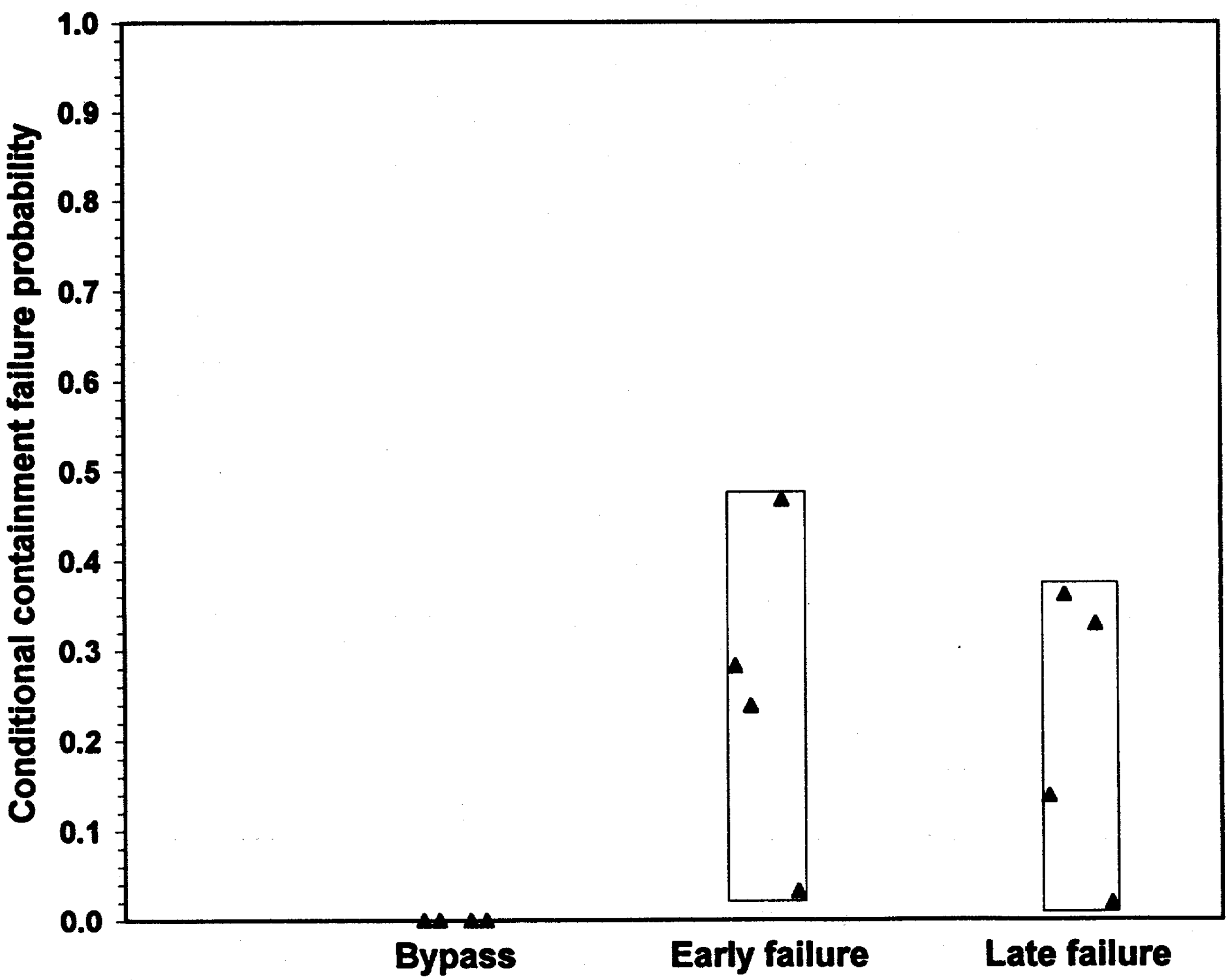


Figure 4.5 Reported IPE CCFPs for BWR Mark III containments.

Table 4.8 Summary of Mark III containment performance perspectives.

Failure mode	Failure mode importance	Important design features, operator actions, and model assumptions	Suggested plant improvements
Early failure	Quite high for a number of plants Isolation failures not important	FCIs and hydrogen burns are found to be important causes ATWS is the leading contributor in one analysis Venting through the MSIVs is a significant contributor in one analysis	Ensuring igniter reliability
Bypass	Not important for this plant group	Low frequency is reported in the CDF analysis	None identified
Late failure	Smaller probabilities than for Mark I and Mark II, but still significant	Late combustible gas burns and phenomena associated with core-concrete interaction are principal contributors Late venting has a significant probability for some plants in this group In-vessel recovery plays an important role for most plants in this group	Enhanced operator training on severe accident phenomena

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. The following mechanisms have been identified in the IPEs as important for early containment failure in Mark III containments:

- early containment failure is primarily caused by energetic events such as FCIs and hydrogen burns.

- one plant identified ATWS sequences as the leading contributors to early containment failure.

- primary reactor system venting via the MSIVs, as found in the procedures, was used in one plant and resulted in a significant release mode.

As noted for other containments, early failure mechanisms are important to risk because of the relatively short time available for radioactivity decay, natural deposition processes, and accident response actions. If the magnitude of the release is significant, the relatively short time to radionuclide release means that these failure mechanisms have been found to be important to all risk measures in past PRAs. As defined in the IPEs, early radionuclide release results from early containment structural failure before or shortly after vessel breach, as well as containment isolation failure, containment bypass, and some cases of early venting. With the exception of one plant, accidents that bypass containment (such as ISLOCAs) or involve containment isolation failure are not important contributors to the CDF in the IPEs for Mark III plants. These accidents are also not important to the likelihood of containment failure because their frequencies of occurrence are so much lower than the frequencies of early structural failure caused by other accidents that dominate the CDF. The exception is Clinton, where early structural failure is calculated to be so small that isolation failure becomes relatively significant.

Hydrogen combustion events or energetic events at vessel failure are the leading contributors to early containment failure for Mark III containments. All of the IPEs for plants with Mark III containments considered early containment failure modes caused by overpressurization from noncondensible gases, hydrogen combustion processes, direct containment heating (DCH), and missile and pressure loads resulting from steam explosions. Hydrogen combustion processes are important because there is a large inventory of zirconium in a BWR core. For example, the Grand Gulf core, which contains approximately 80,000 kg of zirconium, has nearly four times as much zirconium as a PWR core such as the one in Zion. Large amounts of hydrogen are produced from the oxidation of this metal during the core damage process. If the glow plugs making up the hydrogen igniter system (HIS) are not working, the hydrogen will accumulate in the containment. In addition, for accidents in which the suppression pool is subcooled, the steam released from the reactor vessel is condensed in the pool. This lack of steam in the containment atmosphere, in combination with the large amount of hydrogen released during the core degradation process, allows mixtures to form that have a high hydrogen concentration. Subsequent ignition of the hydrogen by either random sources or the recovery of AC power can result in loads that not only threaten the containment but can also pose a significant challenge to the drywell structure. While the causes of early containment failures are not discussed in detail in most IPE submittals for Mark III plants, early containment failure seems to be caused primarily by energetic events, such as FCIs or hydrogen burns.

The conditional early failure probability varies among the four Mark III plants from about 0.03 to 0.5. However, this wide variability in the data mainly results from the small failure probability assigned to Clinton. With the Clinton data excluded, early failure probability among the other three plants varies from about 0.2 to 0.5. The Clinton analysis is discussed further below.

ATWS loads are identified in one IPE as the only mechanism capable of causing an early containment failure. The Clinton IPE discussed and dismissed containment failure mechanisms associated with vessel blowdown forces, invessel and ex-vessel steam explosions, thermal attack of penetrations, DCH, and core-concrete interactions. Only overpressurization from steam generation during ATWS and hydrogen combustion during SBO were found to have the capability to raise containment pressure to the failure point within the first 48 hours after accident initiation. Only ATWS sequences, causing overpressurization of the containment before vessel breach as a result of excessive SRV discharge to the suppression pool, were identified as leading to early failure. The dismissal of other failure mechanisms may be attributable in part to design differences between Clinton and other Mark III containments. These differences are discussed further below.

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 MSIVs results in an early release that is the most severe release mode for Grand Gulf. According to the Grand Gulf IPE, the BWR emergency procedure guidelines direct MSIV venting for containment flooding in response to a loss of RPV level indication. The procedure requires establishment of a vent path to the RPV as containment flooding proceeds beyond the top of the drywell weir wall. This vent path is realized by bypassing the containment interlocks and opening the MSIVs regardless of potential releases. This results in a release that bypasses the containment.

Principal contributors to late failures in Mark III containments are late combustible gas burns and phenomena associated with core-concrete interaction. The phenomena analyzed in the IPEs for Mark III plants that may cause late containment failures include (1) overpressurization with high temperatures associated with noncondensible gases and steam, or combustion processes; or (2) containment basemat melt-through caused by basemat penetration of core debris; and (3) vessel structural support failure caused by core debris erosion. Most of the IPE submittals did not provide a detailed discussion on the causes of late containment failure; therefore, specific contributions to late containment failure as a result of the above containment phenomena are not known. Late failures can be inferred to result primarily from late combustible gas burns and pressure and temperature increases, as well as erosion, from core-concrete interaction. High drywell temperatures are identified as leading to drywell leakage in some of the IPEs. Excluding containment venting, the probability of late containment failure is in the 0.1 to 0.2 range for most of these plants. As in the case of early failures, the exception is the Clinton IPE, for which the probability of late containment failure was calculated to be very low.

One IPE for a Mark III plant reported a significantly lower containment failure probability than the IPEs for the rest of the plants in this group. As noted above, both early and late containment failure probability in the Clinton IPE analysis was much below the values found for other plants with Mark III containments. According to the Clinton IPE, the small containment failure probability is attributed to the large size and greater strength of this containment compared to other Mark III plants. The total free containment volume-to-thermal power ratio is more than 600 ft^3/MWt for Clinton, while it varies from about 400 to 500 ft^3/MWt for other Mark III plants.

According to the Clinton IPE, containment failure will not occur for transient and LOCA events. Although containment venting could be used during transient and LOCA events, the IPE did not provide any release classes for this venting since the Clinton calculations showed that the venting pressure is not reached in these events, and, consequently, containment venting is not carried out. Furthermore, even if the containment were vented, the releases would be small. As a result, early failure occurs only for ATWS sequences that overpressurize the containment, and late containment failure and isolation failures occur only in SBO sequences in the Clinton IPE. According to the Clinton IPE, late containment failure is caused by hydrogen combustion, which occurs when power is recovered 24 hours or more following accident initiation. In addition, the Clinton IPE indicated that decay heat power levels alone will be insufficient to cause failure of the containment as a result of overpressurization within the period covered by the containment analysis (48 hours after event initiation). In general, the significant difference in containment failure probabilities between the Clinton IPE and other IPEs for Mark III plants is attributable, in part, to plant-specific features (e.g., large containment volume) and in part to the differing assumptions used.

Late containment venting is calculated to have a significant probability in some Mark III IPEs. Containment venting is used to prevent containment failure by permitting a controlled release of the containment atmosphere if the containment pressure approaches a predetermined limit (the PCPL). The MSIV venting scheme used in the Grand Gulf IPE analysis has been discussed and is not considered in the other IPEs. Most of the venting described in the IPEs for Mark III plants is scrubbed by the suppression pool; therefore, the releases associated with this venting are small even if the venting time is early. With the exception of the Grand Gulf MSIV venting, early containment venting, (i.e., venting before vessel breach) does not play a significant role in the IPE analyses.

The conditional probability of late containment venting for the Mark III IPEs varies considerably and a zero probability was assigned to venting in the radiological release logic for the River Bend IPE. This is because the vent for River Bend consists of a 3-inch line through the steel containment, which is too small to prevent prompt containment overpressure failure. As a result, venting is credited only if CHR is lost. According to the River Bend IPE, the CHR system is quite reliable and, in those sequences in which CHR would likely to be unavailable (e.g., loss of offsite power), venting would also not be available.

Although the probability of late venting was less than 0.02 in the Clinton IPE submittal, the total venting probability indicated in the Clinton containment event trees (CETs) was more than 0.1. However, as noted earlier, most of this probability was assigned to the "no release category" in the submittal because analysis results showed that the venting pressure would not be reached for some CET sequences in which venting is assumed to occur in the CET quantification.

Invessel recovery plays an important role in the accident progression analyzed in the IPEs for Mark III plants. After core damage, vessel breach can be prevented if coolant injection is restored to the RPV. With the exception of the Grand Gulf IPE, where the probability of invessel recovery is not reported, invessel recovery is significant for the IPEs for Mark III plants. The conditional probability of avoiding vessel breach after initial core damage for Clinton is about 0.6, and, according to the results presented, these cases do not involve containment failure. However, containment failure or containment venting are predicted in the accident progression of other Mark III IPEs, even if there is no vessel breach. For example, the total invessel recovery probability of 0.5 for Perry is split into 0.05 involving early containment failure, 0.2 involving containment venting, and 0.25 involving no containment failure. Similarly, the 0.6 recovery probability for River Bend, is split into 0.2 for early failure, 0.1 for late failure, and 0.3 for no containment failure. Even with containment failure, the release associated with invessel recovery is generally small because, in the majority of the cases in which vessel breach is averted, the releases are scrubbed as they pass through the suppression pool. Furthermore, if the vessel does not fail, there are no ex-vessel releases such as from core-concrete interaction.

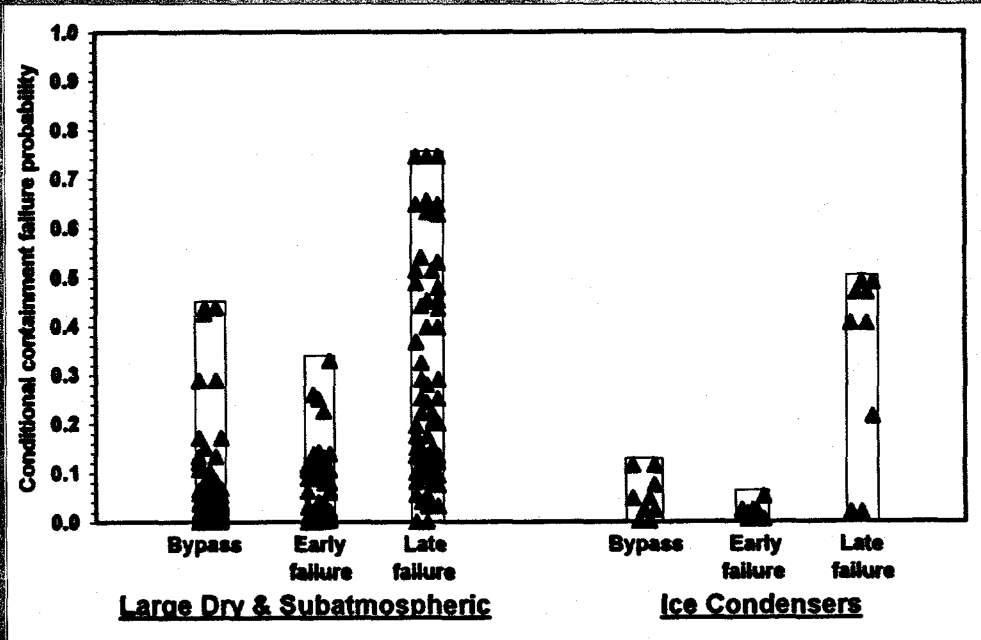
Containment bypass and containment isolation failures are small for most IPEs of plants using Mark III containments. The primary contribution to containment isolation failures comes from SBO events, but isolation failures were small or negligible in all IPEs, with the exception of Clinton. For both River Bend and Clinton, isolation failures were calculated to be less than 5% of CDF. While this is small compared to early containment failure for River Bend, it is of the same order as the small early structural failure calculated in the Clinton IPE. The probability of containment bypass was found to be negligible in all IPEs. However, the MSIV venting discussed in the Grand Gulf IPE leads to a release that bypasses containment and is the most important failure mode in that analysis with respect to radionuclide release.

4.3 PWR Containment Performance Perspectives

As indicated in Section 1.4, for the purpose of identifying containment performance perspectives from the IPEs submitted for PWR plants, the PWRs were separated into two groups according to containment type. Specifically, PWRs are classified as large dry containments (including those operating 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. Big Rock Point is discussed at the end of Section 4.3.1.

Significant variability exists in the contributions of the different failure modes for both containment groups. This variability results from plant-specific design features, as well as the modeling assumptions used in the different IPE analyses. The uncertainty of the phenomena associated with HPME, for instance, is reflected in the variation in the likelihood and magnitude of HPME loads found in the IPEs. Differences in assigning credit for invessel recovery of the core after core damage also play a role in broadening the range of the reported containment failure results. Other reasons for the variability are discussed in the following sections.

Containment performance results for all PWRs in the two groups are shown in Figure 4.6 and the related perspectives are summarized in Table 4.9. The results indicate that most of the containments in both PWR groups have relatively low conditional probabilities of early failure.



- The low early structural failures for ice condensers relative to the other PWRs appear to be driven more by analysis assumptions than by plant features.
- Bypass, especially SGTR, is important relative to early structural failure for both PWR containment types.
- Isolation failures are found to be signficant in a number of large, dry and subatmospheric containments.
- The likelihood of late failures often depends on the mission times assumed in the analysis.

Figure 4.6 Reported IPE CCFPs (given core melt) and key perspectives on containment performance for PWR plants.

Table 4.9 Summary of containment performance perspectives for PWRs.

Failure mode	Failure mode importance	Important design features, operator actions, and model assumptions	Important plant improvements
Early failure	Relatively unimportant for PWRs, but with some important exceptions Isolation failures relatively important for some large, dry and subatmospheric containments	Phenomena associated with HPME are important for large, dry and subatmospheric containments. Importance depends on RCS pressure at vessel breach, cavity geometry, and modeling assumptions Rapid steam generation and hydrogen burns, as well as direct debris impingement, are important in some of the analyses Susceptibility to direct impingement of core debris on the containment in the seal table room is important to some ice condensers	Adding limited barriers to protect against direct core debris impingement on the containment Emphasizing operator training on manual closure of isolation valves Changing to motor-operated isolation valves
Bypass	Relatively important for most PWRs	Bypass can occur as a result of the high operating pressure and large interface between high-and-low pressure systems SGTR is an important bypass mode for most PWRs	Adding training for procedures to cope with SGTR Implementing primary side depressurization to reduce induced SGTR Alternative, independent source of feedwater to reduce induced SGTR
Late failure	Importance varies for PWRs, ranging from unimportant to very important	The dominant late containment failure mode is overpressurization, which occurs when CHR is lost The limited mission time assumed in some of the analyses is an important reason for some of the low late failure probabilities	Emphasis on increasing the likelihood of maintaining a coolable debris bed

Important factors that impact the probabilities of the failure modes in Figure 4.6 are discussed for each containment group in Sections 4.3.1 and 4.3.2. In general, different factors influence the failure modes for the two groups. For instance, while HPME events often play an important role for early failure in the IPEs with large dry containments, this is not the case for IPEs with ice condenser containments. However, the fact that the early failure probability for the ice condenser containments as a group appears to be lower than that of the large dry containments as a group is more likely a result of the modeling assumptions used in the five ice condenser IPEs rather than any phenomenological or design reasons. In addition, the small number of ice condenser IPEs compared to the large number of IPEs for large dry and subatmospheric containments results in a less diverse analysis range for the ice condensers.

Containment bypass, especially associated with SGTR, is important for most of the PWR containments and is the major contributor to large, early releases in many of the IPEs. Induced SGTR is also a significant contributor to bypass in a few of the PWR IPEs.

Late containment failure can result from gradual pressure buildup caused by non-condensible gas release, basemat melt-through, or hydrogen combustion events in conjunction with existing elevated containment pressure. Gradual pressure buildup caused by continued steam production and/or CCI is found to be an important contributor to late failure in the PWR containments.

4.3.1 PWR Large, Dry and Subatmospheric Perspectives

Table 4.10 lists 64 PWR reactor units and 1 BWR unit (described in forty-four submittals) housed in large dry containments.

Table 4.10 Plants (per IPE submittals) in large dry and subatmospheric containment group.

Arkansas Nuclear One 1	Arkansas Nuclear One 2	Beaver Valley 1	Beaver Valley 2
Big Rock Point	Braidwood 1&2	Byron 1&2	Callaway
Calvert Cliffs 1&2	Comanche Peak 1&2	Crystal River 3	Davis-Besse
Diablo Canyon 1&2	Farley 1&2	Fort Calhoun 1	Ginna
Haddam Neck	Indian Point 2	Indian Point 3	Kewaunee
Maine Yankee	Millstone 2	Millstone 3	North Anna 1&2
Oconee 1,2&3	Palisades	Palo Verde 1,2&3	Point Beach 1&2
Prairie Island 1&2	Robinson 2	Salem 1&2	San Onofre 2&3
Seabrook	Shearon Harris 1	South Texas 1&2	St. Lucie 1&2
Summer	Surry 1&2	TMI 1	Turkey Point 3&4
Vogtle 1&2	Waterford 3	Wolf Creek	Zion 1&2

For seven of the PWR units (four submittals) the containments are kept at an internal pressure that is somewhat below atmospheric pressure. All of these containments rely on structural strength and large internal volume to maintain containment integrity during an accident. The IPE results are shown in Figure 4.7 and summarized in Table 4.11.

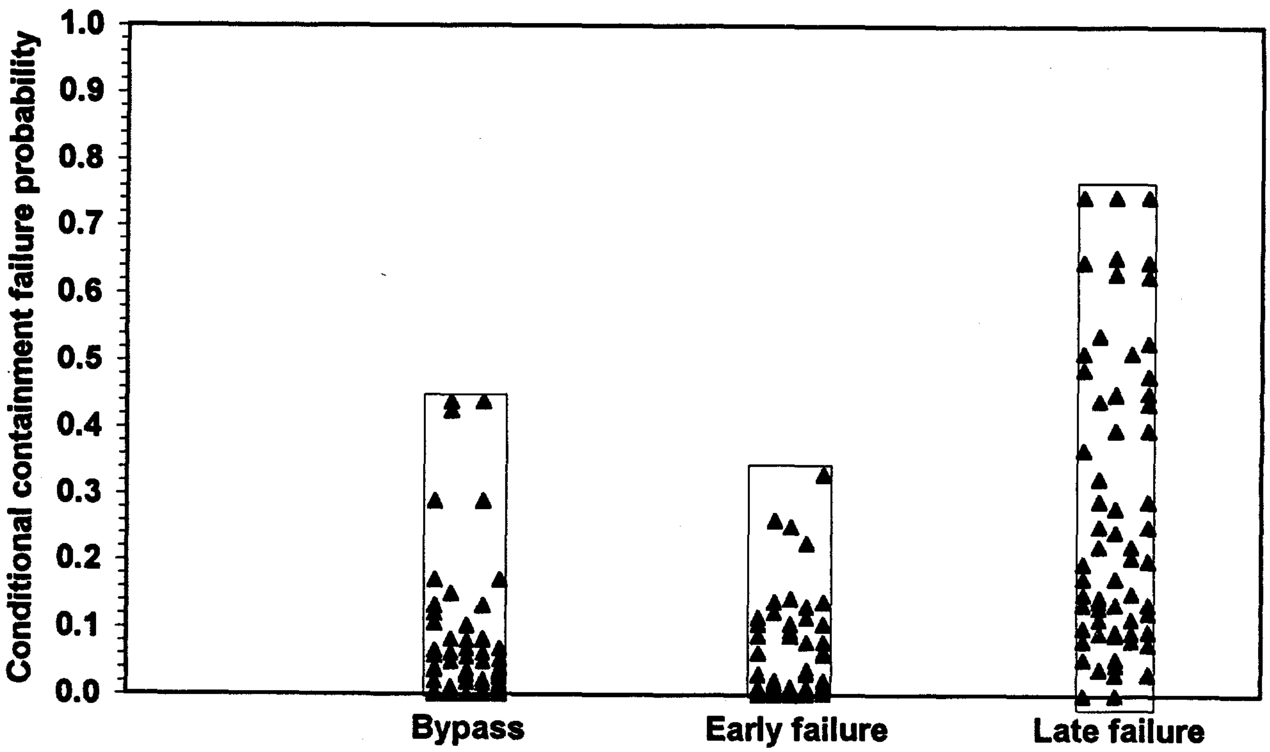


Figure 4.7 Reported IPE CCFPs for PWRs in large dry and subatmospheric containments.

Table 4.11 Summary of large dry and subatmospheric containment performance perspectives.

Failure mode	Failure mode importance	Important design features, operator actions, and model assumptions	Suggested plant improvements
Early failure	Not very important for most plants in this group, but with some notable exceptions Isolation failures important in a number of plants, especially if no credit is given for manual isolation in the analysis, but releases are usually calculated to be small	The leading early challenges for most of the plants in this group are associated with phenomena occurring with HPME. Assumptions made in the analyses regarding these phenomena often determine the early failure probability For a few plants, specific design features lead to unique and significant failure modes. In a number of cases, these involve direct contact of the containment boundary with core debris	Adding procedures for direct RCS depressurization Adding limited barriers to protect against direct core debris impingement on the containment Emphasizing isolation procedures in operator training
Bypass	Relatively important for most plants in this group	Because of the greater pressure differential between primary and secondary systems in PWRs, and the relatively large interface between high and low pressure systems provided by the steam generators, the probability of containment bypass resulting from ISLOCA or SGTR is as large as or larger than early structural failure in many PWR IPEs	Adding training in procedures to cope with SGTR Primary side depressurization to reduce induced SGTR Alternative, independent source of feedwater to reduce induced SGTR
Late failure	Considerable variation among plants in this group, ranging from unimportant to very important	The dominant late containment failure mode is overpressurization, which occurs when CHR is lost The limited mission time assumed in some of the analyses is an important reason for some of the low late failure probabilities	Emphasis on increasing the likelihood of maintaining a coolable debris bed

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

- phenomena associated with HPME pose the most significant early threat for these containments.
- in a few cases, specific design features lead to unique and significant failure modes.
- containment bypass, especially SGTR, is an important source of significant early release.

PWR dry containments are not required to have the intentional ignition systems that are required in PWR ice condenser containments (discussed in Section 4.3.2) since global hydrogen burns, by themselves, are unlikely to cause the failure of these large, robust containments. However, as part of the NRC's CPI program, licensees with large, dry containments were requested to evaluate (as part of their IPE) containment and equipment vulnerabilities to both local and global hydrogen combustion. Structural failures associated with long-term pressure and temperature buildup or penetration of the containment basemat by core debris are not likely as early failure mechanisms but are both possibilities for late failure mechanisms in these containments. The likelihood of these failures depends on the containment strength calculated, the absence or presence of decay heat removal systems, whether the core debris is coolable, and the length of the mission time considered in the analysis. In some large, dry containment IPE analyses, even with decay heat removal systems inoperable, structural failure may never occur in the mission time frame considered.

Isolation failure probability is found to be small for most of these containments, but a number of IPEs report a significant probability for the failure of the containment isolation system. The various containment failure mechanisms are discussed in more detail below.

The most important challenges to containment integrity before or at vessel breach are those associated with HPME. The containment loads associated with HPME are generated by the addition of mass and energy to the containment atmosphere from the following sources:

- blowdown of reactor coolant system steam and hydrogen inventory into the containment
- combustion of hydrogen released before and during HPME
- interaction between molten core debris and water in the reactor cavity or on the containment floor
- direct energy transfer between finely dispersed core materials and the containment atmosphere (i.e., DCH)

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 these energetic events associated with HPME. The pressure of the 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, determine the estimated pressure rise at vessel breach. 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 that used similar analysis methods, the estimated early containment pressure loads were less than the containment pressure capability: therefore, early containment structural failure was assumed not to occur. These IPEs argued that early containment failure modes, such as those discussed above, are not expected to challenge the containment. Bounding hydrogen burn static pressure increases and DCH pressure increases are estimated in these IPEs and compared to a lower bound containment failure pressure, defined in these IPEs as the fifth percentile value of containment pressure capability. The estimated containment pressure obtained in these analyses is about 110 psia for hydrogen burns and below 100 psia for DCH. By contrast, the calculated lower bound of containment failure pressure ranges from 100 to 140 psig. Consequently, early containment failure is considered not credible for these IPEs.

The predicted containment pressure loads are higher in IPEs that reported relatively higher early containment failure probabilities (i.e., from 0.05 to 0.1) than the IPEs discussed above that predict no early containment failure. In these analyses, the containment failure pressure is usually reached when the pressure before vessel breach (the "base" pressure) is combined with the pressure increase at vessel breach. Depending on the individual submittal, the higher pressure loads may be attributed to a high containment base pressure before vessel breach, or a greater pressure increase at HPME, or both. The primary cause for a high base pressure is usually the loss of CHR with successful core injection. A typical example is Arkansas Nuclear One Unit 2 where major contributors to early containment failure (conditional probability of about 0.1) are sequences in which core injection is successful in the injection mode but fails in the recirculation mode, and CHR is not available. Without CHR, the containment pressure is expected to be already high at the time of vessel failure, and is further expected to exceed the containment failure pressure when vessel failure occurs. Sometimes, design features such as reactor cavity layout also play an important role, as at Millstone 2, where early containment failure was dominated by DCH. According to the Millstone 2 IPE, the large DCH failure probability can be partially attributed to the tight reactor cavity, lack of water in the reactor cavity, and lack of an instrument tunnel, typical of some other Combustion Engineering/Bechtel designs.

It should be noted that, for the above cases where the containment base pressure becomes high as a result of loss of CHR, it usually takes many hours for this pressure to build to a significant level. The early failures in these IPEs are defined relative to the time of vessel breach, not relative to the time of accident initiation.

Because of their high containment pressure capabilities and large containment volume-to-thermal power ratios[4.7], large dry containments as a group are not likely to fail before vessel breach by slow pressurization from noncondensible gases and steam or early hydrogen combustion. However, in some IPEs, containment failure is reported to occur before vessel breach either because of containment pressurization from steam and noncondensibles, or because of a combination of containment pressurization and an early hydrogen burn. For example, according to Modular Accident Analysis Package (MAAP) calculations cited in the IPE submittal, the containment failure pressure of approximately 170 psig may be reached before vessel breach at the Palo Verde plant, which has a conditional early failure probability of about 0.1. The DCH peak pressure used in the Palo Verde 1,2&3 IPE is also higher than that estimated in other IPEs, varying from around 130 to 180 psig for cases without sprays, and from around 120 to 140 psig for cases with sprays. These values are considerably higher than those estimated in the IPEs that predict no early containment failure (i.e., about 60 to 100 psig). Another example is Maine Yankee with an early failure probability of about 0.1, where one of the major contributors to early containment failure is hydrogen combustion before and at vessel breach. Maine Yankee's analysis also predicted a high DCH load. The DCH load distribution has a median value of about 115 psia.

As previously noted, an obvious parameter that affects the probability of early containment failure from DCH load is the RCS pressure at vessel breach. For those IPEs that assumed small DCH loads, early failure from DCH is negligible. As a result, the RCS pressure at vessel breach is irrelevant. By contrast, RCS pressure at vessel breach is important for IPEs that predicted significant DCH loads and higher early failure probabilities. RCS pressure at vessel breach depends on the RCS pressure at core damage and any RCS depressurization mechanisms between core damage and vessel breach.

According to the IPE results, the RCS pressure at core damage for PWRs with large dry containments is most likely high or intermediate (the pressure range where DCH is possible). RCS depressurization between core damage and vessel breach can occur as a result of operator actions, because of a stuck open valve, or as a consequence of a temperature-induced hot leg or surge line break. The likelihood of temperature-induced hot leg or surge line break usually used in the IPEs reflects that used in NUREG-1150[4.8] (i.e., about 70% when the RCS is at the

[4.7]The median containment failure pressure (from the containment fragility curve) for large dry containments varies from about 90 psig to over 150 psig, and has an average value of about 135 psig. The containment volume-to-thermal power ratio for large dry containments varies from about 630 to 1220 ft^3/MWt. The average value is approximately 800 ft^3/MWt.

[4.8]USNRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990.

pressurizer PORV setpoint pressure and less than 5% if the RCS is at about 2000 psia). Sensitivity analyses in one IPE indicate that RCS depressurization before vessel failure can reduce the probability of a large early release by as much as 50%. In some IPEs, like Seabrook, added procedures for direct depressurization of the RCS in case of a core melt are listed under CPIs. Other plants for which the IPEs show a relatively high likelihood of DCH-related failure, (like Beaver Valley 2) state that RCS depressurization will be explored further under accident management.

Among PWR IPEs, the conditional probability reported for early containment failure associated with containment overpressurization is exceptionally high for Waterford, at a value of about 0.3. This high probability may be attributed largely to the unusual containment pressure capability curve (or fragility curve) used in this IPE. On the basis of this curve, the containment failure probability is about 0.3 for a 90-psia containment pressure load. This is a high value when compared with that used in other IPEs. With a similar median containment pressure capability, (135 psig for Waterford), other IPE analyses using more conventional fragility curves estimate that containment failure probability is only about 0.05 at pressures of about 100 psig.

In a number of IPEs, specific containment features lead to unique and significant failure modes. For instance, the large probability values of early containment failures found in the IPEs for both Palisades and Davis-Besse[4.9], do not result from the high pressure loads associated with HPME discussed above. Instead, the values are attributed to the special features of the particular containment designs of the plants. The conditional early containment failure probability for Palisades, which is about 0.3, comes primarily from a containment failure mode that is apparently unique to Palisades. The plant feature that contributes to this failure mode is the location of the engineered safety features (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 eventually melt through the pipe wall 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.

For Davis-Besse, the largest fraction of early containment failure is associated with the potential failure of the containment side wall via direct contact with core debris. Although this failure mode is generally unlikely for plants with large dry containments, it contributes significantly to early containment failure for Davis-Besse, one of the few PWR plants that have large dry containments of steel construction. According to the IPE, side wall failure could occur in the event that a significant portion of the core debris is transported from the reactor cavity up to the basement level of the containment at the time of vessel failure. The debris would be dispersed to an area adjacent to the steel containment wall, where the wall is protected by a concrete curb that is 1.5 ft thick and 2.5 ft high. If the debris is not cooled, the concrete could be ablated, leading to a containment failure several hours after vessel failure. This failure mode is defined in the Davis-Besse IPE as one that would result in an early source term.

