NUREG-1560
Vol. 2, Parts 2 – 5

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

Parts 2 – 5

Draft Report for Comment

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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Division of Systems Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
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FOREWORD

In its "Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants," the U.S. Nuclear Regulatory Commission (NRC) 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 Commission issued Generic Letter 88-20, 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."*

In concert with the objectives of the policy statement, a memorandum from the Deputy Executive Director for Nuclear Reactor Regulation, Regional Operations and Research to the Office of Nuclear Regulatory Research, dated May 12, 1993, 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."* This draft report fulfills that recommendation by documenting the insights gained by reviewing the IPE submittals. As such, this report provides perspectives on the following major objectives:

- 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
  - the impact that the improvements have had on plant safety
  - whether any of the improvements have generic implications for all or a class of plants
- plant-specific features and assumptions that play a significant role in core damage frequency (CDF) estimation and containment performance analysis
  - the important design and operational features that affect CDF and containment performance, with regard to the different reactor and containment types
  - the influence of the IPE methodology and assumptions on the results, with regard to the different reactor and containment types
  - the significant plant improvements in reducing CDF and increasing containment performance, with regard to the different reactor and containment types
- the importance of the operator's role in CDF estimation and containment performance analysis
  - operator actions that are consistently important in the IPEs
  - operator actions that are important because of plant-specific characteristics
  - the influence of modeling assumptions and different methodologies on the results
- IPEs with respect to risk-informed regulation

— the quality of the IPEs, compared to a quality probabilistic risk assessment and, therefore, the potential role of the IPEs in risk-informed regulation

In addition to the above objectives, this report provides perspectives on the following items:

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

- 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

- the results of the IPEs compared with the perspectives gained from NUREG-1150

As noted above, the perspectives presented in this report are derived from results presented solely in the IPE submittals. Consequently, comments on the interpretation and accuracy of the IPE results as presented in this report are particularly important. All comments should be addressed in writing within 90 days to:

Mary Drouin
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
MS T10E50
Washington, DC 20555

This report will be revised on the basis of comments received. The final version of this report is expected to be issued in 1997.

M. Wayne Hodges

M. Wayne Hodges, Director
Division of Systems Technology
Office of Nuclear Regulatory Research

REPORT ORGANIZATION

As a result of Generic Letter 88-20, the U.S. Nuclear Regulatory Commission received 75 separate IPE submittals covering 108 nuclear units, each submittal containing a wealth of information. Table 1 lists the general type of information contained in a single submittal.

Table 1 General information contained in IPE submittals.

Core damage frequency general information	Containment analysis general 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

In examining the information from the IPE submittals, the staff adopted the following viewpoints:

- impact of the IPE Program on reactor safety
- reactor and containment design and operational perspectives
- IPEs with respect to risk-informed regulation
- additional IPE perspectives

The report is arranged in five parts. Part 1 provides an overall summary of the key perspectives gained in each of the above areas. Parts 2 through 5 provide a more in-depth discussion of the perspectives gained by reviewing the IPE submittals. The contents of these parts and associated chapters (as shown in Figure 1) are described in more detail below.

Glossary and Index —

Many terms used in this report are dependent on the technical context and, therefore, can vary in definition. Glossaries are 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.

In addition, the staff anticipates that this report will be used by many different readers, each with different interests. To further aid the reader, the staff has provided an index to quickly point the reader to specific items of interest.

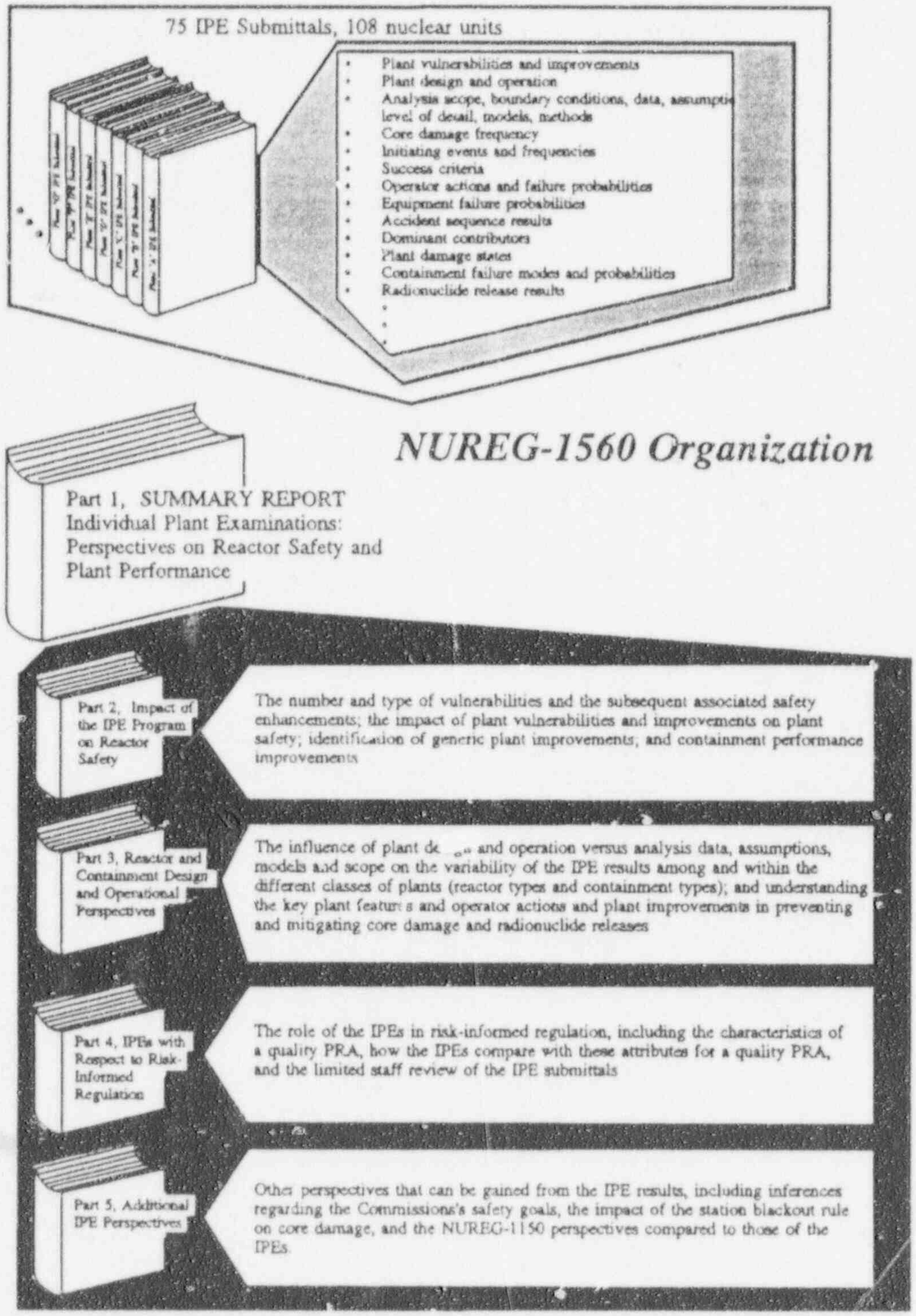

Figure 1 IPE NUREG report roadmap.

NUREG-1560 Organization

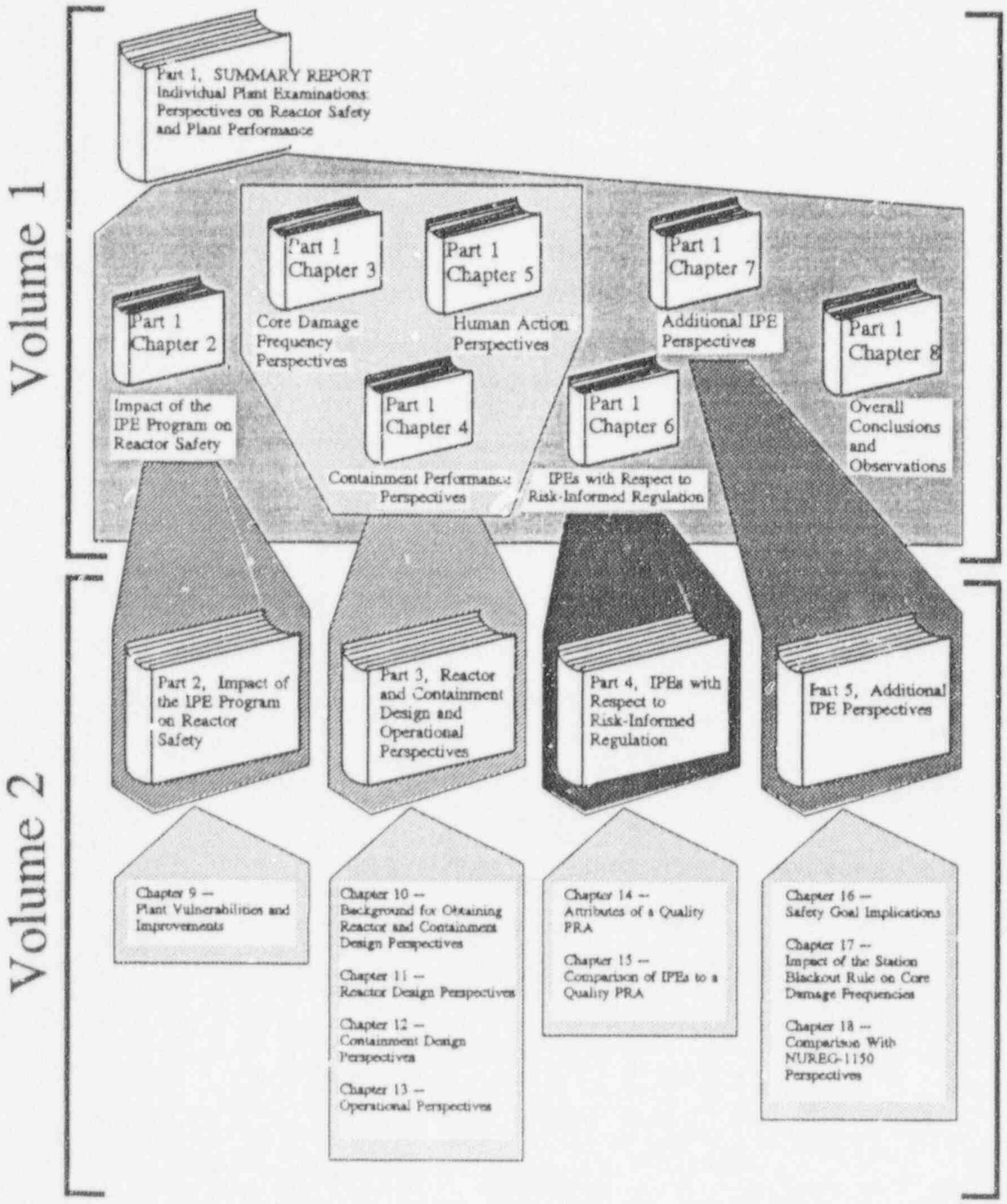

Figure 1 IPE NUREG report roadmap (continued).

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 implications.

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

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

- Chapter 3 a 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 2 and 3, respectively. The perspectives within the different classes are discussed relative to the different accident classes as defined in Table 4.

- Chapter 4 summarizes of 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, the perspectives in Chapter 4 are discussed relative to the different containment failure classes (as defined in Table 5). As in Chapter 3, Chapter 4 is also divided into sections with the perspectives provided for the different BWR and PWR containment classes as defined in Tables 6 and 7, respectively. In addition, this criteria discusses perspectives on the reported radionuclide releases resulting from containment bypass or early containment failure.

Table 2 Summary of BWR plant classes and associated nuclear power plants.

Class	IPE submittals
BWR 1/2/3	• Big Rock Point • Dresden 2&3 • Millstone 1 • Nine Mile Point 1 • Oyster Creek 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.
BWR 3/4	• 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 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.
BWR 5/6	• Clinton • Grand Gulf 1 • LaSalle 1&2 • Nine Mile Point 2 • Perry 1 • River Bend • WNP 2 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.

Table 3 Summary of PWR plant classes and associated nuclear power plants.

Class	IPE submittals
Babcock & Wilcox (B&W)	• ANO 1 • TMI 1 • Crystal River 3 • Davis Besse • Oconee 1,2&3
	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)	• ANO 2 • Millstone 2 • St. Lucie 1&2 • Calvert Cliffs 1&2 • Palisades • Waterford 3 • Fort Calhoun 1 • Palo Verde 1,2&3 • Maine Yankee • San Onofre 2&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 • Robinson 2 • Turkey Point 3&4 • Beaver Valley 2 • Shearon Harris 1 • Farley 1&2 • Summer • North Anna 1&2 • Surry 1&2
	This group is similar in design to the Westinghouse 2-loop group. The RCPs are a Westinghouse design.
Westinghouse 4-loop	• Braidwood 1&2 • Comanche Peak 1&2 • Indian Point 2 • Salem 1&2 • Vogtle 1&2 • Byron 1&2 • DC Cook 1&2 • Indian Point 3 • Seabrook • Watts Bar 1 • Callaway • Diablo Canyon 1&2 • McGuire 1&2 • Sequoyah 1&2 • Wolf Creek • Catawba 1&2 • Haddam Neck • Millstone 3 • South Texas 1&2 • 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 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.*

Table 5 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.

Table 6 Summary of BWR containment classes and associated nuclear power plants.

Class	IPE submittals
Mark I	• Browns Ferry 2 • Duane Arnold • Hope Creek • Oyster Creek • Vermont Yankee • Brunswick 1&2 • Fermi 2 • Millstone 1 • Peach Bottom 2&3 • Cooper • Fitzpatrick • Monticello • Pilgrim 1 • Dresden 2&3 • Hatch 1&2 • Nine Mile Point 1 • Quad Cities 1&2
	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 • WNP 2 • Limerick 1&2 • Nine Mile Point 2 • Susquehanna 1&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 7 Summary of PWR containment classes and associated nuclear power plants.

Class	IPE submittals			
Large dry and Sub-atmospheric	• ANO 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	• ANO 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	• 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 • TMI 1 • Vogtle 1&2	• 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
	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.			
Ice condensers	• Catawba 1&2 • Watts Bar 1	• DC Cook 1&2	• McGuire 1&2	• Sequoyah 1&2
	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.			

*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.

- 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 IPEs with respect to risk-informed regulation. This chapter is divided into four sections as follows:

  — Section 6.1 summarizes the role of the IPEs.
  — Section 6.2 summarizes the characteristicsthat comprise a current quality probabilisitc risk assessment (PRA).
  — Section 6.3 summarizes the comparison of the IPEs against the characteristicsof a quality PRA.
  — Section 6.4 summarizes perspectives regarding their potential role in risk-informed regulation.

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

  — Section 7.1 discusses the NUREG-1150[1] 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 (SBO) rule) in light of their impact on CDF.

  — 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 regarding how NUREG-1560 can be used, and is divided into four sections as follows:

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. Chapter 9 is divided into the following sections:

- Section 9.1, Vulnerability Definition
- Section 9.2, Plant Vulnerabilities
- Section 9.3, Plant Improvements
- Section 9.4, Containment Performance Improvements
- Section 9.5, Impact on Reactor Safety As a Result of Plant Enhancements

Part 3 — Reactor and Containment Design and Operational Perspectives

Part 3 provides a more in-depth discussion of the information in Part 1, Chapters 3, 4 and 5, regarding reactor and containment design and operational 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 discussed in this report. In addition, this chapter makes the reader aware of the plant and containment characteristics, as well as the boundary conditions, assessments, and

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

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 the subsequent chapters.

- Chapter 11, Reactor Design 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 Design 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, Operational 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 — IPEs with Respect to Risk-Informed Regulation

Part 4 provides a more in-depth discussion of the information discussed in Part 1, Chapter 6, IPEs with Respect to Risk-Informed Regulation, and is divided into the following two chapters:

- Chapter 14 provides a detailed and explicit description of acceptable attributes of a quality PRA. These attributes cover the entire scope of a PRA (Levels 1, 2, and 3) for internal events (excluding internal fire) at full power. This discussion does not include the scope of a PRA covering internal fire, external events (such as seismic) and other modes of operation (such as shutdown).

- Chapter 15 provides a detailed comparison of the IPEs, collectively, against the acceptable attributes for a quality PRA (as defined in Chapter 14). This discussion identifies where the IPEs meet the attributes and where (and to what degree) they deviate from the attributes.

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 16 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 17 provides further information related to the impact of the SBO rule on CDFs. This information includes details on the approach used to address the impact of the SBO rule, including a discussion of the type of coping methods used by various plants to comply with the SBO rule. Chapter 17 also provides further details (beyond those in Section 7.2) on the factors affecting the SBO CDF for the groups of plants that accounted for implementation of the SBO 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 SBO CDF.

- Chapter 18 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.

ABBREVIATIONS

AC	Alternating Current
AAC	Alternate AC
ADS	Automatic Depressurization System
ADV	Atmospheric Dump Valve
AEOD	Office of Analysis and Evaluation of Operational Data
AFW(S)	Auxiliary Feedwater (System)
AMSAC	ATWS Mitigating System Actuation Circuitry
ANL	Argonne National Laboratory
ANO	Arkansas Nuclear One
ARI	Alternate Rod Insertion
ARFS	Air Return Fun System
ASEP	Accident Sequence Evaluation Program
ATWS	Anticipated Transient Without Scram
BAAT	Battery lifetime
BE	Basic Event
BOP	Balance of Plant
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
CD	Core Damage
CDF	Core Damage Frequency
CE	Combustion Engineering
CET	Containment Event Tree
CVCS	Chemical and Volume Control Tank
CHR	Containment Heat Removal
CPI	Containment Performance Improvement
CRAC	Calculations of Reactor Accident Consequences
CRD	Control Rod Drive
CRDHS	Control Rod Drive Hydraulic System
CS	Core Spray
CSS	Containment Spray System
Cs	Cesium
CST	Condensate Storage Tank
CGCS	Combustible Gas Control System
CW	Circulating Water
DC	Direct Current
DCH	Direct Containment Heating
DDT	Deflagration to Detonation Transformation
DHR	Decay Heat Removal
DIS	Distributed Igniter System
EAC	Emergency Alternating Current
EC	Emergency Condenser
ECCR	Emergency Core Cooling Recirculation

ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
ECF	Early Containment Failure
ECF/B	Early Containment Failure or Bypass
ECT	Emergency Cooling Tower
EFW	Emergency Feedwater
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure
EPG	Emergency Procedure Guideline
EPRI	Electric Power Research Institute
ESF	Engineered Safety Feature
ESFA(S)	Engineered Safety Feature Actuation (System)
ESW	Emergency Service Water
EVSE	Ex-vessel Steam Explosion
FCI	Fuel-Coolant Interaction
FLIM	Failure Likelihood Index Methodology
FMEA	Failure Mode and Effects Analysis
FSAR	Final Safety Analysis Report
FWCI	Feedwater Coolant Injection
GL	Generic Letter
GE	General Electric
GSI	Generic Safety Issue
HCR	Human Cognitive Reliability
HCU	Hydraulic Control Units
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, Ventilation, and Air Conditioning
I	Iodine
IA	Instrument Air
IC	Isolation Condenser
IE	Initiating Event
IORV	Inadvertently Open Relief Valve
IPE	Individual Plant Examination
IPEP	Individual Plant Examination Partnership
IREF	Individual Risk Of Early Fatality
ISGTR	Induced Steam Generator Tube Rupture
ISLOCA	Interfacing System Loss-of-Coolant Accident
kV	Kilovolt
LOCA	Loss-of-Coolant Accident
LER	Licensee Event Report

LOOP	Loss of Offsite Power
LOSP	Loss of Station Power
NMLPCI	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
MTC	Moderator Temperature Coefficient
NMP	Nine Mile Point
MWt	Megawatt
NPSH	Net Positive Suction Head
NRC	US Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NSW	Normal Service water
NUMARC	Nuclear Management and Resources Council
ORCA	Operator Reliability Characterization and Assessment
ORE	Operator Reliability Experiments
PARV	Power Actuated Relief Valves
RBCCW	Reactor Building Closed Cooling Water
RCFC	Reactor Containment Fun Cooler
RPT	Recirculation Pump Trip
RWCU	Reactor Water Cleanup
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

Abbreviations

SAMG	Severe Accident Management Guidelines
SAR	Safety Analysis Report
SBO	Station Blackout
SBOR	Station Blackout Rule
SDC	Shutdown Cooling
SEGR	Steam Explosion Review Group
SER	Staff Evaluation Report
SG	Steam Generator
SGFP	Steam Generator Feedwater Pump
SGTR	Steam Generator Tube Rupture
SGTS	Standby Gas Treatment System
SHARP	Systematic Human Action Reliability Procedure
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SLIM	Success Likelihood Index Methodology
S-LOCA	Seal LOCA
SNL	Sandia National Laboratories
SORV	Stuck Open Relief Valve
SPC	Suppression Pool Cooling
SPMU	Suppression Pool Makeup
SRV	Safety Relief Valve
SSMP	Safe Shutdown Makeup Pump
SSW	Standby Service Water
SW	Service Water
TAF	Top of Active Fuel
TBCCU	Turbine Building Closed Cooling Water
Te	Tellurium
T-I	Time-Induced
THERP	Technique for Human Error Rate Prediction
TMI	Three Mile Island
TRC	Time Reliability Correlation
TSC	Technical Support Center
TVA	Tennessee Valley Authority
U.S.	United States
USI	Unresolved Safety Issue
W	Westinghouse
WNP2	Washington (State) Nuclear Power, Unit 2
VB	Vessel Breach

PART 2

IMPACT OF THE IPE PROGRAM ON REACTOR SAFETY

9. PLANT VULNERABILITIES AND IMPROVEMENTS

The primary goal of the Individual Plant Examination (IPE) program as delineated in Generic Letter (GL) 88-20 (Ref. 9.1) is "identifying plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements." This goal is a product of the systematic examination required by GL 88-20. The Generic Letter indicates that "It is expected that during the course of the examination, the utility would carefully examine the results to determine if there are worthwhile prevention or mitigation measures that could be taken to reduce the core damage frequency or poor containment performance with the attendant radioactive release." To help the utilities address improvements related to preventing containment failure, Supplements 1 and 3 to GL 88-20 were issued and contain specific containment performance improvement (CPI) recommendations. However, it is recognized in the Generic Letter that the potential benefits from any plant improvement are plant specific and are dependent upon on the frequency and consequences of the accidents contributing to core damage and containment failure.

While only a small fraction of the licensees identify what is explicitly called vulnerabilities in their submittals, nearly all of the licensees identify areas warranting investigation of potential improvements, both design and procedural. The resulting equipment and procedural changes to the plants have been a benefit to 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.

No specific definition for what constitutes a "vulnerability" is provided in GL 88-20 or in the subsequent Nuclear Regulatory Commission (NRC) IPE submittal guidance documented in NUREG-1335 (Ref. 9.2). Instead, the licensees are asked to decide if a specific vulnerability or weakness exists at their plant and whether some plant improvement is needed. Hence, there is considerable variability among the submittals regarding what is a vulnerability. A problem that is considered a vulnerability at one plant may not be considered a vulnerability at another plant. 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 serious issue worthy of attention but not necessarily a "vulnerability." As a result, this report attempts to differentiate those cases where the licensee appears to *explicitly* define the issue as a "vulnerability" from other identified areas considered for plant improvements. The various definitions of "vulnerability" used by the licensees and the plant improvements identified in the submittals to address some of these vulnerabilities and other issues not explicitly identified as vulnerabilities, (including containment performance improvements) are discussed in detail in this chapter. A summary of the key perspectives on the impact of the IPE program on reactor safety is provided in Chapter 2 of Part 1.

9.1 Vulnerability Definitions

One of the reporting guidelines presented in NUREG-1335 is that each licensee present "a concise discussion of the criteria used by the utility to define vulnerabilities, and the fundamental causes of each vulnerability." Most of the licensees clearly identify criteria for identifying "vulnerabilities" or other areas worthy of a potential plant improvement. The identified criteria are discussed below.

The definitions for vulnerability used in many of the submittals are based on one of two sets of quantitative criteria: (1) the criteria provided in NUMARC Severe Accident Issue Closure Guidelines Document 91-04 (Ref. 9.3), and (2) NRC's Safety Goal Policy Statement (Ref. 9.4) defining a core damage frequency (CDF) subsidiary objective of 1E-4 per reactor year (ry) and a large release subsidiary objective of 1E-6/ry. A third criterion utilized in many submittals is based on using importance measures or the results of sensitivity studies to determine which components or systems are the most vital to the plant. Several variations and combinations of these quantitative criteria are identified in the submittals and are discussed below. No specific definition of vulnerability can be identified for a third of the plants. However, for a significant number of these plants, some sort of criterion is utilized to identify areas for improving the plant safety.

Approximately 20% of the plants report using some variation of the NUMARC guidelines to identify what they explicitly call vulnerabilities. The NUMARC guidelines, shown in Table 9.1, consist of a graded review process to identify plant-specific vulnerabilities. Application of the NUMARC methodology requires that the accident sequences be grouped into functional groupings suggested by the guidelines. The quantitative results for each functional grouping are then compared with the NUMARC closure guidelines. The closure guidelines suggest possible licensee responses to identified vulnerabilities, ranging from hardware or procedural modifications to treatment in a severe accident management plan, that are a function of the core damage frequency or percentage of core damage due to an accident functional group. As indicated in Table 9.1, two sets of closure guidelines are provided, each with four levels; one set of criteria is for core damage sequences and one is for containment bypass sequences. The core damage closure guideline values range from 1E-4/ry for the top level to 1E-6/ry at the bottom level and are an order of magnitude higher than the containment bypass values.

Table 9.1 NUMARC vulnerability guidelines.

Mean CDF per sequence group	Mean containment bypass frequency	Potential actions to be taken
Greater than 1E-4/ry or greater than 50% of total CDF	Greater than 1E-5/ry or greater than 20% of total CDF	1. Find a cost-effective plant administrative, procedural, or hardware modification with emphasis on eliminating or reducing the likelihood of the source of the accident sequence initiator. 2. If unable to do 1, treat in Emergency Operating Procedures (EOPs) or other plant procedures with emphasis on prevention of core damage. 3. If unable to do 1 and 2, ensure Severe Accident Management Guideline (SAMG) is in place with emphasis on prevention/mitigation of core damage, vessel failure, and containment failure.
1E-4/ry to 1E-5/ry or 20% to 50% of total CDF	1E-5/ry to 1E-6/ry or 5% to 20% of total CDF	1. Find a cost-effective treatment in EOPs or other plant procedures or make minor hardware changes with emphasis on prevention of core damage. 2. If unable to do 1, ensure SAMG is in place with emphasis on prevention/mitigation of core damage, vessel failure, and containment failure.
1E-5/ry to 1E-6/ry	1E-6/ry to 1E-7/ry	Ensure SAMG is in place with emphasis on prevention/mitigation of core damage, vessel failure, and containment failure.
Less than 1E-6/ry	Less than 1E-7/ry	No specific action required.

A summary of the vulnerability criteria used in the IPEs based on the NUMARC 91-04 guidelines is presented in Table 9-2. Many licensees who adopt the NUMARC guidelines define a vulnerability as a functional sequence exceeding only the top evaluation criteria (greater than 1E-4/ry CDF or 50% of the total plant CDF). Additionally, most licensees using the NUMARC guidelines define a containment bypass vulnerability as any such functional sequence of this type with a CDF greater than the NUMARC top criterion of 1E-5/ry or contributing greater than 20% to the total CDF. Some licensees, when using the NUMARC guidelines, identify vulnerabilities associated with sequences meeting any of the graded NUMARC criteria (not just the top criteria). In many of these cases, resolution of a "vulnerability" meeting the lower tier NUMARC criteria is addressed simply by incorporating the issue into future accident management strategies. Other licensees use slightly modified versions of the NUMARC top criteria to define a vulnerability. When this is done, usually the modification is that the percent contribution forms of the criteria are not used (on the basis that a large percentage of a small absolute frequency should not be used to identify a vulnerability).

Table 9.2 Summary of vulnerability criteria in IPEs using NUMARC 91-04.

Criteria used to define "vulnerabilities"	Plant type	Plant name
NUMARC 91-04.	Combustion Engineering (CE) pressurized water reactor (PWR) 2-Loop	Calvert Cliffs 1&2, Fort Calhoun 1
	Westinghouse (W) PWR 3-Loop	Summer
	W PWR 4-Loop	Callaway
NUMARC 91-04 (adjusted to address systemic sequences).	W PWR 4-Loop	South Texas 1&2
NUMARC 91-04 (modified). Generally, only the top criterion was used. Some licensees also only used the absolute CDF criterion, arguing the percentage contribution criterion is not appropriate when the CDF is small.	Boiling water reactor (BWR) 6	Grand Gulf 1, Perry 1
	Babcock and Wilcox (B&W) PWR 2-Loop	ANO 1
	CE PWR 2-Loop or 3-Loop	ANO 2
	W PWR 3-Loop	Farley 1&2
	W PWR 4-Loop	Diablo Canyon 1&2, Vogtle 1&2
NUMARC 91-04 (modified); also, any source-term analysis bin representing containment failure or impairment, with a frequency greater than 1E-5 and in which a function, system, operator action, or other element substantially contributes to total frequency.	BWR 4	Hatch 1&2
NUMARC 91-04; also, total plant CDF exceeds 1E-4/ry and sequence(s) indicate that a plant-specific feature is an outlier to comparable BWR probabilistic risk assessments (PRAs).	BWR 5	WNP 2
NUMARC 91-04 (modified); also, single or common-mode component failure, support system failure, or operator action with significant impact on CDF.	CE PWR 2-Loop	Waterford 3

Some licensees use the NUMARC guidelines in combination with additional criteria for identifying vulnerabilities. For example, some licensees have added a Level 2 criterion related to the frequency of a source term bin exceeding 1E-5/ry. Waterford has explicitly added criteria related to single failures, common cause failures, support system failures, and operator errors that have a significant impact on the core damage frequency in their vulnerability screening. The vulnerability screening criterion for Washington Nuclear Power Unit 2 also requires that the total CDF be within the NRC safety goal of 1E-4/ry and includes a search for sequences that are outliers when compared to similar plants because of a plant-specific feature.

The CDF and large release subsidiary objectives from the NRC Safety Goal Policy Statement, used by approximately 25% of the licensees (see Table 9.3), are nearly equivalent to some of the NUMARC criteria in that these objectives focus on just the absolute frequencies for CDF (but in this case the total plant CDF instead of an accident grouping frequency) and a large significant release. Any sequences contributing significantly to exceeding either or both criteria are examined by licensees for those design or operational aspects which cause such a vulnerability, and resolutions are investigated to lessen the potential for such a vulnerability. Some plants use modified safety goal subsidiary objectives frequencies in their definitions (e.g., 5E-4/ry for CDF and 5E-5/ry for an early release frequency). One plant (Oyster Creek) applies the criteria at the systemic sequence level instead of for the total plant CDF. Another plant (Palisades) has changed the large release criteria from 1E-6/ry to 10% of the plant CDF.

Table 9.3 Summary of vulnerability criteria in IPEs using NRC Safety Goal subsidiary objectives.