The IPE for Arkansas Nuclear One Unit 1 is another IPE in which a relatively high early failure probability was not primarily associated with containment overpressurization. According to the Arkansas Nuclear One, Unit 1, IPE, ex-vessel steam explosions and especially debris impingement on the containment liner are significant contributors to early containment failure. The threat from debris impingement is associated with the Arkansas Nuclear One Unit 1 cavity configuration, which provides access to the containment liner through the incore instrument tunnel. As a plant improvement, the IPE suggested the design of a protective barrier inside the incore

[4.9]The probability of early failure for Davis-Besse was significantly reduced when, in response to a request for additional information from the NRC, the licensee found a logic error in the original analysis.

instrument tunnel or along the containment liner just beyond the tunnel.

In general, the IPEs report small contributions to early containment failure by other containment failure modes, such as those associated with invessel steam explosion (alpha mode) and vessel thrust forces (rocket mode). However, in IPEs with a very small overall early failure probability, an alpha mode contribution of a fraction of a percent (based on NUREG-1150 data, for instance) can be as large as the contributions from other early failure mechanisms.

The distribution of conditional probabilities of early containment failure for all large dry containments, presented in Figure 4.7, shows a range from zero to more than 0.3. This range reflects the considerable uncertainties associated with early containment failure phenomena, and includes the effects of some unique containment features in some plants as well as the different assumptions used in the analyses.

Figure 4.7 also shows that large dry containments as a group are quite robust in response to severe accident challenges. These containments are not very susceptible to containment overpressure challenges because of their large-volumes and high structural strengths. The probability of containment failure is further reduced by invessel recovery actions. (That is, gross damage and vessel failure are prevented if sufficient coolant injection becomes available after core damage has occurred and the core is cooled invessel.) A few invessel recovery mechanisms are considered in the IPEs. For cases where LPI is available but the primary system pressure is above the shutoff head of the LPI system, LPI initiation can succeed if RCS pressure can be reduced below the LPI shutoff head. As noted above, RCS depressurization can be achieved by operator actions, or it may occur if there is a temperature-induced hot leg or surge line failure. The induced failure is usually assumed to result in a break size in the RCS equivalent to a large-break LOCA, which will rapidly reduce the pressure, allowing for LPI injection.
Invessel recovery can also occur in loss of AC power sequences if AC power is restored before reactor vessel failure. Another scenario involves large LOCA sequences in which accumulators are required and fail to inject, resulting in core damage. In these sequences, if LPI is operating and continuously injecting water into the vessel, eventual invessel cooling and prevention of vessel failure is likely. Individual IPEs include the above invessel recovery mechanisms in their models in varying degrees. The Beaver Valley IPEs, for instance, take no credit for recovery of AC power or CHR after the time of core damage. Some IPEs that take little or no credit for recovery actions state the intention to further explore invessel recovery within their accident management studies. If the core geometry permits, a number of IPEs mention the possibility (some without taking credit for it) of cooling the core in the vessel via ex-vessel flooding (i.e., filling the cavity with water to submerge a good portion of the reactor vessel and remove heat through the vessel wall).

The IPE results show that the dominant late containment failure mode is containment overpressurization, which occurs when CHR capability is lost. Phenomena that may cause late containment failure include (1) overpressurization with high temperatures caused by combustion process or noncondensible gases and steam, (2) containment basemat melt-through associated with basemat penetration by core debris, and (3) vessel structural support failure caused by core debris erosion.

Basemat melt-through may happen if CCI is not terminated, either because there is no water in the reactor cavity or because the corium is not coolable even if water is available. Since the basemats of most PWR containments are quite thick, eventual penetration of the basemat by core debris is not certain even if a large fraction of the core is involved in CCI and water is not available.

Containment failure as a result of reactor vessel support structure failure is not likely for large dry containments since the reactor vessel is usually away from the containment walls and there are structures located between the reactor vessel and the containment boundaries. Consequently, even if the vessel support structure fails, this will not cause a containment failure in large dry containments.

Therefore, in most PWR IPEs, late failure results from overpressurization. One exception is Haddam Neck, which has one of the largest containment volume-to-thermal power ratios (1220 ft^3/MWt) but a relatively weaker containment structure. (The median containment pressure capability is about 90 psig.) According to the Haddam Neck IPE, late containment failure is dominated by containment basemat melt-through because of the relatively thin basemat in the reactor cavity (about 5 ft compared with approximately 10 ft for most other plants), and a relatively dry cavity floor as a result of the high setpoint for manual initiation of the containment spray system. The IPE for Millstone 2, another plant where the cavity is likely to remain dry, also cites basemat melt-through as the dominant mode of late containment failure.

As indicated in Figure 4.7, late containment failure for the large dry containments considered in the IPEs ranges from zero to about 0.7. This variability is caused by many factors, including what systems are credited for CHR. In some of the IPEs, fan coolers that are not designed as engineered safety features are credited for CHR.

One important reason for the low late failure probabilities found in some IPEs is the use of a 48-hour mission time. In these IPEs, the probability of reaching the containment failure pressure during the 48-hour mission time is very low. This is because containment pressurization is due largely to the generation of noncondensible gases as a result of extended CCI in the cavity, and therefore proceeds relatively slowly. That is, pressurization from steam generation is small, either because of lack of water or because CHR is working. However, in one IPE, there is no late containment failure even if CHR is not functioning. The 48-hour cutoff used in these IPEs also excludes basemat melt-through because penetration of the basemat usually takes longer than 48 hours. These IPEs anticipate that beyond 48 hours, actions such as providing an alternative water source for which emergency procedures may already be in place, along with accident mitigation strategies developed at the Emergency Operations Facilities and the Technical Support Center, would mitigate the basemat melt-through sequences and result in a stable configuration within the intact containment.

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 ISLOCA, SGTR, or temperature-induced SGTR. The probability of ISLOCA and SGTR is determined in the CDF analyses of the IPE. The probability of temperature-induced SGTR is calculated as part of the accident progression analysis. This failure typically occurs if one or more steam generator tubes experience a creep rupture caused by the flow of high-temperature 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. For example, SGTR leads to the most serious releases reported in the North Anna 1&2 and Prairie Island 1&2 IPEs. An exception is the St Lucie 1&2 IPE, where ISLOCA is two and three times more likely than SGTR for Units 1 and 2, respectively.

For temperature-induced SGTR, the probability value used in the IPEs is about 0.01; therefore, temperature-induced SGTR is generally not deemed significant in the IPEs. The exceptions are the Prairie Island 1&2 and Shearon Harris IPEs. In the Prairie Island 1&2 IPE, the conditional probability of bypass is almost 45% of total CDF, of which two-thirds is attributed to temperature-induced SGTR and one third to SGTR-initiated events. In the Shearon Harris IPE, the conditional probability for containment bypass is about 0.1, half of which is attributed to temperature-induced SGTR. The high probability of temperature-induced SGTR in these IPEs results from their consideration of reactor coolant pump (RCP) restart and the high value used for the probability of steam generator tube thermal failure for cases where the RCP is on (0.5 versus 0.01 used in other IPEs).

According to these submittals, the procedural guidance requires the operators to restart the RCPs when inadequate core cooling conditions are indicated. This restart clears the RCP seals and establishes a natural circulation path, resulting in increased steam generator tube heating and the potential for a temperature-induced SGTR. Secondary side depressurization, also included in the procedures for restoring heat removal, can increase the pressure differential across the tubes and thus may further increase the potential for failure. On the other hand, some of the IPEs cite primary side depressurization as a way to reduce the probability of temperature-induced SGTR. In the Seabrook IPE, the addition of an alternative independent emergency feedwater pump, that could be used during high-pressure core melt sequences is also listed as an improvement for reducing temperature-induced SGTR. Most IPEs do not consider the effect of RCP restart, however, and in some that do, a low probability of temperature-induced SGTR is used based on the expected limited duration of RCP operation. This variable treatment of temperature-induced SGTR in the IPEs indicates the large uncertainty associated with this issue.

After the Prairie Island IPE, the highest bypass probability is predicted in the Ginna IPE (~0.4), the majority of which results from the CDF analysis. Ginna has the highest bypass frequency found in any of the IPEs for plants with large, dry containments, almost 4E-5/ry. Another IPE with a high bypass conditional probability (almost 0.3) is the Zion IPE, nearly all of which is attributed to SGTR in the CDF analysis. The IPEs for Braidwood and Byron show a low SGTR likelihood because credit is taken for a new steam generator design that uses smaller diameter tubes. These tubes reduce the leakage from primary to secondary side in the event of a rupture, and reduce the likelihood of core damage during the initial 24 hours.

Isolation failure is important in some IPEs. Isolation failure is assumed to be negligible in some IPEs, and is assumed to have a large conditional probability in others. A large probability of isolation failure is most likely in IPEs that assume a lack of operator actions to locally or remotely close the isolation valves if no containment isolation signal is provided. For example, the conditional probability of isolation failure in the Diablo Canyon 1&2 IPE is about 0.1; this is primarily because little credit is taken for operator action to locally or remotely close the isolation valves. In the H.B. Robinson IPE, the probability of containment isolation failure is about 10% of the total CDF. Here, isolation failure is dominated by a plant damage state involving an SBO followed by an RCP seal LOCA with no injection, and a leakage path back through the containment spray lines. According to the H.B. Robinson submittal, this failure mode has a low release potential because of resistance and possible plugging of the spray nozzles and plate-out in the piping. No operator action to isolate the pathway is credited in the IPE.

Pre-existing isolation failures could be expected to be more readily detected in the subatmospheric containments (i.e., those few PWR dry containments that are kept somewhat below atmospheric pressure). However, one of the largest probabilities for isolation failure is found in the IPEs for Beaver Valley 1 and Beaver Valley 2, plants with subatmospheric containments. These probabilities are large because in these IPEs isolation failure always occurs for SBO sequences. Again, the IPE model does not take credit for operator actions to manually isolate the containment building for these sequences. However, since the leak area associated with this isolation failure is small, it does not significantly contribute to radionuclide releases.

The same holds true in most IPEs where isolation failure has a significant probability. (The leak area associated with the failure is usually small; therefore, the failure does not contribute significantly to radionuclide releases.) One exception is the South Texas Project IPE. According to that submittal, the most important single cause of large early release given a core damage event is a large containment isolation failure. This includes failure to isolate the large supplemental purge penetrations in the unlikely event that a purge is in progress during the accident, and large undetected pre-existing leaks that have been introduced since the last integrated containment leak rate test.

The IPEs for the plants with large dry or subatmospheric containments show that the probabilities for isolation failure vary from zero to about 0.1. Plant improvements to reduce isolation failure probability are discussed in most IPEs where the likelihood of isolation failure was found to be relatively high. For instance, the Ginna IPE cited emphasis of operator training on manual closure of isolation valves upon failure of automatic isolation as an improvement. The IPE for the South Texas Project noted that one plant improvement, based on the early results from the IPE, was the changeover from motor-operated to air-operated containment isolation valves in some containment penetrations.

Big Rock Point is the only BWR plant that has a large dry containment. This containment is a large, steel sphere with a volume of 940,000 ft^3. The containment volume-to-thermal power (240 MWt) ratio for Big Rock Point (about 4000 ft^3/MWt) is significantly greater than those of other plants that use large dry containments (about 1000 ft^3/MWt).

Big Rock Point uses an emergency condenser for decay heat removal. The containment management systems that can be used during accident conditions include an enclosure spray system and the containment isolation system. For this plant, the IPE considers an accident management strategy known as 'Fill-the-Ball'. This strategy is used to provide water to the containment for reactor heat removal in the event of a post-accident system failure. In this strategy, water is continuously provided to the containment such that the lower portion of the reactor vessel is covered. Procedures directing the operators to fill the containment vessel with water are in place for this strategy.

The containment failure probabilities for Big Rock Point are small, about 0.01 for containment bypass and about 0.05 for early containment failure. The probability for no containment failure (with a radionuclide release that is less than or equal to the containment design-basis leakage) is about 0.9. No late failures were found. The probability of early containment failure includes 0.005 from containment isolation failure. Early structural failure primarily comes from ATWS events with failure to inhibit the reactor depressurization system. Containment bypass failure primarily results from failure to isolate the main steam line in sequences involving spurious bypass valve opening or a steam line break outside the containment. Although less important, ISLOCAs also contribute to containment bypass. The low initiator frequency for ISLOCAs, compared to that for spurious operation of the bypass valve, diminishes its importance for the containment bypass category.

4.3.2 PWR Ice Condenser Perspectives

Table 4.12 lists nine PWR units (described in five IPE submittals) with ice condenser containments.

Table 4.12 Plants (per IPE submittal) in ice condenser containment group.

Catawba 1&2	DC Cook 1&2
McGuire 1&2	Sequoyah 1&2
Watts Bar 1	

Seven of the nine ice condenser units have a cylindrical steel containment surrounded by a concrete secondary containment. The remaining two units (the D.C. Cook plants) feature reinforced concrete containments with steel liners, and lack secondary containments. All of these plants use 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 accidentally released from the reactor coolant system. The ice condenser containment consists of an upper compartment, a lower compartment, and the ice condenser chamber through which blowdown steam is forced to pass during a LOCA. 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.

Figure 4.8 shows the failure probabilities for this group. Table 4.13 summarizes the findings. Among the five ice condenser IPE analyses, the most important causes of early containment failure are as follows:

- direct impingement of core debris on the containment in the seal table room
- rapid steam generation, DCH, and hydrogen burns
- overpressurization when CHR is not available

Although the accident progression models in the majority of the ice condenser IPEs used data from the NUREG-1150 Sequoyah analysis, additional plant-specific models result in lower failure probabilities than those 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 (RWST) 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 found temperature-induced SGTR to be dominant because of the restart of the RCPs when inadequate core cooling conditions exist.

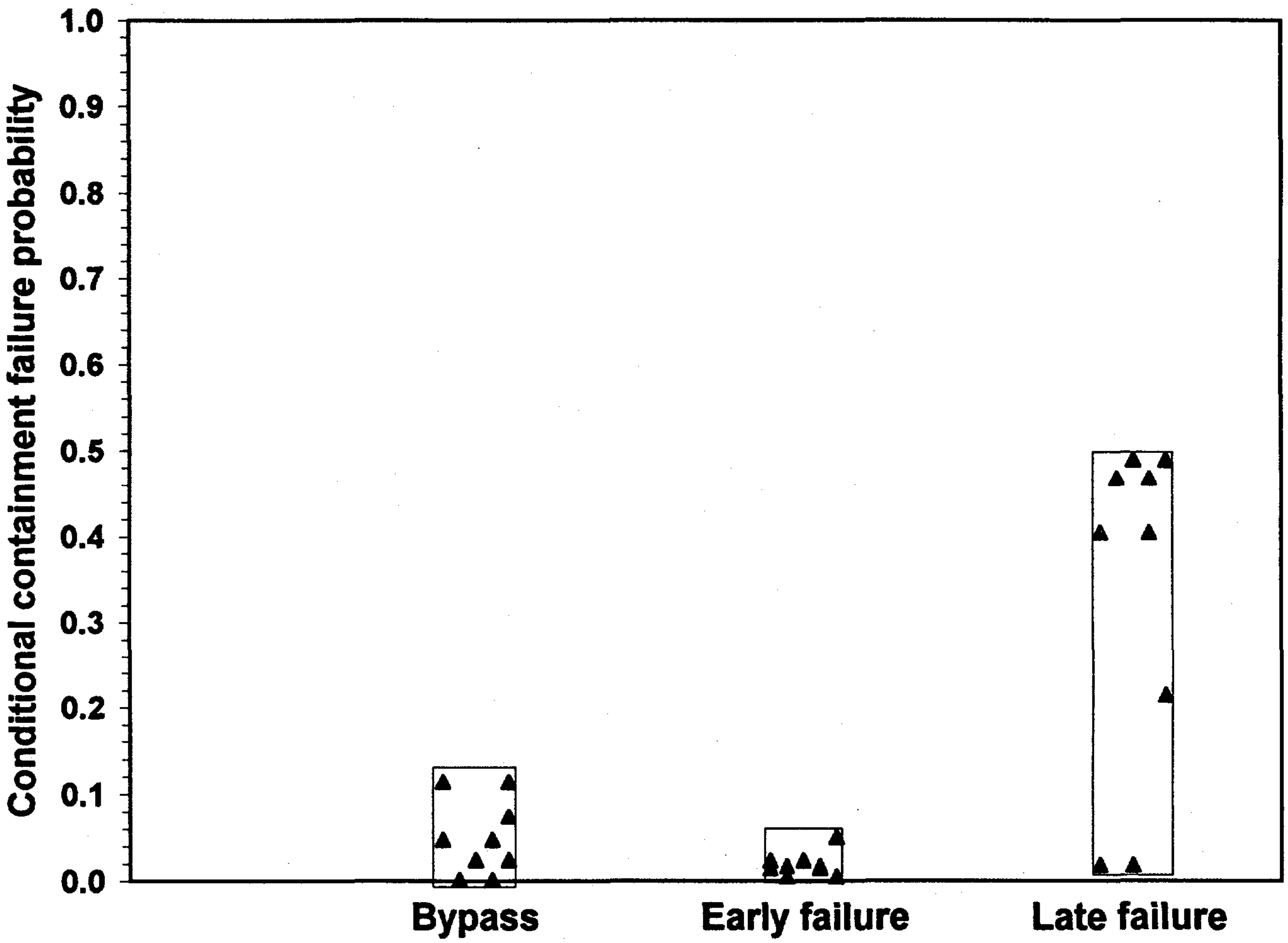


Figure 4.8 Reported IPE CCFPs for PWR ice condenser containments.

Table 4.13 Summary of ice condenser containment performance perspectives.

Failure mode	Failure mode importance	Important design features, operator actions, and model assumptions	Suggested plant improvements
Early failure	Unusually unimportant for the plants in this group Isolation failures found to be unimportant	Causes vary among analyses: direct core debris impingement, DCH, rapid steam generation, and hydrogen burns. Analysis assumptions concerning load magnitude play an important role in the low probabilities found for early failures. Also, the ice condenser is credited with considerable energy-absorbing ability in some of the analyses. Depressurization and deeply flooded cavities are also credited	
Bypass	Significant when compared to early structural failure in these IPEs	Because of the higher operating pressures in PWRs, and the relatively large interface between high and low pressure systems provided by the steam generators, the probability of containment bypass is relatively large in these analyses Induced SGTR is a major contributor in one analysis because of the restart of the RCPs	Procedural changes to cope with SGTR Feedwater flow to faulted steam generator maintained during SGTR Additional procedural guidance on RCP restart
Late failure	Variable from unimportant to more than 50% likelihood	The dominant late containment failure mode is overpressurization, which occurs when CHR is lost Limited mission time is a principal reason for low failure probability in one analysis	Emphasis on increasing the likelihood of maintaining a coolable debris bed

The principal causes of early failure vary among the IPEs for plants with ice condenser containments. All five ice condenser submittals have small probabilities of early containment failure (excluding isolation failure). Although each ice condenser IPE evaluated the containment against similar challenges, the most important causes of early containment failure vary among the five ice condenser IPE analyses. Three somewhat different groups of mechanisms are identified as leading contributors to early failure:

- The leading cause of early containment failure in the Sequoyah and Watts Bar IPEs is direct impingement of core debris on the containment cylinder wall in the seal table room of the containment. In this scenario, core debris is swept out of the reactor cavity during an HPME and comes in contact with the containment boundary in the seal table room. Other important causes of early failure in these two IPEs are invessel steam explosion and HPME/hydrogen burns at vessel breach.

- For Catawba and McGuire, principal contributors to early containment failure include rapid steam generation, DCH, and hydrogen burns. These two IPEs assumed that containment failure caused by the DCH load is unlikely (only a 0.1 probability) if the ice condenser is available to absorb a significant amount of energy. Since the ice condenser is available in most of the sequences for these two IPEs, this assumption probably contributes to a significant reduction in the probability of HPME/DCH failure.

- In the Cook IPE, the effect of HPME is considered for long-term sequence progression in terms of its effect on debris distribution. Early containment failures caused by DCH, steam explosion, and vessel blowdown thrust forces are discounted after brief discussions. Failure caused by hydrogen generation and combustion is also found to be unlikely in the Cook IPE. Early containment failure for Cook is primarily attributed to containment overpressurization when CHR is not available.

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 (80 psig versus 130 psig). No single reason for the lower (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 5 IPEs for ice condenser plants, a relatively small sample, while there are 45 IPEs for plants with either large, dry or subatmospheric containments. Therefore, much greater variation exists in the likelihood of early failure in this larger group.

Depressurization before vessel breach and a flooded reactor cavity reduce the likelihood of early failure in the IPE models. One way to reduce the threat of early containment failure is to depressurize the RCS before vessel breach. The effective mechanisms for RCS depressurization include temperature-induced hot leg or surge line failure, temperature-induced failure of the RCP seals, and the sticking open or deliberate opening of the power operated relief valves (PORVs). Successful RCS depressurization may allow the LPI system to inject to the RCS and thus avoid vessel breach, or eliminate the challenges associated with HPME if vessel breach is not avoided. These depressurization mechanisms are considered in all IPEs except the D.C. Cook 1&2 IPE, in which the loading conditions associated with HPME are not judged to be major concerns. Another factor that may limit the probability of early containment failure is the high likelihood of a deeply flooded reactor cavity. The presence of a large amount of water inhibits the dispersal of debris from the cavity and thus lowers the threat from DCH at vessel breach. This factor is also considered in the IPEs. The D.C. Cook 1&2 IPE mentions additional operator training on the importance of a wet reactor cavity, emphasizing maximum injection from the RWST before switchover to recirculation. While water in the cavity increases the possibility of an ex-vessel steam explosion, the IPEs deem this to be a minor threat.

Although some of the ice condenser IPEs include much data from the NUREG-1150 Sequoyah analysis, their early failure probabilities are less than the NUREG-1150 value for Sequoyah. The IPEs for Sequoyah 1&2 and Watts Bar are similar. Both were prepared by the Tennessee Valley Authority (TVA). While most of the data used in these two IPEs are derived from those presented in the NUREG-1150 analysis for Sequoyah, the early failure probabilities in the Sequoyah 1&2 and Watts Bar IPEs are less than that obtained in NUREG-1150 for Sequoyah. The reasons for this are not apparent from the IPE submittals; however, some data are based on plant-specific calculations. Besides containment loading phenomena, the containment pressure capabilities used in the two IPEs are also different (i.e., greater than those used in NUREG-1150).

Another factor that affects containment failure probabilities is the type of core damage sequences previously obtained. One significant difference is the inclusion of the loss of support system initiators in both the Sequoyah 1&2 and Watts Bar IPEs. The data presented in the IPEs show that the combination of transient and loss-of-support-system initiated events contributes more significantly to the total CDF for the IPEs (0.7 for Sequoyah 1&2 and 0.3 for Watts Bar) than for NUREG-1150 (0.04). Since, NUREG-1150 results indicate that an early containment failure is less likely for transient sequences than for other sequences, the smaller early containment failure probabilities for the IPEs may be partially attributed to the greater fractions of sequences initiated by plant transient or loss of support system initiators.

Furthermore, a review of the IPE submittals shows that while the pressure increase at vessel breach is primarily based on the NUREG-1150 data, the baseline pressures immediately before vessel breach are obtained in the IPEs from MAAP analyses and are smaller than those used in NUREG-1150. Combined with the greater containment pressure capabilities, this leads to smaller containment failure probabilities from the phenomena associated with HPME. Besides the containment baseline pressure, the data used in the IPEs for the calculation of debris impingement, the dominant early failure mode, are based on NUREG-1150 data and should not cause significantly different results.

The IPEs for Catawba 1&2 and McGuire 1&2 are also very similar. Both were prepared by the Duke Power Company. Although the CET structures and the quantification processes for the CETs are similar, the conditional probabilities of early containment failure obtained from the quantification are different. This difference may be attributed to the much higher loss of offsite power and thus SBO, probability for McGuire 1&2. Although the total CDFs are similar for the two plants (about 4E-5), the contribution to the total CDF from loss of offsite power is more than 25% for McGuire 1&2 and less than 5% for Catawba 1&2.

The quantification methods used in the McGuire 1&2 and Catawba 1&2 IPEs are different from those used in the Sequoyah 1&2 and Watts Bar IPEs (and therefore in NUREG-1150). The probability of containment failure from direct contact of core debris seems to be less likely in these IPEs than in some others. According to the McGuire 1&2 and Catawba 1&2 IPEs, this failure mode occurs only if there is a sufficient amount of corium making contact with the containment wall; even if this condition is met, there is a 0.1 probability that the containment will not fail. The likelihood of this condition depends on the configuration of the reactor cavity and the obstructions in the corium flow path. It is assumed in these IPEs that the cavity geometry is "likely" to limit the amount of corium reaching the seal table such that this failure mode will not occur. Therefore, the probabilities of containment failure from debris impingement obtained in the two IPEs seem to be smaller than those obtained in the Sequoyah 1&2 and Watts Bar IPEs that used NUREG-1150 data for this failure mode.

The probability values used in the Catawba 1&2 and McGuire 1&2 IPEs for RCS depressurization may also be higher than those used in other analyses. In addition to the usual depressurization mechanism considered, these IPEs included depressurization by the operators using steam generator PORVs. The probability of RCS depressurization caused by temperature-induced hot leg or surge line failure is considered to be "likely" in the IPEs. Additionally, the application of this depressurization mechanism seems to be less restrictive than in some other analyses. The probability of operator depressurization using the pressurizer PORVs also seems to be more likely and less restrictive in these IPEs.

According to the Catawba 1&2 and McGuire 1&2 IPEs, the probability values used for containment failure from DCH are primarily based on the pressure load developed in NUREG-1150. However, it is assumed in the IPEs that there is a probability of 0.9 that the containment will remain intact in a DCH event if ice is available in the ice condenser. This may cause a lower probability of DCH failure in these IPEs than in NUREG-1150. In addition, invessel steam explosions (alpha mode) are not considered in the Catawba 1&2 and McGuire 1&2 IPEs as a potential failure mode. According to the Catawba 1&2 IPE, hydrogen burns are the primary cause of early containment failure, and events which result in the loss of all ac power dominate this containment failure mode. The high reliability assigned to the hydrogen igniter system, because it can be powered by either offsite or onsite emergency power, is an important factor in keeping the probability of containment failure low. The possibility of providing power to the igniters from another independent source of AC power, the safe shutdown facility onsite, is being investigated at Catawba.

The treatment of the accident progression analysis in the Cook 1&2 IPE is significantly different from that in the other ice condenser IPEs. The CET used in the Cook 1&2 IPE is small, having only eight top events. The quantification process is also significantly different. The CET quantification assigns each core damage sequence to a particular CET end state. Since DCH, steam explosion, vessel thrust force, and hydrogen combustion are assumed in the Cook 1&2 IPE as not likely to cause containment failures, they are not included in the quantification of containment performance. The IPE states that providing additional back up power to the hydrogen igniters would not noticeably decrease containment failure at Cook. Containment failure is primarily caused by overpressurization associated with steaming and/or generation of noncondensible gases.

The dominant late containment failure mode found in the ice condenser IPEs is overpressurization when CHR is lost. The containment phenomena that may cause late containment failures as described in NUREG-1335 include (1) overpressurization with high temperatures due to noncondensible gases and steam or due to combustion processes and (2) containment basemat melt-through due to basemat penetration by core debris. The results of the IPEs show that the dominant late containment failure mode is containment overpressurization, which occurs when CHR capability is lost. One factor that contributes to the high probability of this failure mode is the draining of the RWST following reactor vessel failure. According to the IPEs, the RWST water is likely to drain into the failed vessel and thereby into the reactor cavity after vessel failure, if it has not been injected before vessel failure. The subsequent boiloff of the water leads to ice melt and eventual containment overpressurization.

The probabilities of late containment failure reported in the IPEs for the five ice condenser containments vary significantly. The extremely low failure probability for Cook is due to the use of a 48-hour "mission time" for the accident progression analysis. This means that the IPE containment performance analysis is carried out for an accident progression of 48 hours. If the containment did not fail in the first 48 hours, no containment failure is reported. Dismissal of molten core-concrete attack as a containment failure mechanism in the Cook 1&2 IPE is a result of the assumption of the 48-hour cutoff for containment evaluation. It is assumed in the IPE that this time is sufficient to take action to stop further concrete erosion.

Isolation failures are small for most of the ice condenser IPEs. With the exception of Watts Bar, the probabilities of isolation failure obtained in the IPEs are in general small. The Catawba 1&2 IPE mentions a procedure change to more clearly establish the priority of isolation pathways which must be manually isolated. According to the Watts Bar IPE, the total isolation failure probability obtained from the CDF analysis is about 5 percent of total CDF. The dominant sequence in the plant damage state group that contributes significantly to isolation failure is an SBO sequence in which the operator fails to isolate the containment by failing to close the seal cooling return line after a seal LOCA has developed. In the Watts Bar IPE, part of the isolation failure is binned to a bypass plant damage state in the accident progression analysis.

ISLOCA and SGTR (as an initiator) are the major bypass contributors. In one IPE induced SGTR dominates. Containment bypass failures include those from ISLOCAs, SGTRs, or temperature-induced SGTRs. For the various containment bypass modes, ISLOCA and SGTR are determined in the CDF analyses, and temperature-induced SGTR is an accident progression phenomenon. In all five IPEs, SGTR is the major contributor to the bypass category from the CDF analysis. Some IPEs (D.C. Cook 1&2

for instance) investigated procedural changes to maintain feedwater flow to the faulted steam generator during an SGTR event to reduce releases through a stuck-open safety valve or PORV.

Temperature-induced SGTR occurs if one or more SGTR tubes have a creep rupture due to the flow of high temperature gases from the core to the steam generators when the RCS is at high-pressure. In NUREG-1150 it was assumed that an induced hot leg or surge line break is much more likely than an induced SGTR if such high temperature conditions exist. Consequently, the probability of induced SGTR was assigned a low probability in NUREG-1150. Considerations similar to that used in NUREG-1150 are used in most of the IPEs and, as a result, the contribution from induced SGTR is not significant in the ice condenser plants except for McGuire 1&2 , where the majority of containment bypass is due to temperature-induced SGTR. This high probability for McGuire 1&2 is due to the restart (per procedures) of the RCPs when inadequate core cooling conditions exist. The probability of induced SGTR is assumed to be significantly higher after RCP restarts due to the transport of the hot gases from the core region to the steam generator by forced circulation. Additional procedural guidance, permitting a pump startup only when the steam generator tubes are covered, is recommended in the McGuire 1&2 IPE to eliminate this concern. Induced SGTR is not considered in the Cook 1&2 IPE. The treatment of induced SGTR in the other IPEs is similar to that in NUREG-1150.

4.4 Radionuclide Release Perspectives

It is useful to review the results presented in the IPE submittals regarding radionuclide release, especially early release. Perspectives on the reported IPE early release results are provided below. Chapter 12 discusses the releases calculated in the IPEs in more detail.

Following the usual convention, the source term which defines the severity of radionuclide release is expressed in the IPEs in terms of the fractions of the radionuclides released to the environment to their total inventories initially in the reactor system. These release fractions are predicted in the majority of IPEs using either the MAAP computer code or the parametric source term prediction code developed in NUREG-1150. In some IPEs, results from the calculations of both codes (i.e., MAAP or parametric source term) are presented. Early release is of particular concern because of the potential for severe consequences due to the short time allowed for radioactivity decay and natural deposition, as well as for accident response actions such as evacuation of the population in the vicinity of the plant.

Not all early failures lead to significant release. The containment failure modes that result in an early release of radionuclides to the environment are containment bypass, isolation failure, and early containment structural failure. In BWR pressure suppression containments early containment venting could also lead to an early release. Not all early failures lead to a significant release, since the amount of the release depends on the failure size as well as the removal or "scrubbing" (if any) of some of the radionuclides within the containment that is assumed to take place. What is considered to be a significant release varies among the IPEs.

In many IPEs significant releases includes those release cases that involve a release fraction of volatile radionuclides equal to or greater than 0.1 (i.e., the release fraction of either the iodine and or cesium group is greater than 0.1 of core inventory). This definition can be used to screen the results reported in most of the IPE submittals, and is used for purposes of the discussion in this section, and the additional information on releases presented in Chapter 12. However, in some IPEs release fractions are predicted to be below 0.1 for all containment failure modes. Since there are considerable uncertainties in source term predictions, it seems inappropriate to characterize these IPEs as having zero significant early release. Instead, for these IPEs the probability of containment bypass and the part of early failure that involves a large failure size is used as the probability of early release in the discussion below. Figure 4.9 shows the conditional probability for significant early release of radionuclides by containment type as reported in the IPEs. The reporting of release results in the IPEs varied in the type and detail of the information provided *so that in* some cases the results discussed below have had to be inferred or estimated.

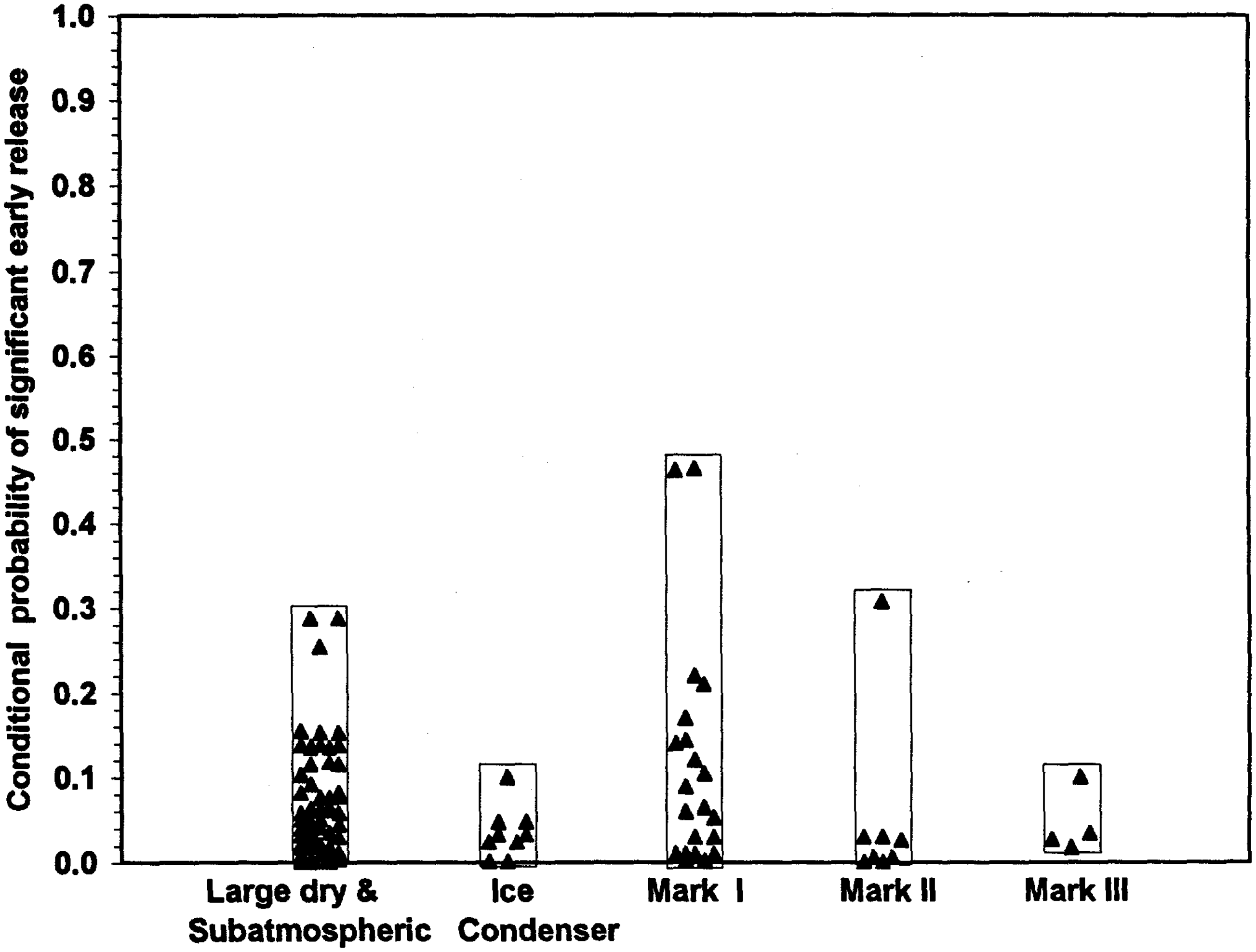


Figure 4.9 Reported IPE conditional probabilities of significant early release by containment type.

Among the BWR plants with pressure suppression containments, those with Mark I containments show the largest variation in the probability and frequency of significant early release reported in the IPEs. As indicated in Figure 4.9, the conditional probability for significant early release reported in the IPEs for Mark I containments varies from less than 0.01 to about 0.5. As discussed above, for some IPEs, even the most severe release sequences are predicted to have release fractions of I and Cs less than 0.1. For example, in the Brunswick IPE, the release fractions for I and Cs are predicted to be less than 0.01 for early containment failure and close to, but still less than 0.1 for bypass releases. The small release fraction for Brunswick is partly due to the use of a concrete structure for the containment. According to the Brunswick IPE, leakage through the cracks of the concrete walls is the most likely containment failure mode and the release associated with this containment failure mode is small. In Figure 4.9, only the probability of bypass is used for significant early release for Brunswick. Consequently, Brunswick has the lowest probability value for significant early release among the Mark I containments. On the other hand, Browns Ferry and Fitzpatrick have the highest probability values (nearly 0.5) for significant early releases. The primary contributor to significant early release for these two plants is shell melt-through. The probability of significant early release for other Mark I plants is equal to or less than 0.2.

With the exception of one Mark I plant, the frequency of significant early release reported for BWR plants is less than 1E-5/ry. The frequency of significant early release reported for Mark I plants varies from less than 1E-8/ry to 2E-5/ry. The highest frequency for significant early release is from the Browns Ferry IPE, due to a combination of high conditional probability and high CDF. It should be noted that the original Unit 2 Browns Ferry IPE has been updated by the licensee to a multi-unit analysis, but no accident progression analysis was carried out for the update, so the results shown are from the original submittal. Except for Browns Ferry, the frequencies of significant early release for all other Mark I IPEs are less than 1E-5/ry.

For Mark II containments Figure 4.9 shows that the conditional probability for a significant early release varies from less than 0.01 for Limerick 1&2 to about 0.3 for WNP2. According to the WNP2 IPE, the three dominant source term categories have release fractions for volatile radionuclides greater than 0.1. It should be noted, however, that although these three release classes are defined as occurring with early failure in the IPE, containment failure and radionuclide release occurs at more than 15 hours after accident initiation. For the LaSalle 1&2 IPE the probability of large containment failure at vessel breach is used as the significant early release probability in Figure 4.9. The frequencies of significant early release reported for Mark II containments vary from less than 3E-8/ry for Limerick 1&2 to about 5E-6/ry for WNP2.

Figure 4.9 shows that the conditional probabilities reported in the IPEs for Mark III containments for significant early release approximately range over an order of magnitude. The highest value, 0.1, is reported for River Bend where the data presented in Figure 4.9 for significant early release includes the early failure cases with large failure size (i.e., containment dome or anchor failure) since the IPE does not report any I or Cs release fractions greater than 0.1. The conditional probabilities of significant early release reported in the other Mark III IPEs are about an order of magnitude lower than River Bend. The frequencies of significant early release vary from about 2E-7/ry for Perry to about 2E-6/ry for River Bend.

The IPE results show that for PWR plants, containment bypass sequences, usually dominated by SGTR sequences, are important contributors to total early as well as significant early radionuclide release. As discussed above, not all early failures involve significant releases. Isolation failure for some of the IPEs involves only a small leak area, and consequently, results in only small releases and consequences. For instance, the isolation failures reported for Beaver Valley 1 and Beaver Valley 2 have high conditional probabilities, but involve only small leak areas, and consequently, result in only small releases and consequences. Even for some of the bypass cases reported in the IPEs, the release point may be submerged under water and the release is thus scrubbed. In SGTR sequences, radionuclide release is more significant if the safety valves or the atmospheric dump valves in the steam line of the faulted steam generator are stuck open rather than cycling. Furthermore, the operation of containment sprays will attenuate radionuclides released to the containment atmosphere and greatly reduce the source term.

Figure 4.9 shows that the conditional probabilities of significant release reported for large dry and subatmospheric containments in the IPEs range over an order of magnitude. Since containment bypass usually causes high releases, the IPEs that have high probabilities of significant early release are those that have high probabilities of containment bypass. For example, the probabilities of significant early release are about 0.1 for St. Lucie Units 1&2, and about 0.3 for Ginna and Zion 1&2. The smallest early release probability is reported in the Wolf Creek IPE submittal, where the probability of bypass is low and the probability of early failure is assumed to be zero. It should be noted that since the frequencies for the release categories are not provided in the Beaver Valley 1 and Beaver Valley 2 IPEs, the conditional probabilities for Release Type I (for large early containment failures and bypass) are used in Figure 4.9.

Release Type I has a conditional probability of about 0.05 for Beaver Valley Units 1 and 2. According to these IPEs, Type I release involves scenarios that would result in potentially life-threatening doses in the same time frame as needed to implement protective actions like sheltering or evacuation. For Release Type I, over 80% is due to containment failure at vessel breach caused by containment loads from a HPME event, about 10% is due to bypass, and 5% due to alpha mode failure. SGTR is categorized in the Beaver Valley IPEs as Release Type II, i.e., small early containment failures or bypasses. Since the release fractions for volatile radionuclides for SGTR events are usually greater than 10%, if SGTR sequences are considered as resulting in significant early releases, the conditional probabilities for significant early release for Beaver Valley 1 and Beaver Valley 2 are increased by about 0.05. The frequencies of significant early release reported in the IPEs for large dry and subatmospheric containments vary from 1E-8/ry to about 2E-5/ry.

Figure 4.9 also shows the conditional probabilities of significant early release reported in the IPEs of PWR plants with ice condenser containments. According to the Catawba 1&2 and the McGuire 1&2 IPEs, only bypass sequences satisfy this criterion, and the probability of significant early release is that of containment bypass. For Cook 1&2, the probability of significant early release is about 0.05, with about equal contributions from containment bypass and early containment failure. The probability values use in Figure 4.9 for Sequoyah 1&2 and Watts Bar are not consistent with the large early failure category defined in the IPEs.

In the Sequoyah 1&2 and the Watts Bar IPEs, Release Category I is defined as encompassing large early containment failure and large bypasses. However, the release fractions for volatile radionuclides obtained by MAAP calculations for this category are smaller than those for Category II, defined in the IPE as encompassing small, early containment failures and small bypasses. For example, the release fraction for iodine is about 0.05 for Category I and 0.2 for Category II. This is because in the Sequoyah 1&2 and Watts Bar IPEs, Category II is primarily due to containment bypass. To be consistent with the definition used in this report, the probabilities in Category II are used in Figure 4.9 as the probabilities of significant early release for Sequoyah 1&2 and Watts Bar. The frequencies reported for significant early release in the IPEs for the plants with ice condenser containments vary from less than 1E-7/yr to 8E-6/yr for the five IPEs.

5. IPE RESULTS PERSPECTIVES: HUMAN PERFORMANCE

An important aspect of the Individual Plant Examination (IPE) program, as described in Generic Letter (GL) 88-20 (Ref. 5.1), is to identify human actions important to severe accident prevention and mitigation. In this context, the human reliability analysis (HRA) is expected to be a critical component of the probabilistic risk assessments (PRAs) for the IPEs. The determination and selection of human actions for incorporation into the event and fault tree models and the quantification of their failure probabilities can have an important impact on the resulting estimates of core damage frequency (CDF). Not surprisingly, results from the submittals indicated not only that human error can be a significant contributor to CDF, but that correct human action can substantially reduce the overall CDF.

The purpose of this chapter is to summarize the human actions important in the IPEs and to address the degree of variability in the results of the HRAs across the different IPEs. Of particular concern is the degree of variability in the quantification of similar human actions across different plants. The degree of variability is important because of the potential impact human error probabilities (HEPs) can have on which human actions and accident sequences are found to be important. After discussing the human actions important for boiling water reactors (BWRs) and pressurized water reactors (PWRs), this chapter addresses some of the potential causes for variability in HRA results, and examines the extent to which variability in HEPs across different plants appears reasonable.

In the process of identifying the important human actions from the submittals, the staff found that both the methods used to identify important human actions and their documentation are inconsistent across the IPEs. For example, some licensees used Fussel-Vesely or similar measures to identify important actions, while others used a sensitivity analysis in which all HEPs less than 0.1 were set to 0.1 and the sequences were requantified. Selected human actions were then systematically restored to their original values and reductions in CDF were examined to determine which actions have the greatest impact. Still other licensees identified the human actions that reduce CDF by an order of magnitude and reported these as the important human actions. In some cases, licensees reported the percent contribution to core damage, while in others cases, licensees presented risk achievement worth or risk reduction values. In some instances, licensees provided a list of important human actions, but did not discuss the basis for the list. Nevertheless, most licensees attempted to provide some indication of which actions are important.

The results and perspective presented below are based on the original IPE submittals forwarded by the licensees to the U.S. Nuclear Regulatory Commission (NRC) (refer to Appendix A for sources of IPE information used in this report). Many licensees have updated their IPEs/PRAs since the original submittal and have reported bottom-line changes in CDF or release frequency; but in most case, detailed information was not provided. Updated IPE information received by the NRC is listed in Appendix B.

5.1 Important Human Actions for BWRs

This section identifies and summarizes the human actions important in a relatively high percentage of BWR IPE submittals. The staff then provides perspectives regarding the impact of specific plant classes and unique plant designs and characteristics on the importance of human actions.

5.1.1 Human Actions Generally Important for BWRs

Table 5.1 lists the most important human actions identified in the staff's review of all 27 BWR IPE submittals (covering 35 units), along with the percentage of all BWR IPEs finding the action important, and the percentage of IPEs finding the action important as a function of BWR class. Of the 27 submittals reviewed, five are in the BWR 1/2/3 class (covering six units), 15 are in the BWR 3/4 class (covering 21 units), and seven are in the BWR 5/6 class (covering eight units).

Table 5.1 Important human actions and percentage of BWR IPEs finding the action important.

Important human actions	Percentage of BWR IPEs finding the action important			
	All BWR IPEs	BWR 1/2/3s	BWR 3/4s	BWR 5/6s
Perform manual depressurization	~80%	~80%	~80%	~60%
Containment venting	~55%	~35%	~60%	~60%
Align containment or suppression pool cooling	~55%	~70%	~50%	~50%
Initiate standby liquid control (SLC)	~50%	~70%	~50%	~40%
Level control in anticipated transient without scram (ATWS)	~25%	~50%	~30%	0%
Align/initiate alternative injection	~25%	~30%	~30%	~15%
Recover ultimate heat sink	~20%	~20%	~20%	~25%
Inhibit automatic depressurization system (ADS)	~20%	~20%	~20%	~25%
Miscalibrate pressure switches	~15%	~20%	~15%	~10%
Initiate isolation condenser	N/A	~85%	N/A	N/A
Control feedwater events (e.g., loss of instrument air)	~15%	~15%	~20%	~15%
Manually initiate core spray or other low-pressure system	~15%	~20%	~20%	0%
Miscalibrate low-pressure core spray permissive	~10%	~20%	~15%	0%
Provide alternative room cooling (in the event of a loss of heating, ventilation, and air conditioning (HVAC))	~10%	0%	~5%	~25%
Recover injection systems	~10%	0%	~15%	~15%

Only a few specific human actions are regularly found to be important across all the BWR IPEs. That is, while many different events are indicated as being important, relatively few are important to most of the IPEs. Thus, the staff attempted to group the operator actions according to the function to be accomplished. For example, events related to aligning an alternative injection source during transients, loss of coolant accidents (LOCAs), and station blackouts (SBOs) are considered important to several licensees. Even though the alternative systems used ranged from firewater to suppression pool cleanup, the function accomplished by performing the action is similar. In order to help capture the general types of events that are important to BWRs, the staff grouped these actions with similar functions and presented them in Table 5.1 along with other important individual operator actions.

Manual depressurization of the vessel[5.1] so that low-pressure injection systems can be used after a loss or unavailability of high-pressure injection systems is important in most BWR IPE submittals. This action is particularly important in some plants for long-term SBO sequences where depressurization is required to allow injection from firewater systems, after loss of steam-driven systems such as reactor core isolation cooling (RCIC). This human action is important largely because of the fact that most plant operators are directed to inhibit automatic actuation of the ADS by the plant emergency operating procedures (EOPs). Thus, operators must manually depressurize the vessel when injection from low-pressure systems is required to cool the core. The percentage of total CDF accounted for by cutsets including this event ranged from one percent to 45%.

While human actions related to an ATWS are frequently found in the licensees' lists of the top ten important events, the contribution of ATWS events to overall CDF is usually relatively small. The human action to inhibit the ADS is important in the ATWS sequences of several submittals. In fact, some licensees assume that because of the instabilities created under low-pressure conditions during an ATWS, core damage will occur if the operators fail to inhibit the ADS. Given this position, it is somewhat surprising to find that only ~20% of the BWR licensees identify inhibition of the ADS as being important. The low percentage results in part from how licensees model ADS inhibition. Many licensees assume that failure to perform this action has a very low probability, or they do not model it at all. Other licensees model the failure to inhibit the ADS as resulting in core damage only if it occurs in conjunction with a second failure (e.g., failure of SLC or failure of low-pressure injection flow control). Such a model can reduce the importance of this type of accident sequence and thus the importance of the related human errors. The remaining licensees model the failure to inhibit the ADS during an ATWS as directly resulting in core damage. This human error is noted as being important for approximately 50% of the licensees that model ADS inhibition in any fashion.

Two other ATWS-related events are found to be important by several licensees. The operator action to initiate boron injection during an ATWS is important in ~50% of the BWRs, and ~25% identify level control as being important. As with ADS inhibition, the modeling of these events partially impacts their importance to core damage. For example, some licensees model early initiation of SLC, while others consider both early and late initiation times. The initiation times (important in calculating the HEPs) are based on avoiding adverse conditions, such as high suppression pool temperatures, and are somewhat variable (ranging from one minute to 45 minutes). Some licensees take credit for alternative means of injecting boron, while others take credit for level control as a means of reducing core power to acceptable levels following SLC failure. All of these variables (and associated dependencies between actions) can contribute to the importance of the failure to manually initiate SLC. Modeling of level control is highly variable, with several different

[5.1]Section 5.3 discusses the variability in HEPs for this event across the BWR IPEs.

factors influencing the modeling. Whether these actions are important for particular licensees is, to some extent, a function of the contribution of the ATWS sequences to overall CDF. The contribution of these events to CDF is usually in the range of 1 percent to 3 percent.

Many licensees identify human actions related to decay heat removal as being important. Two of the most frequently identified important actions in BWRs relate to decay heat removal (DHR) sequences in transients and LOCAs. With a loss of the power conversion system and safety relief valves (SRVs) open, containment temperature and pressure must be controlled. The actions to provide some form of containment or suppression pool cooling, or to vent containment when adequate cooling can not be provided, are important in more than 50% of the IPE submittals. Plant characteristics and modeling differences are important factors in determining the impact of these human actions.

Plants require DHR actuation before adverse conditions are reached. These conditions can range from reaching a high suppression pool temperature that results in a loss of emergency core coolant system (ECCS) pumps, to reaching a high containment pressure that results in closure of SRVs that are required to remain open to maintain the vessel at low-pressure (for coolant injection from low-pressure systems). However, some licensees did not model the failure of DHR as leading to a failure in the ability to inject water into the vessel from the ECCS or from alternative injection systems. In addition, some licensees identified the steam released following containment failure as having a negative impact on the operability of injection systems. In addition, some licensees do not model venting at all. They either do not have reliable venting systems, do not have a strong need to vent, or simply do not take credit for venting. The contribution from these events to CDF generally ranges from 1 percent to 5 percent, with one licensee indicating about a 10% contribution.

Recovery of service water or some other ultimate heat sink is important in approximately 20% of the BWRs. The importance of cooling water systems is variable because of the wide spectrum of cooling water systems that are available in the plants. The need to recover such systems is dependent on the cooling requirements of mitigating equipment and the time before it fails upon loss of cooling. When found to be important, these events are usually relevant to multiple classes of accident sequences. In general, the data provided on these events is insufficient to assess their contribution to CDF.

Alignment/initiation of an alternative coolant injection source is important. This action is important in approximately 25% of the submittals reviewed. The need for an alternative injection system is relevant to loss of DHR and injection sequences (primarily where all normal low- and high-pressure systems are lost). Different plants have different sources for alternative injection, including service water (SW), firewater, the control rod drive (CRD), a unique standby feedwater system, a safe shutdown makeup pump, feed booster pumps, and suppression pool cleanup. Some licensees credit alternative high-pressure injection systems such as the CRD, but most only credit low-pressure systems such as SW. In addition, alternative injection systems are generally credited only in long-term scenarios when sufficient time is available to perform required alignments or when the required coolant injection flow is within the injection system capacity. However, some licensees credit alternative systems in short-term scenarios when the capacities are within makeup requirements and when the alignment is proceduralized and can be accomplished from the control room. CDF contribution for these events ranges from 1 percent to 4 percent.

Pre-initiator human actions were important in some submittals. Miscalibration of various instruments is important in approximately 20% of the BWRs reviewed. Moreover, ~15 percent of the licensees note the importance of failures to restore certain systems after testing or maintenance. The most important events related to instrumentation miscalibration (or failure to restore instruments after testing) involve the calibration of various pressure switches. Examples include miscalibration of the vessel low-pressure permissive required to open the low-pressure core spray and low-pressure coolant injection valves (noted by ~11% of the licensees) and miscalibration of pressure switches that allow automatic initiation of an isolation condenser. CDF contribution for these types of events is found to be as high as 6% in several cases. Failure to restore critical systems includes events such as failure to correctly align SLC valves after testing, failure to restore SW after test, and failure to restore high-pressure core spray (HPCS) after testing or maintenance. The fact that these events are not important for more licensees may be attributable in part to the apparent failure of some licensees to model pre-initiator events. Another factor may be the assignment of either unrealistically low or unrealistically high HEP values for these events. Finally, modeling considerations also impact the importance of some of these events. In spite of the importance of pre-initiator events in some instances, there is no evidence that such events are major drivers in determining overall CDF.

5.1.2 Relationship Between BWR Class and Important Human Actions

Table 5.1 presents the percentage of the different classes of BWRs for which a particular humanaction is found to be important as a result of the licensees' IPEs. The reason for extracting this information is to assess whether any interesting relationships exist between the classes of BWRs and the events found to be important (that is, whether certain events are important for some classes of plants but not others). A review of Table 5.1 suggests several instances in which the importance of particular events may be related to BWR class. However, most of the differences in the human error listing by BWR class are believed to relate to modeling or plant-specific differences rather than to differences in general BWR vintage designs. Potential relationships are discussed in Chapter 13.

5.1.3 Human Actions Important at Selected BWRs

Table 5.2 lists human actions identified as being important for a single plant or because they apparently relate to a unique plant design or characteristic. A discussion of several of these human actions and why they are unique to a given plant is presented following the table.

Table 5.2 Human actions important at selected BWR plants.

Plants	Human action
Big Rock Point, BWR1	Post-incident recirculation (switchover to recirculation)
Big Rock Point, BWR1	Tripping of condensate pumps on low hotwell level (prevents loss of pumps)
Big Rock Point, BWR1	Alignment of firewater system for makeup to the hotwell (refill of hotwell permits continued use of condensate and feedwater as injection source)
Nine Mile Point 1, BWR2	Recovery of screen house intake (loss of lake intake)
Nine Mile Point 1, BWR2	DC load shedding (also see Clinton)
Nine Mile Point 1, BWR2	Load shedding emergency diesel given LOCA conditions and SBO
Millstone, BWR3	Initiation of isolation condenser before SRVs open following SBO (avoids problems with stuck-open relief valves (SORVs)

Table 5.2 Human actions important at selected BWR plants.

Plants	Human action
Monticello, BWR3	Alignment of bottled nitrogen to SRVs or restoration of motor control center (MCC) 41 on loss of offsite power (LOSP)
Pilgrim, BWR3	With both DC buses lost, if operator follows the loss of DC power procedure, both feed pumps will be lost
Pilgrim, BWR3	4160-bus breaker maintenance errors would prevent breakers from changing state, resulting in loss of feedwater, low-pressure injection, DHR, etc.
Quad Cities 1&2, BWR3	Initiation of safe shutdown makeup pump - serves as a backup to RCIC for fire scenarios and can be used as an alternative injection source with suction from the condensate storage tank (CST) or firewater system (has redundant electrical supplies from each unit)
Fermi 2, BWR4	Manual initiation of unique standby feedwater system
Perry, BWR6	Alignment of feed booster pump or suppression pool cleanup for alternative injection
Perry, BWR6	Reopening of motor feed pump control valves and manually depressurizing injection during ATWS, possibly using condensate from the condenser hotwell (procedures instruct operators to terminate HPCs injection during an ATWS)
Clinton, BWR6	DC load shedding (also see Nine Mile Point 1)
River Bend, BWR6	Operator starts standby cooling tower fans

Millstone 1 is the only isolation condenser (IC) plant where failure of the manual actuation of the IC before SRVs open is identified as a human failure event. Successful IC actuation before SRV opening eliminates the potential for SORVs that defeat the use of ICs and lead to the need for another source of coolant injection. Automatic actuation of the IC is also modeled and occurs at a setpoint below the first SRV setpoint. It appears that licensees with other IC plants do not model manual actuation. While there is no apparent reason why modeling of this human action is unique to Millstone 1 and why it would not apply to other IC plants, it is certainly possible that analyses by other licensees indicated that this human action should not be modeled (e.g. Oyster Creek indicated manual initiation of an IC before SRV actuation occurs in not likely). By taking credit for this action, the licensee for Millstone 1 may effectively reduce the occurrence of transients with SORVs for which the ICs can not be used for mitigation.

Big Rock Point is the only BWR 1 currently being operated in the United States. The plant has unique features that require modeling of at least one human error that applies only to that plant. Big Rock Point is like a PWR in that it is housed in a dry containment and, during a LOCA, it requires a switchover from a coolant injection mode to a coolant recirculation mode. Decay heat removal is provided in the recirculation mode by passing the recirculated coolant through a heat exchanger. Failure to perform the switchover to recirculation is modeled as a human error that eventually results in continued injection from exterior sources. Eventually, injection from exterior sources must be terminated to prevent containment failure from high static heads of water.

The licensee for Big Rock Point also takes credit for condensate injection for some LOCAs, as do other plants. However, Big Rock Point is the only plant where the licensee identifies a human error in the failure to prevent cycling of the condensate pump breaker caused by condenser hotwell level signals. These signals are generated as the hotwell level is replenished by firewater and subsequently drained again by the condensate pump. This cycling is assumed to lead to failure of the breaker and loss of condensate and can be prevented by the operator tripping the pump on low hotwell level. This failure mode is not believed to be unique to Big Rock Point and could be modeled by other plant licensees.

As shown in Table 5.2, many of the remaining human actions relate to either load shedding or actuation of alternative injection systems. These actions apply to most plants; however, design and procedures are plant specific. For example, the actions required to initiate the alternative systems, the time available to perform the actions, and the circumstances in which the systems can be used are all plant specific. Therefore, the importance of these human errors varies among the different IPEs.

5.2 Important Human Actions for PWRs

This section identifies and summarizes the human actions important in a relatively high percentage of PWR IPE submittals. The staff then provides perspectives regarding the impact of specific plant classes and unique plant designs and characteristics on the importance of human actions. Consistent with the rest of this report, the PWRs are classified as Babcock and Wilcox (B&W) plants; Combustion Engineering (CE) plants; and Westinghouse plants with two, three, and four loops (i.e., W-2s, W-3s, and W-4s). Of the 48 submittals reviewed, five of the plants are B&Ws (covering 7 units), ten are CEs (covering 15 units), four are W-2s (covering 6 units), nine are W-3s (covering 13 units), and 20 are W-4s (covering 32 units).

5.2.1 Human Actions Generally Important for PWRs

Table 5.3 lists the most important human actions identified in the staff's review of all 48 PWR IPEs submittals, along with the percentage of all PWR submittals finding the action important, and the percentage of submittals finding the action important as a function of PWR class.

Table 5.3 Important human actions and percentage of PWR IPEs finding the action important.

Important human actions	Percentage of IPEs finding event important					
	All PWRs	B&W	CE	W-2	W-3	W-4
Switchover to recirculation (plants with manual or semi-automatic switchover)	~80%	~85%	N/A	100%	~55%	~90%
Feed-and-bleed	~60%	~45%	~60%	~70%	~45%	~70%
Depressurization and cooldown	~50%	~60%	~30%	100%	~70%	~50%

Table 5.3 Important human actions and percentage of PWR IPEs finding the action important.

Important human actions	Percentage of IPEs finding event important					
	All PWRs	B&W	CE	W-2	W-3	W-4
Use of backup cooling water systems	~40%	~45%	~30%	~35%	~60%	~30%
Makeup to tanks for water supply	~35%	~30%	~20%	~35%	~40%	~40%
Restoration of room cooling (HVAC)	~30%	~15%	~50%	~35%	~30%	~30%
Restoration of main feedwater (MFW) or condensate to steam generators (SGs)	~30%	~30%	~35%	~35%	~50%	~30%
Proper control of auxiliary feedwater (AFW) or emergency feedwater (EFW)	~25%	~30%	~40%	~35%	0%	~30%
Reactor coolant pump (RCP) Trips	~25%	~45%	~35%	~35%	~15%	~20%
Pre-initiators	~25%	0%	~50%	0%	~25%	~20%
ATWS reactivity control	~20%	0%	~20%	0%	~10%	~35%
Water supply for AFW or EFW	~15%	0%	~40%	~35%	~10%	~5%
Initiation of AFW or EFW	~15%	0%	~50%	0%	~10%	~10%

As with BWRs, only a few human actions are regularly found to be important across all PWR submittals. The human action most consistently important for PWRs is the switchover to recirculation during LOCAs. Other human actions frequently important include feed and bleed, and actions associated with depressurization and cooldown. Only these three actions are important in more than 50% of the submittals. They are discussed in more detail below, along with several other actions frequently found to be important by the licensees. Identified relationships between PWR plant classes and important human actions are discussed in Section 5.2.2, while Section 5.2.3 addresses events important only in selected IPE submittals.

Switchover to recirculation on low ECCS level is important for LOCA sequences in most submittals for plants with semi-automatic or manual switchover. All ten CE plants (15 units) have an automatic switchover, as do four of the other plants. For the 35 plants (58 units) that require operator actions (either completely manual or semi-automatic) to complete the switchover, ~80% of the submittals find this action to be important. One possible reason some licensees fail to find this action important (even in sequences where cooldown and depressurization have failed) may be the fact that the sizes of refueling water storage tanks (RWSTs) vary from plant to plant. Licensees with plants that have larger RWST capacities may model the small LOCA and long-term transient sequences as not requiring the switchover to recirculation cooling, thereby lessening the importance of the recirculation function and hence human actions related to recirculation cooling. Additionally, some licensees model RWST refill as the action preferred over recirculation cooling, particularly in small LOCA and long-term transient cooling situations. This again lessens the overall importance of recirculation cooling and the corresponding related human actions. For licensees that find the switchover to recirculation to be an important operation (and report the related contribution to total CDF), the contribution to CDF ranges from less than one percent in several cases to as much as ~16%, with an average contribution of ~6%.

Many licensees identify the initiation of the feed-and-bleed operation as being important. This event is important in transient and steam generator tube rupture (SGTR) sequences when all feedwater has failed. In addition, a few licensees find the establishment of a reactor coolant system (RCS) bleed path with one power operated relief valve (PORV) to be important in small LOCAs. In all, about 60% of the submittals indicate that feed-and-bleed is one of the more important events. Some licensees may fail to find feed-and-bleed important for a variety of reasons that are interrelated and not easily discernible. For instance, the relative reliability of each plant's AFW or EFW system is a factor since it is only in sequences where AFW or EFW has failed that feed-and-bleed becomes another important action in the in-depth defense to provide core cooling. Thus, accident sequences involving AFW/EFW failure (and thus the need to use the feed-and-bleed function) can vary considerably in frequency, thereby affecting the overall importance of the feed-and-bleed function. Specific support system dependencies can also be important to the overall feed-and-bleed reliability and hence the importance of this human action. For plants with a higher susceptibility of failing feed-and-bleed because of support system failures, this mode of cooling is less reliable, and the human action of feed-and-bleed operation can be less important.

Additionally, many licensees spent considerable effort to model the ability to depressurize the plant and use condensate as yet another way to achieve core cooling. Taking credit for such action further lessens the overall importance of feed-and-bleed function and the related human action. Other factors related to the success criteria for feed-and-bleed, as well as the HEPs themselves, can contribute to the relative importance of this mode of cooling and the related human action. The CDF contribution for this event ranges from less than one percent to ~10%, with most submittals showing relatively small contributions from this event, resulting in an average total CDF contribution of about four percent.

The depressurization and cooldown operation, in order to use available sources of core cooling (and in many cases to lessen SGTR leakage), is found to be important by more than half of the licensees. This action usually (but not always) involves depressurizing the steam generators to cool the RCS and is found to be important in all types of sequences except ATWS. It is most frequently deemed important in SGTR sequences. As a result, 52 percent of the licensees find this human action important. As discussed above regarding the feed-and-bleed function, licensees may neglect to find depressurization and cooldown important for numerous interrelated reasons (including those described for the feed-and-bleed event). Additionally, not all of the plants model this mode of cooling, in some cases because of the relatively low capacity to depressurize the Sgs in some scenarios (depending on PORV, atmospheric dump valve, or other equipment sizes). The CDF contribution for this event ranges from less than one percent to approximately seven percent, and is similar to the contributions for feed-and-bleed. Most submittals show relatively small contributions from this event, resulting in an average total CDF contribution of approximately three percent.

None of the remaining human actions are important in more than 40% of the submittals, and none of them consistently contributes significantly to CDF. As shown in Table 5.3, the remaining human actions are not important in a large percentage of the submittals. Recovering and using backup cooling systems, supplying makeup for injection sources, and recovering loss of room cooling are important for accident sequences in approximately one-third of the submittals. Several actions related to restoration and appropriate use of MFW and AFW systems are found to be important in several submittals, and manual RCP trips upon loss of seal cooling are important in about 25% of the submittals. Similar to the BWRs, pre-initiator events, including both miscalibration and restoration errors, are found important in some submittals. The miscalibration errors tend to involve the traditional instruments such as level, pressure, and temperature sensors and transmitters, but the restoration errors tend to vary across submittals. Examples of important restoration errors include those associated with AFW and EFW systems, diesel generators, and several unique events such as leaving a nitrogen station manual valve closed and removing a jumper in the reactor protection system after refueling.

5.2.2 Relationship Between PWR Class and Important Human Actions

Table 5.3 presents the percentage of the different classes of PWRs for which the submittal reports a particular human action to be important. The information is presented in this way to assess whether any human actions appear particularly important to one class of PWRs as opposed to others. Any observations of this type may suggest the possibility of plant-class differences that cause certain human actions to be important and not others. While several interesting observations can be made regarding trends that appear to be related to plant-class definitions, the differences are believed to relate to modeling preferences or plant-specific differences rather than to differences in general PWR vendor or vintage designs. Relevant observations on these differences are covered in Chapter 13.

5.2.3 Human Actions Important at Selected PWRs

Table 5.4 lists human actions identified as being unique for a single plant (or both units at a single site) or because they apparently relate to a unique plant design or characteristic. While the importance of these actions is presented in the licensee submittals using a variety of methods, these actions typically contribute one percent to ten percent or more to the total CDF presented in each submittal. A discussion of several of these human errors is presented following the table.

Table 5.4 Human actions important at selected PWR plants.

Plants	**Human action**
Arkansas Nuclear 1, B&W-2	Locally open service water jacket cooling valves to diesel generator upon failure of these normally closed valves to open
Beaver Valley 1&2, W-3	Prematurely secure high-pressure safety injection
Beaver Valley 2, W-3	Prevent or otherwise recover from automatic switchover of high-pressure suction from the volume control tank to the RWST if the RWST pathway fails

Table 5.4 Human actions important at selected PWR plants.