Criteria used to define "vulnerabilities"	Plant type	Plant name
Any core damage sequence exceeding 1E-4/ry or any containment bypass sequence or large early containment failure sequence exceeding 1E-6/ry.	BWR 2	Oyster Creek
	BWR 6	River Bend
	W PWR 2-Loop	Ginna
	W PWR 4-Loop	Seabrook
Total plant CDF exceeds 1E-4/ry or large release frequency exceeds 1E-6/ry? If so, are any plant-specific design/operating characteristics dominant contributors?	BWR 4	Vermont Yankee
	W PWR 2-Loop	Point Beach 1&2
	W PWR 4-Loop	Indian Point 2
Results suggest core damage frequency would not meet NRC Safety Goal subsidiary objectives? Are there any new/unusual core damage containment failure mechanisms compared to other PRAs?	BWR 3	Monticello, Pilgrim 1
Is there adequate assurance of no undue risk to public health and safety? Are there any new/unusual core damage containment failure mechanisms compared to other PRAs?	W PWR 2-Loop	Prairie Island 1&2
Results suggest core damage frequency would not meet NRC Safety Goal subsidiary objectives? New/unusual core damage containment failure mechanisms compared to other PRAs? Systems, components, operator actions that dramatically affect core damage?	BWR 2	Nine Mile Point 1
	BWR 4	Duane Arnold
	BWR 5	Nine Mile Point 2
	BWR 6	Clinton

Table 9.3 **Summary of vulnerability criteria in IPEs using NRC Safety Goal subsidiary objectives.**

Criteria used to define "vulnerabilities"	Plant type	Plant name
Level 1: New/unusual core damage containment failure mechanisms, compared to other PRAs? Results suggest core damage frequency would not meet NRC Safety Goal subsidiary objectives? Any systems, components, or operator actions that control core damage? Level 2: Containment capability acceptable? Unusually poor containment response performance? Containment isolation system reliability acceptable? Containment bypass frequency acceptable? Unusually poor performance of containment mitigating systems?	BWR 4	Fermi 2
Do IPE results meet NRC Safety Goal subsidiary objectives for core damage frequency? Are results for core damage sequences or containment performance consistent with other PRAs? Does probability of sequences characterized as having large releases exceed 10% of CDF?	CE PWR 2-Loop	Palisades
If CDF exceeds 1E-4/ry, are one or a few plant features/operating practices responsible? If CDF is acceptable, are plant features/operating practices relatively high contributors? Is CDF very sensitive to highly uncertain aspects of plant response?	B&W PWR 2-Loop	Davis-Besse
Does CDF exceed 5E-4/ry or large, early release frequency exceed 5E-5/ry? If so, vulnerability identified if common function, system, operator action, or other common element contributes substantially to total CDF.	BWR 4	Browns Ferry 2
	W PWR 4-Loop Ice Condenser	Sequoyah 1&2, Watts Bar 1
"Defense-in-Depth Criteria": Sequences with "high calculated frequencies are not acceptable . . .sequences having low-calculated frequencies must also have 'defense-in-depth' for both equipment and procedures."	BWR 4	Susquehanna 1&2

Some licensees using the NRC Safety Goal subsidiary objectives in their vulnerability screening also use additional criteria. The most common criterion is a comparison to similar plants for the purpose of identifying any new or unusual core damage or containment failure mechanisms specific to their plant. Other licensees also include a criterion that requires that any systems, components, or operator actions that significantly impact the core damage frequency be listed as vulnerabilities. One plant (Davis-Besse) considers that a vulnerability might exist if the frequency of core damage is sensitive to a highly uncertain aspect of the plant response, but states that further evaluation to reduce the uncertainty would be a more appropriate response than a change to the plant. Finally, several plants have added Level 2 vulnerability criteria that address the performance of containment mitigating systems and the containment itself during severe accidents.

For approximately 25% of the plants, the percent contribution to CDF is used as the base criterion for screening vulnerabilities, as indicated in Table 9.4. Some plants also include the percent contribution to containment failure in their vulnerability screening. The licensees usually rely on the relative contribution of systemic sequences, plant damage states, containment failure modes, and release categories to identify the important contributors to the plant risk. These important contributors are equated to areas where vulnerabilities may exist. Importance measures and sensitivity studies are also generally utilized to identify the fundamental causes or plant features contributing to these potential vulnerabilities.

Table 9.4 Summary of vulnerability criteria in IPEs using percent contribution to CDF.

Criteria used to define "vulnerabilities"	Plant type	Plant name
Relative contribution of systemic sequences to CDF; relative contribution of containment failure modes.	W PWR 2-Loop	Kewaunee
	W PWR 3-Loop	Beaver Valley 1&2, North Anna 1&2, Surry 1&2
	W PWR 4-Loop	Millstone 3
	W PWR 4-Loop Ice Condenser	D.C. Cook 1&2
Single or common-mode component failure, support system failure, or operator action with significant impact on CDF; mode of early containment failure with relatively high probability (>10%) of occurrence in core melt accident.	BWR 3 (Isolation Condenser)	Millstone 1
	CE PWR 2-Loop	Millstone 2
	W PWR 4-Loop	Haddam Neck
Failure mode, single failure, or combination of small number of failures not used to create a support state that disproportionately contributes to overall CDF.	BWR 4	Limerick 1&2, Peach Bottom 2&3
Plant features which contribute disproportionately large percentage to core damage frequency, which are, in turn, higher than those of similar plants.	BWR 4	Hope Creek
Plant features which contribute disproportionately large percentage to either core damage or significant release frequencies, which are, in turn, higher than those of similar plants.	W PWR 4-Loop	Salem 1&2
	CE PWR 2-Loop	St. Lucie 1&2, San Onofre 2&3
Vulnerabilities limited to issues where there was "high-confidence" in the results of the IPE (plant change/countermeasure may be recommended). For "low-confidence" issues, additional analysis may be recommended.	W PWR 3-Loop	Turkey Point 3&4
	BWR 3	Quad Cities 1&2
	BWR 4	Brunswick 1&2
	W PWR 3-Loop	H.B. Robinson 2, Shearon Harris 1
	W PWR 4-Loop	Braidwood 1&2, Byron 1&2, Wolf Creek, Zion 1&2

Generally quantitative thresholds, as exist in the NUMARC guidelines, are not established for screening vulnerabilities based on the percent contribution to CDF or containment failure. Instead, the licensees apply qualitative thresholds using terms such as "significant" or "disproportionately high." Some licensees indicate that a plant feature will only be considered a vulnerability if it is a proportionately higher contributor or outlier when compared to similar plants. Thus a 50% contributor to CDF might not be a vulnerability if it has a similar contribution at similar plants. One licensee (Turkey Point 3&4) states that vulnerabilities will only be considered for issues where they had the highest confidence in the results of their submittal (i.e., where all possible credit is taken for recovery actions).

For the remaining 30% of the plants (see Table 9.5), no vulnerability screening criteria can be explicitly identified. However, all of these plants use their submittal results to help identify plant improvements. Half of the plants use the NUMARC criteria to help identify areas for plant improvements but do not explicitly state that the NUMARC criteria are being used to identify vulnerabilities. The other half of this group of plants appear to use the percent contribution to CDF and sensitivity studies to help identify and evaluate the impact of plant improvements.

Table 9.5 List of plants with no vulnerability criteria defined in IPE.

Criteria used to define "vulnerabilities"	Plant type	Plant name
Criteria not defined; insights from IPE developed with objective of identifying plant improvements.	BWR 1	Big Rock Point
	BWR 4	Fitzpatrick, Cooper
	B&W PWR 2-Loop	Crystal River 3, Oconee 1,2&3, TMI 1
	W PWR 4-Loop	Indian Point 3, Comanche Peak 1&2
	BWR 5	LaSalle 1&2
	CE PWR 2-Loop or 3-Loop	Maine Yankee, Palo Verde 1,2&3
	W PWR 4-Loop Ice Condenser	Catawba 1&2, McGuire 1&2
Criteria not defined; IPE insights developed with objective of identifying plant improvements. NUMARC 91-04 criteria used to identify sequences that could lead to significant CDF reduction and/or to assess appropriateness of potential plant improvements.	BWR 3 (Isolation Condenser)	Dresden 2&3
	BWR 3	Quad Cities 1&2
	BWR 4	Brunswick 1&2
	W PWR 3-Loop	H.B. Robinson 2, Shearon Harris 1
	W PWR 4-Loop	Braidwood 1&2, Byron 1&2, Wolf Creek, Zion 1&2

9.2 Plant Vulnerabilities

One of the reporting guidelines in NUREG-1335 is that each licensee present "a list of any vulnerabilities identified by the review process, a concise discussion of the criteria used by the utility to define vulnerabilities, and the fundamental causes of each vulnerability." Most of the licensees clearly identify a criterion for identifying vulnerabilities. In addition, approximately 20% of the licensees *clearly state they have vulnerabilities according to their definitions* and go on to identify potential improvements in equipment, procedures, or training programs to address these vulnerabilities. The identified vulnerabilities and suggested plant improvements are discussed below.

9.2.1 Boiling Water Reactor Vulnerabilities

Using the various definitions of vulnerability, approximately 20% of the plants explicitly identify "vulnerabilities" in their submittals. The vulnerabilities tend to be plant-specific features that the licensee decides require resolution or, at least, further investigation. It should be noted that while only a fraction of the submittals actually identify vulnerabilities using their respective definitions, nearly all the plants go further and identify other areas warranting investigation for additional improvements. These other improvements are discussed in the next section of this report.

Only four licensees with boiling water reactors (BWRs) explicitly state that they have vulnerabilities. A summary of the BWR vulnerabilities is provided in Table 9.6. Although no common vulnerabilities are identified, some of the vulnerabilities can be considered generic to many BWRs. These potentially generic vulnerabilities are identified at three plants: Millstone 1, Hope Creek, and Susquehanna 1&2. The resolutions to the vulnerabilities identified in the submittals are also listed in Table 9.6. In some cases, the vulnerability was resolved before the IPE was completed and the resolution is reflected in the results, while in other cases, no resolution is suggested for the particular vulnerability except to follow research developments concerning the issue and accident management strategies in general.

Table 9.6 Summary of BWR plant vulnerabilities identified by licensees.

Plant name	Vulnerability description	Licensee approach to resolve vulnerability
BWRs 1/2/3s (Isolation Condensers)		
Millstone 1	Failure of isolation condenser makeup from city water supply and diesel fire-water pump, resulting in isolation condenser failure.	Procurement of portable diesel pump; implementation of procedures for supplying isolation condenser shell-side makeup following fire-water system failure.
	Operator failure to initiate isolation condenser to prevent safety relief valves (SRVs) from lifting in station blackout.	Not identified by licensee.
	Operator failure to restore/maintain RPV level following various accident scenarios.	Not 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 further.

Table 9.6 Summary of BWR plant vulnerabilities identified by licensees.

Plant name	Vulnerability description	Licensee approach to resolve vulnerability
BWRs 3/4s		
Fitzpatrick	Loss of 3/4 RHR loops (directly or through loss of RHR service water) due to catastrophic failure of either one of the 4.16 kV alternating current (AC) safety buses.	Consider procedure modification and operator training to allow manual alignment of fire protection system to the RHRSW system; installation of an RHRSW header cross-tie; installation of a tap for fire protection system on RHRSW loop B; and provision of a portable generator to charge safety DC batteries (to prevent SRV closure from battery depletion following loss of a 4.16 kV AC safety bus).
Hope Creek	Loss of switchgear or Class 1E Panel Room HVAC result in delayed loss of power and heat sinks.	Developed recovery procedure to supply alternate ventilation to prioritized rooms.
Susquehanna 1&2	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.	HPCI/RCIC backpressure trip setpoints raised 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 operator, severe power excursion will occur.	Considering deletion of, or installation of override switch on LPCI control delay to allow for immediate operator control of LPCI injection.
	During loss of offsite power or station blackout condensate storage tank (CST) keepfill function is lost; occurrence of waterhammer could cause failure of suppression pool cooling, causing containment failure, unless CST available for injection. Failure of the fire main as an injection source during station blackout will also result in vessel and containment failure.	Considering installation of independent, mobile diesel-powered diesel AC generator power supply for CST pumps.
BWRs 5/6s		
None	-----	-----

The Millstone 1 submittal identifies isolation condenser issues involving failure of the water supplies to the isolation condensers and failure of the operator to initiate the isolation condensers in time to prevent safety relief valves from lifting and subsequently sticking open (effectively causing the loss of isolation condenser operation) as vulnerabilities. The proposed resolution of the first vulnerability involves procurement of a portable diesel pump and corresponding procedural changes to supply the isolation condenser. The Millstone 1 submittal also identifies drywell steel shell melt-through (a containment performance issue) as a generic Mark I containment vulnerability. These issues could be applicable to the other BWR 1/2/3 plants. The licensee also identifies failure of the operator to restore and maintain reactor pressure vessel (RPV) level as an important operator action that meets the criteria for vulnerability. This issue is likely to be important at *all* BWRs.

The Hope Creek submittal identifies an electrical switchgear room heating, ventilation, and air conditioning (HVAC) vulnerability that will result in a delayed loss of power and available heat sinks. During the IPE analysis process, the utility developed a recovery procedure to address this vulnerability by aligning alternate means of cooling. Credit for this recovery procedure is taken in the final results and reduces the CDF by two orders of magnitude. The Susquehanna submittal identifies potentially generic vulnerabilities related to operation of the coolant injection system (high-pressure coolant injection (HPCI) system and reactor core isolation cooling (RCIC) system) during sequences with loss of containment heat removal. In addition, vulnerabilities related to the automatic injection from the low-pressure coolant injection (LPCI) system that can result in power excursions during an anticipated transient without scram (ATWS) accident are also identified in the Susquehanna submittal. These are issues that can be applicable to other BWR 3/4 plants.

No vulnerabilities are identified by the licensees with BWR 5/6 plants.

The Fitzpatrick submittal identifies a vulnerability that is unique to that plant. The vulnerability results from a previous plant modification to delete the residual heat removal (RHR) system loop selection logic that realigns RHR-related components to different safety-related buses. The electrical realignment results in a vulnerability involving loss of three out of four RHR loops (either directly or through the RHR service water (RHRSW) system) when either one of two safety-related 4.16 kV buses is lost. Fitzpatrick is considering procedure modifications and training for using firewater as a backup to RHR service water and installation of a cross-tie between RHRSW trains.

9.2.2 Pressurized Water Reactor Vulnerabilities

A summary of the vulnerabilities identified in the submittals for pressurized water reactors (PWRs) and the proposed resolutions are provided in Table 9.7. Among the 15 PWR licensees that identify vulnerabilities, certain vulnerability issues are common to more than one plant and can have generic implications. 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, are defined as vulnerabilities for Calvert Cliffs, Turkey Point, Summer, and Beaver Valley. The 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 due to the loss of the CCW system. For the licensees identifying this vulnerability, resolution of the issues involves implementation or consideration of alternate RCP seal cooling capabilities, inclusion in severe accident management guidelines, or consideration of new pump seal materials.

Table 9.7 Summary of PWR plant vulnerabilities identified by licensees.