Plants	Human action
Beaver Valley 2, W-3	Manually align/start safety injection upon failure of both trains of the solid state protection system used for automatic safety injection
Calvert Cliffs 1&2, CE-2	Block spuriously opened PORV with blocking motor operated valve
Crystal River 3, B&W-2	Locally isolate high-pressure makeup pump individual recirculation line when common recirculation valves have failed
Crystal River 3, B&W-2	Manually close open valve between the borated water storage tank (BWST) and the containment sump to prevent drainage of BWST used for high-pressure injection suction, or use recirculation cooling early
Haddam Neck, W-4	Isolate core deluge line break to ensure sufficient ECCS recirculation
Haddam Neck, W-4	Locally control charging flow given loss of air
Kewaunee, W-2	Manually isolate makeup valve in path between CST and condenser to ensure initial suction for AFWs
Maine Yankee, CE-3	Close the reactor coolant system stop valves to stop primary-to-secondary leakage before steam generator overfill (also useful for isolating reactor coolant pump thermal barrier rupture)
North Anna 1&2, W-3	Identify and correct safety injection flow diversion (backflow) through inoperative charging pump discharge check valve
Seabrook, W-4	Recover from a loss of emergency safety features actuation system (ESFAS) by manually starting equipment from the control room
Sequoyah 1&2, W-4	Isolate the component cooling train from the spent fuel heat exchanger to ensure adequate cooling during the recirculation phase of the accident
Summer, W-3	Align and start standby charging pump and chiller upon failure of one chilled water train
Summer, W-3	Restore one chilled water train within 12 hours to support charging
Three Mile Island 1, B&W-2	Throttle high-pressure injection flow to avoid PORV demands and possible stuck-open PORV
Three Mile Islant 1, B&W-2	Open DHR dropline valves to prevent boron precipitation

Nearly all of these important human actions involve the failure to respond to a degraded condition of certain systems or components so as to overcome the failure and successfully achieve the desired result. Which actions are important at which plants appears in many cases to depend on the specific design details at each plant. The specific faults and corresponding responses or recoveries are somewhat dependent on plant-specific design configurations for many of the unique human actions listed above. Nevertheless, it appears that many of these important human actions may have applicability at other plants (to the extent configurations and potential faults are similar), although the relative importance of each action at other plants will vary. For instance, the action listed above for Arkansas Nuclear One, Unit 1 clearly depends on the normal state of the diesel generator jacket cooling valves during normal plant operation. Other plants may operate with a similar configuration. Vital bus loads and design specifics for automatic switchover of high-pressure suction can make the Beaver Valley 2 automatic switchover action similarly applicable at other plants. The Calvert Cliffs 1 & 2 action concerning the use of the PORV block valves to mitigate the effects of a stuck-open PORV seem applicable to many plants, yet only the Calvert Cliffs submittal identifies this specific event among the "important" human actions.

Some actions, are similar in nature among a few of the plants listed above, and may also apply at other plants as well. For example, the Crystal River 3 action associated with closing off a potential drainage path from the BWST to the containment sump, and the Kewaunee action associated with closing off a drainage path from the CST to the condenser are similar in nature. Other plants may have similar configurations, potential failures, and corresponding human actions. The Beaver Valley 2 action concerning manual alignment and startup of safety injection upon loss of the automatic start signals, and the similar Seabrook action associated with manual startup of equipment upon loss of the ESFAS, may have similar applicability at other plants depending on control system design specifics and the relative reliability of such systems at each plant.

One action of particular interest is the Beaver Valley 2 action concerning prematurely securing high-pressure injection, particularly during small LOCAs when the status indication regarding the continued need for injection may be confusing or otherwise uncertain. Whether the source of such "uncertainty" is specific to Beaver Valley 2, or derives from a conservative modeling approach, or whether such an action applies to other plants can not be determined on the basis of the submittal contents.

The Maine Yankee action concerns closure of reactor coolant loop stop valves to mitigate an SGTR or a rupture of an RCP thermal barrier. Although some other plants also have reactor coolant loop stop valves, only the Maine Yankee IPE included this action in the PRA. Note that this action was considered for inclusion in the generic Westinghouse Owners Group procedures, but was rejected.

In all of the cases discussed above, specific design configuration details and the time and status information available for performing these actions is plant-specific. This results in variability concerning the importance of these actions and likely accounts for the unique importances identified at the plants listed above. However, these important human actions may apply to other plants as well, although the relative importances will probably vary.

5.3 Variability in Human Error Probabilities

Numerous factors can influence the quantification of HEPs and introduce significant variability in the resulting HEPs, even for essentially identical actions. General categories of such factors include plant characteristics, modeling details, sequence-specific attributes (e.g., patterns of successes and failures in a given sequence), dependencies, HRA method and associated performance shaping factors (PSFs) modeled, application of HRA method (correctness and thoroughness), and the biases of both the analysts performing the HRA and the plant personnel from whom selected information and judgments are obtained. Although most of these factors introduce appropriate variability in results (i.e., the derived HEPs reflect "real" differences such as time availability and scenario-specific factors), several have the potential to cause invalid variability. A discussion of both appropriate and inappropriate influences is presented below, followed by a discussion of the variability in the HEPs for a specific event.

5.3.1 HRA Influences

An important factor that potentially leads to differences in results is the HRA method used to derive the HEPs. In quantifying HEPs, several licensees use a single HRA methodology, while others use a combination of HRA methods to address different aspects of the analyses. In general, it appears that the different HRA methods can be grouped into six basic categories with the following distinctive attributes:

(1) ***A <u>modified</u> version of the Success Likelihood Index Methodology (SLIM) (Ref. 5.2) that relies on subjective estimates of the impact of various performance shaping factors on the operator's likelihood of failure.*** This method (often referred to as the failure likelihood index methodology), explicitly addresses approximately 23 different PSFs ranging from fairly common factors such as operator training and experience, available procedural direction, and relevant plant indications to relatively unique factors such as the impact of preceding and concurrent unrelated actions on potential operator confusion. This is the only method that consistently relies directly on subjective estimates by experts to derive the HEPs for the post-initiator human actions. In addition, this method is distinguished by the fact that the impact of time on the performance of a task is usually determined on the basis of subjective estimates rather than the time reliability correlations (TRCs) used by most other HRA methods. A characteristic of SLIM-based methodologies is that the process provides a basis for interpolating between upper and lower bounds of the failure probability. These upper and lower bounds must be provided by the analyst separate from the SLIM analysis. Inappropriate selection of these bounds can bias the calculated probabilities of failure.

(2) ***The Electric Power Research Institute (EPRI) cause-based decision tree method described in EPRI-TR-100259[5.2].*** This decision tree method considers a number of potential failure mechanisms and associated PSFs in determining the failure probability for the detection, diagnosis, and decisionmaking phases of an operator action. Values from the Technique for Human Error Rate Prediction (THERP) are used to quantify the execution of the human actions. This method generally accounts for time by limiting the consideration of recovery factors (such as verification by additional crew members or availability of the emergency response facility) for only those actions where the available time exceeds some criterion. In some applications of this method in the submittals, time-critical actions were quantified using the TRCs from the Accident Sequence Evaluation Program (ASEP) (Ref. 5.3) or THERP (Ref. 5.4). Only non-time critical actions are quantified using the decision tree method.

(3) ***The Human Cognitive Reliability (HCR) method (Ref. 5.5) or the Operator Reliability Experiments (ORE) based modification of the HCR method (EPRI NP-6560-L)[5.3].*** These methods are essentially TRC methods that may also use THERP to quantify the execution of the action. The HCR method itself considers whether an action is "skill-based, rule-based, or knowledge-based" in determining which TRC to use to assess failure probability. The ORE method apparently no longer considers this distinction, but does include a detailed approach that incorporates data collected from

[5.2] G.W. Parry, "An Approach to the Analysis of Operator Actions in PRA," , EPRI TR-100259, Electric Power Research Institute, Palo Alto, CA., June 1992.

[5.3] A.J. Spurgin, P. Moieni, and G.W. Parry, "A Human Reliability Analysis Approach Using Measurements for Individual Plant Examinations," EPRI NP-6560-L, Electric Power Research Institute, Palo Alto, CA, 1989.

simulator exercises for deriving HEPs as a function of time. In most applications of the ORE method, consideration of *plant-specific* performance shaping factors is limited.

(4) ***The operator reliability characterization and assessment method described by Dougherty and Fragola (Ref. 5.6).*** In one submittal based on this method, the licensee stated that the method functionally combines Systematic Human Action Reliability Procedure (Ref. 5.7) and HCR, and therefore may in some ways be similar to method (3) above. However, as documented by some licensees, the method separates post-initiator human actions into slips and mistakes. It is generally assumed that slips are actions in which the execution of the task is the dominant failure mode, while mistakes are where the cognitive part of the task is less reliable. Thus, for most actions, either the execution or the cognitive portion is quantified (not both). Slips are quantified using a simplified THERP method, and mistakes are quantified with a set of TRCs that vary as a function of whether the actions are taken inside or outside the control room and whether they are based on procedures (i.e., rule-based) or represent recovery actions (which may be trained but are not documented in procedures). In addition, the influence of burden on operators in terms multiple tasks and conflicting demands is considered in quantifying mistakes.

(5) ***The THERP method or the ASEP HRA method derived from THERP.*** Among other PSFs, these methods explicitly consider whether a task to be performed is "dynamic" or "step-by-step" and they have explicit HEP adjustment factors for several different levels of stress. In the application of ASEP (and usually THERP) licensees use a TRC to quantify the diagnosis portion of an operator action. Upper and lower bounds of the TRC are used to account for various PSFs, such as frequency of training. In some applications of these methods, the number of PSFs actually considered is limited.

(6) ***The Individual Plant Examination Partnership (IPEP) methodology which (at least nominally) is a modified version of THERP.*** This method, apparently only documented in the IPEP IPE submittals, is distinguished by its lack of emphasis on modeling the diagnosis portion of a task, while creating a PSF (referred to as "slack time") that allows substantial credit for potential recovery of initially failed operator actions. In at least some applications of this method, other limitations include arbitrary reductions in failure probabilities, consideration of a limited number of failure modes, and a lack of consideration of dependencies.

One intent of the brief descriptions provided above is to point out that while different HRA methods naturally have many commonalities, they may also consider different factors (including PSFs) in deriving HEPs. This situation creates the possibility that essentially the same operator actions can be assigned somewhat different HEPs as a function of the different factors considered by different methods. While some of the accepted methods have been "benchmarked" to some degree in order to assess the values they produce relative to other methods, it is clear that the potential exists for method-based differences.

The fact that different methods have the potential to produce different HEPs for similar actions does not necessarily imply that they will. Essentially all HRA methods attempt to consider relevant variables and to systematically factor in their impact in determining HEPs. To the extent that the methods consider a reasonable set of variables, and to the extent that those variables are systematically and appropriately addressed, realistic and consistent HEPs can be expected across different methods. It might seem reasonable to compare HEPs for similar actions across the different methods used in the IPEs (in order to determine the impact of each method). Nonetheless, the results of such a comparison could be misleading. For example, as noted above, analyst bias can influence HEPs, as well as plant characteristics, sequence-related factors, and modeling details. In addition, given the sparse documentation of many of the submittals with regard to the necessary information, in most cases it is difficult to achieve a valid comparison regarding the specific influence of a given HRA method. Suffice it to say that at least some "unexplained variability" in resulting HEPs can be created as a function of the differences in HRA methods and in particular the ways in which they are applied.

In addition to the basic HRA methodology and the associated PSFs used to quantify the post-initiator HEPs, a number of other factors related to how the analysis is conducted can have an impact on the results. Many of these factors may have a direct impact on the derivation of HEPs, but reflect the nature and extent of the analysis performed for the HRA or on how the HRA is incorporated into the PRA. Thus, their influence can be quantitative, qualitative, or both. Several of these factors and their treatment in the IPEs are discussed below.

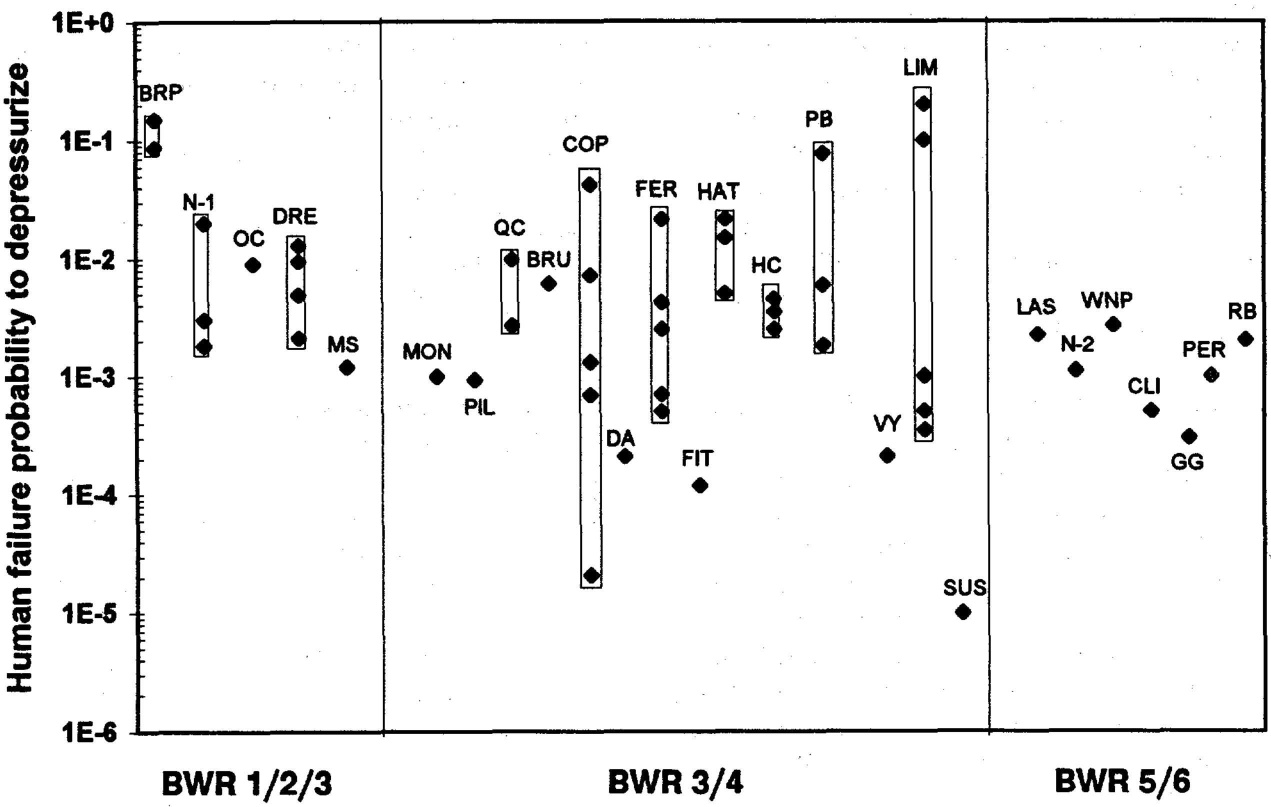
One potentially important factor concerns the extent to which accident progression and context effects are taken into account in determining the HEPs. For example, an operator action indicated by the EOPs can be called for in the context of a variety of different initiators and after different patterns of previous operator and system failures or successes. Therefore, in order to realistically quantify the human potential for failure or success, context and effects and dependencies across a given accident sequence should be considered. While most licensees clearly consider context and dependencies in analyzing post-initiator actions, some do not. Two licensees analyzed operator actions only to the extent needed to determine the conditions that yield the highest failure probability for a given human action event. The HEP for the action in that context only is then quantified, and the resulting "bounding" value is assigned in all cases where the event occurs. Other licensees addressed context only in cases where extreme differences in HEPs was expected, and several either failed to consider context or dependency at all, or at least failed to provide any evidence that they did so in their documentation. Obviously, these types of decisions can lead to variability in quantifying HEPs and in the knowledge gained by the licensees about the importance of operator actions.

Other factors having a potential impact on the HRA results include whether the analysts conduct simulator exercises to assess the performance of the control room crews in responding to important accident sequences and whether the analysts conduct walkthroughs of important operator actions that must be performed outside the control room during emergency situations. Conducting simulator exercises and directly evaluating the demands placed on operators who are carrying out actions inside and outside the control room provide the HRA analysts with important information regarding PSFs that is likely to bear on the probability of successfully completing a given task. To the degree that some licensees do or do not evaluate simulator exercises and conduct walkthroughs of specific actions, differences might be expected in HRA results.

5.3.2 Example of Variability in HEPs

In order to examine the variability in HRA results from the IPEs and to assess the extent to which variability in results is caused by valid factors versus unintended effects (e.g., analyst bias, inappropriate assumptions etc.), the staff examined HEPs from several of the more important human actions appearing in the submittals across plants. However, since the staff reached the same general conclusion after examining several important human actions for the BWRs and PWRs, this summary report presents the results from the examination of a single important human action. Discussions of the variability in HEPs for several other human actions from BWRs and PWRs are presented in Chapter 13.

Figure 5.1 presents the HEPs used in various BWR submittals for failure to depressurize the vessel during transients. Note that some submittals provided additional values for LOCAs, but they are not included here. As shown in the figure, a relatively large variability exists across the submittals for this event. However, there appears to be reasonable explanations for much of the variability in the HEPs. For values on the high end of the continuum, the events modeled appear to be special cases of depressurization. For example, the high value for Nine Mile Point 1 (N-1) involves depressurization using main steam isolation valves and the condenser, which is apparently not typically modeled. The high value for Peach Bottom 2&3 (PB) and the next to the highest value for Limerick 1&2 (LIM) pertain to the case in which a controlled depressurization is needed to allow use of the condensate system. The highest value for Limerick 1&2 (LIM) pertains to a recovery of a failed automatic depressurization. While the justification for the high values for Big Rock Point (BRP) is not apparent, it is unique relative to the other BWRs in that the plant has some characteristics similar to PWRs. The reason for the high value for Cooper (COP) is also not obvious, but the large range of values for that plant apparently relates to the number of SRVs to be used for depressurization.



(HEPs shown on figure are on a submittal basis, and not necessarily a plant-unit basis.)

Figure 5.1 HEPs for depressurization failure during transients by BWR class.

The explanations for the large difference (approximately one and one-half to two orders of magnitude) between the HEP values in the middle range appear to be related, at least in part, to dependencies and initiator- and sequence-specific factors. Several licensees, such as Nine Mile Point 1 (N-1), Dresden 2&3 (DRE), Fermi 2 (FER), and Limerick 1&2 (LIM), conducted relatively detailed analyses and apparently derived multiple values in order to account for specific conditions. These specific conditions include LOOPs, SBOs, loss of DC power, use of turbine bypass valves for depressurization, and loss of feedwater and standby feedwater. Nevertheless, while much of the variability in the middle range of values is clearly explainable, some differences are less clear. For example, the generally lower values for Fermi 2 (FER) and Limerick 1&2 (LIM) relative to those from Nine Mile Point 1 (N-1) and Dresden 2&3 (DRE) are not explainable in a straightforward manner, but may very well result from valid, plant-specific characteristics.

Finally, the reasons for the relatively low HEP values for depressurization during transients at Cooper (COP), Duane Arnold (DA), FitzPatrick (FIT), Vermont Yankee (VY), and Susquehanna 1&2 (SUS) are not clear. It could be argued that at least the top three or four values from these submittals fall within an acceptable range i.e., they are not necessarily outliers. It may also very well be the case that plant-specific characteristics support the HEPs on the lower end of the continuum. For example , the relatively low value for Cooper (COP) is for a long-term DHR sequence in which operators have up to 4 hours to depressurize. The lowest value, from Susquehanna (SUS), is clearly an outlier, but this value is consistent with many of that plant's HEP values and is a direct function of the HRA methodology used in the Susquehanna IPE.

At least some of the variability in HEP values can arise inappropriately as an artifact of the way in which HRA methods are applied (e.g., analyst bias, inappropriate assumptions, or superficial analysis, etc). Nonetheless, the main point to be derived from examining the HEPs for specific actions across plants, is that, in most cases, it also appears that there are explanations for much of the variability in HEPs and in the results of the HRAs across the different IPEs. However, such an assertion does not necessarily imply that the HEP values are generally valid. Reasonable consistency can be obtained in HRA without necessarily producing valid HEPs. An HEP is only valid to the extent that a correct and thorough application of HRA principles has occurred. For example, if a licensee simply assumes (without adequate analysis) that their plant is "average" in terms of many of the relevant PSFs for a given event, but appropriately considers the time available for the event in a given context, the value obtained may be similar to those obtained for other plants with similar time frames for the event. Yet, the resulting value may be optimistic or pessimistic relative to the value that would have been obtained if the licensee had conducted a detailed examination of the relevant plant-specific factors. Thus, to reiterate, consistency does not necessarily imply validity. In addition, because many of the licensees failed to perform thorough HRAs, it is possible that the licensees obtained HEP values that are not appropriate for their plants.

5.4 Similarities and Differences in Operational Observations Across BWRs and PWRs

Given the basic differences between BWRs and PWRs, the preceding discussion has for the most part provided separate observations regarding the submittals for the two different plant types. Nevertheless, the obvious commonalities across the plant types, prompt an examination of potential similarities or differences in the operational and HRA-related observations:

- Neither BWR nor PWR submittals show a broad consistency in terms of which human actions are found to be important. Given the numerous factors that can influence the IPE results, and the fact that functional redundancy creates the opportunity for quite a few operator actions to be taken to mitigate an accident scenario in both BWRs and PWRs, there is no reason to expect more consistency in what is found to be important for one type of plant as opposed to the other.

- Of the events frequently found to be important in BWRs and PWRs, the only somewhat similar actions are those related to depressurization and cool down, and even these actions have different cues and are performed for different purposes.

- Events related to aligning or recovering backup cooling water systems (e.g., service water) are found to be important in approximately one-third of both BWRs and PWRs.

- In both BWRs and PWRs, no individual human action appears to account for a large percentage of the total CDF across multiple submittals. Taken together, however, human actions are clearly important contributors to operational safety.

- With the exception of the licensees using the IPEP methodology, there is no indication that particular HRA methods are applied more frequently to one type of plant than another. (The IPEP methods are primarily applied to PWRs.) Thus, except for the IPEP plants, there is no reason to expect that any general differences in the results of the PRAs for the two different plant types is related to HRA method (or to any of the more general influencing factors).

In summary, it seems that most of the differences in the HRA results from the BWR and PWR submittals relate (not surprisingly) to the differences in the systems used in the two types of plants. In terms of more methodological aspects, general patterns of results, and the overall importance of humans in operating the plants, BWRs and PWRs are reasonably similar.

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6. IPE MODELS AND METHODS PERSPECTIVES

In Generic Letter (GL) 88-20 (Ref. 6.1) the U.S. Nuclear Regulatory Commission (NRC) requested that licensees perform an Individual Plant Examination (IPE) to identify any plant-specific vulnerabilities. In that request, the NRC indicated that a probabilistic risk assessment (PRA) is an acceptable approach to use in performing the IPE. The NRC further stated, *"In addition to being an acceptable method for conducting an IPE, there are a number of potential benefits in performing PRAs on those plants without one."* Representative benefits enumerated in the GL included (1) support for licensing actions, (2) license renewal, (3) risk management, and (4) integrated safety assessment. Further, in the PRA Policy Statement (Ref. 6.2), the NRC stated that *"the Commission believes that the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the quality in PRA methods and data..."*

As a result of the IPE program, licensees elected to perform PRAs for their IPEs. A PRA of a nuclear power plant is an analytical process used to quantify the potential risk to the health and safety of the public as a result of the design, operation, and maintenance of the plant. A full-scope PRA is used to quantify the risk from all events (both internal and external) that can occur at the plant and under either full-power or low-power/shutdown conditions. The risk evaluation involves three sequential parts or "levels":

Level 1 — identification and quantification of the sequences of events leading to core damage

Level 2 — evaluation and quantification of the mechanisms, amounts, and probabilities of subsequent radioactive material releases from the containment

Level 3 — evaluation and quantification of the resulting consequences to both the public and the environment.

A full-scope PRA, as currently defined, does not include evaluation of sabotage events, the risk from events that lead to releases from other radioactive material sources (such as the spent fuel pool), or the risk to plant personnel from any accident. The analytical work involved in a PRA is only part of the effort required. Documentation and peer review are also essential elements.

Perspectives on the PRAs used in the IPEs provides information on where the models are sophisticated versus where potential development may be needed. Developing these perspectives requires consideration of several issues:

- What are the characteristics of a PRA?
- How do the IPEs/PRAs compare to these characteristics?
- What can be said about the IPE analyses, given the scope of the IPEs and the scope of the staff's IPE reviews?

To allow a meaningful comparison of the analyses performed in response to GL 88-20 against the characteristics of a PRA, one must first describe current PRA practices applied in the nuclear industry. Key characteristics are summarized in this chapter. A more comprehensive discussion of the characteristics of a PRA is provided in Part 4, Chapter 14 of this report.

Although this chapter describes key characteristics of of a PRA, this report ***does not*** address the implementation of these characteristics for specific regulatory applications. Such implementation issues are addressed in the regulatory guides (Ref. 6.3) being developed for that purpose. However, the insights reported in NUREG-1560 were incorporated into the regulatory guides. In addition, it should also be emphasized that it was not necessary for the IPEs to include all of the characteristics described in this report in order to meet the intent of GL 88-20.

Second, the IPE models and methods as described in the individual submittals (refer to Appendix A for sources of IPE information used in this report) were compared against the characteristics of a PRA. Many licensees have updated their IPEs since the original submittal and have reported bottom-line changes in CDF or release frequency; but in most cases, detailed information was not provided. Updated IPE information received by NRC is listed in Appendix B. Some of the comments received on draft NUREG-1560, and discussed in Appendix C, also refer to updated IPE results. The characteristics identified in this report, therefore, reflect the scope of GL 88-20. That scope encompasses the core damage and containment performance impact of internal events (excluding internal fire) at full power. The GL does not address external events (such as earthquakes) and other modes of operation (such as shutdown), or the associated consequences. Therefore, this report provides perspectives on a PRA addressing core damage frequency (CDF) and containment performance analysis (Level 1 and Level 2 PRAs) and considering only internal events (including internal floods) at full power. However, this report also provides perspectives for a Level 3 PRA because a few licensees examined offsite consequences (Level 3 PRA).

Third, some general conclusions can be drawn that would apply to the majority of the IPEs, based on the limited reviews that have taken place. However, as in many other forms of engineering analyses, a PRA evolves over time. Higher levels of modeling sophistication and technical rigor become possible as PRA practitioners develop new, innovative analysis techniques and generate new or refined data for model quantification. Consequently, the characteristics for performing a PRA today represent a level of analytical sophistication that (in some areas) surpasses the state-of-the-art of only a few years ago. For example, some procedures reflected in NUREG/CR-2300 (Ref. 6.4) represented the state-of-the art in PRAs in the early 1980s, but are now considered inadequate. Publications that reflect enhanced practices in PRA technology beyond NUREG/CR-2300 include NUREG/CR-2815 (Ref. 6.5), NUREG/CR-4550 (Ref. 6.6), and NUREG/CR-4551 (Ref. 6.7). An attempt is made to indicate where recent advances in the state-of-the-art have occurred and whether or not these advances are reflected in the IPE submittals.

6.1 Level 1 PRA

A Level 1 PRA comprises three major elements as follows:

(1) delineation of event sequences that, if not prevented, could result in core damage and the potential release of radionuclides

(2) development of models that represent core damage sequences

(3) quantification of the models in the estimation of the core damage frequency

Each of the above elements involve several analytical tasks. The characteristics of a Level 1 PRA are summarized for each task in the following section. Perspectives drawn from a comparison of the IPE analyses (models and methods) with these characteristics are provided in Section 6.1.2.

6.1.1 Characteristics of a Level 1 PRA

The first element of a Level 1 PRA delineates event sequences that, if not prevented, could result in core damage and the potential release of radionuclides. This process is typically divided into two tasks, including identification of the initiating events and development of the potential core damage accident sequences associated with those events.

The identification of initiating events pertains to events that challenge normal plant operation and require successful mitigation in order to prevent core damage. Since there can be tens to hundreds of events that can challenge the plant, part of this task involves grouping the individual events into initiating event classes that have similar characteristics and require the same overall plant response.

The development of accident sequences involves delineating the different possible sequences of events that can evolve as a result of each initiator group. The resulting sequences depict the different possible combinations of functional and/or system successes and failures (and operator actions) that lead either to successful mitigation of the initiating event or to the onset of core damage. Determination of what constitutes success (i.e., success criteria) to avert the onset of core damage is a crucial part of the accident sequence analysis task.

The second element of a Level 1 PRA involves developing models for the mitigating systems or actions delineated in the core damage accident sequences. This process typically comprises a single task referred to as systems analysis. This task involves modeling the failure modes of the plant systems that are needed to prevent core damage, as defined by the core damage accident sequences. This modeling process, which usually involves the use of fault trees, defines the combinations of equipment failures, equipment outages (such as for testing or maintenance), and human errors that cause failure of the systems to perform the desired functions, and considering the dependencies and interfaces within and among the systems.

The third element of a Level 1 PRA involves estimating of the plant's core damage frequency and the associated statistical uncertainty. This process is typically divided into three tasks, including data analysis, human reliability analysis, and quantification and uncertainty analysis.

The data analysis involves determining the initiating event frequencies, equipment failure probabilities, equipment maintenance unavailabilities, and common cause failure rates. Plant maintenance and other operating records are evaluated to derive plant-specific equipment failure rates and the frequencies of the initiating events. Where insufficient plant experience exists, failure rates and initiating event frequencies based on industry-wide "generic" databases are used to complete the database used in the risk analysis.

The human reliability analysis involves evaluating the human actions that are important in preventing core damage. This evaluation involves identifying operator actions and quantifying the associated error probabilities. Human reliability analysis is a special area requiring unique skills to determine the types and likelihoods of human errors germane to the sequences of events that could result in core damage.

The quantification and uncertainty analysis involves integrating the initiating event frequencies, event probabilities, and human error probabilities into the accident sequence models to calculate the average annual core damage frequency and its associated uncertainty. The uncertainty analysis reflects the lack of precision in the data or a lack of detailed understanding of the modeled phenomena.

A PRA covering internal flooding analysis uses much of the same processes and meets the same attribtues provided under the various tasks described above for a full-power internal event PRA. However, an internal flooding analysis requires additional work to define and screen the most important flood sources and possible scenarios for further evaluation.

Table 6.1 summarizes the key characteristics that would be in a Level 1 PRA. A detailed discussion and description of a Level 1 PRA is provided in Part 4, Chapter 14, of this report.

Table 6.1 Summary of characteristics of a Level 1 PRA (only considering full power and internal events, excluding internal fire).

PRA element and task	Key task activities	Key characteristics
Delineation of accident sequences		
Accident Sequence Initiating Events	• Identification of initiating events • Exclusion of initiating events • Grouping of initiating events	• Thorough identification and grouping of loss-of-coolant accidents (LOCAs), transients (including transient-induced LOCAs and support systems), and internal flood sources • Systematic review with justified elimination of possible initiators • Applicable actual experience is used, augmented by analytical techniques to identify initiators • Grouping of initiating events is sufficiently discriminating
Accident Sequence Analysis	• Accident sequence model selection • Success criteria development • Accident progression modeling	• Defined in terms of hardware and timing requirements • Based on best-estimate engineering analyses applicable to the actual plant design • Includes all necessary and sufficient equipment (safety and non-safety) reasonably expected to be used to mitigate initiators • Includes functional, phenomenological, and operational dependencies and interfaces
System modeling		
Systems Analysis	• Selection of systems analysis models • Establishment of system analysis boundaries • Modeling of system dependencies and interfaces • Screening/exclusion of components and failure modes	• Generally robust level of detail regarding equipment that is modeled, as well as the types of failures and failure modes that are included • High level of assuredness that the as-built/as-operated equipment configurations are modeled via walkdown of the plant • Includes design, operational features, and testing and maintenance aspects under both normal and accident environment conditions • Systematic determination of system dependencies and potential interactions • Includes hardware, operator, and common cause failures and relevant failure modes that can be supported by relevant data

Table 6.1 Summary of characteristics of a Level 1 PRA (only considering full power and internal events, excluding internal fire).

PRA element and task	Key task activities	Key characteristics
Quantification		
Data Analysis	• Identification of data sources and models • Data input selection • Data parameter quantification • Identification and quantification of common cause failures	• Use of plant-specific experience collected from plant records • Bayesian updating of generic data with plant-specific data • Includes all incidents that affect performance of equipment • Incidents involving component reliability include all modes of operation • Incidents involving component testing and maintenance availability and event initiators only involve full-power operation
Human Reliability Analysis (HRA)	• Model/method selection • Human event identification and selection • Screening/exclusion of human events • Evaluation and quantification of human events • Integration into sequence quantification	• Human actions selected are consistent with level of detail in accident sequence delineation and systems analysis • Includes both pre-initiator and post-initiator errors • Significant errors of omission and response/recovery failures are included, while errors of commission are covered on an exception basis • Screening considers dependencies and does not pre-eliminate significant accident sequences • Uses plant-specific performance shaping factors • Treats dependencies among human error events and addresses dependencies from accident sequence progression • Uses plant-specific procedures, talk-throughs, simulator exercises, walk-throughs of ex-control room actions, and careful consideration of timing
Sequence Quantification	• Quantification model/code selection • Selection of truncation values • Integration of HRA into quantification process • Uncertainty analysis • Sensitivity analysis	• Treatment of success states and probabilities, where appropriate, as well as elimination of mutually exclusive events • Includes uncertainty, sensitivity, and importance measure results • Sufficiently low truncation limit such that 95 percent of the CDF is captured • Quantification yields a total mean plant CDF with identification of dominant accident sequences and associated mean CDFs

Table 6.1 Summary of characteristics of a Level 1 PRA (only considering full power and internal events, excluding internal fire).

PRA element and task	Key task activities	Key characteristics
Internal Flood Analysis	• Flood source and propagation pathway identification and screening • Flood scenario identification and screening • Flood-related core damage estimation	• Identifies both water and steam sources • Considers leakage, rupture, and human-induced failures • Propagation pathways examined include all sources through which water may pass • Isolation and screening based on engineering analysis • Identifies all core damage prevention and mitigation equipment identified • Quantification accounts for both flood-induced and random failures

6.1.2 IPE Level 1 Perspectives

The core damage frequency analysis of the IPEs generally compare well to the PRA characteristics listed in Table 6.1. The IPEs are generally robust with respect to the identification of dominant accident sequences. This is not to say that particular accident sequence frequencies have been verified or even that the ranking of the sequences is precise, but rather that most of the important accident types (those comprising the bulk of the CDF) have been captured. Each of the PRA elements are discussed in more detail below.