Plant name	Vulnerability description	Licensee approach to resolve vulnerability
B&W PWR 2-Loop		
None	-----	-----
CE PWR 2-Loop		
Calvert Cliffs 1&2	Turbine driven pumps significant pair failure frequency due to 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).
Calvert Cliffs 1&2	Inadvertent actuation of emergency safeguard features actuation system, reactor protection system, or auxiliary feedwater (AFW) actuation system resulting from loss of two vital AC buses, which causes 2/4 actuation channels to fail to their actuated state.	Improved awareness through documentation in corrective action system, review of procedures, additional operator training.
	Reactor coolant pump seal and safety injection failure on loss of component cooling water (CCW).	Consideration of third CCW, pump, power modification, and reduction in likelihood of CCW leakage. Capping of downstream piping on all normally isolated drain/vent valves.
	Loss of switchgear HVAC, resulting in failure of both safety-related 4 kV buses with minimal time for start-up of standby unit or compensatory actions on running unit.	Pre-staged portable fans were put in place; development of procedure for switchgear HVAC recovery actions.
	Limited alternates to depressurization of reactor coolant system (RCS) during a steam generator tube rupture (SGTR) (primarily operator actions to depressurize RCS using main or auxiliary pressure spray).	Development of third depressurization procedure to depressurize pressurizer vent path.
	Minimal surveillance on critical condensate supply manual valve. This valve is necessary for operation when alternate water sources are needed for the auxiliary feedwater pumps.	Development of surveillance test, preventive maintenance, and performance evaluation procedures to periodically cycle critical condensate supply manual valves.
	Loss of main feedwater on a reactor trip. When steam generator feedwater pump (SGFP) control system failed to reduce pump speed, the SGFPs would trip on high-discharge pressure.	Addition of digital feedwater system to rapidly reduce pump speed on a trip, avoiding the high-discharge overpressure trip.
Millstone 2	Interfacing system LOCA (ISLOCA) RCP thermal barrier tube rupture	Modification planned for April 1997 to install relief valves to limit pressure build up in the reactor building closed cooling water system

Table 9.7 Summary of PWR plant vulnerabilities identified by licensees.

Plant name	Vulnerability description	Licensee approach to resolve vulnerability
Westinghouse PWR 2-Loop		
Kewaunee	In RHR system, normally open motor-operated valves (MOVs) in low-pressure system injection (LPSI) lines connected to reactor coolant system provide interfacing systems LOCA path during normal operations. Four pressure isolation valves (not leak tested) provide interfacing systems LOCA path in the RHR system.	Leak testing of additional four valves serving as boundary between reactor coolant system and a low-pressure system.
	Procedural guidance for determining where LOCA occurs not complete for failure of RHR pump suction valves.	Modification of emergency operating procedure to improve guidance to the operators in identifying and mitigating an ISLOCA.
Kewaunee	Internal flooding propagation from turbine building basement to adjoining areas containing safeguards equipment. Doors that swing out of the affected room cannot withstand the flooding forces and fail.	Modification of swing direction of doors separating turbine building basement from areas containing safeguards equipment.
	Internal flooding due to failure of circulating water expansion joint at main condenser. Routine inspections of expansion joints were not conducted.	Improvement of inspection method for rubber expansion joints to identify potential flooding problems.
	Upon loss of offsite power or station blackout, 3/6 air compressors are unavailable, reducing reliability of instrument air. No procedural steps for maintaining a swing bus energized for two of the remaining compressors.	Modification of emergency operating procedures to ensure power is available to 2 instrument air compressors.
	Makeup valve to condenser fails open on loss of instrument air (IA) or control power, causing condensate diversion from storage tanks to main condenser, reducing quantity available to auxiliary feedwater pumps.	Design information being reviewed to determine basis for current fail safe position of makeup valve.
	Failure of the auxiliary feedwater system contributes approx. 30% to total CDF; reliability of turbine-driven auxiliary feedwater pump directly relates to approx. 20% of CDF.	Modifications to improve reliability of turbine-driven auxiliary feedwater pump are scheduled.
	Air compressors are subject to frequent maintenance outages, making the station and instrument air system less reliable.	Design modification initiated to remove the two older air compressors and replace them with air-cooled air compressors.
	Charging pump relief valves opening can divert flow back to volume control tank, affecting ability of pumps to provide reactor coolant system makeup and reactor coolant pump seal cooling.	Actions being investigated to correct problem of diversion of chemical and volume control system water.

Table 9.7 Summary of PWR plant vulnerabilities identified by licensees.

Plant name	Vulnerability description	Licensee approach to resolve vulnerability
Westinghouse PWR 3-Loop		
Beaver Valley 1	Upon loss of all emergency switchgear ventilation, operator fails to promptly provide alternate cooling.	Possible change to procedures to provide more explicit guidance on how to establish sufficient alternate cooling in event that both emergency switchgear ventilation fan trains fail.
	Failure of breakers that perform transfer of 4.16 kV non-emergency buses from unit station service transformers to system station service transformers, leading to loss of emergency AC power (i.e., in conjunction with failures of the diesel generators).	Development of procedure to repair or change out failed breakers and provide training.
	Limited recovery time upon loss of AC power and subsequent battery depletion at 8 hours, followed by steam generator level instrumentation loss and turbine-driven AFW pump failure.	Considering providing more explicit guidance on battery load shedding or providing some means of battery charging during loss of all AC power.
	Reactor trip breaker failure makes it unlikely that operators can remove power to control rods prior to RCS pressure peaking during ATWS scenarios initiated by loss of main feedwater.	Considering adding capability for operator to remove power from bus.
	In a station blackout, diesel generators of other unit cannot be connected to emergency buses of affected unit, since 4.16 kV emergency AC buses between units are not cross-tied.	Cross-tie connecting 4.16 kV normal buses of both units will be installed; existing procedures will be revised and training will be provided 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 alternate seal cooling systems. Also considering modifications to address RCP seal integrity for loss of all seal cooling.
	Loss of offsite power delays reactor trip, resulting in power-operated relief valves (PORVs) lifting and possibly sticking open, potentially causing small LOCA (which shortens time for power recovery).	Considering eliminating PORV challenge by defeating 100% load rejection capability.
	Containment bypass sequences dominated by steam generator tube rupture, resulting in core damage and ISLOCAs.	Changes to plant procedures and training to enhance operator response to such bypass sequences. Improve guidance to the operators to close key valve during an ISLOCA.
	Containment overpressurization resulting from RCS blowdown, early hydrogen burns, and direct containment heating.	Considering extending procedures to all core damage sequences for reducing RCS pressure prior to vessel breach. Procedures will instruct on alignment for recirculation from containment sump back to vessel even if core damage has occurred. Also considering alternate modes for injecting water into reactor cavity and conserving reactor water storage tank (RWST) inventory.

Table 9.7 Summary of PWR plant vulnerabilities identified by licensees.

Plant name	Vulnerability description	Licensee approach to resolve vulnerability
Beaver Valley 1	Pressurizer PORV block valve alignment provides 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.	Considering extending procedures to align recirculation from containment sump back to reactor vessel. Considering alternate modes for injecting water, including natural flow of water to reactor cavity and use of diesel-driven fire system pump. Throttling of quench spray pumps also considered.
	Same as Beaver Valley 1.	Same as Beaver Valley 1.
	Loss of both primary and secondary heat removal in injection phase primarily due to failure of the turbine-driven pump during a station blackout (unavailability due to test/maintenance on pump).	The licensee considered development of Severe Accident Management Guidelines to be sufficient to address vulnerability.
	Induced RCP seal LOCAs with loss of primary coolant makeup or adequate heat removal in injection or recirculation phase.	Same as above.
	Small LOCA with loss of primary coolant makeup or adequate heat removal in 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 recirculation phase. 85% of frequency is related to failure of low-pressure recirculation (due to 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 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 recirculation phase. Failure of low-pressure recirculation (due to failure of RWST signal) following successful high-pressure injection for medium LOCAs.	Same as above.
	Failure of reactivity control primarily due to reactor trip failure following total loss of IA.	Same as above.
	Steam generator tube ruptures with loss of effective inventory makeup. This category consists of SGTR events that result in containment bypass.	Replacement of steam generators with new design that should lower the expected SGTR frequency.

Table 9.7 Summary of PWR plant vulnerabilities identified by licensees.

Plant name	Vulnerability description	Licensee approach to resolve vulnerability
Surry 1&2	Internal flooding in turbine building due to ruptures of 1 of 4 circulating water (CW) inlet motor-operated valves.	Considering flood mitigation procedural and training improvements. These include inspection/maintenance improvements, use of submersible MOV operators, improved sump pump capacity/reliability, and back flow prevention in drain lines.
	Internal flooding in turbine building due to rupture of 1 of 4 service water expansion joints in valve pits.	Same as above.
	Internal flooding in turbine building due to severe ruptures of 1 of 4 service water isolation motor-operated valves in valve pits.	Same as above.
	Internal flooding in turbine building due to rupture of service water pipe in valve pit on CW inlet pipe.	Same as above.
Turkey Point 3&4	Loss of CCW, combined with "B" charging pump unavailability, leading to reactor coolant pump seal LOCA. The high-pressure recirculation function is predicted to fail and result in core damage.	Charging pump operation with supplemental cooling; installation of service water hose connections on "A" and "C" pumps, allowing operation independent of CCW.
Westinghouse PWR 4-Loop		
Haddam Neck	Operator failure to transfer to sump recirculation following large or medium LOCA, due to limited time available prior to draindown of refueling water storage tank level below high-pressure safety injection (HPSI) pump net positive suction head (NPSH) requirements.	Analysis is being performed to justify stopping LPSI pumps much earlier, giving operators more time to transfer. Also, increased emphasis in operator training on timely sump transfer emergency operating procedure.
Millstone 3	Station blackout major contributor to public risk.	Prioritized recovery of offsite power steps in procedure training; developed procedure for severe weather conditions; air-cooled diesel generator to be added; numerous activities in response to station blackout rule.
	ISLOCA major contributor to public risk.	Open-valve alarm to be added as part of RHR autoclosure interlock removal; 1988 emergency exercise involved ISLOCA in RHR pump suction line; RHR system walkdown in emergency safeguard feature building to determine characteristic of potential releases.
	AFW and feed and bleed failures are in many accident sequences.	Prioritized operator training on AFW system, recovery of main feedwater, and primary feed and bleed procedure.
	Failure of containment sump recirculation found in dominant sequences.	Implemented design change for cold leg recirculation array; prioritized training on transfer to sump recirculation steps in emergency operating procedure; prioritized maintenance of service water to containment recirculation cooler MOVs.

Table 9.7 Summary of PWR plant vulnerabilities identified by licensees.

Plant name	Vulnerability description	Licensee approach to resolve vulnerability
Millstone 3	Seismic-induced station blackout major risk contributor; dominated by diesel generator oil cooler anchor bolt failure.	Anchor bolts replaced.
Salem 1&2	RHR valves direct initial leakage to 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 possibility of ISLOCA).	Considering revision of emergency operating procedures related to ISLOCAs.

Auxiliary feedwater (AFW) system turbine-driven pump reliability is a common issue defined as a vulnerability at Calvert Cliffs, Summer, Millstone 3, and Kewaunee. Calvert Cliffs identifies a plant-specific design problem which results in removal of both turbine-driven AFW pumps from service any time maintenance is required on one of the pump steam admission valves. A modification has been made in which additional valves have been added on the steam lines that allow continued operation of one AFW pump when the other is out for maintenance. The Kewaunee submittal identifies a flow path that diverts condensate from the condensate storage tank (CST) to the main condenser and therefore reduces the quantity available to the AFW pumps for secondary cooling. The diversion path, which appears to be unique to this plant, 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 fails closed upon loss of air or control power (the basis for the current fail safe position of the valve was being reviewed).

The Summer, Haddam Neck, and Millstone 3 submittals identify common vulnerabilities related to the failure of the operator to switch over from the injection phase to the recirculation phase of coolant injection. This vulnerability is generic to many other PWRs which require this operator action. At Summer, the switchover is partially automated as the sump recirculation valves are opened upon a low reactor water storage tank (RWST) level. The operator must manually close the RWST suction valves. Failure of the RWST low-level signal is identified as the vulnerability even though manual operator action to close the valves is not credited in the submittal. Based upon this conservatism, the resolution listed in the Summer IPE was to address accidents during the recirculation phase in the accident management guidelines. In the Haddam Neck IPE, failure of the operator to transfer suction during a large or medium LOCA represents a significant contributor to the CDF. A high-operator error probability is assigned to these scenarios due to limited time available to establish sump recirculation prior to draindown of the RWST level below HPSI net positive suction head requirements. Since the RWST inventory is primarily depleted by the operation of the low-pressure safety injection (LPSI) pumps in the injection phase, analysis is being performed to justify stopping the LPSI pumps at a much earlier step in the sump transfer procedure, which will afford the operators more time to perform the switchover procedure.

Another common vulnerability identified in the Calvert Cliffs and Beaver Valley submittals is the loss of critical switchgear HVAC equipment, resulting in loss of emergency AC power. This vulnerability may be applicable to additional plants. Beaver Valley is reviewing alarm response procedures to determine if they can provide more explicit guidance on how to establish sufficient alternate cooling. Calvert Cliffs has implemented a procedure to use staged portable fans for alternate cooling.

The Surry, Kewaunee, and Salem submittals identify internal flooding issues as vulnerabilities. For example, at Surry, flooding from failed cooling water components is the most significant vulnerability at the plant since the source of water is gravity fed and there is little means of isolating the failure. Proposed resolutions to the identified vulnerabilities at these plants include revision of flooding procedures and training, periodic inspection and replacement of components identified as potential flood initiators, and improvement of sump pump protection from flood effects.

Interfacing system LOCAs occurring from multiple valve failures or through normally open valves are identified as vulnerabilities at Salem, Kewaunee, Millstone 3, and Beaver Valley. The resolutions in these submittals include procedure improvements already made or under consideration to address improved valve testing, LOCA identification and isolation, and a modification to change normally open valves to close.

Other vulnerabilities are defined, but by only one PWR, and involve such things as inadequate surveillances 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. Some of these vulnerabilities can be considered generic to many PWRs. Millstone 3 also identifies an external-event-related vulnerability--a seismic-induced station blackout scenario that is dominated by diesel generator oil cooler anchor bolt failure. Further discussion of these vulnerabilities and proposed resolutions is provided in Table 9.7.

9.3 Plant Improvements

As previously discussed, a major goal of the IPE process is to identify plant-specific vulnerabilities to severe accidents that can be fixed with low-cost improvements. It is clear from the submittals, however, that most licensees went beyond this goal and identified other improvements (over 500 were identified by the plants) worthy of consideration or implementation, even though no specifically associated vulnerabilities were identified. Many of the plant improvements are potentially generic, with the most often cited BWR improvements addressing station blackout concerns and the PWR improvements addressing both loss of power and loss of reactor coolant pump seal cooling concerns. Changes aimed at improving core cooling or injection reliability, particularly for those systems or portions of systems that can operate during loss of AC power, are often identified in submittals for both PWR and BWR plants. Improvements to address internal flooding and interfacing systems LOCAs (ISLOCAs) are identified more often in PWR IPEs than in BWR IPEs. Other less frequently cited and plant-specific improvements are identified to address a number of other accident class issues at individual plants. A summary of the general areas where plant improvements were listed by the licensees is provided in Tables 9.8 and 9.9, respectively.

Table 9.8 Areas of BWR plant improvement identified by licensees.

Plant	Area of improvement										
	AC reliability	DC reliability	Coolant injection systems	Decay heat removal (DHR) systems	Support systems	ATWS	RCP seal LOCAs	Internal flooding	Interfacing system LOCAs	Containment	Miscel-laneous
BWR 1/2/3											
Big Rock Point											✓
Nine Mile Point 1		✓	✓	✓	✓		✓				✓
Oyster Creek	✓	✓		✓	✓						✓
Dresden 2&3			✓	✓							
Millstone 1	✓	✓	✓	✓	✓			✓	✓		
BWR 3/4											
Monticello		✓	✓	✓	✓	✓			✓		✓
Pilgrim											
Quad Cities 1&2	✓			✓							
Browns Ferry 2											
Brunswick 1&2	✓	✓		✓							
Cooper		✓	✓		✓						✓
Duane Arnold				✓							✓
Fermi 2				✓	✓						✓
Fitzpatrick	✓		✓	✓	✓			✓			✓
Hatch 1&2		✓		✓	✓						
Hope Creek					✓						
Limerick 1&2	✓		✓	✓	✓						
Peach Bottom 2&3	✓			✓							
Susquehanna 1&2		✓	✓	✓	✓	✓					✓
Vermont Yankee	✓	✓	✓	✓	✓	✓					✓
BWR 5/6											
LaSalle 1&2			✓								
Nine Mile Point 2	✓				✓			✓			✓
WNP 2	✓		✓			✓		✓			✓
Clinton	✓		✓		✓	✓				✓	✓
Grand Gulf	✓		✓		✓						✓
Perry	✓	✓	✓	✓		✓		✓			✓
River Bend	✓	✓	✓	✓	✓					✓	

Table 9.9 Areas of PWR plant improvement identified by licensees.