Beginning with initiating events, the modeling of all general plant transient events appears to be complete for almost all of the plants, although there is variability in whether controlled shutdown events were included. This completeness should not be surprising since these transients have occurred at most plants and are well documented. All of the licensees modeled LOCAs of various sizes (typically small, medium, and large), determined primarily by the requirements of mitigating systems. Some licensees also differentiated the LOCAs by location (e.g., steam line versus water line, and inside containment versus outside containment). The modeling of support system transient initiators, however, is more variable than other initiating events. Most licensees presented justifications for modeling or eliminating various support system initiators. However, a small fraction of the licensees simply eliminated some support system initiators without any documented justification. In addition, the basis for elimination, in many cases, was qualitative (i.e., calculations were not provided to support the inclusion or exclusion). A common example of a support system initiator that is not rigorously analyzed is the loss of the control room heating, ventilating and air conditioning (HVAC) system.

The success criteria used for mitigating systems responding to the various initiating events generally is more realistic than that credited in the analyses documented in the licensees' final safety analysis reports. The basis for emergency core cooling system and alternative coolant injection system success criteria is generally well established through references to existing thermal-hydraulic analyses or other PRAs (including the NUREG-1150 (Ref. 6.8) studies). Some licensees performed plant-specific calculations to determine whether systems, such as the control rod drive system in boiling water reactors (BWRs), could be used early to mitigate transients. However, a number of licensees did not provide a documented basis for the use of some systems. For example, the use of a single power-operated relief valve (PORV) to depressurize the reactor coolant system (RCS) at some pressurized water reactors (PWRs) may be in doubt because of the size of the PORV. Similarly, the use of firewater for coolant injection at BWRs may be limited because of the low head of the pumps. For most IPEs, the success criteria for each individual mitigating system appear to be reasonable based upon past PRAs and what is credited in other IPEs. However, a few licensees used success criteria for which the bases did not appear to be adequately justified. In addition, much of the observed variability in the success criteria is more attributable to the boundary conditions imposed on the analysis by the licensee. Where one licensee may have defined a more expanded boundary condition in crediting (with appropriate justification) plant systems for mitigation, another licensee may have elected to not perform the necessary analysis, and therefore, not credit the system(s) in the success criteria.

Because the IPE submittals do not usually provide the system models, the staff could not thoroughly assess the adequacy of these models. However, the submittals generally addressed the failure modes of components, and their dependencies on support systems. The failure modes included in the system models are generally complete and included hardware failures, testing and maintenance outages, human errors, and common cause failures. The greatest variability relates to the treatment of common cause failures or human error. In addition, the IPEs generally addressed and modeled the dependencies of systems on support systems. However, the basis for eliminating some dependencies (primarily HVAC dependencies), in some cases, was qualitative and made without apparent calculational support. The dependencies are included in the models through either fault tree linking or the use of support state event trees. Continued operability of systems in light of phenomenological impacts caused by the accident progression (e.g., harsh environments), were also not always justified in the IPEs, or were treated qualitatively at best.

A variety of approaches and methods were used in conducting the HRAs for the IPEs. Given that some methods rely more heavily than others on subjective estimation techniques and that different methods often consider somewhat different operator performance shaping factors, the possibility exists that different human error probabilities could be obtained for identical human actions using these different techniques. In addition, all of the methods are somewhat vulnerable to the biases of the analysts performing the HRA. In particular, several instances of the application of one of the HRA methods were determined to be inconsistent with the intent of GL 88-20. As far as completeness in modeling human errors, pre-accident errors are often given screening values and subsequently screened out or just argued to be insignificant. Some plants excluded pre-initiators by claiming that those failures are already included in the component random failure data. This assumption is inaccurate because some of the most important pre-initiators such as common cause miscalibration occur rarely and, therefore, are not represented in the random failure data. Thus, pre-initiators should always be included in a PRA. Some licensees only considered failure to restore components after maintenance and dismissed calibration errors; others included both types of errors. Most licensees included post-initiator operator actions in their IPEs, but consideration of proper timing, plant-specific performance shaping factors, and dependencies between different operator actions was somewhat variable. In a few cases, licensees assumed that the operator would not make any errors in performing a particular action. Plant-specific procedures are generally used in the HRA evaluations, but the use of simulator exercises and operator talk-throughs is not always evident.

The data analysis performed in the IPEs varies. For component hardware failures, most licensees used a combination of plant-specific and generic data to quantify their IPE models. Some licensees performed Bayesian updates of generic data while others directly used plant-specific or generic data. The degree of plant-specific data analysis also varies, with some licensees only generating plant-specific data for components assumed to be major contributors to core damage, and other licensees performing a comprehensive analysis for all components modeled in the IPE. A few licensees used only generic data in their IPEs. (The generic data sources used in most IPEs are from the nuclear industry).

In general, the licensees modeled common cause failures; however, the submittals considered only components that are required to change state. Further, the actual components modeled vary among the IPEs. As is common PRA practice, most licensees only treated common cause failures within a system; a few modeled failures across systems. The treatment of common cause failure data differed somewhat because of their relatively rare occurrences. Most licensees used only generic data; however, some licensees adjusted the generic data to be more applicable to their plants by reviewing the events contributing to the generic data values and eliminating those events that were deemed not applicable. Unusually low common cause failure rates are used in a few IPEs. These rates are generally a result of an optimistic examination of the common cause database, leading to the conclusion that the failures of interest are not applicable to that particular plant. This approach fails to recognize that common cause failures are, by nature, rare events that tend to have some plant-specific features. The approach used to eliminate inapplicable failures needs to also consider possible as yet undiscovered failures relevant to that particular plant.

Details of the quantification process used in the IPEs are not always available. However, the submittals generally discussed the methods and computer codes used. Because these methods and codes are adequate to treat the Boolean reduction process, it is expected that the core damage sequences are appropriately generated. The truncation limits are not provided for many of the IPEs; for the ones that provided this information, an adequately low cutoff appears to have been used, such that ~90 to 95 percent of the CDF would have been captured. In general, the IPEs do not adequately document the elimination of accident sequences as unimportant because of their application of human recovery actions. In addition, the CDFs reported in the IPE submittals appear to be point estimates with no uncertainty propagation.

The majority of the IPEs appeared to have comprehensively considered possible internal flooding sources consistent with the characteristics of a current PRA. However, one source that was not always considered in the IPEs involved inadvertent fire suppression system actuation. Also, it was not always clear whether or not the potnetial for human-induced flooding was included in the IPE analyses. In general, the modeling of flood scenarios in the IPEs also appeared to be consistent with a current PRAs. However, some variability in the modeling was obvious. For example, the majority of the IPEs appeared to include submergence impacts on equipment. However, it was not always clear if spray impacts were incldued in some IPEs. Many of the IPEs performed screening evaluations in which all the equipment in a flood location was assumed to fail by either submergence or spray impacts. A more detailed evaluation was performed in some IPEs for those areas surviving the screening evaluation. Other IPEs ended their internal flooding evaluation with the screening analysis. The flood zone definitions appeared from the limited information provided int he IPE submittals to consider the existence of barriers such as walls, doors, and curbing. In addition, consideration of drains that would limit a lfood to a particular area was also generally consdiered int he flood zone definitions. Flood propagation pathways from one flood zone to another through stairwells, drains, hatches, and doors were also generally considred int eh IPEs. The failure of barriers to propagation was not a singificant factor in the IPE anlayses. Finally, isolation of floods was often included in teh IPE evaluations. However, the methods of detecting a flood and the time available for performing such io\solations were not always provided with a firm basis.

6.2 Level 2 PRA

The primary objective of the Level 2 portion of a PRA is to characterize the potential for, and magnitude of, a release of radioactive material to the environment given the occurrence of an accident that sufficiently damages the core as to cause the release of radioactive material from fuel. To satisfy this objective, a Level 2 PRA couples two major elements of analysis to a completed Level 1 PRA:

(1) structured, comprehensive evaluation of containment performance in response to the accident sequences identified from the Level 1 analysis

(2) quantitative characterization of radiological release to the environment that would result from accident sequences that involve leakage from the containment pressure boundary

The specific tasks performed in either area can vary substantially from one PRA to another depending on the specific objectives of the Level 2 analysis. The approach taken in this report is to identify all of the characteristicss that could be expected in a Level 2 PRA. These characteristics are provided in the following section. Perspectives drawn from a comparison of the IPE analyses with these characteristics are provided in Section 6.2.2.

6.2.1 Characteristics of a Level 2 PRA

The first element of a Level 2 PRA involves an evaluation of containment performance. This process is typically divided into four major tasks as follows:

- assessment of core damage accident sequences into corresponding plant damage state bins
- assessment of the full range of credible challenges to containment integrity (i.e., determination of failure mechanisms and corresponding structural challenges)
- characterization of the capacity of the containment to withstand various challenges (i.e., determination of performance limits);
- organization and integration of the uncertainties associated with these two evaluations to estimate the *probability* that containment would fail (or be bypassed) for a given accident sequence.

An assessment of the type and severity of challenges to containment integrity acknowledges the dependence of containment response on details of the accident sequence that resulted in core damage and on the performance of containment safeguard systems in response to that sequence. Therefore, the first task involves developing a structured process for defining the specific accident conditions to be examined. Because it is impractical to evaluate severe accident progression and resulting containment loads for each cut set generated by a Level 1 analysis, it is common practice to group the Level 1 cut sets into a sufficiently small number of "plant damage states" (PDSs) to render the analysis more tractable. The process by which this simplification of the Level 1 results is performed involves a systematic procedure to group cut sets that would be expected to result in similar severe accident progression and containment response (and ultimately produce a similar environmental source term). The end result is a smaller list of accident sequences to be evaluated which covers the *full range* of initial and boundary conditions for severe accident progression represented by the original list of sequence cut sets.

The second task involves assessment of full range of credible challenges to containment integrity. The accident sequence analysis performed in a Level 1 PRA does not always incorporate the reliability of systems that primarily function to safeguard containment integrity during accident conditions. Such systems may include containment isolation, fan coolers, distributed sprays, and hydrogen igniters. An assessment of the reliability of these systems is, therefore, incorporated in a Level 2 analysis to account for their failure probability in the analysis of containment response during core damage sequences.

The aspect of a Level 2 PRA that often receives the greatest attention is the evaluation of severe accident progression and the attendant challenges to containment integrity. This is because considerable time and effort can be spent performing computer code calculations of dominant accident sequences. Further, exercising broad-scope accident analysis codes (such as MAAP (Ref. 6.9) or MELCOR (Ref. 6.10)) provides the only framework within which the important interactions among severe accident phenomena can be accounted for in an integrated fashion. Consequently, the results of these calculations typically form the principal basis for estimating the timing of major accident events and for characterizing maximum containment loads.

Although code calculations are an essential part of an evaluation of severe accident progression, their results do *not* form the sole basis for characterizing challenges to containment integrity in current PRAs. This is because certain assumptions are made (either by the model developers or the code user) regarding physical processes and the appropriate formulation of representative models that cannot be demonstrated to be robust in light of the numerous large uncertainties that persist in severe accident and source-term phenomena. In addition, none of the integral severe accident codes contain models to represent *all* accident phenomena of interest.

As a result, the process of evaluating severe accident progression involves a strategic blend of plant-specific code calculations, applications of analyses performed in earlier PRAs or severe accident studies, focused engineering analyses of particular issues, and experimental data. Although engineering judgment and expert opinion continue to play an important role in defining a reasonable balance among these varied sources of information, minimum analysis characteristics can be defined to ensure that the results of a Level 2 PRA reflect a reasonable range of views in the technical community. These characteristics are discussed in Part 4, Chapter 14, of this report.

The third task of an evaluation of containment performance is the determination of the limits (or capacity) of the containment to withstand challenges to its integrity that may occur during core damage sequences. These challenges take many forms, including elevated static internal pressure, high temperatures, thermo-mechanical erosion of concrete and steel structures, and (under some circumstances) localized dynamic loads such as shock waves. Realistic estimates of the limits at which the containment structure can no longer withstand these challenges must be generated to provide a metric against which the likelihood of containment failure can be estimated. In a full-scope Level 2 PRA, the characteristics of the analyses necessary to characterize containment performance limits are consistent with those of the containment load analyses against which they will be compared, as follows:

- They focus on *plant-specific* containment performance; application of reference plant analyses is generally inadequate.

- They consider design details of the containment structure, such as containment type, the spectrum and distribution of penetration sizes and types, penetration seal configuration and materials, and discontinuities in the containment structure.

- They consider interactions between the containment and neighboring structures, including the reactor vessel and pedestal, auxiliary building(s), and internal walls.

The final task of a probabilistic evaluation of containment performance is an explicit and quantitative recognition given to uncertainties in the individual processes and parameters that influence severe accident behavior and attendant containment response. The treatment of uncertainty is a distinguishing characteristic of different approaches to evaluating containment performance. The assignment and propagation of statistical distributions for uncertain aspects of the analysis is the most sophisticated approach. The result is a quantitative estimate of the *uncertainty* in containment performance. Containment performance can be expressed as a single value for the probability of containment failure for any accident sequence, and the mean, median, 5th and 95th percentile values can be derived as well. A point estimates of probability are not universally applied to the probabilistic logic model (i.e., containment event tree) supplemented by sensitivity studies is a less sophisticated approach.

The second element of a Level 2 PRA is a quantitative characterization of radiological release to the environment resulting from each accident sequence that contributes to the total core damage frequency. In many *limited scope* Level 2 analyses, this information is used solely as a semi-quantitative scale to rank the relative severity of accident sequences. In such circumstances, a rigorous quantitative evaluation of radionuclide release, transport and deposition may not be necessary. Rather, estimates of the order-of-magnitude of release for a few representative radionuclide species provide a satisfactory scale for ranking accident severity. In a full-scope Level 2 PRA, however, the characterization of radionuclide release to the environment provides sufficient information to completely define the "source term" for calculating offsite health and economic consequences for use in a Level 3 PRA. Further, the level of rigor required in the evaluation of fission product release, transport, and deposition directly parallels that used to evaluate containment performance, as follows:

- Source term analyses (deterministic computer code calculations) reflect *plant-specific* features of system design and operation.

- Calculations of fission product release, transport, and deposition represent *sequence-specific* variations in primary coolant system and containment characteristics.

- The effects of uncertainties in the processes governing fission product release, transport, and deposition are quantified.

Table 6.2 summarizes the key characteristics of a Level 2 PRA. A detailed discussion and description of the characteristics for a Level 2 PRA is provided in Part 4, Chapter 14, of this report.

Table 6.2 Summary of characteristics of a Level 2 PRA (only considering full power and internal events, excluding internal fire).

PRA element and task	Key task activities	Key characteristics
	Containment Performance	
Assessment of plant damage state bins	• Determination of accident sequences to be assessed	• Grouping of cut sets (of entire core damage frequency) based on accident characteristics that influence either subsequent containment performance or resulting fission product source term • Containment systems models are linked directly with accident sequence models from Level 1 to fully account for mutual dependencies
Assessment of Credible Challenges to Containment Integrity	• Containment system performance analysis • Severe accident progression analysis	• Long-term performance of containment safeguard systems accounts for harsh environments associated with severe accidents • Integrated computer code simulations form *one* (not an exclusive) basis for characterizing accident timing and containment response; extensive sensitivity analyses are performed to quantify the effects of alternative, credible code modeling assumptions • Independent, specialized engineering analyses are performed to characterize the effects of modeling simplification in lumped-parameter, integral severe accident analysis codes • Application or adaptation of analyses from surrogate plants is accompanied by analyses to demonstrate applicability • Importance of phenomena not represented in integral severe accident codes evaluated by some other means

Table 6.2 Summary of characteristics of a Level 2 PRA (only considering full power and internal events, excluding internal fire).

PRA element and task	Key task activities	Key characteristics
Characterization of Containment Performance Limits	• Engineering analyses of structural capacity to withstand loads • Systematic search for plant-specific containment failure modes	• Engineering analyses of performance limits account for plant-specific design features, such as discontinuities in containment shell, penetrations, and interactions with neighboring structures • Evaluation of ultimate pressure capacity is performed using a finite-element model of sufficient detail to capture the effects of localized design features (listed above) • Engineering analyses are performed to characterize likely failure sizes for each postulated failure mode
Probabilistic Characterization of Containment Performance	• Development of a probabilistic logic structure to trace severe accident progression (e.g., containment event tree) • Assignment of probabilities to uncertain parameters and events in the logic model	• Logic structure includes explicit recognition of importance time phases in severe accident progression • Interdependence of accident phenomena is treated consistently such that a time-line of accident progression can be created for each accident sequence • Uncertainty in the quantification of events and phenomena is addressed through the assignment of probability density functions to individual events (logic model provides statistical mechanism for combining the distributions in a consistent manner)
Radiological release		
Characterization of Radionuclide Release	• Source term definition • Coupling of source term and severe accident progression analysis • Treatment of source term uncertainties	• Radiological source term(s) are defined for each accident sequence in the Level 2 model • Uncertainties in the radiological source term for a given accident sequence are quantified such that the distribution of plausible environmental releases associated with any sequence can be defined • Correlations between uncertain parameters associated with fission product release, transport, and deposition and with other aspects of severe accident behavior are identified and treated consistently • Source term is characterized by sufficient information to calculate environmental consequences in a Level 3 PRA

6.2.2 IPE Level 2 Perspectives

The IPEs exhibit greater variation in the methods and scope of the accident progression analyses than that found in the core damage frequency analyses. This is commensurate with the guidance of GL 88-20 and NUREG-1335 (Ref. 6.11), which allowed significant diversity in the ways licensees could conduct their containment performance analyses. In many of the IPEs, the containment performance analysis and the source term calculations are more simplified than one would expect to find in a current PRA.

Most licensees used a PDS approach to initiate the containment performance analysis and link the core damage frequency analysis with the accident progression. In most cases, the PDS characteristics were selected so that the impact of initiators, RCS conditions, and availability of mitigating systems were adequately considered. However, in some IPEs, important information like the availability of power is not explicitly carried over to the PDSs and can only be inferred, making the actual availability and operability of mitigating systems less clear. This is the case in some IPEs where a previously existing Level 1 PRA (i.e., core damage frequency analysis) was updated and an accident progression analysis added for the IPE exercise. Also, in a few IPE submittals, the information presented does not sufficiently describe the linkage between the core damage frequency and accident progression analysis, and it is not possible for the staff to judge whether information was adequately carried over.

In most of the IPEs, the CDF was adequately conserved in the accident progression analysis (i.e., CDF and PDS frequency closely agree). However, there are some important exceptions. In a number of IPEs, the CDF from internal flooding was not carried over to the accident progression. Also, in some IPEs accident sequences below a certain frequency, included in the core damage frequency analysis, were not carried over to the Level 2 analysis and it is not always clear that their contribution to early release was insignificant. Finally, in some IPE accident progression analyses, licensees argued that success criteria used in the Level 1 analysis were too pessimistic and a number of sequences were dropped from further consideration because they were not believed to actually lead to core damage.

Most licensees considered the appropriate spectrum of known severe accident phenomena that could threaten their containment in the accident progression analysis. However, not infrequently, licensees dismissed a number of these phenomena with little quantitative justification in the submittal. In other cases, based on submittal information, licensees treated phenomena without properly considering the uncertainty attached to their magnitude and timing, and sufficiently varied sensitivity analyses were not performed. Additionally, phenomena effects on prolonged use of containment safeguard systems were not always considered.

The majority of the licensees relied on reference calculations for parts of their analyses, such as quantifying certain split fractions in their containment event trees. A significant number of licensees relied completely on reference plant methods and calculations to carry out their accident progression analyses. While most of these sufficiently justify their similarity with the reference plant and adequately account for differences, the justification is unsatisfactory in a few cases, and appropriate sensitivity analyses addressing differences from the reference plant were not performed. In any case, complete reliance on reference plants for the accident progression analyses is not appropriate for a PRA.

The majority of licensees performed a thorough and detailed containment failure analysis for their plants; in cases where a reference plant calculation was used in this area also, the analysis cannot be considered up to the characteristics of a PRA.

The IPE results obtained for the accident progression analysis are point estimates with no uncertainty propagation. The MAAP code was widely used in the IPEs to quantify parts of the accident progression. While some licensees recognized the limitations in the treatment of severe accident phenomena by MAAP and supplemented their analyses with information from other sources, many did not and, therefore, did not consider scenarios outside the range of MAAP results. Also, a number of severe accident phenomena were eliminated in some IPEs, based on the viewpoint of set of industry position papers. A more comprehensive treatment of such phenomena would be to use sensitivity studies supplemented by experimental data.

The logic and level of detail in the containment event trees found in the IPEs varies considerably. Many licensees provided good delineation and logic in the trees in their IPEs, while others were too simplified to be considered adequate for a PRA. In addition, many licensees deliberately avoided considering the impact of operator performance on the accident progression in their IPEs, especially those conducted for PWR plants. For BWRs, the necessary operator actions in the Level 2 analysis were usually included, although their quantification was sometimes inadequately justified. In addition, in a number of IPEs, the effect of severe accident environmental conditions on system operability is apparently not considered or dismissed.

The mission time considered in the accident progression analyses in some IPEs was only 24 hours. In many others, a 48 hour mission time was analyzed, while still others took the analysis to a point where either a release occurred or a safe stable state was assured. In some analyses that terminated after 24 or 48 hours, sequences which would lead to a late containment failure, beyond the mission time, were binned with no containment failure categories.

The source term analyses in the IPEs usually relied on a limited number of selected MAAP runs, and in most cases, did not include a robust sensitivity analysis. In some IPEs, the source term binning did not adequately differentiate the timing of the release or the containment failure type. Release results were usually presented in terms of the fission product groups in MAAP. Some analyses, however, presented only noble gas fractions and those for Cesium and Iodine, while other IPEs reported only ranges of releases, rather than actual core inventory fractions. In general, the source term analyses found in the majority of IPEs were not adequate for a PRA.

6.3 Level 3 PRA

The analysis tasks performed in a Level 3 PRA consist of two major elements:

- accident consequence analysis
- computation of risk by integrating the results of Level 1, 2, and 3 analyses

The characteristics of a Level 3 PRA are summarized for each of the above elements in the following section. Perspectives drawn from a comparison of the IPE analyses with these characteristics are provided in Section 6.3.2.

6.3.1 Characteristics of a Level 3 PRA

The consequences of an accident release of radioactive material from a nuclear power plant can be expressed in several forms such as impacts on human health, the environment, and economics. The consequence measures of interest to a Level 3 PRA focus on impacts to human health. These measures are estimated both in societal terms and in terms of the most-exposed individual. The MACCS computer code (Ref. 6.12) is acceptable for use in a Level 3 PRA to calculate the consequences based on the radiological source terms generated in the Level 2 portion of the PRA.

Table 6.3 summarizes the key characteristics of a Level 3 PRA. Additional discussion and description of the characteristics for a Level 3 PRA and the process of computing risk is provided in Part 4, Chapter 14, of this report.

Table 6.3 Summary of characteristics of a Level 3 PRA (only considering full power and internal events, excluding internal fire).

PRA element and task	Key task activities	Key characteristics
Consequence Analysis	• Collection of demographic and weather-related data • Reduction of source terms from Level 2 analysis for consequence calculations • Consequence calculation	• Data regarding local meteorology and terrain, site demographics, and local land use represent current, plant-specific conditions • Source terms used to calculate offsite consequences preserve the full range of early (mechanistic) and late (stochastic) health effects that would result from actual Level 2 source terms • Variability in weather are addressed as major uncertainty in consequences

6.3.2 IPE Level 3 Perspectives

Only a handful of utilities carried out a consequence analysis as part of their IPE. A few utilities briefly referred to Level 3 analyses done for a previous PRA, but without presenting the associated results in the IPE submittal. IPEs for which the licensee conducted a Level 3 analysis used either the MACCS or CRAC2 code (Ref. 6.13) to calculate their results. Sufficient review has not been performed by the staff to compare these Level 3 PRAs to the above characteristics because the focus of GL 88-20 was on a Level 1/simplified Level 2 analysis.

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7. ADDITIONAL INDIVIDUAL PLANT EXAMINATION PERSPECTIVES

The results and perspective presented in this chapter are based onthe original Individual Plant Examination (IPE) submittals forwarded by the licensee to the U.S. Nuclear Regulatory Commission (NRC), (refer to Appendix A for sources of IPE information used in this report). Many licensees have updated their IPEs since the original submittal and have reported bottom-line changes in core damage frequency (CDF) or release frequency; but in most cases, detailed information was not provided. Updated IPE information received by the NRC is listed in Appendix B.

7.1 Safety Goal Implications

The purpose of this section is to use the results of the IPEs to assess how existing plants compare with the safety goals of the NRC. More detailed discussions of this subject are presented in Chapter 15. The Safety Goals Policy Statement (Ref. 7.1) established two qualitative safety goals that are supported by two quantitative health objectives (QHOs) as follows:

> "The risk to an average individual[7.1] in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed."
>
> (Based on the risk of accidental death in the U.S. this implies a prompt fatality QHO of 5E-7 per year).
>
> "The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes."
>
> (Based on the occurrence of cancer fatalities, this implies a latent cancer fatality QHO of 2E-6 per year).

It is important to note that the safety goals do not represent regulatory requirements that all plants must meet. Uncertainties in the probabilistic risk assessment (PRA) process are sufficiently large that such a rigorous implementation of the safety goals is not warranted. Rather, the policy statement indicates that the goals should serve as "aiming points or numerical benchmarks." This interpretation of the safety goals should be kept in mind while reading the remainder of this chapter.

During implementation of the Safety Goals Policy Statement (Ref 7.2), several subsidiary objectives arose, including a total CDF of 1E-4 per reactor-year (ry) and a total conditional containment failure probability (CCFP)[7.2] of 0.1. A "large" release frequency of 1E-6/ry was subsequently dropped. (Ref. 7.2)

When comparing the IPE results with the above objectives, it is important to note that the scope of the IPE program is limited to accidents initiated by internal events (excluding internal fires) that occur during full-power operation. Therefore, the risk estimates inferred from the IPE results may reflect

[7.1] The calculation of individual risk is discussed in Chapter 15 where individual risk is also contrasted with societal risk.

[7.2] Conditional containment failure probability is defined as the probability of containment failure conditional on core damage having occurred. Chapter 14 provides a more detailed discussion concerning the definition and estimation of conditional failure probability.

only a fraction of the total risk of operating the plant. The results of other PRAs that include external events (and internal fires) and other modes of operation (e.g., low-power and shutdown) indicate risk levels comparable with those obtained for internal events during full-power operation.

The issuance of the Safety Goals Policy Statement was followed in May 1988 by an "Integration Plan for Closure of Severe Accident Issues" (Ref. 7.3), which has several elements, including the IPEs. In response to Generic Letter (GL) 88-20 (Ref. 7.4), the licensees were not requested to calculate offsite health effects (although some did); thus, the IPE results, in general, cannot be directly compared against the health objectives recounted above. However, certain subsidiary objectives can be obtained directly from the IPEs. These numerical objectives are presented in Figures 7.1 (CDF) and 7.2 (CCFP). The values (CDF and CCFP) from the IPEs are generally point estimates.

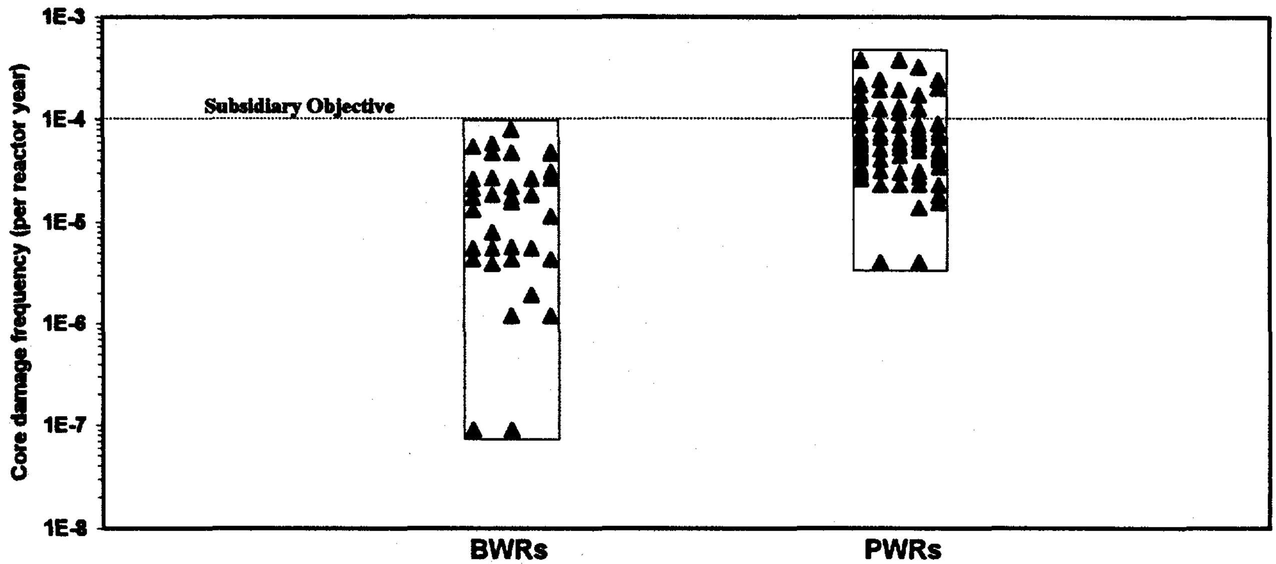

Figure 7.1 Core damage frequency for BWR and PWR IPEs.

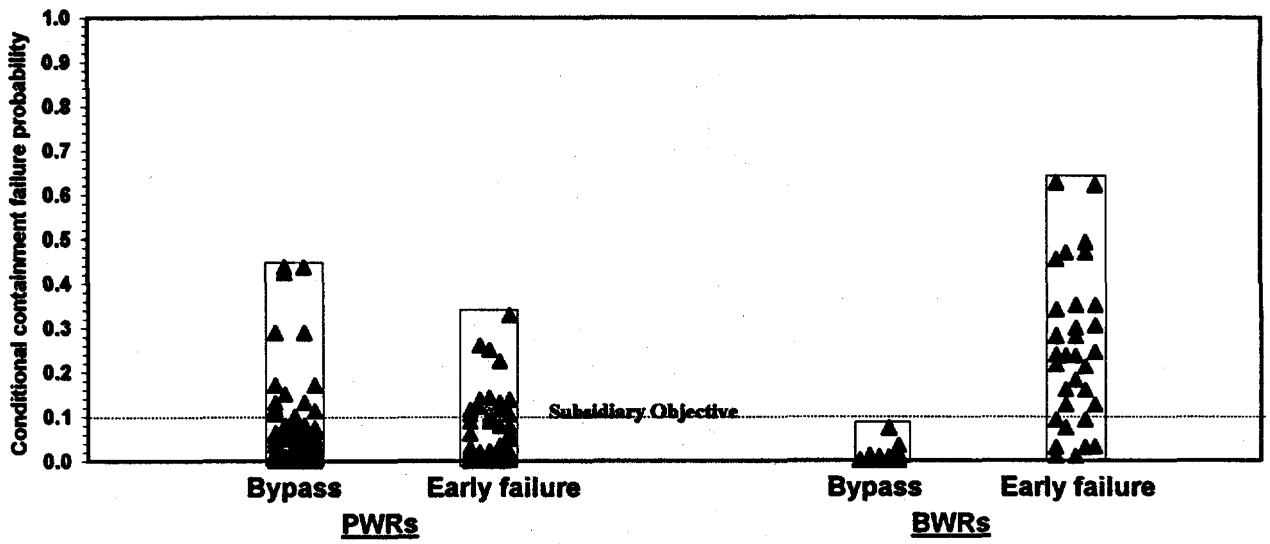

Figure 7.2 Conditional containment bypass and early failure probabilities for all BWRs and PWRs.

The CDFs for all boiling water reactors (BWRs) and most pressurized water reactors (PWR) fall below 1E-4/ry; however, nine licensees representing 15 PWR units reported CDFs above 1E-4/ry. CCFPs for bypass and early containment failure are below 0.1 for most of the PWRs. All of the CCFPs for bypass events in BWRs are below 0.1; however, most of the CCFPs for early containment failure are above 0.1. This result is expected because of the nature of BWR pressure suppression containments.

Since several plants exceed the subsidiary objectives, it is useful to compare the IPE results with the QHOs in order to determine how the existing plants compare with the safety goals. A direct comparison is not possible since, as noted above, offsite consequences and health effects generally were not calculated in the IPEs. Offsite health effects were reported, though, for five plants in NUREG-1150 (Ref. 7.5). The NUREG-1150 results can be used to obtain risk insights from the IPE results using a two-step process:

(1) NUREG-1150 identified the key containment failure modes contributing to offsite health risks for several plants. Therefore, in the first step, these results are compared with the frequencies of key containment failure modes (such as early structural failure and bypass) as reported in the IPEs. This comparison permits an inference regarding how the IPE results might compare to the health objectives of the safety goals.

(2) In the second step, plants that have relatively high frequencies of key containment failure modes are examined in greater detail to more accurately assess potential contributors to early fatality risk.

One objective of the NUREG-1150 study was to gain and summarize perspectives regarding risk to public health from severe accidents at five commercial nuclear power plants. Consequently, the study included calculation of several risk measures, including individual fatality risks, for comparison with the NRC's safety goals. The results reported in NUREG-1150 indicated that the early fatality and latent cancer fatality risk estimates for accidents at all five plants studied are below the NRC's safety goals. On the basis of the NUREG-1150 results, the mean individual early fatality risk at Surry would have to increase by a factor of about 30 in order to approach the early health objective. For Sequoyah or Zion, individual latent cancer fatality risk would have to increase by more than two orders of magnitude to approach the corresponding health objective. This margin does not account for the risk associated with external events plus internal fire and flood and other modes of operation.

NUREG-1150 also reported the contributions of the various accident progression bins to mean early and latent cancer fatality risk for the five plants. Inspection of the NUREG-1150 results clearly suggests that accidents resulting in containment bypass or early containment failure are dominant contributors to the risk of both early fatalities and latent cancer fatalities. Consequently, by comparing the NUREG-1150 results with the frequencies of early containment failure or bypass reported in the various IPE submittals, one can draw conclusions regarding how the existing plants might compare with the NRC's QHOs. Figure 7.3 compares the early containment failure and bypass frequencies reported in each of the IPEs with those obtained in the NUREG-1150 study. The IPE results presented are the mean or point estimate frequencies taken directly from the submittals and do not include estimates of uncertainty. The NUREG-1150 results include the mean, 5th percentile, and 95th percentile frequencies for each of the five plants studied.