Plant	Area of improvement											
	AC reliability	DC reliability	Coolant injection systems	Decay heat removal (DHR) systems	Support systems	ATWS	RCP seal LOCAs	Steam generator tube ruptures	Internal flooding	Interfacing system LOCAs	Containment	Miscellaneous
B&W PWRs												
ANO 1	✓	✓	✓	✓						✓	✓	✓
Crystal River 3					✓							✓
Davis-Besse	✓	✓	✓	✓	✓		✓	✓		✓	✓	✓
Oconee 1,2&3			✓				✓		✓			
Three Mile Island 1				✓			✓	✓				✓
CE PWRs												
ANO 2	✓		✓	✓						✓	✓	
Calvert Cliffs 1&2				✓	✓	✓	✓	✓				✓
Fort Calhoun									✓	✓		
St Lucie 1&2				✓								
Maine Yankee		✓			✓							✓
Millstone 2		✓	✓	✓	✓			✓		✓		
Palisades					✓				✓		✓	✓
Palo Verde 1,2&3	✓			✓								✓
San Onofre 2&3					✓							
Waterford 3	✓	✓	✓	✓	✓						✓	
Westinghouse 2-Loop PWRs												
Ginna			✓		✓			✓			✓	
Kewaunee				✓	✓		✓		✓	✓		
Point Beach 1&2	✓	✓		✓					✓			✓
Prairie Island 1&2	✓		✓		✓			✓	✓			✓
Westinghouse 3-Loop PWRs												
Beaver Valley 1	✓	✓	✓		✓	✓	✓	✓				✓
Beaver Valley 2	✓	✓	✓		✓	✓	✓	✓			✓	✓
Farley 1&2	✓			✓	✓		✓					
Robinson 2	✓	✓		✓	✓		✓	✓	✓	✓		
North Anna 1&2			✓		✓				✓	✓		
Shearon Harris	✓	✓										
Summer	✓		✓	✓	✓							✓
Surry 1&2	✓				✓				✓			✓
Turkey Point 3&4	✓				✓							✓
Westinghouse 4-Loop PWRs												
Braidwood 1&2	✓											
Byron 1&2	✓											
Callaway	✓		✓	✓	✓		✓		✓			✓
Catawba 1&2					✓		✓				✓	✓
Comanche Peak 1&2			✓	✓	✓		✓					

Table 9.9 Areas of PWR plant improvement identified by licensees.

Plant	Area of improvement											
	AC reliability	DC reliability	Coolant injection systems	Decay heat removal (DHR) systems	Support systems	ATWS	RCP seal LOCAs	Steam generator tube ruptures	Internal flooding	Interfacing system LOCAs	Containment	Miscel-laneous
D.C. 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	✓		✓	✓			✓					
Wolf Creek				✓	✓		✓		✓		✓	
Zion 1&2			✓									

Using the plant grouping pattern mentioned throughout this report, the following sections summarize the plant improvements documented in the submittals. The status of these improvements (implemented, planned, or under evaluation) is as of the dates when each IPE submittal was completed. In many cases, a few years have passed since the submittal date. Hence, some of the planned improvements or those under evaluation may or may not have been implemented as of the date of this report.

9.3.1 BWR Plant Improvements

Many of the plant improvements identified in the BWR submittals are procedural/operational changes (approximately 40%). Most of the BWR procedural/operational improvements address station blackout scenarios or operation of coolant injection systems. A significant number also address operation of support systems since they can impact different accident classes. Only a small percentage specifically address accident classes that are minor contributors to core damage (e.g., internal flooding, ATWS, LOCAs, and ISLOCAs). The most common procedural/operational plant improvements identified in the BWR submittals include incorporating the IPE results into operator training programs, developing or improving procedures for battery load shedding or cross-tieing electrical buses, and arranging for alternate room cooling upon loss of HVAC. Of the procedural/operational improvements identified, approximately 45% have been implemented, but not all have been credited in the BWR IPEs. Approximately 20% have been implemented and credited in the BWR IPEs.

Most of the plant improvements identified in the BWR submittals are design/hardware changes (approximately 50%). As is the case with procedural/operational improvements, most of the BWR design/hardware improvements address station blackouts scenarios, operation of coolant injection systems, or operation of support systems. Only a small percentage specifically address accident classes that are minor contributors to core damage (e.g., internal flooding, ATWS, LOCAs, and ISLOCAs). The most common design/hardware plant improvements identified in the BWR submittals include improving or replacing diesel generators, establishing new offsite power lines, establishing a hard pipe vent path, purchasing a portable generator for charging station batteries, and establishing flow paths for firewater injection into the vessel. Of the design/hardware improvements identified, approximately 65% have been implemented, but not all have been credited in the BWR IPEs. Approximately 25% have been implemented and credited in the BWR IPEs.

Few of the identified BWR plant improvements are maintenance-related changes. Most of the BWR maintenance improvements address operation of support systems, with a smaller number addressing prevention of internal flooding events. The maintenance-related plant improvements identified in the BWR submittals are more diversified than the other plant improvements. The most common improvements are related to instrumentation calibrations and inspection of piping and seals. Of the maintenance improvements identified, approximately 50% have been implemented and few have been credited in the BWR IPEs. Approximately 10% have been implemented and credited in the BWR IPEs.

More details concerning the BWR plant improvements identified for each group of plants are provided in the following sections. A summary of the implementation status (as of the IPE submittal dates) for the BWR plant improvements is provided in Table 9.10.

Table 9.10 BWR plant improvement implementation by licenses as of the date of IPE submittal.

Plant	Number implemented	Plant	Number implemented
BWR 1/2/3			
Big Rock Point	unknown	Nine Mile Point 1	some
Oyster Creek	most	Dresden 2&3	some
Millstone 1	most		
BWR 3/4			
Monticello	most	Pilgrim 1	*
Quad Cities 1&2	all	Browns Ferry 2	*
Brunswick 1&2	most	Cooper	none
Duane Arnold	some	Fermi 2	none
Fitzpatrick	some	Hatch 1&2	all
Hope Creek	unknown	Limerick 1&2	all
Peach Bottom 2&2	all	Susquehanna 1&2	some
Vermont Yankee	some		
BWR 5/6			
LaSalle 1&2	all	Nine Mile Point 2	some
WNP 2	none	Clinton	some
Grand Gulf 1	none	Perry	some
River Bend	some		

* No plant improvements identified because of IPE process.

9.3.1.1 BWR 1/2/3s with Isolation Condensers

Most of the licensees in this group have implemented improvements to address station blackout, and indicate they are planning or evaluating other changes to further address this accident class. These changes are introduced or supported by the findings of the IPEs and past PRA analyses. In most cases, these improvements are also a result of evaluations in response to the station blackout rule.

As indicated in Table 9.8, three of the licensees in this plant group identify specific improvements, many of which have been implemented, to address the power reliability issue for station blackout. These improvements include improving both the AC and DC reliability and could be applied at many other plants. For AC improvements, Oyster Creek has provided an interconnection to an alternate offsite power source. Millstone 1 has upgraded an offsite power line and improved its multiunit site cross-tie capability, as well as introducing a number of onsite AC equipment changes to improve reliability and performance. DC power reliability improvements and changes to extend battery life during station blackout are also being considered. Millstone 1, Oyster Creek, and Nine Mile Point 1 specifically mention in their IPE submittals that the addition of portable chargers is under evaluation. Nine Mile Point 1 has apparently added DC power capacity to reduce the requirements of load shedding but is also examining additional load shedding provisions. Millstone 1 is making some maintenance changes to decrease the outage time for some batteries.

Another general area of improvement addressing station blackout, but also of value in other types of accident scenarios, involves improved decay heat removal (DHR) capability and reliability. For instance, three of the licensees (one for a dual unit site) specifically mention improvements made to the isolation condensers in their submittals. One of these sites, Millstone 1, has made valve operator replacements and apparently added a portable diesel pump for extended shell-side water supply to the isolation condenser during station blackout. The other site, Dresden 2&3, cites a planned change to procedures so the operators will have better guidance for maintaining operation of the isolation condensers during an extended blackout. Additionally, Oyster Creek has apparently implemented a firewater backup spray capability for use during an extended blackout condition. Two plants, Nine Mile Point 1 and Oyster Creek, specifically mention the hard pipe improvements made and credited in the IPE in response to Generic Letter 89-16. Nine Mile Point 1 and Millstone 1 also specifically mention emergency operating procedure (EOP) changes either being implemented or under evaluation to improve overall containment venting reliability. Most of these improvements could be applicable to other plants in this plant group.

Several licensees also mention plant improvements related to coolant injection systems that could also be applied to other BWRs. Millstone 1 has apparently replaced the motors on the LPCI and core spray pumps with air-cooled motors to reduce the support system dependencies. Dresden 2&3 has implemented procedure and training changes to realign low-pressure cooling suction from the condensate storage tank (CST) whenever suppression pool cooling cannot be established. This is to eliminate possible pump net positive suction head (NPSH) concerns during any accident class resulting in a high-pool temperature condition. Nine Mile Point 1 is considering improved maintenance guidance on calibration of the reactor low-pressure permissive logic and a design change to provide local manual capabilities to some air-operated valves.

Other improvements are specifically cited in the submittals that address other classes of accidents. These improvements can be plant specific or can be applied to the entire plant group. For example, a plant-specific improvement at Oyster Creek involves the implementation of an improved reactor overfill protection system to reduce high-level excursions. More generic improvements include a new surveillance procedure to help reduce the probability of an ISLOCA in the reactor water cleanup (RWCU) system and an equipment qualification boundary program to reduce the probability of equipment failure during various high-energy line breaks. Both of these improvements were implemented at Millstone 1. Oyster Creek and Nine Mile Point 1 also mention the broader use of IPE results in future operator training. Finally, Nine Mile Point 1 implemented and credited replacement of recirculation pump seal cartridges to lessen recirculation pump seal effects.

Regarding the quantitative benefit of most of these changes, one of the most noteworthy was that Dresden 2&3 indicated that the low-pressure suction realignment mentioned above should reduce the estimated total plant CDF by about 2E-5/ry. This level of reduction is similar to each unit's CDF, which is reported as 2E-5/ry. All of the station blackout improvements have certainly made the estimated core damage frequencies lower than they would be without the improvements. Since some of these changes are under evaluation, further reduction of the station blackout contribution is possible.

9.3.1.2 BWR 3/4s

Like the BWR 1/2/3s, nearly all the plants in this group have made changes to address the commonly identified concern of station blackout. No specific items are identified in the Browns Ferry submittal, and Pilgrim, which had already made a number of PRA-supported modifications even before the IPE, does not highlight any *new* changes. In the past, Pilgrim had added a third diesel generator and also a backup nitrogen supply to extend the use of safety relief valves during blackout and loss of air events. These blackout-related changes are often, but not always, part of the licensees' station blackout rule responses and are further identified and supported by the findings in the IPEs. As with the BWR 1/2/3s, these improvements take various forms and include changes to address AC, DC, and other system reliabilities and enhancements. At least half the sites (including the dual unit sites for Quad Cities 1&2, Brunswick 1&2, Limerick 1&2, and Peach Bottom 2&3) cite both procedural and hardware upgrades to improve AC system reliability that could be applied at most BWRs. Examples of these improvements are implementation of bus cross-tie and bus recovery enhancements at many of the sites, startup auxiliary transformer upgrades implemented at Brunswick 1&2, procedure changes being evaluated at Fermi 2 to address partial offsite power loss, two new diesels installed at Quad Cities 1&2, evaluation of cross-connecting fuel oil sources at Vermont Yankee, and implementation of an underground and more reliable source of offsite power at Peach Bottom 2&3. Potentially generic DC power improvements identified in the IPE submittals include new DC power restoration procedures implemented at Brunswick 1&2, DC load shedding procedure enhancements being evaluated at Cooper and Vermont Yankee and implemented at Monticello, and replacement of station battery chargers implemented at Hatch 1&2.

The addition of firewater backup capability is another potentially generic improvement identified in at least four of the submittals. Monticello and Limerick 1&2 apparently implemented changes so as to be able to inject firewater into the RHR paths for injection, which should be beneficial in loss of RHR accidents. Vermont Yankee was evaluating a similar change. The Fitzpatrick and Cooper licensees state that the use of firewater as a backup to diesel cooling is being planned or evaluated, most likely to deal with perceived service water weaknesses.

Other blackout-related changes mentioned in individual submittals include procedure and hardware improvements for loss of ventilation at Hatch 1&2, the switch to DC-backed instrument buses for safety relief valve power at Monticello, and the switch to a DC-powered bus for RCIC room exhaust fans being planned at Fitzpatrick. Hope Creek is planning a detailed reevaluation of service water pump requirements. This is more of an analytical change rather than an actual change to plant hardware, but could yield a 50% reduction in the station blackout contribution in the plant's submittal.

Another common area of improvement identified in the BWR 3/4 IPE submittals is coolant injection systems. Monticello is apparently evaluating procedure and training changes for the use of alternate low-pressure injection systems to avoid pump cavitation concerns under loss of RHR conditions. The Fitzpatrick submittal indicates that changing the control rod drive (CRD) flow control valves so as to fail in their current position or in the open position upon loss of air is under evaluation. The Vermont Yankee submittal indicates that changes are being evaluated to enhance the use of CRD whenever HPCI and RCIC might be simultaneously unavailable. At least Monticello and Vermont Yankee are evaluating improvements for replenishing the CST in long-term emergency situations. The Fitzpatrick and Vermont Yankee submittals mention the evaluation of procedure improvements to lessen the chance of loss of HPCI and RCIC when performing emergency depressurization. All of these improvements could potentially be made at additional plants.

Numerous other improvements are being considered by licensees. For example, the Fermi 2 submittal indicates that standby system realignment procedure enhancements are being planned. The Hatch 1&2 submittal indicates that some loss-of-HVAC design and procedural improvements have been implemented for pump rooms and the intake structure. A few of the IPE submittals mention IPE-related operator training as being planned or implemented, and over half of the submittals specifically mention implementation of a hardened containment vent path and crediting it in the IPE. Vermont Yankee indicated that cross-connecting the two standby liquid control (SLC) system trains was being considered to increase the system redundancy in responding to an ATWS event. The Fitzpatrick submittal indicates that measures for protecting some equipment from internal flooding effects were being evaluated. Finally, the Monticello IPE submittal indicates that interlocks to prevent opening a shutdown cooling suction valve on the RHR system when its associated torus return valve is open have been installed.

Many of the station blackout improvements have certainly made the estimated core damage frequencies lower than they would be without the improvements. Explicit quantitative evaluations of these effects are not given. However, since many of the changes summarized above are under evaluation, some reduction of plant CDFs may be achieved.

9.3.1.3 BWR 5/6s

Like the previous groups of BWRs, station blackout improvements are commonly identified in the submittals for the BWR 5 and 6 designs. In this case, most are identified as planned or under evaluation while a few were apparently implemented at the time of the submittals. A commonly identified improvement involves enhancing the use of the high-pressure core spray (HPCS) diesel (a Division III power source) for powering Division I or II whenever these two latter divisions are lost. Nine Mile Point 2 has apparently implemented procedure changes to this effect, Grand Gulf is planning such a change, and Perry 1 has implemented a permanent cross-tie arrangement. The River Bend licensee reports a service water valve change to address the loss of service water flow to the HPCS diesel to decrease the station blackout contribution to CDF.

Other AC power improvements identified in the submittals as implemented or being evaluated or planned included changes to offsite and onsite power maintenance and recovery procedures at most of the sites. Both the River Bend and Perry 1 submittals identify specific DC power improvements. The Perry 1 submittal indicates a cross-tie capability for the batteries has been implemented, while the River Bend submittal indicates the use of a portable charging capability is under evaluation. Firewater backup capability improvements are also identified as under evaluation at Perry 1.

Improvements related to coolant injection systems were identified by almost all of the plants in this group. A "sneak circuit" problem at LaSalle, which can potentially cause the loss of RCIC whenever AC power is lost and then subsequently restored, has received attention through a procedure change and additional operator training. Other potentially generic improvements were also identified by other licensees. The licensee for WNP 2 indicated that raising the RCIC low-vessel level actuation setpoint was being evaluated. Revision of procedures to address bypassing of the RCIC high-steam tunnel trips was implemented at Perry and was under consideration at Grand Gulf.

The Perry submittal also indicated that implementation of automatic vessel depressurization for non-ATWS events was being evaluated. The Grand Gulf submittal indicated that alternate operation of low-pressure core spray and RHR pumps during a loss of standby service water was being evaluated as a means of extending the time to seal failure. Finally, both Perry and River Bend listed plant improvements related to using firewater to provide coolant to the vessel.

Other plant changes are also identified, including a variety of procedure changes to address pump room and control building HVAC loss implemented at Nine Mile Point 2 and River Bend, and hydrogen igniter power source and procedure changes being evaluated at River Bend for station blackout some. ATWS-related improvements are identified as planned or under evaluation in three submittals. These improvements involve training for scram hardware failures at Clinton, the automatic inhibiting of the automatic depressurization system (listed in both the WNP 2 and Perry submittals), and a possible alternate boron injection capability at Perry. The Perry submittal also indicates the planned use of plant-specific ATWS information to improve operator performance. Finally, both Perry and River Bend stated that a containment vent system was being evaluated. Most of these improvements could be applicable to other BWRs.

Regarding the quantitative effects of some of these improvements, the WNP 2 submittal indicates that a reduction of approximately 50% in its CDF might be realized for its backfeed power modification. The licensee for Perry 1 estimates a reduction of approximately 1E-5/ry for ATWS changes if implemented. At River Bend, an estimated reduction in station blackout from its current contribution of approximately 90% to the total CDF to less than a 60% contribution might be realized from implementation of all blackout-related changes being evaluated. The "sneak circuit" procedure and training changes at LaSalle 1&2 are estimated to reduce the contribution to the total CDF for sequences involving this issue from approximately 20% to 1%.

9.3.2 PWR Plant Improvements

Most of the plant improvements identified in the PWR submittals are procedural/operational changes (approximately 50%). The PWR procedural/operational improvements address station blackout, or loss of reactor coolant pump seals; steam generator tube rupture scenarios, or operation of coolant injection, DHR, or support systems. Only a small percentage specifically address accident classes that are minor contributors to core damage (e.g., internal flooding, ATWS, LOCAs, and ISLOCAs). The most common procedural/operational plant improvements identified in the PWR submittals include incorporating the IPE results into operator training programs or developing or improving procedures for battery load shedding, switching over emergency coolant injection to the recirculation mode, arranging for alternate RCP seal cooling, and performing feed and bleed operations. Of the procedural/operational improvements identified, approximately 35% have been implemented, but not all have been credited in the BWR IPEs. Approximately 25% have been implemented and credited in the BWR IPEs.

Many of the plant improvements identified in the PWR submittals are design/hardware changes (approximately 40%). Most of the PWR design/hardware improvements address station blackouts scenarios, RCP seal cooling, and operation of coolant injection, DHR, or support systems. Only a small percentage specifically address accident classes that are minor contributors to core damage (e.g., internal flooding, ATWS, LOCAs, and ISLOCAs). The most common design/hardware plant improvements identified in the PWR submittals include improving or replacing diesel generators and auxiliary feedwater pumps, replacing RCP seal material, replacing battery chargers, purchasing a portable generator for charging station batteries, and establishing flow paths for alternate RCP seal cooling. Of the design/hardware improvements identified, approximately 45% have been implemented, but not all have been credited in the BWR IPEs. Approximately 30% have been implemented and credited in the BWR IPEs.

Few of the identified PWR plant improvements are maintenance-related changes. Most of the PWR maintenance improvements address operation of support systems preventing internal floods or ISLOCAs. The maintenance-related plant improvements identified in the PWR submittals include inspection of piping and other components, valve leak testing, and improved maintenance procedures for HVAC systems and diesel generators. Of the maintenance improvements identified, approximately 60% have been implemented and few have been credited in the BWR IPEs. Approximately 45% have been implemented and credited in the PWR IPEs.

More specific details concerning the PWR plant improvements identified for each group of plants are provided in the following sections. A summary of the implementation status (as of the IPE submittal dates) for the PWR plant improvements is provided in Table 9.11.

Table 9.11 PWR plant improvement implementation by licenses as of the date of IPE submittal.

Plant	Number implemented	Plant	Number implemented
B&W PWRs			
ANO 1	some	Crystal River 3	all
Davis-Besse	some	Oconee 1,2&3	none
Three Mile Island 1	none		
CE PWRs			
ANO 2	some	Calvert Cliffs 1&2	some
Fort Calhoun	some	St. Lucie 1&2	all
Maine Yankee	unknown	Millstone 2	all
Palisades	none	Palo Verde 1,2&3	all
San Onofre 2&3	all	Waterford 3	none
Westinghouse 2-loop PWRs			
Ginna	most	Kewaunee	most
Point Beach 1&2	most	Prairie Island 1&2	most
Westinghouse 3-loop PWRs			
Beaver Valley 1	some	Beaver Valley 2	some
Farley 1&2	most	Robinson 2	most
North Anna 1&2	all	Shearon Harris	some
Summer	most	Surry 1&2	all
Turkey Point 3&4	all		
Westinghouse 4-loop PWRs			
Braidwood 1&2	none	Byron 1&2	none
Callaway	some	Catawba 1&2	some
Comanche Peak 1&2	all	D.C. Cook 1&2	none
Diablo Canyon 1&2	none	Haddam Neck	some
Indian Point 2	some	Indian Point 3	none
McQuire 1&2	some	Millstone 3	most
Salem 1&2	none	Seabrook	none
Sequoyah 1&2	none	South Texas 1&2	*
Vogtle 1&2	all	Watts Bar 1	none
Wolf Creek	none	Zion 1&2	unknown

* No plant improvements identified because of IPE process.

9.3.2.1 B&W PWRs

The IPEs for this group of plants identify improvements that typically address a variety of different concerns as opposed to a single common issue such as station blackout for the BWRs. Improvements related to RCP seal failure, however, are identified in three of the five submittals. TMI 1 is evaluating training enhancements for maintaining seal injection flow and tripping of the RCPs to reduce the chance of RCP seal failure. Oconee 1,2&3 is evaluating procedural actions to supply RCP seal water during loss of power conditions. The Davis-Besse submittal addresses the evaluation of procedure enhancements for isolating the seal return flow when seal cooling to the RCPs has been lost and for dealing with loss of injection during a seal LOCA.

The licensees for Davis-Besse and ANO 1 either have implemented or are evaluating ISLOCA procedure and training modifications which will lessen the chance of interfacing system LOCAs at these plants. These two licensees also address station blackout concerns, with ANO 1 implementing the use of a fifth swing air-cooled diesel and added battery capacity, and Davis-Besse evaluating procedural changes to add redundancy capability to power supplies, provide for DC load shedding, and guide the replenishment of diesel fuel oil for long-term diesel use.

Additional individual improvements being evaluated for this plant group are related to staggering HPSI pump use during loss of service water events at ANO 1, maintaining service water operation with loss of HVAC at Davis-Besse, a procedure change at Oconee to reduce the effects of internal flooding, and procedural changes to refill the borated water storage tank during SGTRs at ANO 1, Davis-Besse, and TMI 1. All of these plant improvements could be applicable to other PWRs

One of the more noteworthy and explicitly documented quantitative effects of these improvements is that ANO 1, which is implementing the fifth swing diesel and additional battery capacity, estimates a 23% and 14% reduction, respectively in the plant CDF from the current value in the IPE of almost 5E-5/ry.

9.3.2.2 CE PWRs

With the exception of power-related improvements, including power equipment HVAC improvements in a number of submittals, most improvements identified in the submittals for this plant group are somewhat unique and not common to many of the CE plants. Of note, RCP seal LOCA improvement issues are not as pervasive in many of the B&W and Westinghouse plant submittals. Only Calvert Cliffs identified a plant improvement in this area. The implemented improvements in the CCW system should reduce the frequency of an RCP seal LOCA.

Several licensees identified plant improvements related to improving the reliability of AC and DC systems. As discussed above for ANO 1 (the B&W unit), the licensee for ANO 2 added a swing diesel generator. Palo Verde has installed two 4.5 MW gas turbine generators. The Waterford submittal indicates that the installation of a portable diesel generator for charging the station batteries was being evaluated. Maine Yankee has replaced and added spare battery chargers and inverters and improved the cross-tie capabilities of these inverters and chargers. Millstone 2 has also added a new battery charger. All of these changes could be applied at other plants.

Support system improvements, particularly with respect to loss of HVAC events, were identified by many licensees in this group. The Maine Yankee licensee has installed a high-temperature alarm and revised procedures related to a loss of switchgear room ventilation. Calvert Cliffs has also implemented procedural upgrades to improve mitigation of loss of switchgear room cooling. Millstone 2, which made DC room temperature alarm improvements, has also implemented a loss of engineered safeguard feature room ventilation procedure. At least three other submittals covering a total of five units state that temperature alarms and procedure changes have been implemented for DC rooms and inverter rooms.

Other improvements include AFW relay improvements for a backup water supply and maintenance improvements for AFW at Calvert Cliffs. Palo Verde has implemented power source changes involving main steam isolation valve (MSIV) and feedwater isolation and a backup power source for AFW control. Both St. Lucie and Waterford listed procedures to refill the CST as plant improvements. Calvert Cliffs was considering enhancing the EOPs to include the use of the pressurizer vent valves as a means for depressurizing the vessel during a SGTR event. Millstone 1 revised their EOPs to direct that a faulted steam generator be isolated.

Additional improvements include firewater injection to steam generators being evaluated at Waterford 3, additional valve checks or testing to decrease ISLOCAs at ANO 2 and Maine Yankee, various AFW improvements, and a variety of valve equipment or testing improvements. At Fort Calhoun, a number of flooding improvements are being implemented or being evaluated. These involve the prevention or mitigation of a rupture in the RCP seal cooler of the component cooling water system, an ISLOCA in a shutdown cooling line, and an AFW flood involving the need to possibly remove a water-tight door.

Two of the submittals identify noteworthy quantitative effects of the improvements. The Palo Verde submittal indicates that all the power-related changes implemented and credited in the IPE made a reduction of almost 1E-3/ry in the total plant CDFs which is very significant. The San Onofre 2&3 IPE indicates a 6E-5/ry reduction in the CDF for its implemented and credited installation of inverter room temperature alarms.

9.3.2.3 Westinghouse PWRs

Like the other groups of PWRs, all of the licensees in this group identify a variety of different types of improvements addressing a number of different issues. The most common issues have to do with power-related improvements for both station blackout and non-blackout accident scenarios (including loss of HVAC issues), AFW changes, and RCP seal cooling including related loss of component cooling water scenarios. Not quite as common but still identified for many plants are improvements regarding plant flooding. Other miscellaneous improvements are also identified. These improvements are fairly consistent among 2-, 3-, and 4-loop Westinghouse PWRs. The different loop plants tend to "look the same" from the standpoint of improvements that were implemented or are being planned or evaluated. Hence no significant difference is apparent among the different loop plants with regard to the identified improvements in the IPEs.

Power-related upgrades take a variety of forms and include diesel and gas turbine upgrades, battery and associated charging capability improvements, related loss of HVAC issues, load changes, and bus cross-tie improvements, among others. At least a dozen submittals indicate that the plants have apparently implemented onsite AC power improvements by either adding to or replacing existing power sources with new diesels, upgrading gas turbines, providing procedures for use of "backup" diesels, incorporating improvements in diesel reliability and maintenance programs as well as monitoring diesel operation (e.g., monitoring temperature for loss of jacket cooling), and providing procedural enhancements for 4 kV bus cross-tieing. Many of these improvements are done in coordination with the station blackout rule. At least six submittals indicate that additions of new or replacements of old diesels or gas turbines are under evaluation. The Indian Point 2&3 submittal indicates that potential changes to diesel functional testing are being considered, including room fan testing. The Beaver Valley 1&2 submittal indicates that a manual river backup for diesel cooling is being considered. The majority of these improvements could be applied to all PWRs.

DC power improvements, most of which are being evaluated, are identified in the Shearon Harris 1, H.B. Robinson, Seabrook, Point Beach 1&2, and Sequoyah 1&2 submittals. An apparently implemented DC power improvement at Point Beach 1&2 involves battery upgrades and a new swing battery. The improvements under evaluation are all different in form and include adding battery capacity, improving load shedding procedures, modifying DC loads on the batteries, providing independent battery charging capability, adding possible cross-tie features, and reducing present battery unavailability. All of these improvements could be applied at other plants as required.

Many licensees, including Diablo Canyon 1&2, Sequoyah 1&2, Indian Point 2&3, Salem 1&2, and Wolf Creek, identify improvements associated with room ventilation issues, including the diesel fan issue mentioned above. A few of these improvements are under evaluation while the rest are planned or have been implemented. Proposed ventilation changes involve adding two trains of ventilation for a 480V room, modifying the ventilation system for a 480V switchgear room, adding a switchgear room high-temperature alarm, evaluating possible changes to procedures for the loss of HVAC for switchgear rooms, and implementing a general "open doors" policy for area losses of ventilation.

Auxiliary feedwater system improvements are numerous. Three submittals, Indian Point 3, Point Beach, and Haddam Neck, identify planned or implemented backup sources of water for the AFW, including a new CST, the use of firewater, and the use of backup city water. The Ginna submittal identifies the use of firewater to cool the steam-driven AFWS pump during station blackout as an improvement, while the Robinson submittal indicates that the modification of its steam-driven pump to be self-cooled is under evaluation. Vogtle 1&2 and Comanche Peak 1&2 have apparently implemented procedural guidance for local manual operation of AFW when control power is lost. The Kewaunee submittal indicates a planned reliability improvement program for its steam-driven train, and the Indian Point 3 submittal indicates that the implementation of a procedure to open a roll-up door on loss of ventilation makes a significant change in its plant CDF. Lastly, the Haddam Neck submittal indicates that adding a motor train to its current steam trains is under evaluation. These improvements can be applied to each Westinghouse plant.

The RCP seal LOCA issue is also the source of a number of identified improvements for many plants. For instance, the Comanche Peak 1&2, Wolf Creek, Vogtle 1&2 and Farley 1&2 submittals specifically mention the changeout to high-temperature seals while others indicate this same change is under evaluation. At least eight submittals identify improvements to provide additional capability to supply RCP seal cooling when all normal means of cooling is lost. Some of the descriptions of these changes provide little detail, but others indicate more specifically how this might be accomplished. Examples include the use of swing component cooling water pumps, addition of a new pump with a possible backup diesel, use of firewater, and providing a means to not only cool the seals but simultaneously charge the primary system. These improvements, like those listed above, could be applied to other PWRs.