A comparison of the frequencies of early containment failure or bypass given in Figure 7.3 indicates that most of the IPE results are similar to or lower than the NUREG-1150 results. This implies that most plants have risk levels similar to those reported in NUREG-1150. However, there are a number of plants that have early containment failure or bypass frequencies that are an order of magnitude higher than the Surry results. If the Surry risk results are assumed to be applicable to the higher frequencies for these plants, the risk estimates are unlikely to approach the individual latent cancer fatality health objective because of the margin between the NUREG-1150 results and the cancer fatality QHO. However, because of the smaller margin between the NUREG-1150 results and the prompt fatality QHO, the risk estimates for plants with higher frequencies of early containment failure and bypass could approach this health objective.

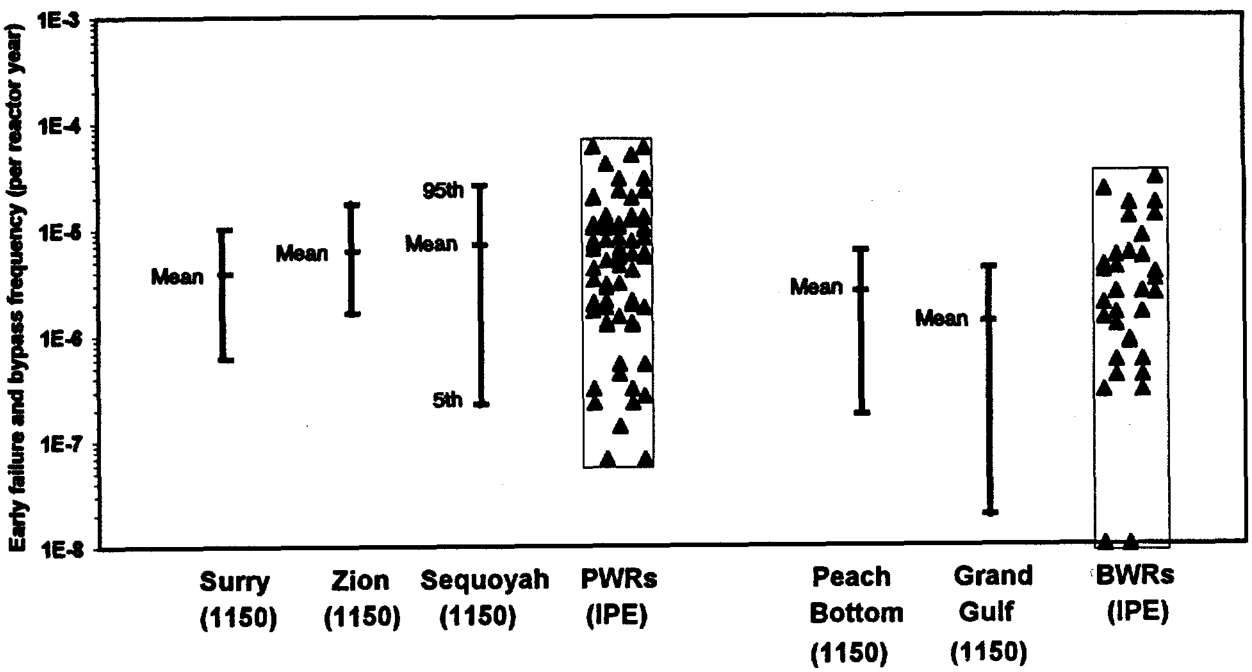

Figure 7.3 Comparison of NUREG-1150 and IPE results concerning early containment failure and bypass frequencies for PWRs and BWRs.

Assuming that the Surry risk estimates can be used as surrogates for risk at other plants is potentially misleading. The predicted individual early fatality risk is a complex function of the quantity of radioactive materials released, the timing of the release, and site-specific characteristics such as meteorology and the population distribution in the vicinity of the plant. Consequently, the higher frequencies of early containment failure and bypass for the IPE results in Figure 7.3 may not translate into equivalent increases in early health risk. Therefore, plants with relatively high frequencies of early containment failure and bypass are examined in more detail as discussed below.

In the IPE submittals reviewed by the staff, 18 licensees (representing 29 units) reported early containment failure and bypass frequencies higher than 1E-5/ry. The staff reviewed each of these submittals to determine the frequencies of source terms that would be expected to give rise to early fatalities within a one mile radius of the plant. One licensee (for La Salle 1&2) submitted a Level 3 IPE and was eliminated on the basis of the reported results. For all other submittals, source terms with early releases of iodine (I), cesium (Cs), and tellurium (Te) isotopes (which mainly impact early health risk) greater than three percent are assumed to have the potential to result in early fatalities within a one mile radius of the plant. These frequencies, in turn, were adjusted by a factor to reflect the fraction of the population actually at risk given a release. For most weather conditions and given an approximately uniform population distribution in the plant's vicinity, the plume characteristics will place approximately two percent of the population out to one mile at risk of an early fatality. (Refer to Chapter 16 for a more detailed discussion on this factor.) Therefore, the individual risk of an early fatality was estimated by examining the frequencies of the early containment failure and bypass source terms, with release fractions of the core inventory of I, Cs, Te greater than three percent, multiplied by a factor of 0.02 to reflect plume characteristics.

After this second step, 14 units were estimated to have the potential for relatively high individual early fatality risk within a factor of two higher or lower than the early fatality QHO of 5E-7/ry. Of these 14 units, two are BWRs with Mark I containments. For those plants, shell melt-through is the primary cause of early containment failure. This failure mode was identified as a Mark I specific failure mode before GL 88-20 was issued, and has been addressed separately through experimental research and analysis.

Of the remaining 12 units, eight are PWRs with large dry containments and relatively high CDFs and CCFPs. The relatively high CCFP (approximately 0.3) for one plant (Palisades) results from a unique failure mode in which core debris is predicted to melt through a pipe and enter the auxiliary building. The CCFP for five other units, Calvert Cliffs 1&2 and Palo Verde 1,2,&3 is lower (at approximately 0.1) than Palisades and driven by assumptions regarding early overpressurization failures associated partly with direct containment heating. This failure mode also was identified before GL 88-20 was issued, and is also being addressed by research activities. Two other units, Ginna and Haddam Neck, have relatively high containment bypass frequencies.

The remaining four units are PWRs with subatmospheric containments. The CCFP for two of these units (Beaver Valley 1&2) is also relatively high (approximately 0.3) and isolation failure is an important contributor. The probability of isolation failure is large because, for station blackout sequences, the IPE model does not take credit for operator actions to manually isolate the containment. The IPE results for Surry 1&2 are driven mainly by containment bypass. (While containment bypass was also identified as the major contributor to the risk of early fatality from internal initiators at Surry in the NUREG-1150 study, the mean bypass frequency in NUREG-1150 was estimated to be more than one order of magnitude smaller than to the IPE result. Hence, the mean distribution of the individual risk of early fatality reported in NUREG-1150 lies well below the early fatality QHO.)

While these preliminary estimates of early fatality risk based on the IPE results are approximate, they point to the need to more carefully examine site-specific characteristics at individual plants. For example, the conditional consequences of early failures may be more severe at sites such as Beaver Valley that have a relatively high population density (compared to, say, Ginna) in the plant vicinity. Site meteorology and plant size (i.e., the radionuclide inventory) can also influence offsite consequences. In addition, as noted earlier these results are based on the original IPE submittals. Many licensees have updated their IPEs and reported significantly lower large early release frequencies. In some cases, these new results imply lower early risk estimates than those given above. (Refer to Appendix B for updated IPE information.)

In summary, by comparing the IPE results with the NUREG-1150 study, the staff concluded that, with a few exceptions, most of the IPE results are likely to meet the NRC's QHOs. The IPE results, which only reflect accidents initiated by internal events at full power, imply risk levels below the individual latent cancer fatality health objective. In addition, with the possible exception of a few plants, the IPE results also suggest risk levels below the individual early fatality health objective. Although relatively more plants exceeded the proposed subsidiary objectives, only a fraction of these are found to have the potential for individual early fatality risk levels that could approach the corresponding QHO.

7.2 Impact of the Station Blackout Rule on Core Damage Frequencies

A station blackout (SBO) is the total loss of AC power to the essential and nonessential equipment at a nuclear power plant. If an SBO occurred and AC power was not recovered, it would ultimately result in core damage since nuclear reactors require AC power for decay heat removal. The SBO rule (Ref. 7.6) requires that nuclear power units perform an analysis to establish a method to cope with an SBO for a specified duration without core damage occurring (coping method). In some instances, licensees implemented unit modifications to improve the unit's ability to endure an SBO. The goal of the SBO rule is to limit the average so contribution to CDF to about 1E-5/ry. This goal should be interpreted in the same manner as the safety goals, that is, as an aiming point or numerical benchmark, rather than as a hard and fast requirement. This section provides perspectives on the SBO rule impact on CDF as reported in IPE submittals and on the achievement of the goal of 1E-5/ry. In addition, this section identifies and characterizes the important variables that contributed to the variation in SBO CDF results.

SBO-related information contained in the IPE submittals was used to determine the reported impact on CDF. The submittals were categorized by considering whether the SBO rule was addressed, whether the SBO rule coping method was modeled, and whether the impact of the SBO rule on CDF was reported. Those submittals with insufficient information to categorize the impact of the rule were excluded.

In all, the staff reviewed 75 IPE submittals (covering 108 plant units) and various licensee responses to requests for additional information were reviewed to obtain information addressing the SBO rule. For licensees that modeled the SBO rule coping method, the staff compared the average SBO CDF with the rule's goal to determine how well it was achieved. For licensees that did not model the SBO rule coping method, the staff compared the average SBO CDF with the rule's goal to provide insight into the margin for improvement in CDF by implementing the SBO rule. The staff also performed a regression analysis to assess the factors that are directly relevant to variations in the SBO CDF.

Ten licensee IPE submittals (covering 15 plant units) reported estimates of the reduction in total CDF that resulted from implementing the SBO rule. The average reported reduction was ~2E-5/ry, ranging from ~7E-6 to ~6E-5/ry. The average reported percent reduction in total CDF was about 20%, ranging from about 10 to 50%. Licensees that met the SBO rule using existing equipment were not included in the average CDF reduction calculation.

Of the 108 plant units reviewed by the staff, 54 modeled the SBO rule coping method. The variability of the reported SBO CDF was large, ranging from negligible to ~3E-5/ry. The average reported SBO CDF was ~9E-6/ry. The variability of the reported percent SBO contribution was also large, ranging from 0 to about 90%. The average reported percent SBO contribution was about 20%. Figure 7.4 summarizes the reported SBO CDFs for these units.

The IPE submittals exhibited a wide range of SBO CDFs relative to the SBO rule goal. Some licensees that modeled the SBO rule coping method reported SBO CDFs about two orders of magnitude lower than the goal, while others reported SBO CDFs about three times higher than the goal.

Licensees reporting the lowest SBO CDFs attributed the values to their plants having highly redundant and reliable emergency diesel generator configurations, having intradivisional crosstie capability backed up by an alternative AC power source, having emergency diesel generators that do not depend on room cooling; having a low loss of offsite power (LOSP) initiating event frequency, and having a battery depletion time of 6 hours or more. Licensees reporting high SBO CDFs relative to the SBO rule goal or high percent SBO contributions, attributed the values to having only a short time in which to recover offsite power[7.3], having a battery depletion time of 4 hours or less, having less emergency diesel generator redundancy, and having a limited supply of water for the auxiliary feedwater system.

[7.3]Having only a short time in which to recover offsite power means that the probability of not recovering power is higher than if a longer recovery time was available.

Of the 108 plant units reviewed by the staff, 27 did not model the SBO rule coping method. The units exhibited large variability in the reported SBO CDFs, ranging from ~1E-7 to ~6.5E-5/ry. The estimated average SBO CDF (~1E-5/ry) was comparable to the SBO rule goal. The 27 units also exhibited large variability in the percent SBO contribution, ranging from about 2% to 70%. The average estimated percent SBO contribution was about 20%. Figure 7.5 depicts the reported SBO CDFs.

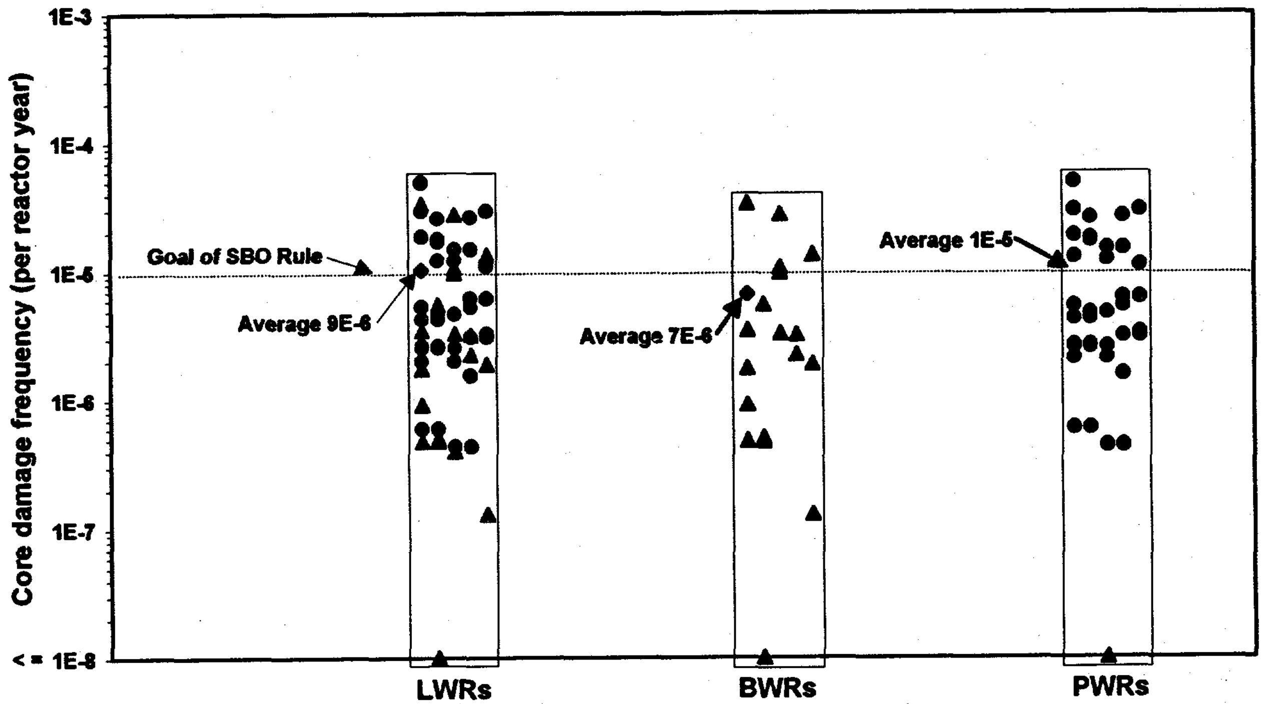

Figure 7.4 Units where licensees modeled the coping method.

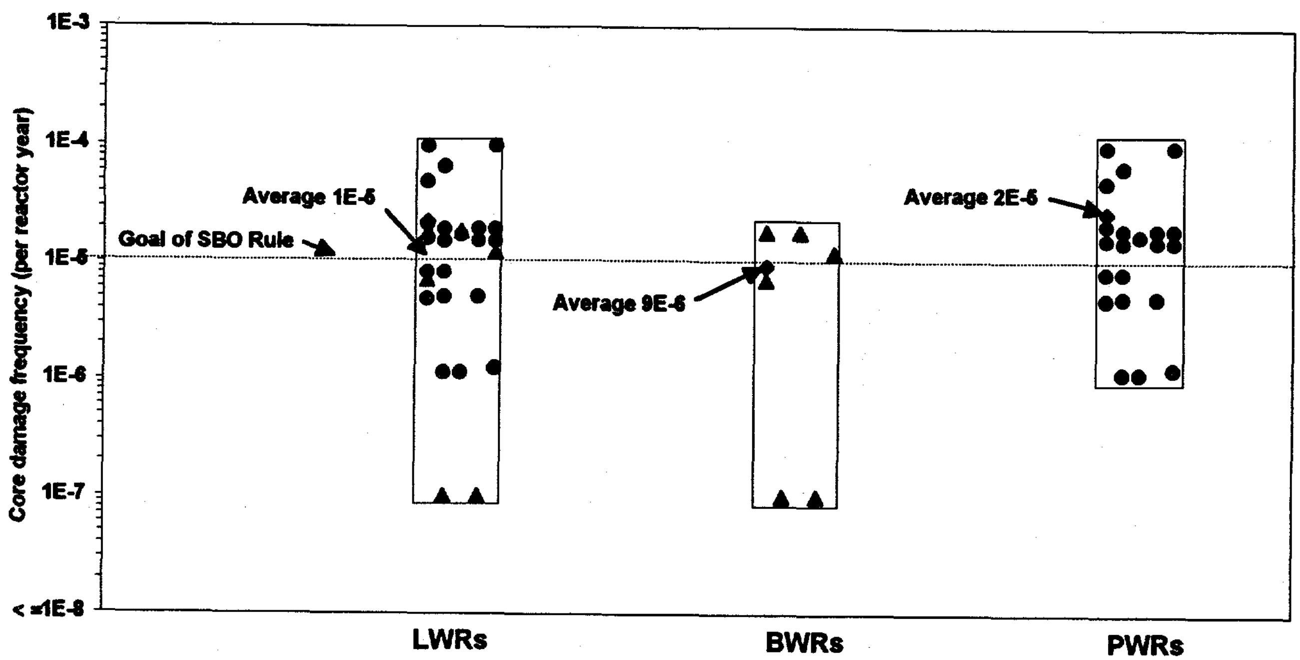

Figure 7.5 Units where licensees did not model the coping method.

The units reviewed by the staff exhibited a wide range of SBO CDFs relative to the SBO rule goal. Some licensees reported SBO CDFs two orders of magnitude lower than the SBO rule goal without modeling the SBO rule coping method. Others reported SBO CDFs close to an order of magnitude higher than the goal.

Licensees reporting low SBO CDFs attributed the low values to their plants having highly redundant and independent emergency diesel generator configurations, having a low LOSP initiating event frequency, having a battery depletion time of eight hours or more, having operator action to manually control auxiliary feedwater flow following battery depletion, and having a low likelihood of reactor coolant pump (RCP) seal loss of coolant accidents (LOCAs). Licensees reporting high SBO CDFs relative to the SBO rule goal or high percent contributions, attributed the high values to their plants having a high likelihood of an RCP seal LOCA during the SBO or a short battery lifetime.

For BWRs, a simple regression of the SBO CDFs indicates that the variation in SBO CDFs is caused by several factors. These factors include the LOSP initiating event frequency, battery lifetime, diesel generator configuration, emergency AC power configuration, site weather characteristics, and the independence of offsite power systems. The emergency AC power configuration is the most important.

For PWRs, the regression of the SBO CDFs indicates that the variation in SBO CDFs is caused by a combination of factor. The emergency AC power configuration and the RCP seal LOCA are the most important.

The average reported reduction in total CDF (~2E-5/ry, of which a significant portion was due to the reduction in the SBO CDF) is consistent with the average reduction in SBO CDF (3E-5/ry) from the backfit analysis of the SBO rule (Ref. 7.7). The average SBO CDF for all plant units considered in the evaluation (1.5E-5/ry) is comparable to a "typical" estimate (on the order of 1E-5/ry) from an evaluation of SBO accidents at nuclear power plants (Ref. 7.8). The large variability in the SBO CDF results for all the plant units evaluated (negligible to ~6.5E-5/ry) is also consistent with the variability in the SBO CDF results (~1E-6 to ~1E-4/ry) from the evaluation of SBO in other studies.

In summary, the results in the IPE submittals indicated that implementing the SBO rule measurably reduces total CDF, largely because of the reduction in SBO CDF. However, there are exceptions in which licensees used existing equipment to satisfy the SBO rule. The SBO CDF is also impacted by a combination of factors (i.e., design features, site characteristics, and/or modeling assumptions and techniques) unique to each unit.

7.3 Comparison with NUREG-1150

In 1990, the NRC published NUREG-1150, which assessed risk for five nuclear power plants covering both PWR and BWR designs. While these five plants only represent a small sample of designs, it is possible to consider whether the NUREG-1150 results and perspectives are consistent with those found in the IPEs. This section only compares the global perspectives discussed in NUREG-1150 with the overall results of the IPEs. This section does not provide perspectives regarding a plant-specific comparison between NUREG-1150 and the applicable IPE analyses. Perspectives of this nature are provided in the individual SERs on the five IPEs.

The average CDFs from the NUREG-1150 PWR analyses fall within the ranges of the CDFs estimated for the PWRs in the IPEs. Similarly, the average CDFs for the NUREG-1150 BWR analyses fall within the ranges of the BWR IPE values. The mix of relative contributions of accident sequences in the IPE results is consistent with the NUREG-1150 results. For the PWRs, station blackout, transients, and LOCAs are usually the more important contributors. For the BWRs, LOCAs and anticipated transients without scram (ATWS) are generally less important than SBO and transients.

The staff also compared the conditional probabilities of early containment failure in NUREG-1150 results with the IPE results. The NUREG-1150 results (mean values) fall within the range of the IPE results for each containment type. NUREG-1150 identified that the conditional probability of early failure is significantly lower for PWRs with large dry or subatmospheric containments than for other PWRs and BWRs with pressure suppression containments. This trend is not as apparent in the IPE results. However, the IPE results indicated that BWR containments generally have higher conditional probabilities for early containment failure than PWR plants. For those IPEs with extremely low or high early containment failure probabilities, the probabilities are mainly driven by modeling assumptions and plant-specific features. On the basis of absolute frequency, early containment failures for BWRs are similar to those for PWRs because the higher values for the conditional early containment failure probabilities for the BWR containments are compensated by the lower values for the BWR CDFs.

NUREG-1150 provides general perspectives for BWRs and PWRs based on the five plants analyzed. Generally, the perspectives discussed in NUREG-1150 are consistent with the IPE results; however, there are some notable exceptions. Table 7.1 summarizes each perspective discussed in NUREG-1150 with a comparison to the IPE analyses. More details on these comparisons are contained in Chapter 17.

Table 7.1 Comparison of NUREG-1150 perspectives with IPE results.

NUREG-1150 perspectives	Comparison with IPE perspectives
BWRs tend to have lower CDFs than PWRs	This finding is less pronounced for the IPEs. LOCAs in BWRs are not as important as in PWRs because BWRs have more makeup systems than PWRs
Support systems are crucial to the CDF	This finding holds for IPEs. Support systems (electrical power; service water (SW); instrument air; and heating ventilating and air conditioning) failures reduce redundancy of front-line systems
Operator recovery actions significantly reduce the CDFs	This finding holds for IPEs. Improvements in emergency operating procedures have enhanced the effectiveness of operator recovery actions
Properly designed crossties between systems can substantially decrease the CDF	This finding holds for IPEs. Many plants can crosstie a few important systems. Administrative control is important to prevent incorrect crosstie
Station blackout is important at both PWRs and BWRs	This finding holds for IPEs. SBO contributes relatively more to the CDF for BWRs than for PWRs. However, SBO tends to give a higher absolute frequency at PWRs than at BWRs, partly because of RCP seal LOCAs at PWRs
Containment venting can reduce CDF at BWRs	This finding holds for IPEs. Injection systems can fail from high containment pressure, high suppression pool temperature, loss of net positive suction head, or harsh environment in the reactor building following containment failure. Venting can eliminate these failures at some plants
Loss of SW or component cooling water can be dominant at PWRs	This finding holds for IPEs. Loss of component cooling water or SW may lead to loss of high-pressure injection and RCP seal LOCA. RCP seal LOCA is more important at Westinghouse plants with old seal designs than at other PWR plants

Table 7.1 Comparison of NUREG-1150 perspectives with IPE results.

NUREG-1150 perspectives	Comparison with IPE perspectives
Feed-and-bleed is an important safety strategy at many PWRs	This finding holds for IPEs. Feed-and-bleed requires power-operated relief valves (PORVs) to be unblocked at most plants. However, many PWRs have PORVs blocked because of PORV leakage problems
Switchover to recirculation is important to LOCA CDFs at PWRs	This finding holds for IPEs. Substantial variation exists among the PWR designs for recirculation switchover. Plants with manual switchover, a smaller refueling water storage tank, and a low containment spray setpoint tend to have higher LOCA CDFs
Large dry and subatmospheric containments are highly likely to maintain integrity during a severe accident	This finding holds for most IPEs, but with some notable exceptions. For some of these plants, phenomena associated with high-pressure melt ejection were important causes of loss of integrity. Isolation failures and bypass accidents were found to be important in other plants. For a few plants, specific design features lead to unique and significant failure modes
The likelihood of early containment failure is higher for ice condenser designs than for large dry and subatmospheric designs	This finding does not hold for IPEs. The opposite trend appears to be driven by the modeling assumptions made in the five ice condenser IPEs rather than any phenomenological or design-related reasons
There is a substantial likelihood of early failure in BWR Mark I containments as a result of direct attack of the drywell shell by molten core debris	This failure mode was also found to be important in most of the IPEs for Mark I plants. GL 88-20 gave licensees the option whether to address this issue. Some licensees did not consider this issue and the potential for early containment failure was found to be low. In other IPEs, the likelihood of early failure was high even if shell melt-through was neglected
Venting can eliminate some sequences that would otherwise result in gradual overpressure failure of Mark I containments	Most IPE results for Mark I containments include late containment venting as a way to prevent late containment failure
Hydrogen deflagration is the principal mechanism for early containment failure in BWR Mark III containments	The conditional probabilities of early failure in the IPEs were less than the NUREG-1150 values because of modeling assumptions and high reliability of the igniter system in the IPEs
If core damage is arrested invessel, the likelihood of containment failure is small for all containment types	Not all IPEs considered this effect. For those that did, the impact on the progression results was significant
Containment bypass events represent a large fraction of high-consequence accidents for PWR containments	This finding holds for most of the IPEs and, in some cases, bypass events dominated the frequency of high-consequence accidents

REFERENCES FOR CHAPTER 7

7.1 USNRC, "Safety Goals for the Operation of Nuclear Power Plants: Policy Statement," *Federal Register*, Vol. 51, p. 30028, August 21, 1986.

7.2 USNRC, SECY-89-102, "Implementation of Safety Goal Policy," March 30, 1989.

7.3 USNRC, SECY-88-147, "Integration Plan for Closure of Severe Accident Issues," May 25, 1988.

7.4 USNRC, "Individual Plant Examination for Severe Accident Vulnerabilities—10 CFR§50.54(f)," Generic Letter 88-20, November 23, 1988.

7.5 USNRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990.

7.6 USNRC, "Loss of All Alternating Power," *Code of Federal Regulations*, Title 10, Section 50.63, July 21, 1988.

7.7 USNRC, "Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout Accident," NUREG-1109, June 1988.

7.8 USNRC, "Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to Unresolved Safety Issue A-44," NUREG-1032, June 1988.

8. OVERALL CONCLUSIONS AND OBSERVATIONS

In considering the perspectives summarized in Chapters 2 through 7 (and discussed in greater detail in Chapters 9 through 17), the staff drew certain conclusions and observations regarding the results reported in the Individual Plant Examinations (IPEs). These conclusions and observations address the following areas, as discussed below:

- Generic Letter (GL) 88-20 objective (including improvement of plant safety)
- regulatory follow-up activities
  - — plant safety enhancements
  - — containment performance improvements
  - — potential future uses of IPEs/PRAs
  - — plants with relatively unfavorable IPE results
- safety issues
  - — unresolved safety issue (USI) A-45
  - — other USIs and generic safety issues (GSIs)
  - — potential GSIs
- Commission's Safety Goals
- use of NUREG-1560
  - — plant inspection activities
  - — accident management
  - — maintenance rule
  - — risk-informed regulation
  - — areas for research
  - — miscellaneous issues
- probabilistic risk analysis (PRA)

In reading the following conclusions and observations, the reader should consider the scope and limitations of both the individual IPEs and the IPE Insights Program which are discussed in Section 1.3 of Chapter 1 of this report. Further, the conclusions and observations presented in this chapter are based on the original IPE submittals forwarded by the licensees to the U.S. Nuclear Regulatory Commission (NRC) (refer to Appendix A for sources of IPE information used in this report). Many licensees have updated their IPEs/PRAs since the original submittal and have reported bottom-line changes in core damage frequency (CDF) or release frequency; but in most cases, detailed information was not provided. Updated IPE information received by the NRC is listed in Appendix B.

8.1 GL 88-20 Objective (Including Improvement of Plant Safety)

GL 88-20 (Ref. 8.1) established that *"systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low cost improvements."* To realize this benefit, GL 88-20 requested that each licensee implement an IPE program to *"perform a systematic examination to identify any plant-specific vulnerabilities to severe accidents."* Further, GL 88-20 identified the following general purposes for performing an IPE:

(1) Develop an appreciation of severe accident behavior.

(2) Understand the most likely severe accident sequences that could occur at its plant.

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

(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.

In addition, GL 88-20 noted that the *"maximum benefit from the IPE would be realized if the licensee's staff were involved in all aspects of the* *examination to the degree that the knowledge gained from the examination becomes an integral part of plant procedures and training programs."*

Overall, the licensees have met the intent of GL 88-20, conducting systematic examinations of 108 nuclear power plant units. Through these examinations, the licensees have identified existing "vulnerabilities" (or "weaknesses") and have considered more than 500 plant improvements (including hardware modifications and procedural changes) that would potentially help to prevent or mitigate severe accidents. Although the quality and level of detail varied, each licensee elected to perform Level 1 and 2 PRAs for their IPE and completed the following tasks:

- developed a model of the severe accident behavior of their plant

- identified the most likely severe accident sequences based on their PRA model

- estimated the CDF and radionuclide release frequencies based on their PRA model

- identified the most likely features needed to prevent and mitigate core damage and radionuclide releases based on their PRA model

In addition, almost every licensee performed the majority of the analysis associated with the IPE, relying on contractors primarily for those specialized, technical areas where licensees typically do not have in-house expertise. In many cases, it appears that this technology was transferred from the contractors to the licensees.

As a result of the IPE program, it is clear that licensees have generally developed in-house capability with an increased understanding of severe accidents and PRA. This increased understanding has indirectly increased plant safety. Further, while only a small fraction of licensees actually identified vulnerabilities, nearly all licensees identified areas warranting investigation for potential improvements. (As noted above, licensees collectively identified more than 500 potential improvements.) Although it is not possible to quantify the overall, industry-wide impact on plant safety that resulted from these improvements, CDF reductions as high as two orders of magnitude have been reported for certain improvements. In this regard, the IPE program has served as a catalyst to further improve the overall safety of nuclear power plants: therefore, the GL 88-20 initiative has clearly proved to be a success.

8.2 Regulatory Follow-up Activities

The results of the IPE program, represented by the perspectives discussed in this report suggest the following areas and issues for which the NRC staff plans to pursue some type of follow-up activities (as described below):

- plant safety enhancements
- containment performance improvements
- additional review of IPEs/PRAs
- plants with relatively unfavorable IPE results

8.2.1 Plant Safety Enhancements

Chapters 2 and 9 of this report identify the plant improvements planned or implemented by the licensees. Many of these improvements, although identified on a plant-specific basis, have the potential for applicability to other plants and, therefore, the potential to further increase reactor safety, as demonstrated by the following significant examples:

- *Using firewater as an alternative injection source for boiling water reactors (BWRs).* Some licensees have implemented system and procedural modifications to permit use of the firewater system as a backup source of low-pressure coolant injection (LPCI) to the reactor vessel. The system would be used in the event of long-term accident sequences, primarily aiding the plant during station blackout conditions.

- *Replacing the seal material for Westinghouse reactor coolant pumps (RCPs).* Some

Westinghouse licensees have replaced the o-rings in the RCPs with a more temperature-resistant design that is less susceptible to leakage when seal cooling is lost. This modification reduces the potential for seal-related loss of coolant accidents (LOCAs) sequences that can lead to core damage if inventory makeup is not available.

- *Supplying the RCPs with alternative cooling.* At some plants, licensees have added backup cooling for RCPs (e.g., by powering charging pumps off the technical support diesel generators). This additional capability reduces the likelihood of sequences without RCP cooling, thereby reducing the threat from RCP seal LOCAs.

- *Adding procedures and portable fans for alternative room cooling upon loss of heating, ventilating, and air conditioning (HVAC).* Some licensees have addressed a susceptibility to loss of HVAC by adding procedures such as opening doors to provide ventilation upon indication of loss of HVAC. Similarly, some licensees have added backup room cooling capabilities by making portable fans available.

- *Revising containment spray initiation criteria in boiling water reactor (BWR) Mark I plants.* Some BWR Mark I licensees have indicated that they are considering revised criteria that would permit earlier initiation of containment spray to increase the likelihood of having a flooded drywell floor and the benefits associated with that condition.

- *Revising procedures to preclude RCP restart in pressurized water reactors (PWRs) under core damage conditions, thereby reducing the likelihood of induced steam generator tube rupture (SGTR).* Some PWR licensees have modified their emergency operating procedures to preclude RCP restart under conditions that could lead to an induced SGTR.

The staff (Office of Nuclear Reactor Regulation) plans to conduct follow-up activities with each licensee to monitor the status of plant improvements. For those planned improvements not yet implemented, the specific actions undertaken by the staff will depend on the actual improvements and their potential impact on plant safety (while considering the backfit rule).

8.2.2 Containment Performance Improvements

In SECY-87-297 (Ref. 8.2), the NRC staff presented its Containment Performance Improvement (CPI) Program as a plan for evaluating generic vulnerabilities related to severe accident containment. This was predicated on the conclusion that there are generic severe accident challenges to each light water reactor (LWR) containment type that should be assessed to determine the adequacy of existing Commission policy, and to ascertain whether additional regulatory guidance or requirements are warranted with regard to containment features.

The NRC's "Integration Plan for Closure of Severe Accident Issues," SECY-88-147 (Ref. 8.3), refers to the close integration of the CPI Program, and the IPE Program. Specifically, SECY-88-147 states that the CPI Program *"focuses on resolving hardware and procedural issues related to generic containment challenges,"* while the IPE program, *"will be examiningplant-specificaccidentvulnerabilitieswhich could threaten containment integrity."*

SECY-88-147 further clarified that the NRC staff would first focus its efforts on the BWR Mark I containments. Technical insights arising from the CPI program with regard to these containments were discussed in SECY-89-017, "Mark I Containment Performance Improvement Program" (Ref. 8.4), and summarized in Supplement 1 to GL 88-20 (Ref. 8.5).

The enclosure to Supplement 1 stated that the Commission expects that licensees of BWR Mark I plants to seriously consider the following three improvements during their IPEs:

(1) alternate water supply for drywell spray/vessel injection

(2) enhanced reactor pressure vessel (RPV) depressurization system reliability

(3) emergency procedures and training (implementation of Rev. 4 of the BWR Owner's Group Emergency Procedure Guidelines (BWROG EPGs))

Supplement 1 further stated that the staff planned to communicate directly with each Mark I licensee regarding the use of a hardened vent path. Specifically, the staff contended that licensees should consider the three improvements listed above, in addition to improvements stemming from the evaluation and implementation of the hardened vent.

In Supplement 3 to GL 88-20 (Ref. 8.6), the staff announced the completion of the CPI Program and forwarded the resulting insights to licensees of BWR Mark II and Mark III containments, as well as PWR containments. Supplement 3 noted that the technical information conveyed might be useful as these licensees conducted their IPEs for severe accidents vulnerabilities.

With regard to BWR Mark II containments, Supplement 3 stated that, for events where inadequate containment heat removal (CHR) could cause core degradation, licensees are expected to consider (as part of the IPE process) additional CHR capability using plant-specific hardware procedures. In addition, Mark II licensees are expected to consider the applicability of the Mark I improvements reported in Supplement 1 to GL 88-20.

Similarly, Supplement 3 stated that licensees with BWR Mark III containments are expected to evaluate (as part of the IPE) the vulnerability of their plants to interruption of power to the hydrogen igniters. In addition, Mark III licensees are expected to consider the applicability of the Mark I improvements reported in Supplement 1 to GL 88-20, as well as the additional CHR capabilities suggested for Mark II containments.

Further, Supplement 3 noted that the same situation could occur in PWR ice condenser containments as in Mark III containments with regard to hydrogen detonations following restoration of power. Therefore, licensees with ice condenser containments are expected to evaluate (as part of the IPE) their plants' vulnerability to interruption of power to the hydrogen igniters.

For PWR dry containments, Supplement 3 stated that licensees' IPEs are expected to evaluate the need for improvements (including accident management procedures) related to containment and equipment vulnerabilities to localized hydrogen combustion.

In NUREG-1335 (Ref. 8.7), the NRC requested that licensees use their IPE submittals to *"...develop strategies to minimize the challenges and the consequences such severe accident phenomena may pose to the containment integrity and to recognize the role of mitigation systems while awaiting their generic resolution."*

In the IPEs, most licensees responded to the CPI Program recommendations related to their type of containment. However, the licensees varied widely in their responses to the CPI recommendations. In several cases, the licensees indicated that they are considering the CPI recommendations, but did not identify them as commitments. In other cases, they are not addressed at all. Section 9.4 of this report provides an overview, grouped by containment type, of how the licensees' addressed the CPI concerns in their IPEs.

The CPI Program accomplished the objective of identifying generic issues for each containment type (briefly summarized above), and bringing those issues to the attention of the licensees. Many licensees examined their plants to assess the relevance of the generic issues in light of their plant's particular features and procedures. These licensees then indicated in their IPE submittals either that they had adopted modifications to hardware and/or procedures, or reasons why changes were not necessary. For some issues, such as alternative water supplies in BWRs or adoption of Revision 4 of the BWROG EPGs, many licensees indicated that they had already implemented the needed measures before conducting their IPEs. For other issues, such as enhanced RPV depressurization in BWRs or alternative igniter power supplies in BWR Mark III containments or PWR ice condensers, most of the affected licensees stated that no modifications were needed. However, there were a number of licensees that did not address some of the CPI items in their submittal.

It is important to note that, in general, the IPE process did not identify any new generic containment issues (i.e., issues not previously identified in other safety assessments or the CPI Program). However, some plant-specific containment-related insights derived from the IPEs may be worth pursuing (under the backfit rule) for possible applicability to other plants. Examples include the failure modes related to core debris contact with the containment boundary (found in some IPEs for PWR plants) and the containment bypass caused by isolation condenser failure (identified in one BWR IPE). The staff plans to conduct follow-up activities with each licensee on those CPI items not yet implemented and those not addressed in the IPE submittal. The specific actions undertaken by the staff will again depend on the actual CPI and its potential impact on plant safety.

8.2.3 Potential Future Uses of IPEs/PRAs

The staff reviewed each licensee's IPE submittal to determine if the submittal met the intent of GL 88-20. This review, therefore, focused on the objectives of the generic letter (whether the licensee's analysis was capable of identifying plant-specific vulnerabilities). The staff's review of the IPEs did not include identifying other potential uses and applications. In fact, it is stated in each staff evaluation report that *"It should be noted that the staff's review primarily focused on the licensee's ability to examine [plant] for severe accident vulnerabilities. Although certain aspects of the IPE were explored in more detail than others, the review is not intended to validate the accuracy of the licensee's detailed findings (or quantification estimates) that stemmed from the examination. Therefore, this SER does not constitute NRC approval or endorsement of any IPE material for purposes other than those associated with meeting the intent of GL 88-20."*

Licensees elected to perform PRAs for their IPEs; however, only a small fraction of these analyses were provided to the staff in the licensees' submittals. Therefore, to determine whether each IPE/PRA was capable of identifying vulnerabilities, the staff focused its review on the following aspects:

- "completeness" of the PRA and all necessary elements (e.g., whether the licensee performed an initiating event analysis and addressed support system failures in identifying potential initiators)

- "reasonableness" of the assumptions, boundary conditions, and data (as substantiated by supporting analyses and calculations)

- "reasonableness" of the results (given the design and operation of the plant)

The staff's review did not, however, include verifying or validating the licensees' analyses or the supporting calculations. For additional uses of the PRA, such verification and validation may be necessary for particular applications. Observations concerning the use of IPEs to support risk-informed regulatory changes and other applications are discussed in Section 8.5 of this report. In addition, Chapter 14 of this report defines the attributes for a state-of-the-art PRA and compares the IPE methods and models to those attributes. Perspectives on the IPE analyses are discussed at detail in Chapter 14 and summarized in Chapter 6.

8.2.4 Plants with Relatively Unfavorable IPE Results

There are a number of possible criteria by which the IPE results can be judged. These include:

- CDF
- Conditional containment failure probability (CCFP)
- Individual sequence frequency
- Comparison to safety goals

None of the plants reported results indicating undue risk to the public and the need for immediate regulatory action. However, in reviewing the results reported in the IPE submittals, the staff noted that about half of the plants had CDFs and/or CCFPs that were relatively high. In all, 15 PWR units have CDFs greater than 1E-4 per reactor year (ry) and 23 PWR units have CCFPs greater than 0.1. No BWR units have CDFs greater than 1E-4/ry, but 26 BWR units have CCFPs greater than 0.1. In addition, many plants have individual sequences with frequencies greater than 1E-5/ry, and two BWRs and twelve PWRs have results indicating the potential to approach the early fatality safety goal.

For these plants, the staff is planning follow-up activities to determine whether additional regulatory actions are warranted. In deciding what actions to take, the staff will examine underlying causes of the high values, that is, the staff will determine the necessary regulatory actions on the basis of whether the value is high because of analysis assumptions or because of the design and operation of the plant. For example, the relatively high CCFP (approximately 0.3) for the Palisades plant (a PWR with a large-volume containment) is attributed to plant-specific features. By contrast, the high CCFP reported for Waterford is driven largely by modeling assumptions. A high CCFP caused by plant-specific features may warrant attention, whereas a high CCFP driven by limiting or bounding modeling assumptions could require further evaluation regarding the validity of the assumptions. Given that the results do not indicate undue risk to the public or violation of the regulations, any actions ordered by the NRC must pass the cost-benefit test (currently $2000/person-rem) associated with the Backfit Rule. This fact supports the desire to examine sequences with frequencies above 1E-5/ry. For backfitting purposes, reductions in CDF of 1E-5/ry or greater are often considered to be significant and worthy of further analysis by the staff.

8.3 Safety Issues

In examining the IPEs, the staff gleaned perspectives regarding the following three categories of safety-related issues:

(1) How did licensees address and resolve USI A-45, "Shutdown Decay Heat Removal (DHR)," (GL 88-20 specifically requested that licensees address this issue).

(2) Did licensees' IPE submittals propose resolution of any other USIs or GSIs? If so, were those issues effectively resolved?

(3) Have any new potential safety issues been identified as a result of the IPEs?

These three categories are discussed in the following sections.

8.3.1 Unresolved Safety Issue A-45

GL 88-20 specifically requested that licensees address USI A-45 to determine whether the DHR function at operating plants is adequate, and whether cost-beneficial improvements could be identified. In its evaluation of this issue, the staff concluded that a generic resolution (e.g., a dedicated DHR system) is not cost effective, and this issue could only be resolved on a plant-specific basis. Because the IPEs provide an examination of the DHR systems and their importance, as well as the importance of systems performing other functions, the staff concluded that USI A-45 should be subsumed into the IPE Program.

In NUREG-1289 (Ref. 8.8), the regulatory and backfit analysis of USI A-45, the staff defined the components and systems related to the DHR function as those required to maintain primary and secondary coolant inventory control and to transfer heat from the reactor coolant system to the ultimate heat sink following reactor shutdown for transients (such as loss of feedwater and loss of offsite power) and small-break LOCAs. Support systems (such as standby service water and emergency AC power) required for various modes of DHR were also to be considered. The transition from reactor trip to hot shutdown (excluding the reflooding phase in a large LOCA), the transition from hot shutdown to cold shutdown, and the maintenance of cold shutdown conditions were considered part of the USI A-45 program. However, the USI A-45 program did not consider anticipated transients without scram (ATWS), interfacing system LOCAs (ISLOCAs), or large LOCAs. This definition of DHR sequences is an expanded version of the functional definition used in the IPEs and in this report. By contrast, the IPEs typically defined the DHR function as involving direct heat removal from the reactor coolant system (RCS) and removal of heat transported to the containment.

GL 88-20 directed licensees to pay particular attention to identifying DHR vulnerabilities as part of their IPEs. However, GL 88-20 did not provide any definition of a DHR vulnerability. Many of the licensees chose to use the same definition or criteria used to identify vulnerabilities associated with other functions modeled in the IPE. In addition, many chose to use the interim quantitative design objectives provided in NUREG-1289, which stated that a *"limited-scope, plant-specific PRA could demonstrate the adequacy of the existing decay heat removal function by documenting that its contribution to core damage frequency was relatively low, on the order of 1E-5 per reactor year or less"*. Little, if any, cost-beneficial modifications would be warranted if the core damage frequency was less than this value.

On the basis of experience gained from applying PRA to U.S. light water reactors for USI A-45 and other programs, the staff indicated in NUREG-1289 that the adequacy of the DHR function as calculated in the IPEs and in the IPEs of externally initiated events will fall into three broad categories. As a basis for this categorization, the staff has used the quantitative values presented in Table 8.1.

Table 8.1 DHR vulnerability classification criteria.

Category	DHR vulnerability	Criteria
1	The frequency of core damage due to failures of DHR function acceptably small or reducible to an acceptable level by simple improvements	less than 3E-5/ry
2	DHR performance characteristics intermediate between Categories 1 and 3	less than 3E-4/ry but greater than 3E-5/ry
3	Frequency of core damage so large that prompt action to reduce the probability of core damage related to DHR failure to an acceptable level is necessary	greater than 3E-4/ry

In addition to reviewing the DHR vulnerability assessments provided in the IPEs, the staff has compared the IPE results to the criteria in Table 8.1 on the basis of the DHR definitions provided in NUREG-1289 and the IPEs. None of the BWRs fall into the Category 3; that is, no prompt action would be required to fix any identified vulnerabilities. This is true whether the CDF is assessed according to the expanded DHR definition used in the USI A-45 program or the traditional definition used in PRAs, or even if the total CDF is compared to the criteria in Table 8.1. In addition, all of the BWRs fall into Category 1 when the CDF is assessed according to the traditional definition. A few plants fall into Category 2 when the CDF is assessed according to the expanded DHR definition used in the USI A-45 program. These plants have evaluated improvements as discussed below. Thus, the DHR functions in BWRs are adequate according to the criteria in Table 8.1, regardless of which DHR definition is used.

The above conclusion agrees with the evaluations conducted by BWR licensees. Only two licensees' IPE submittal explicitly identified vulnerabilities associated with DHR systems. The Millstone 1 IPE reported a vulnerability associated with failure of isolation condenser makeup that could apply to all BWRs in this plant group. The licensee proposed to resolve this vulnerability by procuring a portable diesel-fired pump and implementing procedures to use that pump to supply water to the isolation condenser. The FitzPatrick IPE identified a vulnerability (unique to that plant) where failure of either one of the 4.16 kV AC safety buses would result in loss of three of the four residual heat removal (RHR) loops directly or through loss of RHR service water (RHRSW). The licensee was considering installation of an RHRSW header cross-tie, and has implemented procedure modifications to allow alignment of firewater to the RHRSW system.

Loss of DHR is a dominant accident sequence for most BWRs, primarily because such a loss can have a negative impact on continued operation of the coolant injection systems. For some plants, high suppression pool temperatures resulting from a loss of DHR will cause a loss of adequate emergency core cooling system (ECCS) pump, net positive suction head. Containment venting or failure can result in harsh environments in areas surrounding the containment; such environments, in turn, can result in failure of coolant injection systems or required support system components. The dominant contributor to these sequences involves loss of support systems that leads to failure of both DHR and coolant injection systems. In addition, the credit given for DHR systems and alternative injection systems that can function following a loss of DHR also impacts the contribution from loss of DHR sequences.

Because of the importance of sequences involving a loss of DHR, many BWR licensees identified related plant improvements. DHR system improvements included hardware modifications to increase DHR system reliability, procedures and hardware to perform containment venting (credited in most IPEs), replacement of drywell spray valve motor operators with environmentally qualified operators, and additional guidance in emergency operating procedures for enhanced use of containment spray. Improvements to coolant injection systems were identified to ensure continued coolant injection following the loss of DHR. These improvements included establishing procedures for aligning low LPCI or core spray pumps to the condensate storage tank (CST) and replenishing the CST to prevent high-pressure coolant injection (HPCI) from switching suction to the suppression pool. Other improvements include increasing the reactor core isolation cooling (RCIC) turbine exhaust pressure trip setpoint, and preventing HPCI suction switchover to the suppression pool on a high torus level.

Many of the licensees also identified improvements related to support systems that impact the DHR function. These include procedures and hardware for aligning firewater for isolation condenser makeup, procedures for aligning alternative systems for cooling the RHR heat exchangers, revised isolation logic for plant service water and the instrument air system, procedures and hardware modifications for cross-tieing RHRSW trains, and revised procedures and training for losses of support systems.

As with BWRs (discussed above), none of the PWRs fall into Category 3 (see Table 8.1) when CDF is assessed according to the traditional PRA definition of DHR. Furthermore, only two units (Turkey Point 3&4) can be placed in Category 3 when CDF is assessed according to the expanded DHR definition used in the USI A-45 program. The high CDFs for these units primarily derive from RCP seal LOCAs rather than loss of the DHR systems. However, the licensee addressed the vulnerability at Turkey Point 3&4 and installed a hose connection between service water (SW) and charging pumps to increase the reliability for charging pump cooling and RCP seal integrity. This improvement reduced the estimated CDF to 1E-4/ry. Although many of the PWRs fall into Category 1 when the core damage frequency obtained using either of the DHR definitions, the majority of PWRs fall into Category 2. This suggests that immediate action is not required to reduce any DHR vulnerabilities in these PWRs. However, many of these plants did evaluate and implement improvements as discussed below.

As with the BWR licensees, some PWR licensees explicitly reported vulnerabilities associated with the DHR function. All of the plants reporting DHR vulnerabilities are Category 2 plants. Calvert Cliffs identified a plant-specific vulnerability of the turbine-driven auxiliary feedwater (AFW) pumps resulting from isolation of the steam supply to both pumps during maintenance. Manual isolation valves were added on both sides of the turbine-driven pump steam admission valves to allow for maintenance on one pump at a time.

The licensee for Kewaunee indicated that turbine-driven AFW pump reliability was a vulnerability that would be resolved by scheduled modifications. Similarly, Summer identified failure of turbine-driven AFW during a station blackout (SBO) as a vulnerability and indicated, that because of "conservatisms" in the IPE, the vulnerability would be considered in the development of the plant's Severe Accident Management Guidelines. Finally, Millstone 3 indicated that the failure of AFW and feed-and-bleed occurred in many important accident sequences and thus was a vulnerability. In response, the licensee prioritized operator training on the AFW system, recovery of main feedwater, and the primary feed-and-bleed procedure.

Accident sequences involving loss of AFW and feed-and-bleed were important for many PWRs. Loss of containment heat removal systems were generally not important in the PWR IPEs. The following primary factors contributed to the importance of DHR sequences in PWRs:

- failure of support systems (e.g., service water, component cooling water, AC and DC power, instrument air, and HVAC) required for AFW or feed-and bleed operation
- failure to provide long-term makeup to AFW
- failure of AFW and required feed-and-bleed components.

The guidance in NUREG-1335 indicated that PWRs without feed-and-bleed capability should particularly address the capability of the plant to recover from events involving loss of all feedwater. For many of these plants, the IPEs modeled the depressurization of the steam generator secondaries using the steam dump valves to a pressure where the condensate system could provide water for cooling. This action assumes that the condensate system is available for many loss of feedwater events. When modeled, this action reduced the impact from loss of DHR during all transients for plants with and without feed-and-bleed capability.

Many of the PWR licensees identified plant improvements related to the DHR function. These improvements include use of firewater for steam generator cooling, use of alternative water supplies to AFW, changes to feed-and-bleed procedures, equipment modifications to improve AFW reliability, and addition of a new CST.

In summary, the incorporation of the A-45 issue into the IPE program has provided a framework for resolution on a plant-specific basis. Each licensee chose to perform a PRA which by its nature provides a systematic evaluation of a plant's capability to cope with the requirement for decay heat removal. Thus, the IPEs provide the quantitative information necessary to categorize each plant in accordance with the criteria described in Table 8.1. In addition, the IPEs have identified plant-specific vulnerabilities and proposed plant improvements to address DHR and other vulnerabilities. The NRC staff has reviewed this information in the IPEs and have addressed the resolution of the A-45 issue on a plant-specific basis in the individual staff evaluation reports.

8.3.2 Other Unresolved and Generic Safety Issues

GL 88-20 states that *"if a utility (1) discovers a notable vulnerability during its IPE that is topically associated with any other USI or GSI and proposes measures to dispose of the specific safety issue or (2) concludes that no vulnerability exists at its plant that is topically associated with any USI or GSI, the staff* *will consider the USI or GSI resolved for a plant upon review and acceptance of the results of the IPE."* Furthermore, in NUREG-1335, the NRC specifically requested that licensees discuss the following aspects of the resolution of any other USI or GSI:

- ability of the proposed methodology to identify vulnerabilities associated the USI or GSI
- contribution of each USI or GSI to CDF or unusually poor containment performance (including sources of uncertainty)
- technical bases for resolving the issue

Several licensees elected to use their IPEs to resolve other USIs and GSIs. Example of these USEs and GSIs are provided in Table 8.2.

Table 8.2 Plant-specific examples of unresolved and generic safety issues proposed for resolution.

Plant	USI/GSI
ANO-1	GSI-23, RCP Seal Failures GSI-105, ISLOCA at LWRs
Catawba	GSI-105, ISLOCA at LWRs
Davis-Besse	GSI-23, RCP Seal Failures GSI-105, ISLOCA at LWRs GSI-128, Electric Power Reliability GSI-143, Availability of Chilled Water System and Room Cooling GSI-153, Loss of Essential Service Water (ESW) at LWRs USI-A-17, Systems Interaction GSI-77, Flooding of Safety Equipment Compartments by Backflow Through Floor Drains
FitzPatrick	USI-A-47, Safety Implications of Control System Failures
Hope Creek	GSI-105, ISLOCA at LWRs
Indian Point 3	GSI-23, RCP Seal Failures
McGuire 1&2	GSI-105, ISLOCA at LWRs GSI-130, SW Pumps for Multi-Unit Sites
Nine Mile 1	GSI-A-30, Adequacy of Safety-Related DC Power Supply
North Anna 1&2	GSI-23, RCP Seal Failures
Oconee 1,2, &3	GSI-23, RCP Seal Failures GSI-105, ISLOCA at LWRs GSI-153, Loss of ESW
Oyster Creek	GSI-101, BWR Water Level Redundancy GSI-105, ISLOCA at LWRs USI-A-47, Safety Implications of Control System Failures USI-A-17, Systems Interactions
Point Beach 1&2	GSI-23, RCP Seal Failures
River Bend	GSI-23, RCP Seal Failures GSI-105, ISLOCA at LWRs USI-A-17, Systems Interactions
San Onofre 2&3	USI-A-47, Safety Implications of Control System Failures
South Texas 1&2	GSI-15, Radiation Effects on RPV Supports GSI-23, RCP Seal Failures GSI-24, Automatic ECCS GSI-83, Control Room Habitability
Surry 1&2	GSI-23, RCP Seal Failures
TMI 1	GSI-23, RCP Seal Failures

A few utilities have used the IPE process to provide a framework for resolution of safety issues on a plant-specific basis. The IPEs have identified plant-specific vulnerabilities and proposed plant improvements associated with these issues. The NRC staff has reviewed this information in the IPEs and have addressed the resolution of these issues on a plant-specific basis in the individual staff evaluation reports and in a separate technical report (to be issued). Although most licensees chose not to use their IPEs to resolve any USIs or GSIs, among those who did, many selected GSIs 23 and 105. The following paragraphs discuss the background of these two generic issues, as well as the approaches used by the licensees to resolve the issues, and examples of the improvements proposed by the licensees on the basis of their findings.

GSI-23, "Reactor Coolant Pump Seal Failures" —

A LOCA can occur if leakage through the RCP seals exceeds the capacity of the normal makeup systems. RCP seals limit the leakage of reactor coolant along the pump shaft, directing the majority of this flow back to the chemical and volume control system (CVCS), with the remainder being directed to the reactor coolant drain tanks. The RCPs use a series of primary and secondary seals to limit the reactor coolant leakage to containment. Therefore, these seals are part of the RCS pressure boundary.

Certain identified common mode vulnerabilities could result in an RCP seal LOCA, rendering the mitigating systems inoperable, thereby leading to core damage. One such scenario involves the complete loss of the component cooling water (CCW) system, which provides cooling water to the seal thermal barrier heat exchanger, among other systems. In some plants, the CCW system also cools the reactor coolant makeup system pumps, CVCS charging pumps, or high-pressure safety injection (HPSI) pumps that supply RCP seal injection flow. Therefore, for those plants, complete loss of CCW could result in the equivalent of a small-break LOCA caused by seal degradation, with no HPSI pumps available for emergency core cooling. This sequence of events could lead to core damage and could be initiated by the loss of all AC power, as in a SBO.

To resolve GSI-23, licensees performed analyses to ensure adequate core cooling during loss of seal cooling events, including SBO, loss of SW, and loss of CCW. Approaches examined by the licensees included use of a dedicated seal cooling system, alternative seal cooling, an alternative AC power source, and/or a core cooling capability using the fire water system. Some licensees also considered using the high-temperature o-rings recommended by Westinghouse.

For example, in the case of Surry, the licensee argued that GSI-23 should be considered resolved, because the plant's CDF associated with RCP seal failure SBO and non-SBO conditions are lower than the industry-average CDF identified in NUREG-1401 (Ref. 8.9). Virginia Power also asserted that planned improvements in Surry's ability to cope with an SBO would decrease the overall CDF associated with RCP seal failures. In addition, Virginia Power provided instrumentation and procedures to detect seal failures and mitigate their consequences.

GSI-105, "Interfacing Systems LOCA in LWRs" —

An interfacing systems ISLOCA is defined as a breach of the pressure boundary between the RCS and any one of several low-pressure systems. Breaching the pressure boundary consists of the failure or improper operation of two or more pressure-isolation valves (PIVs) that comprise the boundary. The PIVs are typically check valves and/or motor-operated valves. If high-pressure coolant enters the relatively low-pressure interfacing system, the possibility exists of overpressurizing and rupturing the interfacing system; such a rupture usually extends outside containment. If the break is not isolated, core damage is likely, with the subsequent release of radioactive material bypassing containment.

Several NRC-sponsored studies (Ref. 8.10) have analyzed PWR susceptibility to ISLOCA without revealing generic problems. The major insight from the latest studies is that ISLOCA problems, if they exist, are highly plant-specific. Because of that insight, and because the calculated risk is low at the plants analyzed, the Commission concluded (Ref. 8.11) that the most effective course of action for resolution of GI-105 involved plant-specific analyses at each PWR.

To resolve GSI-105 within the scope of the IPEs, the licensees examined major systems of concern, using a screening process to eliminate systems posing insignificant threat from ISLOCA. As a result, the licensees identified the valves in the flowpaths that would pose a threat from ISLOCA and estimated the CDF for the plausible ISLOCA scenarios. For example, the CDF at McGuire Nuclear Station has been demonstrated to be sufficiently low, on the basis of credible analyses, to consider GSI-105 resolved at McGuire. The low CDF was achieved through a combination of good practices and design features. These include independent verification of valve position, valve position alarms, double checks in procedures, and removal of power to motor-operated PIVs before power operation.

8.3.3 Potential Generic Safety Issues

This report identifies issues that play a key role in both prevention and mitigation of severe accidents. The following are some significant examples of issues resulting from the IPE reviews which have the potential for generic applicability and are candidates for further investigation:

- For some PWRs, the IPEs indicated that transient sequences involving RCP seal LOCAs are the largest contributors to CDF. For other PWRs, the IPEs indicated that these sequences have a negligible contribution either because plant design characteristics minimize the potential for unmitigated RCP seal LOCAs, or because the IPE modeling of seal LOCAs predicted lower probabilities of seal leaks or smaller leak rates. Thus, RCP seal LOCAs remain an issue for many, but not all, PWRs. This issue is currently under evaluation by the NRC's Office of Nuclear Regulatory Research. Resolution of this issue will involve careful evaluation of information from the individual IPEs regarding the plants at which RCP seal LOCA is and is not a contributor, the reasons underlying that assessment, and the types of improvements proposed.

- Because of ATWS concerns, the most recent BWR emergency procedures direct the operators to inhibit the automatic depressurization system, with subsequent manual depressurization performed if required. Some licensees have indicated that this inhibition results in a higher estimated probability of the operator failing to depressurize the vessel, resulting in dominant accident sequences at high pressure. In addition, because of the consequence of failing to depressurize (although the operator may have a low probability of failure), there are many BWRs still have a relatively high CDF contribution from transient sequences at high pressure. The issue of depressurization in BWRs may, therefore, be worthy of further investigation.

- Some licensees indicated that accident environments for some sequences (e.g., those producing high containment pressure or temperature) can affect equipment operability in a manner that significantly increases the plant CDF. Such effects are plant specific, but it is not clear from the submittals whether all licensees have adequately addressed the effects of adverse conditions on equipment operability. Some licensees appear to have dismissed this issue without adequate evaluation, while others have included failures for cases that might be shown to be non-threatening with further analysis.

- Some BWR IPEs identified a potential containment failure mechanism involving prolonged discharge through the safety relief valves into a hot suppression pool. However, the majority of the licensees did not discuss the possibility or implications of such a blowdown to a hot pool. While substantial information exists for design-basis accident hydrodynamic loads, there is considerable uncertainty regarding these loads under severe accident conditions.

- Most PWR IPEs assigned a small probability to induced-SGTRs (ISGTRs). A few IPEs, however, considered ISGTR significant because RCP restart under degraded core conditions is considered possible and this would lead to a high probability of ISGTR. Most licensees did not consider the effect of RCP restart, or if they did, they still assigned a small probability to ISGTR on the basis of the expected limited duration of RCP operation. This variable treatment in the IPEs indicates the large uncertainty associated with the ISGTR issue. It should be noted that this issue is being addressed under the proposed rule making on steam generator tube integrity.

- One BWR 2 Mark I IPE identified an issue involving containment bypass as a result of high-temperature creep rupture failure in the isolation condenser tubes. That particular IPE estimated that this failure mode has a relatively low frequency. Nevertheless, failure of the isolation condenser tubes could be a potential safety issue for several other early BWRs that are equipped with isolation condensers but failed to address this scenario in their IPEs.

- Drywell shell melt-through plays a significant role in the early failure probability reported in most Mark I IPEs. Although the reactor safety community appears to be approaching a consensus on when and under what conditions this failure mechanism is important, shell melt-through may still be a safety issue for some of these plants given their mix of accident sequences and containment conditions (i.e., insufficient water on the drywell floor).

For some of these issues, the technical analysis (or phenomena) are generally understood and, in general, there is agreement among the experts. In some cases, such as drywell shell melt-through in BWR Mark Is and direct containment heating in PWR containments, agreement in the expert community has only been recently reached, (i.e., since the completion of the IPE analysis). For other issues, the technical analyses may not be well understood even now, and there may be disagreement among the experts.

For the type of issues discussed above, the NRC is planning follow-up activities to determine whether any actions (e.g., evaluate for continued or further research) are warranted.

8.4 Commission's Safety Goals

The NRC established two quantitative health objectives (QHOs) for the prompt fatality risk and cancer fatality risk to an individual near a nuclear power plant. In addition, the NRC proposed subsidiary objectives related to CDF and CCFP. The IPE results are directly compared against the subsidiary objectives in Section 7.1 and Chapter 15. However, most IPE results cannot directly be compared to the QHOs because licensees, in response to GL 88-20, were not requested to calculate offsite health effects. However, it is possible to link Level 1 and Level 2 results to the safety goals by using surrogate indicators (such as the frequencies of early containment failure and bypass). The results of using Level 1 and Level 2 surrogates are summarized in Section 7.1 and described in detail in Chapter 15.

On the basis of the frequencies of early containment failure and bypass reported in the IPEs, a fraction of the plants have the potential for early fatality risk levels that could approach the QHOs. This subset of plants was further examined using the frequencies of source terms (from the IPEs) with relatively high release fractions (>0.03, I, Cs. Te) adjusted for population. On the basis of this further screening, two BWRs and 12 PWRs remain with the potential for early fatality risk levels that could approach the QHOs.

8.5 Use of NUREG-1560

This report provides information concerning the IPE results and perspectives that can be used by both the NRC staff and the industry to help guide and focus a wide spectrum of activities. This section discusses the use of this information in a variety of activities such as accident management; maintenance rule implementation; risk-informed regulation, and several miscellaneous activities including examination and prioritization of plant improvements, development of risk management programs, development of an enhanced understanding of plant performance, and evaluation of the effectiveness of regulatory activities.

8.5.1 Plant Inspection Activities

This report provides a wealth of information that can be used to augment and support plant inspections. PRA perspectives can help to optimize and prioritize inspection resources by focusing on the high-risk aspects of a plant. Using the results from individual IPEs, the NRC, the industry, and individual licensees can identify the plant-specific systems and components and operator actions that have a significant impact on preventing or mitigating core damage. In addition, the IPE results make it possible to identify and understand the factors underlying the significance of those systems/components and operation actions. That is, it is important to know whether the system, component, or action is critical because its reliability is extremely poor, or because its failure (although rare) has severe consequences. Knowledge of these issues serves as an important aid in inspection planning.

This report also provides information on those systems, components, and actions that are safety significant in preventing and mitigating core damage for the different reactor and containment classes. This information can be used to gain perspectives on important systems, components, and actions for a given plant. For example, SBO sequences are important at almost all of the plants. Systems, components, and human actions that are important in SBO sequences include the following examples:

- turbine-driven cooling systems, such as auxiliary feedwater systems in PWRs and RCIC systems in BWRs
- diesel generators
- operator actions to shed DC loads and extend battery life

Chapters 3 through 5 summarize the important contributors to risk, while Chapters 10 through 13 provide related detail.

In many cases, the information provided about sequences is straightforward to interpret and apply in an inspection situation. For the dominant accident sequences, inspectors seek to understand the accident process and the contributing systems, components, and operator activities. In addition, they subsequently consider operator training, clarity of procedures, and other factors associated with the sequences of interest. For example, in looking at SBO for one licensee, the estimated SBO-related CDF is high (~6E-5/ry and which is above the SBO rule goal of 1E-5/ry). The ability to shed the DC load and extend battery life is a critical aspect in coping with SBO at this plant. Consequently, the key issues to be investigated during inspection activities include how quickly and easily load shedding can be performed, what procedures and training are provided, how knowledgeable the operators are, and how accurate the technical analysis is in predicting the extended battery life.

The use of front-line systems, as described in the Final Safety Analysis Report to mitigate accidents is clear and requires little attention other than to ensure that the systems function properly. However, the majority of the licensees credited the use of alternative systems and equipment beyond their original design function.

These credits have generally included the following non-safety systems and/or non-technical specification systems:

- non-safety systems for coolant injection (such as firewater, control rod drive, service water crosstie)
- alternative AC or DC power sources (such as gas turbine generators, standby shutdown facility diesel generators, or portable power supplies to battery chargers)
- crossties among systems and units (such as AC power, CCW, feedwater)

- room cooling (by opening doors and using portable fans)

While such use is expected and even encouraged in order to achieve a realistic PRA, licensees' training and testing for the systems/equipment associated with these "alternative" functions is sometimes questionable. In many cases, this credit has a significant impact on the results; that is, if the credit were removed, the estimated CDF would substantially increase and the dominant accident sequences and contributors would change.

In general, on the basis of the information presented in the submittals, it appears that the above systems can meet the functional requirements of the accident sequences. However, it is not clear that the licensees always performed all of the necessary "analyses" to credit the use of these systems. For example, for firewater use, it is not clear that the licensees measured the length of the firewater hose to ensure that it has sufficient length to connect, that it has the proper connection, and that sufficient time is available to complete the connection. For use of alternative AC or DC power sources, it is not clear that the sources are always available (e.g., that a portable battery is available and reliable at any given time). For room cooling, it is not clear that a portable fan is necessarily available and functioning under all necessary conditions (e.g., SBO). Furthermore, it is not clear that the training provided for the use of these options ensures as high a degree of operator reliability as is reflected by the licensees. That is, it is not always clear that the operators have sufficient understanding of how to use these alternative systems in different accident situations.

Chapter 5 and 13 discuss important operator actions identified in the IPEs. These actions can be considered during inspection activities related to training and procedures. For example, the operator action to manually depressurize the vessel given failure of high-pressure injection systems was identified as an important action in BWRs. The importance of this action is at least because of the fact that most emergency operating procedures direct operators to initially inhibit the automatic depressurization system, thus requiring the operators to manually depressurize when low-pressure systems are needed.