General flooding improvements are identified as mostly being implemented at many plants including Kewaunee, Point Beach 1&2, Prairie Island 1&2, Surry 1&2, Salem 1&2, H.B Robinson, and North Anna 1&2. A number of these licensees changed door swing-out directions so that the doors will be forced against their door jams based on identified flood sources and the corresponding flow directions. Many of the improvements are procedural changes to improve identification and isolation of flood sources and effects.

The remaining types of improvements vary widely and are typically identified in only a few submittals. These include ISLOCA procedural improvements, steam generator tube rupture procedure improvements, procedures for refilling the CST, a few core cooling recirculation hardware and procedure improvements, a few instrument air and compressor upgrades, and feed and bleed training upgrades. The single most common "other" improvement issue is related to the RCP seal cooling concerns related to the loss of component cooling water and/or, in some cases, service water. Procedural guidance and the use of cross-tied pumps or a swing pump is a commonly identified improvement implemented at some plants and under evaluation at others.

Quantitative effects of some of these improvements include a significant 1E-3/ry reduction in the CDF at Surry 1&2 as a result of flooding improvements and a more realistic evaluation. At Indian Point 3, the AFWS roll-up door improvement mentioned above is estimated to reduce the plant CDF by 6E-4/ry. Numerous power-related changes, including a mixture of those implemented or planned or under evaluation, showed CDF reductions of the order of 1E-5/ry. Similarly, implemented improvements credited in the existing IPEs for the RCP seal cooling issue are estimated to reduce the CDF by 5E-4/ry at Farley 1&2 and approximately 3E-5/ry at Kewaunee.

9.4 Containment Performance Improvements

9.4.1 The NRC Containment Performance Improvement Program

In SECY 88-147 (Ref. 9.5), excerpts of which were attached to GL 88-20, the NRC staff noted that it had undertaken a program to determine what, if any, actions should be taken to reduce the vulnerability of containments to severe accident challenges and to reduce the magnitude of releases that might result from such challenges. This program was referred to as the NRC's Containment Performance Improvement (CPI) program.

The NRC staff also noted that the first focus of its efforts would be the BWR Mark I containments. Technical insights arising from the CPI program for these containments were discussed in SECY 89-017 (Ref. 9.6), and summarized in Supplement 1 to GL 88-20 (Ref. 9.7).

The enclosure to Supplement 1 lists three improvements and states that the Commission expects that licensees of Mark I plants will seriously consider these improvements during their Individual Plant Examinations. Quoted below are excerpts from Supplement 1 regarding each improvement:

(a) Alternate Water Supply for Drywell Spray/Vessel Injection

"An important improvement would be to employ a backup or alternate supply of water and a pumping capability that is independent of normal and emergency AC power. By connecting this source to the low-pressure residual heat removal system (RHR) system as well as to the existing drywell sprays, water could be delivered either into the reactor vessel or to the drywell, by use of an appropriate valving arrangement.

An alternate source of water injection into the reactor vessel would greatly reduce the likelihood of core melt due to station blackout or loss of long-term decay heat removal, as well as provide significant accident management capability.

Water for the drywell sprays would also provide significant mitigative capability to cool core debris, to cool the containment steel shell to delay or prevent its failure, and scrub airborne particulate fission products from the atmosphere.

A review of some BWR Mark I facilities indicates that most plants have one or more diesel driven pumps which could be used to provide an alternate water supply. The flow rate using this backup water system may be significantly less than the design flow rate for drywell sprays. The potential benefits of modifying the spray headers to assure a spray were compared to having water run out of the spray nozzles. Fission product removal in the small crowded volume in which the sprays would be effective was judged to be small compared with the benefit of having a water pool on top of the core debris."

(b) Enhanced Reactor Pressure Vessel (RPV) Depressurization System Reliability

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

Another important benefit is in the area of accident mitigation. Reduced reactor pressure would greatly reduce the possibility of core debris being expelled under high-pressure, given a core melt and failure of the reactor pressure vessel. Enhanced RPV depressurization system reliability would also delay containment failure and reduce the quantity and type of fission products ultimately released to the environment. In order to increase reliability of the RPV depressurization system, assurance of electrical power beyond the requirements of existing regulations may be necessary. Performance of the cables needs to be reviewed for temperature capability during severe accidents as well as the capacity of the pneumatic supply."

(c) Emergency Procedures and Training

"Revision 4 to the BWR Owner's Group [BWROG] EPG [Emergency Procedure Guideline] is a significant improvement over earlier versions in that they continue to be based on symptoms, they have been simplified, and all open items from previous versions have been resolved. The BWR EPGs extend well beyond the design basis and include many actions appropriate for severe accident management.

The improvement to EPGs is only as good as the plant-specific EOP implementation and the training that operators receive on use of the improved procedures. The NRC staff encourages licensees to implement Revision 4 of the EPGs and recognize the need for proper implementation and training of operators."

The GL 88-20 Supplement 1 further states that the staff plans to communicate directly with each licensee who possesses a Mark I plant on the matter of a hardened vent path and that improvements (a), (b), and (c) above should be considered in addition to improvements that stem from the evaluation and implementation of the hardened vent.

In Supplement 3 to GL 88-20 (Ref. 9.8), the NRC staff announced the completion of the CPI program and forwarded the insights arising from this effort for BWR Mark II and Mark III containments and for PWR containments for use in licensee efforts as part of the IPE program. Supplement 3 notes that the technical information conveyed may be useful to licensees during their examinations of their plants for vulnerabilities to severe accidents.

The following statements regarding BWR Mark II and Mark III, as well as PWR containments are quoted from Supplement 3 to GL 88-20:

- "*Mark II Containments* — For events where inadequate containment heat removal could cause core degradation, additional containment heat removal capability using plant-specific hardware procedures is expected to be considered as part of the IPE process. Potential methods of removing heat from containment include, but are not limited to, using a hardened vent or other means of improving reliability of suppression pool cooling. It is expected that the negative as well as the positive benefits of the enhanced containment heat removal capability will be considered. For example, for those events where venting is initiated after core melt and subsequent vessel failure have occurred, the benefit of scrubbing of fission products cannot be assured for Mark II containments to the same degree as in Mark I plants. This is because molten core materials on the floor of the containment may fail downcomers or drain lines and result in suppression pool bypass.

  In addition, the Mark I improvements contained in Supplement 1 to Generic Letter 88-20 dated August 29, 1989 are expected to be considered for applicability to Mark II containments."

- "*Mark III Containments* — A potential vulnerability for Mark III plants involves station blackout, during which the hydrogen igniters would be inoperable. Under these conditions, a detonable mixture of hydrogen could develop which could be ignited upon restoration of power. Licensees with Mark III containments are expected to evaluate the vulnerability to interruption of power to the hydrogen igniters as part of the IPE. A backup power supply meeting the requirements for the Alternate AC option of the Station Blackout Rule would be one method of ensuring uninterrupted operation of the hydrogen igniters.

  In addition, the Mark I improvements contained in Supplement 1 to Generic Letter 88-20 dated August 29, 1989, as well as containment heat removal as discussed for Mark II containments, are expected to be considered for applicability to Mark III containments."

- "*PWR Ice Condenser Containments* — The same situation could occur in ice condenser containments as in Mark III containments relative to hydrogen detonations following restoration of power. Therefore, licensees with ice condenser containments are expected to evaluate the vulnerability to interruption of power to the hydrogen igniters as part of the IPE."

- "*PWR Dry Containments* — Depending on the degree of compartmentalization and the release point of the hydrogen from the vessel, local detonable mixtures of hydrogen could be formed during a severe accident and important equipment, if any is nearby, could be damaged following a detonation. In addition, smaller subatmospheric containments may develop detonable mixtures of hydrogen on a global basis. Licensees with dry containments are expected to evaluate containment and equipment vulnerabilities to localized hydrogen combustion and the need for improvements (including accident management procedures) as part of the IPE.

  It should be noted that currently available computer codes have been shown to overestimate mixing of hydrogen in the containment and may not be adequate to evaluate the potential for high-local concentrations of hydrogen (e.g., ANS Proceedings, 1989 National Heat Transfer Conference, August 6-9, 1989, Philadelphia, PA, Page 233-241). Thus any analyses should be supplemented by judgement as to the adequacy of the results and consideration of the impact of higher than predicted hydrogen concentration due to stratification.

Given an estimate of local concentration of hydrogen, NUREG/CR-5275 provides a discussion of one method that has been used to evaluate the potential for local hydrogen detonations."

NUREG-1335, which provides guidance to licensees on submitting the IPE results, mentions CPI issues in Section 2.3, which deals with the submittal of specific safety features and potential plant improvements. Appendix C of NUREG-1335 provides the NRC response to comments and questions the licensees raised about the IPE process. Some of these questions and their responses address the relationship of the CPI program to the IPE program.

In their IPE submittals most licensees respond to the CPI Program recommendations relevant to their type of containment. These responses are usually contained in a separate section devoted to CPI issues, but sometimes are scattered throughout the containment performance discussion. In those cases where CPI issues are not addressed in the submittals, the review of the submittals by NRC and its contractors usually elicits a response to CPI concerns from the licensee through a request for additional information (RAI). Nevertheless, the licensee's IPE submittals vary widely in their response to CPI recommendations. In several cases the licensees indicate that the CPI recommendations are being considered, but do not identify the recommendations as commitments. The sections below present an overview, grouped by containment types, of how the CPI concerns were treated in the different IPEs, and Tables 9.12 and 9.13 provide a summary of how each licensee addressed CPI in their IPE submittals.

Table 9.12 BWR containment performance improvements.

Plant name	Containment type	CPIs as discussed by licensees in IPEs
Browns Ferry 2	Mark I	- Intend to install hardened vent
Brunswick 1	Mark I	- Hardened vent (installed per GL 89-16)
Clinton	Mark III	- Addressed/no actions deemed necessary - BWROG EPG Rev. 4[1] was implemented
Cooper	Mark I	- None addressed
Dresden 2&3	Mark I	- Considering alternate containment spray
Duane Arnold	Mark I	- Installed hardened vent per GL 89-16 - Incorporated BWROG EPG Rev. 4 - Alternate vessel injection already in EPGs
Fermi 2	Mark I	- Hardened vent to be installed - BWROG EPG Rev. 4 implemented
Fitzpatrick	Mark I	- Installed hardened vent - Implemented BWROG EPG Rev. 4 - Did not install modification for alternate injection and spray
Grand Gulf 1	Mark III	- BWROG EPG Rev. 4 previously implemented

Table 9.12 BWR containment performance improvements.

Plant name	Containment type	CPIs as discussed by licensees in IPEs
Hatch 1&2	Mark I	- BWROG EPG Rev. 4 implemented - EPGs provide for alternate injection methods and also for alternate drywell sprays although not credited in IPE
Hope Creek	Mark I	- Implemented BWROG EPG Rev. 4 - Use of alternate injection systems modeled and included in EPGs - Installed hardened vent GL 89-16
LaSalle 1&2	Mark I	- BWROG EPG Rev. 4 implemented - Fire water as an alternative to RPV injection - EPGs provide diverse means to depressurize using turbine bypass valves, reactor feedwater pump and RCIC - Hardened vent previously installed
Limerick 1&2	Mark II	- No hardened vent - Planned capability for firewater alternate RPV injection; existing capability for injection via RHRSW - Incorporated BWROG EPG Rev. 4
Millstone 1	Mark I	- Installed hardened vent - Implemented BWROG EPG Rev. 4 with some additional evaluations ongoing - Alternate RPV injection and drywell spray provided
Monticello	Mark I	- Alternate RPV injection and drywell spray through RHR SW and fire water cross-ties - Considering modifying power supplies to N2 bottles for enhanced depressurization - Stayed with BWROG EPGs Rev. 3 - Enhanced ADS (SRV power available during loss of offsite power)
Nine Mile Point 1	Mark I	- Installed hardened vent (GL 89-16) - Adopted BWROG EPG Rev. 4 - Raw water available as alternate RPV injection, drywell spray and containment flooding
Nine Mile Point 2	Mark II	- Hardened wetwell vent to be installed during 1993 refueling outage - Adopted BWROG EPG Rev. 4 - Considering evaluation of diesel driven fire pump providing alternate RPV injection
Oyster Creek	Mark I	- Alternate RPV injection in place - RPV enhanced depressurization through alternate AC source - Implemented BWROG EPG Rev. 4 - Planned hardened vent
Peach Bottom 2&3	Mark I	- Capability to provide alternate RPV injection and drywell spray - Implemented BWROG EPG Rev. 4 - Installed hardened vent
Perry 1	Mark III	- Considering passive containment venting, ATWS alternate shutdown and ADS inhibit design

Table 9.12 **BWR containment performance improvements.**

Plant name	Containment type	CPIs as discussed by licensees in IPEs
Pilgrim 1	Mark I	- Installed hardened vent - Alternate RPV injection or drywell spray from firewater cross-tie - Implemented BWROG EPG Rev. 4
Quad Cities 1&2	Mark I	- Installed hardened vent - Implemented BWROG EPG Rev. 4 - Alternate sources (fire protection) for RPV injection and drywell sprays are under consideration
River Bend	Mark III	- Planned permanent installation of diesel-driven air compressor to supply vessel SRVs - BWROG EPG Rev. 4 implemented (prior to IPE) - Addressed need for containment venting and H2 igniters, no modifications deemed necessary
Susquehanna 1&2	Mark II	- Many EOP revisions derived from BWROG EPG, Rev.4 - Thirty-day supply of compressed nitrogen for SRV actuation - Alternate injection to RPV and containment spray via diesel-driven fire pump aligned to RHR service water - Considering wetwell vent procedures (credited in IPE)
Vermont Yankee	Mark I	(All identified prior to issuance of GL 88-20) - Installed hardened vent - Enhanced RPV depressurization reliability - Implemented BWROG Rev. 4 - Alternate RPV injection and drywell spray using river water or fire protection system
WNP 2	Mark II	- Enhanced RPV depressurization through operator training - BWROG EPG Rev. 4 implemented (prior to IPE)

[1] General Electric, et al., "BWR Owners Group Emergency Procedure Guidelines, Rev. 4," NEDO-31333, Class I Document, March 1987.

Table 9.13 **PWR Containment Performance Improvements.**

IPE submittals	Containment type	CPIs as discussed by licensees in IPEs
ANO 1	Large-dry	Discussed in responses to RAI, no action deemed necessary by licensee
ANO 2	Large-dry	Discussed in submittal and responses to RAI. No vulnerabilities for hydrogen accumulation and combustion
Beaver Valley 1	Large-dry (subatmospheric)	Discussed in IPE submittal, no action deemed necessary by licensee
Beaver Valley 2	Large-dry (subatmospheric)	Discussed in IPE submittal, no action deemed necessary by licensee

Table 9.13 PWR Containment Performance Improvements.

IPE submittals	Containment type	CPIs as discussed by licensees in IPEs
Big Rock Point*	Large-dry	Discussed only in responses to RAI, no action deemed necessary by licensee
Braidwood 1&2	Large-dry	Not discussed in IPE submittal (revised IPE not submitted to NRC as of 10-96)
Byron 1&2	Large-dry	Not discussed in IPE submittal (revised IPE not submitted to NRC as of 10-96)
Callaway	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Calvert Cliffs 1&2	Large-dry prestressed post tensioned	Discussed in IPE submittal, no action deemed necessary by licensee
Comanche Peak 1&2	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Crystal River 3	Large-dry	Discussed in responses to RAI, no action deemed necessary
Davis Besse	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Diablo Canyon 1&2	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Farley 1&2	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Fort Calhoun 1	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Ginna	Large-dry	Not specifically discussed, hydrogen detonation is discussed and deemed not a concern by licensee
Haddam Neck	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Indian Point 2	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Indian Point 3	Large-dry	Discussed: no immediate modifications identified; formal implementation of accident management strategies for hydrogen control deferred until SAM guidelines implemented--expected June 1998 (to be based on WOG Guidelines)
Kewaunee	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Maine Yankee	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Millstone 2	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee

Table 9.13 PWR Containment Performance Improvements.