8.5.2 Accident Management

One element of the "Integration Plan for Closure of Severe Accident Issues" described in SECY-88-147 is accident management. SECY-88-147 broadly defines accident management as measures taken by the plant staff to (1) prevent core damage, (2) terminate the progress of core damage if it begins and maintain the core within the reactor vessel, (3) failing that, maintain containment integrity as long as possible, and finally (4) minimize the consequences of offsite releases. In addition, accident management includes certain measures taken before an event occurs (e.g., improved training for severe accidents, and hardware or procedural modifications) to facilitate implementation of accident management strategies.

GL 88-20 emphasized that the results of licensees' IPEs are an essential ingredient in developing a severe accident management program for individual plants. This is because the conclusions drawn from *"the IPE for severe accident vulnerabilities (1) will depend on the credit taken for survivability of equipment in a severe accident environment, and (2) will either depend on operators taking beneficial actions during or prior to the onset of core damage or depend on the operators not taking specific actions that would have adverse effects."* GL 88-20 further states that licensees are not required to develop an accident management plan as an integrated part of their IPE (i.e., they can defer the development of such a comprehensive plan). Nonetheless, GL 88-20 clearly encourages licensees not to defer implementing operator or plant personnel actions, identified in the course of conducting the IPE, that can substantially reduce the risk from severe accidents and that they believe should immediately be implemented in the form of formal guidance. Furthermore, Supplement 2 to GL 88-20, "Accident Management Strategies for Consideration in the IPE Process" (Ref. 8.12), lists accident management strategies that have significant potential for recovering from a wide variety of accident scenarios. These strategies are grouped into three broad categories:

- conserving and/or replenishing limited resources during the course of an accident
- using plant systems and components for innovative applications during an accident
- defeating appropriate interlocks and overriding component protective trips in emergency situations

The purpose of Supplement 2 is to *"forward these strategies to industry so that licensees can evaluate these or similar strategies for applicability or effectiveness at each of their plants as part of conducting the individual plant examination."* Finally, in the "IPE Submittal Guidance" (NUREG-1335), the NRC requested that utilities document any worthwhile strategies that were developed as part of the IPE process to further prevent or mitigate the detrimental effects of severe accidents.

A review of the IPE submittals indicates that, in general, utilities have responded appropriately to the requests in GL 88-20 and NUREG-1335 related to accident management. The submittals indicated that, at most plants, many of the candidate accident management strategies from Supplement 2 to GL 88-20 already existed or were added to operator guidance during the conduct of the IPEs. Where additional strategies were found to be important on the basis of IPE results, licensees usually documented them in their list of improvements as having been implemented. In addition, in many IPE submittals, licensees indicated that they have identified issues involving potential further strategies that they will deal with in their accident management programs. Examples of these strategies include the possibility of ex-vessel flooding to cool core debris in-vessel in some PWRs, or the revision of initiation criteria for containment spray or containment venting in BWRs.

Clearly, licensees have recognized that their IPEs provide a framework for developing their comprehensive and structured accident management plans. That framework can be used to identify where strategies are most needed, as well as to evaluate the efficacy of particular strategies. Future updates with strategies from finalized accident management plans will further improve the quality of the IPEs (PRAs). For example, many PWR IPEs/PRAs use somewhat unrealistic modeling because they fail to consider operator actions during accident progression; this inadequacy would be remedied by the eventual inclusion of the actions developed from the licensee's accident management program.

8.5.3 Maintenance Rule

The IPEs/PRAs can be used (by both NRC and the licensees) to support implementation of the maintenance rule, which requires that licensees establish performance goals commensurate with safety in the following areas:

- determination of the risk significance for all plant systems and functions
- establishment of performance criteria to monitor all selected systems
- control and monitoring of the risk of simultaneous out-of-service equipment

When explicitly modeled in the IPEs, it is possible to calculate the risk importance of systems, equipment, and human actions for prevention and mitigation of specific core damage scenarios. IPEs have demonstrated that some nonsafety systems have the potential to significantly reduce CDF (e.g., CCW crosstie between units) as discussed in Section 8.5.1. Nonsafety systems are systematically modeled in IPEs and, therefore, are directly incorporated into the PRA calculations that can be used to support the determination of risk significant systems, equipment, and functions for the maintenance rule implementation.

Plant-specific data used in the IPEs to calculate the unavailability and failure rate of a component provides a basis for establishing appropriate performance criteria. The component importance measures calculated on the basis of IPE models could provide the basis for determining the response time and calibration parameters for the statistical monitoring techniques that are required for this task.

Some of the IPEs can be used to improve configuration management to calculate the CDF increase that results from multiple equipment being taken out of service simultaneously. The combination of equipment that either causes the plant to enter a limiting condition of operation, or could significantly affect the CDF could be identified and avoided. Control and monitoring of the out-of-service configuration risk relies heavily on the IPE models.

8.5.4 Risk-Informed Regulation

The staff is currently preparing a set of regulatory guides (RGs) and standard review plans (SRPs) that describe the NRC's approach to risk-informed regulation and provide implementation guidance to the utilities. NUREG-1560 provides a variety of information that is being used to support risk-informed regulation. As the IPEs represent a substantial investment by the licensees, the potential use of IPEs in risk-informed regulation is of considerable importance. Chapter 14 describes the characteristics for a "benchmark" PRA; the staff is using this information as it develops guidance for the use of PRA in risk-informed regulation. Chapter 14 also compares the IPE methods and models against the characteristics for the PRA defined. Areas where the analysis is incomplete or simplified, and areas where the analysis exhibits the strength of acceptable methods and appropriate implementation are also identified. Licensees can use the information in Chapter 14 to identify areas that may need additional work. For example, this report identifies a weakness in some IPEs in over-reliance on generic data. This area may need to be examined in greater detail depending on the regulatory application, as discussed in the Rgs and SRPs (Ref. 8.13).

Chapters 3, 4, 11, and 12 of this report discuss the systems, components, and human actions that are important in preventing and mitigating core damage, on a reactor- and containment-type basis. This information can be used to support pilot applications and studies, providing perspectives as to where to focus concerns. The relative importance of systems, components, and human actions in the IPEs indicates areas in which plant changes can significantly reduce risk and areas where relaxations can be made without significantly increasing risk. This information can be used to support estimates of the change in plant risk associated with improvements or relaxations. For example, the South Texas Project IPE/PRA results were used as the basis for several licensee requests for permanent technical specification relaxations.

The discussions in Chapters 3, 4, 11, and 12 on key contributors to CDF and containment performance also provide information that is useful when IPEs/PRAs are used for risk-informed regulation. These chapters cover the key CDF and containment performance issues arising for the various plant groups. If the submittal results differ significantly from the perspectives provided in Chapters 3, 4, 11 or 12, specific plant features responsible for the deviations should be identified. For example, if the PRA results for a Westinghouse 4-loop plant indicate a small contribution from seal LOCAs, then it should be determined that there is adequate cause for the low contribution because of features such as backup cooling for the seals.

8.5.5 Areas for Research

This report indicates numerous areas in which further research would be useful and should be considered regarding both severe accident behavior and analytical techniques. This need, combined with the number of affected plants and the impact of additional knowledge (or enhanced methodology) on improving the prediction of plant risk or core damage frequency and on reducing or more clearly defining the uncertainties, will help to prioritize the research and allocate resources.

It must be recognized that substantial advances in areas important for the risk analysis of nuclear power plants have been made in the time since the IPEs were completed. For example, in the area of severe accident phenomena, an improved understanding has been achieved for drywell shell melt-through in BWR Mark Is and for direct containment heating (DCH) in PWR containments. Such advances in the state-of-knowledge in this and other areas important for PRA analysis will be taken into account when looking to the IPEs as a source of information for identifying and prioritizing research.

The following areas are examples of issues that should be considered for research:

- Human actions are important contributors to plant CDFs in the IPEs, with correct operator actions often significantly reducing the CDFs. However, there are considerable uncertainties associated with determining human error probabilities for operator actions. As a result improved modeling of human actions would significantly improve the understanding of the risk associated with operating nuclear power plants. The NRC's Office of Nuclear Regulatory Research is currently conducting a research program to develop an improved method to address human reliability analysis concerns. Perspectives from the IPEs are being incorporated into this human reliability analysis program.

- Many of the IPEs included core damage prevention strategies (e.g., RCS depressurization without high-pressure injection available during LOCAs so that low-pressure injection can be used) that were not previously analyzed. These strategies could have significant benefit at some plants, but further evaluation is needed to determine their generic applicability.

- The IPEs estimated the CDFs from full power internal events, but did not provide estimates for other operating modes (i.e., low power and shutdown). However, studies to date that have estimated the risk during low power/shutdown modes have required significant effort. Streamlined approaches that draw from the insights of the initial low power/shutdown studies would be useful as a tool for performing a more comprehensive CDF estimate than that provided by the IPEs. While not required, such estimates may be needed for some aspects of risk-informed regulation.

- Implementation of the station blackout rule has reduced the SBO frequency at many plants. Nevertheless, SBO remains a dominant contributor to risk at many plants, even after implementation of the rule. SBO frequencies are above 1E-5/ry for many plants. Given the results, further investigation of strategies for reducing SBO frequencies appears warranted. Such investigation could include an evaluation of the efficacy of various coping strategies implemented by the licensees to determine which strategies achieved the greatest risk reduction.

- For a number of Westinghouse PWRs, sequences involving failure of RCP seals were dominant contributors to risk. However, there was substantial variability in seal LOCA contribution to core damage due in part to differences in modeling assumptions. This is an area that has been debated since the NUREG-1150 studies. Unfortunately, little new information has been added since that time. Since this issue is likely to have an effect on implementation of risk-informed regulations at Westinghouse PWRs, further investigation into mitigation strategies or improved models would be useful.

- In most cases, ATWS was not a dominant contributor to the CDF. However, there was a great variability in the modeling of ATWS events for both BWRs and PWRs. For example, the importance of an unfavorable moderator temperature coefficient at PWRs or the need for standby liquid control at BWRs was treated differently from plant to plant. The fact that differences in ATWS behavior are being predicted, often with no apparent basis, leads to concerns about operator performance during these events. ATWS risk may be higher than reported if operator responses turn out to be incorrect. Further justification for current predictions of ATWS behavior would help alleviate this concern.

- Steam generator tube rupture (SGTR) represents a small contribution to CDF for most PWRs. However, because of the associated containment bypass, the contributions to CCFP and large source terms can be substantial. Current treatments of SGTR are very uncertain and reflect a combination of optimistic (e.g., only one tube fails) and pessimistic (e.g., steam generator safety valves stick open) assumptions. SGTR sequences are important to risk-informed regulation because they can be the dominant contributor when comparing to the safety goals. Therefore, development of more accurate models for estimating the risk from SGTR appears warranted.

- A number of issues related to containment venting and its variable treatment in the BWR IPEs may deserve further evaluation. All BWR Mark I plants have installed, or have committed to install, a hardened vent; however, a number of IPEs from these plants stated that only negligible benefits result from such a capability, while other Mark I IPEs ascribed substantial benefit to the ability to vent through a hardened path. In a few cases, the IPE submittals suggested that the basis for venting procedures should be changed to include containment temperature as well as the current sole reliance on containment pressure, since elevated temperatures can seriously degrade containment capability. In addition, several BWR IPEs called for reevaluation of the containment flooding contingency, found in current EPGs, which involves drywell venting and could lead to significant releases. Finally, one BWR 6 Mark III IPE identified RCS venting through the main stream isolation valves as a very important early release mechanism, while no such venting is discussed in the other Mark III IPE submittals.

- Our understanding of drywell shell melt-through in BWR Mark Is has improved substantially since the IPE process started. For the case of a dry floor, meltthrough is to be expected, while with a foot or so of water on the floor, meltthrough will be prevented, or at least the fission product releases will be substantially mitigated. Unfortunately, most plants do not have the means to introduce sufficient water into the drywell before vessel breach in the accident sequences of interest. Therefore, CCFPs for these plants will remain high for this cause. Further examination of mitigation strategies for the risk dominant sequences appears warranted and such strategies are being investigated by some licensees.

- The uniformly low early containment failure probabilities reported in the IPEs for plants with ice condenser containments are surprising, especially when compared with the results reported for large dry containments. No single reason for the low failure probabilities is apparent from the IPE submittals; however, modeling assumptions clearly play a role. Further evaluation of the reasons for the low probabilities reported in these IPEs appears warranted. This includes evaluation of assumptions concerning ice condenser availability and its capability to absorb energy.

- A number of phenomenological issues are addressed with substantially different approaches in the IPEs of similar plants. For instance, some IPE submittals discussed exvessel fuel-coolant interactions (FCIs) as insignificant for containment failure, while other submittals claim that FCIs play an important role. Debris bed coolability also received disparate treatment in the IPEs. Some submittals indicated that coolability is always assumed when water is present, while others used stricter criteria involving the depth of debris and the amount of water. These issues are already the subject of ongoing research at the NRC and elsewhere.

8.5.6 Miscellaneous Issues

The information contained in this report can support other possible uses by both the NRC and the licensees. These uses include such items as examination and prioritization of plant improvements, development of risk management programs, development of enhanced understanding concerning plant performance, and evaluation of the effectiveness of regulatory activities. The following paragraphs discuss these uses in greater detail.

During their IPEs, licensees identified a variety of plant enhancements to improve reactor safety. Many of these improvements are potentially applicable to other plants. For example, some licensees with BWRs made enhancements to improve the ability to use firewater as an alternative injection source. Other licensees with BWRs might also realize significant improvements in plant safety with similar modifications. Section 8.2.1 discusses some other issues with possible generic applicability .

Some licensees have developed risk management programs (e.g., configuration control) to assess and control a variety of risks, including but not necessarily limited to, severe accident risks. Information provided in this report can be used to help licensees assess the completeness of their risk management programs with respect to severe accidents that are within the scope of the IPE program. For example, licensees can assess whether any accident initiators and sequences have been identified in other IPEs that may also apply to their plant. Further, many suggested plant improvements may be identified to help control the risks in areas where improvements may be warranted. For example, if a licensee wishes to reduce the risks from SBO accidents, the licensees as a whole have identified many different alternatives for reducing such risks.

This report can also provide insights as to how plants may perform in accident situations. A wide range of outcomes is possible for some types of sequences, as discussed in this report. For example, during an ATWS event at a BWR, different outcomes can result from different combinations of system pressure, standby liquid control injection timing, and operator actions to control level and those outcomes involve uncertainties for many scenarios. The containment performance portion of this report, for example, discusses many different phenomena, such as hydrogen burns and DCH. By making operators aware of all of the various possibilities, this report helps to ensure that they will be better prepared to deal with real events that may occur.

By looking at the results for specific classes of plants or the industry as a whole, insights can be gained regarding the effectiveness of certain regulations. For example, the impact of the SBO rule on plant safety can be inferred from the IPE results. Results reported for 15 plant units indicated an average reduction in CDF of 2E-5/ry for these units as a result of SBO rule modifications. Most of the plants that accounted for SBO rule changes now have SBO CDFs below the goal of 1E-5/ry, but a few plants have SBO CDFs above the goal even after implementing the changes. Insights such as these can be obtained for other rules given further analysis of the results.

8.6 Probabilistic Risk Analysis

This report identifies the dominant accident sequences, containment failure modes, plant features, and human actions that play a major role in preventing or contributing to core damage, containment, failure and radionuclide release (as reported in the IPEs/PRAs). The models used in the licensees' PRAs and the results of those analyses (as reported to the staff) can be used to support PRA activities. However, variability in the results are observed for plants with similar design. In addition, although much of the variability is expected (as a result of differences in the plant design and operation), some variability is not supported by the differences in plant characteristics. Therefore, before these results can be used, it is essential to develop an understanding of the bases for this variability. The information presented in Chapters 3 through 5 and 10 through 13 indicates where much of the variability exists (e.g., contributors and core damage frequency) and what has caused the variability. Whether caused by plant differences or modeling assumptions the variability in the IPE results generally falls into one of the following categories:

- differences in definition of scope or boundary conditions (e.g., completeness of initiating events, or accounting for support system dependencies such as instrument air or HVAC)

- differences in plant (system/equipment) response to accident conditions (e.g., effect of containment conditions on injection pumps, ability of power-operated relief valves to reclose, or potential for RCP seal LOCA if pumps continue to run without seal cooling)

- difference in definition of terms (e.g., differences in the definitions of accident sequences, differences in the criteria used to define the onset of core damage)

Standardization of these items would ensure that the variability is driven by plant characteristics rather than "subjective judgements" that are not necessarily supported by plant-specific analyses. However, even allowing for the variability, the models and results presented in the IPEs can provide useful support for ongoing and future PRA activities:

- The IPE results can be used to ensure completeness and consistency among PRAs. For example, new failure modes were reported in some IPEs, but not in other IPEs for plants with similar design features. This may indicate that the other PRA models may need to include these new failure modes.

- Some IPEs used new PRA models that have distinct advantages over previously available models. For example, it is now possible to directly couple the Level 1 and 2 analyses such that core damage accident sequences are directly allocated to containment failure modes. This eliminates the need for binning into plant damage states, and the approximations that can accompany this process.

REFERENCES FOR CHAPTER 8

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8.3 USNRC, "Integration Plan for Closure of Severe Accident Issues," SECY-88-147, May 25, 1988.

8.4 USNRC, "Mark I Containment Performance Improvement Program," SECY-89-017, January 23, 1989.

8.5 USNRC, "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities," Generic Letter 88-20, Supplement 1, August 29, 1989.

8.6 USNRC, "Completion of the Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities," Generic Letter 88-20, Supplement 3, July 6, 1991.

8.7 USNRC, "Individual Plant Examination Submittal Guidance," NUREG-1335, August 1989.

8.8 USNRC, "Regulatory and Backfit Analysis: Unresolved Safety Issue A-45, Shutdown Decay Heat Removal Requirements," NUREG-1289, November 1988.

8.9 S.K. Shaukat, J.E. Jackson, and D.F. Thatcher, "Regulatory Analysis for GI-23: Reactor Coolant Pump Seal Failure," April 1991.

8.10 USNRC, "Interfacing Systems LOCA: Pressurized Water Reactors," NUREG/CR-5102, February 1989.

USNRC, "Assessment of ISLOCA Risk — Methodology and Application to a Westinghouse Four-Loop Ice Condenser Plant," NUREG/CR-5744, April 1992.

USNRC, "Assessment of ISLOCA Risk — Methodology and Application to a Combustion Engineering Plant," NUREG/CR-5745, April 1992.

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8.11 USNRC, "Regulatory Analysis for the Resolution of Generic Issue 105, 'Interfacing System Loss of Coolant Accident In Light-Water Reactors'," NUREG-1463, July 1993.

8.12 USNRC, "Accident Management Strategies for Consideration in the Individual Plant Evaluation Process," Generic Letter 88-20, Supplement 2, April 4, 1990.

8.13 reg guides

GLOSSARY

Accident analysis — steps taken by a PRA analyst to model and quantify the frequency of core damage, containment response, and public risk attributable to a specific accident or class of accidents

Accident class — a grouping of severe accidents with similar characteristics (such as, transients, loss of coolant accidents, station blackout accidents, and containment bypass)

Accident conditions — environmental or operational conditions occurring during events that are not expected in the course of plant operation but are postulated for design or analysis purposes

Accident initiators — initiating events that can challenge plant systems and components

Accident management — strategies and guidance developed for incorporation into the emergency response procedures of a plant to prevent or mitigate events during a severe accident

Accident progression analysis — modeling of that part of the accident sequence which follows the onset of core damage, including containment response to severe accident conditions, equipment availability, and operator performance

Accident sequence analysis — the process of determining the combinations of initiating events, safety functions, and system failures and successes that may lead to core damage

As-built, as-operated — a phrase used to refer to the conformity of the PRA to actual operational and design conditions at the nuclear plant

Availability — the probability that a system or component will function satisfactorily when required to respond to a randomly occurring initiating event or system/component challenge (unavailability is the complement of availability)

Back-end — the portion of the PRA dealing with the containment response to severe accident challenges and the associated radiological release to the environment (also referred to as a Level 2 PRA); can include consideration of consequence to both the public and environment (also referred to as a Level 3 PRA)

Best estimate — the point estimate of a parameter used in a computation which is not biased by conservatism, pessimism or optimism

Boolean algebra — relating to, or being, a logical combinational system that represents symbolically relationships (as those implied by logical operators **AND**, **OR** and **NOT**) between activities

Burden — in human reliability analysis, any of the factors that affect operator performance including such items as time constraints (short available time), diagnosis constraints (confusing indications), factors related to decision making (competing resources), command and control impediments (remoteness between people who need to communicate), and physiological factors (hostile environment)

Common cause event — a subset of dependent events in which two or more component fault states exist at the same time, or within a short time interval, and are the direct result of a shared cause

Common cause failure — a single event that adversely affects two or more components at the same time (also referred to as common mode failure)

Component — an element of plant hardware designed to provide a particular function (for system modeling purposes, a component is at the lowest level of detail in the representation of plant hardware in the models)

Conditional containment failure probability — the likelihood, expressed as a probability, that the containment will fail, given that core damage has occurred.

Conditional probability — the conditional probability of event A occurring given that event B has already occurred is given as: $P(A|B) = P(A \cap B)/P(B)$

Containment bypass — an event which opens a flow path that allows the release of radioactive material directly to the environment bypassing the containment atmosphere

Containment class — a grouping of U.S. containments with similar characteristics (for BWRs, containment classes include Mark I, II, and III containments; for PWRs, containment classes include large dry, atmospheric and subatmospheric, and ice condenser containments)

Containment failure — loss of integrity of the containment pressure boundary (caused by severe accident conditions) which results in leak rates to the environment that exceed the design limits

Containment failure mechanisms — accident conditions that can cause loss of containment integrity (examples for severe accidents include failures resulting from direct containment heating, steam explosions (in-vessel and ex-vessel), hydrogen combustion/detonation and shell melt-through)

Containment failure modes — descriptions used to classify the type of containment failure, such as isolation failure, bypass failure, and early or late failure

Containment isolation failure — failure to isolate all lines that penetrate the containment (the frequency of containment isolation failure includes the frequency of pre-existing unisolable leaks)

Containment performance — a measure of the response of nuclear plant containments to severe accident challenges (containment performance is typically represented by the conditional containment failure probability)

Core damage — uncovery and heatup of the reactor core as a result of a loss of core cooling to the point where prolonged clad oxidation and fuel damage is anticipated

Core damage frequency — the frequency, per reactor year, of an accident leading to core damage

Core-concrete interaction — interaction of molten core material with concrete structures in the containment during a severe accident in which the reactor pressure vessel fails

Core melt — severe damage to the reactor fuel and core internal structures following the onset of core damage, including the melting and relocation of core materials

Core melt frequency — the frequency, per reactor year, of an accident resulting in core melt

Creep rupture — a mechanism of failure resulting from continuous deformation at constant stress; important for metal components at elevated temperatures, such as steam generator tubes or a steel containment boundary in contact with molten core material

Cut set — combination of a set of events (e.g., initiating event and component failures) that, if they occur, will result in an undesirable condition (such as the onset of core damage or containment failure); ***minimal*** cut set is the minimum combination of the set of events that would result in the undesirable condition

Dependency — requirement external to an item and upon which its function depends

Design-basis event — any of the events specified in the nuclear power plant's safety analysis that are used to establish acceptable performance for safety-related functions (events include anticipated transients, design-basis accidents, external events, and natural phenomena)

Diagnosis — examination and evaluation of data to determine either the condition of a structure, system, or component, or the causes of the condition

Dominant contributor — an accident class that has a major impact on the total core damage frequency or a containment failure mechanisms having a major impact on the total radionuclide release frequency

Early containment failure — failure of the containment in a time frame considered short relative to the overall timing of the severe accident (typically, early containment failure is defined as containment failure before or within a few hours of reactor vessel breach)

Early release — a radioactive release from the containment that occurs early, (i.e., occurring within a few hours of vessel breach) and typically before effective implementation of the offsite emergency response and protective actions

Equipment qualification — the generation and maintenance of data and documentation to ensure that the equipment will operate on demand to meet system performance requirements under adverse environments including high temperatures, humidity, and radiation that occur during design basis accidents

Event tree — a quantifiable logical network that begins with an accident initiator or condition and progresses through a series of branches that represent possible system performance, human actions, or phenomena that yield either a safe, stable state or an undesirable one, such as core damage or containment failure

Event tree top event — the conditions (system behavior or operability, human actions, or phenomenological events) that are considered at each branch point in an event tree

External event — an event initiated outside the plant systems that can affect the operability of plant systems (examples include earthquakes, tornados, and floods and fires from sources outside the plant)

Failure — a state that renders a component incapable of performing its specified operation according to established success criteria (the component can fail if it either functions when not required, or does not function when required)

Failure analysis — the systematic process of determining and documenting the mode, mechanism, and root cause of failure of a component or system

Failure mechanism — any of the processes that result in failure, including chemical, electrical, mechanical, physical, thermal, and human factors

Failure mode — manner or state in which a system or component fails (examples include stuck-open valves, motor-bearing seizure, excessive leakage, and failure to produce a signal that drops control rods)

Failure rate — the number of failures of an item within the population per unit measure of life in such terms as the number of demands or operating time

Fault tree — a graphical representation showing the logical relationships among faults; provides a concise and orderly description of the various combinations of possible fault events within a system which could result in some predefined, undesirable event for the system

Fault tree analysis — deductive process used to identify faults required to lead to an undesirable event (fault tree analysis begins with an undesired top event and attempts to identify the sub-events that are necessary to cause the top event; fault tree analysis contrasts with failure modes and effects analysis, which is a bottom-up approach)

Freeze date — the cut-off date for the plant model in an IPE; plant modifications after this date are not included in the model

Frequency — the number of occurrences of an event per unit time

Front-end — the portion of the PRA dealing with the core damage frequency analysis (also referred to as a Level 1 PRA)

Front line system — an engineered safety system used to provide core or containment cooling and to prevent core damage or containment failure (such as ECCS and containment spray systems)

Fuel-coolant interaction — the energetic interaction, by direct contact between water and molten core material, that may result in a steam explosion (fuel-coolant interactions may occur either in-vessel or ex-vessel)

Fussell-Vesely importance — the fractional decrease in total core damage frequency when the plant feature (e.g., a component, train, or system) is assumed to be perfectly reliable (failure rate = 0.0)

Generic Letter 88-20 — a generic letter issued by the U.S. Nuclear Regulatory Commission on November 23, 1988, which requested that U.S. nuclear utilities submit an Individual Plant Examination for severe accident vulnerabilities for each licensed nuclear power plant

Generic failure rate — failure rates that apply generically to a class of equipment rather than specifically to an individual piece of equipment; (rates for equipment from a specific vendor or for a specific application may vary from generic values; generic failure rates, also called "handbook" failure rates, are useful in preliminary design analysis, predictions, and design planning to estimate inherent capability, but should not be preferred to more specific, actual component data, if available)

Harsh environment — an environment expected as a result of the postulated accident conditions appropriate for the design basis or beyond-design basis accidents

High pressure melt ejection — a reactor vessel failure mode that occurs with the reactor coolant system at high pressure and results in rapid dispersal of molten core material, steam, and hydrogen into the containment, challenging it in two ways:

(1) The high temperature core material may come in contact with the containment liner resulting in liner failure

(2) The dispersal of core material and steam into the containment atmosphere may result in direct containment heating and, possibly, hydrogen combustion

Human error probability — a measure of the likelihood that the operator will fail to initiate the correct, required, or specified action or response needed to allow the continuous or correct function of an item of equipment

Human reliability analysis — a structured approach used to identify potential human errors and to systematically estimate the probability of those errors using data, models, or expert judgement

Initiating event — see accident initiators

Individual plant examination — Generic letter 88-20 requested U.S. nuclear utilities to perform an evaluation to identify any plant-specific vulnerabilities to severe accidents; in responding to GL 88-20 most utilities performed the equivalent of a Level 2 PRA, and considered accidents initiated by internal events during full power operation

Internal events — accident initiators originating in a nuclear power plant and, in combination with safety system failures and/or operator errors, leading to core damage accident sequences (see also external events)

Knowledge-based operator action — The mode in which operators may have to act under accident conditions that occur during unfamiliar situations or in an environment for which no know-how or rules for control are available from previous encounters; (various models proposed in the literature emphasize two different aspects of the problem, specifically, the two classes distinguish time-oriented models that emphasize the time available for operator action, and rating-oriented models that rate human actions according to various characteristics, such as difficulty in diagnosis; error rates are developed from these ratings)

Late containment failure — failure of the containment in a time considered long relative to the overall timing of the severe accident (typically, late containment failure is defined as containment failure occurring more than a few hours past reactor vessel breach)

Late release — a radioactive release from the containment that occurs late (i.e., occurring more than a few hours past reactor vessel breach) and typically after effective implementation of the offsite emergency response and protective actions

Level 1 analysis — an identification and quantification of the sequences of events leading to the onset of core damage

Level 2 analysis — evaluation of containment response to severe accident challenges and quantification of the mechanisms, amounts, and probabilities of subsequent radioactive material releases from the containment

Level 3 analysis — evaluation and quantification of the resulting consequences to both the public and environment

Mission time — the time period that a system or component is required to be operable in order to carry out its mission; (for example, a mission time of 24 hours implies that containment sprays are required to be operable for 24 hours in order to prevent containment failure from occurring within that period)

Model— an approximate mathematical representation that simulates the behavior of a process, item, or concept (such as failure rate); (for example, the probability of a system failure is synthesized using models that relate system failures to component failures and human errors; the probability of system failure is then calculated from these more elementary and better understood failures; these models contain parameters, such as the rates of occurrence of various events, that are not known precisely)

Modeling assumption — an assumption on which a model is based (such assumptions may not be valid or universally accepted)

Performance shaping factor (PSF) — an influence on the performance of an operator; (underlying PSFs is the idea that the human error rates for a set of specified actions can be derived by investigating how a small set of PSFs influence the success or failure of the operators; PSFs include such considerations as training, experience, availability and quality of a procedure, stress, interdependence among operators, environment, and timing)

Plant — a general term used to refer to a nuclear power facility; (for example, plant could be used to refer to a single unit or a multi-unit site)

Pool scrubbing — the retention of some of the radioactive material from core debris released into the pool; (for example, the suppression pool of the BWR Mark I containment may provide pool scrubbing for some accident scenarios)

Probabilistic Risk Assessment/Analysis — of a nuclear power plant, is an analytical process that quantifies the potential risk associated with the design, operation and maintenance of a plant to the health and safety of the public; the risk evaluation involves three sequential parts or "Levels" (refer to Level 1 analysis, Level 2 analysis and Level 3 analysis)

Reactor class — a group of nuclear power plants of similar NSSS design with reactors manufactured by the same vendor; (for example, all Westinghouse 4-loop plants belong to the same reactor class)

Reactor year — a period of the reactor operation that accounts for the downtime during a calendar year

Recovery action — an operator action intended to bring failed or unavailable (such as due to test and maintenance) equipment back to operable status

Recovery (non-recovery) factor — a correction factor that is applied either to sequence cut sets or an event tree; (for example, a sequence cut set may be modified by including a new basic event representing the probability of an operator's failure to perform a recovery action; if several cut sets are affected by the same dominant recovery action, it may be more useful to include the recovery action at a higher level in the logic model; for example, actions to recovery offsite power in response to a loss of offsite power initiating event are included in the event tree functions)

Regression analysis — a statistical technique which hypothesizes a model relating a dependent variable to a set of independent variables; (the dependent variable is assumed random and its expected value is expressed as a function of the independent variables with unknown coefficients; the coefficients are estimated based on the observed values of all the variables)

Reliability — the probability that a component performs its specified function and does not fail under given operating conditions for a prescribed time

Risk — typically, the expected value of the consequences per unit time (usually expressed as fatalities/yr or $/yr); defined more broadly using the "set of triplets" $\{ \langle s_i, f_i, x_i \rangle \}$; (in the set of triplets, s_i identifies one of several possible scenarios, f_i is the frequency of that scenario, and x_i is the consequence of that scenario; the risk is the set of all possible scenarios, their frequencies, and their consequences; this definition distinguishes between low-frequency, high-consequence scenarios and high-frequency, low-consequence scenarios)

Risk-informed regulation — a regulation whose decision making criteria integrate probabilistic and conventional deterministic evaluations

Rule-based operator actions — a sequence of actions in which an operator follows remembered or written rules; (for example, performance of written post-diagnosis actions or calibrating an instrument or using a checklist to restore manual valves to their normal operating status after maintenance are classified as rule-based operator actions)

Safety systems/components — those systems/components that are designed for design-basis accident; (technical specifications and administrative controls are required for safety systems/components)

Scope — refers to the extent of initiating events considered in a PRA; a full-scope PRA usually includes accidents initiated by internal and external events during full-power and low power/shutdown conditions; the scope should be distinguished from the PRA Level, which defines the extent of the analysis (refer to Level 1 analysis, Level 2 analysis and Level 3 analysis)

Sensitivity analysis — an analysis in which one or more input parameters to a model are varied in order to observe their effects on the model predictions

Severe accident — an accident that goes beyond the design-basis of the plant and usually involves extensive core damage

Skill-based operator action — the performance of more or less subconscious routines governed by stored patterns of behavior; (for example, the performance of memorized immediate emergency action following an accident initiator)

State-of-the-art PRA — a PRA that reflects the latest improvements in PRA modeling and evaluation

Station blackout — an accident sequence initiated by loss of all offsite power with failure of onsite emergency AC power (diesel generators), and failure of timely recovery of offsite power and onsite emergency AC power

Success criteria — the systems/components and their combinations that are needed to carry out their mission given an accident initiator

Support system — a system that provides a support function (e.g., electric power, control power, and cooling) for another system; (for example, HVAC is often considered as a support system)

Uncertainty Analysis — the quantification of the imprecision in the PRA estimate that results from imprecisely formulated PRA models and imprecisely known input variables

Unit — refers to a single nuclear power reactor with its associated systems and components; most nuclear power plant sites have either one or more units; at multi-unit sites, some support systems can be shared between units

Vessel breach — refers to the failure of the reactor pressure vessel (RPV) boundary and a release of the radioactive material from the RPV

Walk-through/walk-down — inspection of local areas in a nuclear power plant where systems and components are physically located in order to verify the location of the equipment, assess its operating status, and ascertain any environmental effects or system interaction effects on the equipment which could occur during accident conditions