IPE submittals	Containment type	CPIs as discussed by licensees in IPEs
Millstone 3	Large-dry (subatmospheric)	Discussed in IPE submittal, no action deemed necessary by licensee
North Anna 1&2	Large-dry (subatmospheric)	Discussed in IPE submittal, no action deemed necessary by licensee
Oconee 1,2&3	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Palisades	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Palo Verde 1,2&3	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Point Beach 1&2	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Prairie Island 1&2	Large-dry	Discussed: hydrogen burn determined by licensee not to be a threat to containment integrity
H. B. Robinson 2	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Salem 1&2	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
San Onofre 2&3	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Seabrook	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Shearon Harris 1	Large-dry	Discussed: possible procedural change to reduced potential for induced SGTR
South Texas Project 1&2	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
St. Lucie 1&2	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Summer	Large-dry	Discussed: hydrogen detonation not deemed to be a concern by licensee
Surry 1&2	Large-dry, (subatmospheric)	Discussed in response to RAI, no action deemed necessary by licensee
Three Mile Island 1	Large-dry	Referenced Oconee analysis, no action deemed necessary by licensee
Turkey Point 2&3	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Vogtle 1&2	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee

Table 9.13 PWR Containment Performance Improvements.

IPE submittals	Containment type	CPIs as discussed by licensees in IPEs
Waterford 3	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Wolf Creek	Large-dry	Discussed in IPE submittal, no action deemed necessary by licensee
Zion 1&2	Large-dry	Not discussed in detail: steam inerting of containment identified by licensee as a short-term hydrogen control
Catawba 1&2	Ice condenser	Discussed: possibly restore hydrogen igniters in small groups
DC Cook 1&2	Ice condenser	Discussed in IPE submittal, no action deemed necessary by licensee
McGuire 1&2	Ice condenser	Discussed: restore igniters as part of SAMG
Sequoyah 1&2	Ice condenser	Discussed in IPE submittal, no action deemed necessary by licensee
Watts Bar 1	Ice condenser	Discussed in IPE submittal, no action deemed necessary by licensee

* BWR housed in a large-dry containment, therefore listed in this table rather than Table 9.6.

9.4.2 Treatment of CPI Issues in IPEs for Plants with BWR Mark I Containments

Among the IPE submittals for Mark I plants, the responses to the CPI issues are summarized below.

Hardened wetwell vent

In response to the recommendation of the NRC's GL 89-16 (Ref. 9.9), all the utilities with Mark I containments committed to install a hardened wetwell vent system, if one was not already in place. A hardpipe containment vent leading from the torus to outside the containment building provides an independent means for containment heat removal, while allowing a habitable environment to be maintained in the reactor building during venting operation. The licensees with Mark I containments vary considerably in their assessment of the benefit gained from a hardened vent capability. For a number of reasons, including the ability of systems in the turbine building (which is not affected by venting to the reactor building) to provide adequate makeup and because of the limited impact on core cooling systems from the existing (i.e., not-hardened) vent systems predicted in some IPEs, a number of IPE submittals stated that the benefit of installing a hardened vent system was marginal. For example, according to the Millstone 1 IPE, the addition of a hardened vent path would lead to a reduction in CDF of less than 1% since a not-hardened path exists and is used in the EOPs.

In contrast, the Fitzpatrick IPE submittal states that containment venting reduced the core damage frequency by an estimated factor of 14. It is not clear, however, how much of this reduction would have been achieved with a not-hardened path through preexisting piping and ductwork paths.

Alternate Water Supply for Drywell Spray/Vessel Injection

Most of the submittals for plants with Mark I containments discuss external sources for drywell spray or vessel injection. In many cases the submittal states that alternate water sources exist at the plant and have been credited in the IPE. Other IPEs state that while such alternate sources exist and are referred to in their EOPs, they are not credited in the IPE analysis. Some submittals state that providing alternate sources is still under consideration. Usually the external water source is provided via the service water system using river or pond water or the fire protection system. For example, five separate sources for alternate RPV injection or drywell spray are described in the Duane Arnold IPE. One source includes the fire water system, which is capable of providing alternate injection and spray during station blackout conditions with the RPV and drywell depressurized by utilizing the independent diesel driven pump. A few licensees state the results of analysis they have performed indicate that the benefits of arranging for alternate sources are marginal and therefore not implemented. Examples are the IPEs for the Fermi 2 plant and the Brunswick units. The Fermi IPE states that only a 28% reduction would be achieved from connecting the firewater system to the drywell spray system. The Brunswick 1&2 IPE notes that connecting firewater to containment spray has negligible benefit because the low-pressure of the firewater system leads to reduced flow, or no flow, depending on containment pressure. In the Hope Creek IPE, the licensee states that the benefit of connecting the fire protection system to the drywell spray could be very important; nevertheless this improvement will not be implemented.

Enhanced RPV Depressurization System Reliability

Most of the licensees either do not discuss this issue in their submittals or state that the existing system is reliable enough. Some note that other improvements would be more cost effective. A few submittals, like those for Oyster Creek and Vermont Yankee, indicate enhancements to the AC power supply to the ADS valves. A number of submittals note that enhancing power reliability in general will add to the reliability of the depressurization system.

Emergency Procedures and Training

In each case where the use of the BWR owners group EPGs is addressed, the licensee has adopted Revision 4. Several of the licenses indicate that their emergency operating procedures would be revised to better address the human reliability aspects to depressurize the RPV. Several licensees have also indicated that current guidelines, and therefore procedures, for venting, primarily when used in connection with the containment flooding contingency, should be reconsidered.

9.4.3 Treatment of CPI Issues in IPEs for Plants with BWR Mark II and BWR Mark III Containments

As noted above, the CPI recommendations for licensees with Mark II and Mark III containments were that they consider the improvements suggested for Mark I containments, and that licensees with Mark IIIs consider providing an additional means of assuring the power supply to the igniters. The CPI program suggested that hardened vents for Mark II and Mark III containments might be evaluated by the licensees of plants with these containments as part of possible enhancements for containment heat removal. However, unlike licensees with Mark I containments, licensees with Mark II or Mark III containments were not expected to commit to the installation of a hardened vent path by the NRC.

9.4.3.1 Treatment of CPI Issues in IPEs for Plants with BWR Mark II Containments

Among the four IPE submittals for Mark II plants, the responses to the CPI issues are summarized below.

Hardened Vents

The LaSalle and Nine Mile Point 2 submittals state that hardened vents are in place in these plants, while Limerick and WNP 2 use the existing, not-hardened vents. The Limerick submittal acknowledges that use of the existing vent system will lead to a duct work failure and therefore should only be used when there is adequate core cooling. The WNP 2 analysis indicates very limited benefit from a hardened vent path.

Alternate Water Supply

A connection to the fire protection system for the drywell sprays exists at Limerick but no credit was taken in the IPE. Nine Mile Point 2 has implemented and credited a raw water cross-tie as an alternate injection source to the RPV or the containment spray. A fire water connection is under consideration at LaSalle, while the WNP 2 submittal states that alternate sources were not expected to provide any additional benefits.

Enhanced Depressurization

All Mark II IPEs indicate that the existing capability is adequate. LaSalle notes that the EOPs provide several means of depressurization, including the turbine bypass valves, the turbine driven reactor feedwater pump, and the RCIC steam line. WNP 2 mentions additional operator training on depressurization.

Emergency Procedures

All licensees with Mark II plants indicate that Revision 4 of the BWR EPGS is being used. As some other BWR licensees have done, the Limerick IPE suggests that relaxation of the drywell spray initiation criteria should be considered.

9.4.3.2 Treatment of CPI Issues for Plants with BWR Mark III Containments

Among the four IPE submittals for Mark III plants, the responses to the CPI issues are summarized below.

Hardened Vents

None of licensees with Mark III containments find that a hardened vent would have a significant impact on their CDF or containment results. One plant, Perry, evaluates the effect of a passive vent design featuring a rupture disk and an alternate vent line which would open automatically upon containment overpressure. A substantial decrease in the probability of RPV failure and containment failure is observed. This alternate vent path has not been designed, however.

Alternate Water Supply

As with other BWR licensees, fire water is the principal alternate water source considered by licensees of Mark III plants. The Clinton IPE notes that such a connection is under consideration. The Grand Gulf IPE finds that a f water cross-tie to vessel injection has a significant impact on CDF. The River Bend IPE also states that a cross-tie of the fire protection water to the RPV injection has been made subsequent to the submittal of the IPE analysis.

Enhanced Depressurization

In general the licensees deem their existing capabilities adequate. The Clinton IPE notes that backup batteries to extend the duration of the power supply for depressurization valves are under consideration to be evaluated under the accident management plans. The River Bend submittal indicates the installation of an additional diesel compressor.

Emergency Procedures

All the licensees indicate the use of Revision 4 of the BWR EPGs. The Grand Gulf IPE submittal discusses the need to reassess the BWR EPGs on MSIV venting during containment flooding. For Grand Gulf, a major contributor to the source term released to the environment, as identified by the Level 2, results is the "MSIV venting" event. This venting is procedurally called for by the Containment Flooding portion of the Emergency Procedures. If this procedural step were removed or suitably revised, the Grand Gulf IPE results indicate that the source term releases could be reduced. According to the IPE, the BWROG Severe Accident Management Subcommittee is studying severe accident management guidelines and is considering changes to this part of the EPGs for all BWRs. Grand Gulf is involved in this activity and expects to contribute these IPE insights in support of changes to the MSIV venting guidance.

Enhanced Hydrogen Igniter Power Supply

All of the Mark III licensees find that additional enhancements to the igniter power supply are not warranted. As an example, the Clinton submittal states that having an alternate source of power with 90% availability to the hydrogen igniters would reduce the frequency of a containment release from approximately 1E-6 to 9E-7/ry and would have a negligible reduction in the frequency of a large (Class III) release. The River Bend submittal notes that portable DC generators to enhance DC power requirements can be provided and that the station blackout procedure has been revised to instruct operators to turn off igniters if AC power is unavailable.

9.4.4 Treatment of CPI Issues in IPEs for Plants with PWR Large-Dry and PWR Ice Condenser Containments

As discussed in Section 9.4.1 above, the CPI issue for PWR large-dry containments is the evaluation of containment and equipment vulnerabilities to localized hydrogen combustion. For PWR ice condenser containments, the issue is vulnerability to interruption of power to the hydrogen igniters, i.e., the same issue as for BWR Mark III containments.

Regarding the interruption of power to the igniters, each of the five submittals for plants with ice condenser containments indicates that the present arrangements are adequate. Two submittals, those for Catawba 1&2 and McGuire 1&2, mention that sensitivity studies were conducted to evaluate the benefit of a backup power supply. The Catawba submittal reports that reducing the unavailability of igniter power to zero would reduce the probability of early containment failure by about a factor of five (however, early failure probability at Catawba is reported as less than 0.01). The McGuire submittal reports that if igniters were always available the whole-body person-rem risk would be reduced by approximately 6 person-rem/reactor-year. In both cases, as part of their accident management programs, the licensees intend to evaluate a potential strategy for restoring the igniters in small groups as a means of reducing the potential for containment overpressure due to hydrogen combustion.

The D.C. Cook 1&2 analysis looks at cases with hydrogen burns suppressed until maximum levels were attained in the upper containment compartment. Assuming a burn would then occur, the analysis indicates a containment failure at approximately 16 hours, which the licensee did not consider an early failure. The Watts Bar 1 submittal states that in order to demonstrate that Watts Bar has no specific vulnerability to igniter unavailability, it was assumed that all of the CDF associated with plant damage states with the igniters unavailable would result in containment failure at some time.

The response in the IPE submittals to the CPI issue of the effect of hydrogen burns, including localized detonations, varies considerably among the licensees of plants with PWR large-dry containments (one BWR, Big Rock Point, is included in this group). In a considerable number of the submittals the issue is not directly addressed. In most such cases responses to RAI questions provide some information. In general the licensees report that their containments are open enough and that sufficient communication paths exist between compartments, so that good atmospheric mixing occurs and local accumulation of hydrogen is unlikely. Even those submittals which do not specifically address the CPI issue usually provide a description of the containment which emphasizes its large volume and openness. The evidence for the containments' open geometry usually comes from visual inspections carried out during containment walkdowns. In a number of submittals, such as those for Palo Verde 1,2&3, Salem 1&2, and H.B. Robinson, some local pocketing is acknowledged to be possible but in areas which contain no vital equipment and in which a detonation is judged not to severely affect the rest of the containment or any vital equipment.

While the CPI hydrogen issue is dismissed as a concern in most submittals, the level of supporting detail varies considerably with some submittals using only qualitative arguments to dismiss the issue and others referring to detailed analytical methods similar to the method provided as an example in Supplement 3 of the GL 88-20. In the Summer, Vogtle and Wolf Creek submittals, simplified deflagration-to-detonation transitions are discussed. The licensee of Maine Yankee states in the IPE submittal that since hydrogen combustion is a major contributor to early containment failure in this plant, the issue will be considered further in the development of the accident management plan.

9.5 Impact on Reactor Safety Due to Plant Enhancements

Although only a few of the IPE submittals explicitly identify vulnerabilities, almost all of the submittals identified plant improvements to address these vulnerabilities and other issues of concern identified through the IPE process. Many of these plant improvements had been implemented at the time of the IPE submittals or were scheduled for implementation. The quantitative impact of these plant improvements has not generally been calculated, but they commonly improve plant safety. Thus, the IPE program has served as a catalyst for further improving the overall safety of nuclear power plants.

Many of the vulnerabilities explicitly identified in the submittals are applicable to other plants. In addition, many of the identified plant improvements addressed common problem areas and often involved similar solutions. In fact, many of the plant improvements identified for one plant could be applied to similar plants. A summary of the plant improvements that would have the most impact on the core damage frequency is presented in Table 9.14.

These improvements address the dominant accident scenarios identified in the IPEs and are generally applicable to most plants. For BWRs, the dominant accident scenarios are station blackouts, transients with loss of coolant injection, and transients with loss of DHR. Important contributors to these accidents scenarios include AC and DC power reliability, support system dependencies, availability of alternative injection systems, failure to depressurize the reactor vessel, and the operability of coolant injection systems following loss of DHR. For PWRs, the dominant accident scenarios are station blackout, transients, and LOCAs. Important contributors to these accident scenarios include susceptibility to RCP seal LOCAs, AC and DC power reliability, operator action to switch from coolant injection mode to recirculation mode, feed-and-bleed capability, and the support system dependencies.

Table 9.14 Summary of important plant improvements identified by licensees.

Area of improvement	Applicability		Specific improvement
	BWR	PWR	
AC Reliability	✓	✓	• Added or replaced diesel generators
	✓	✓	• Added or replaced gas turbine generator
	✓	✓	• Redundant offsite power capabilities
	✓	✓	• Improved bus/unit cross-tie capabilities
DC Reliability	✓	✓	• Install new batteries, chargers, or inverters
	✓	✓	• Alternate battery charging capabilities
	✓	✓	• Increased bus load shedding
Coolant Injection Systems	✓		• Replace emergency core cooling system pump motors with air-cooled motors
	✓		• Align LPCI or CS 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 ADS for non-ATWS scenarios
		✓	• Improve procedure and training on switch to recirculation
		✓	• Increased training on feed-and-bleed operations
DHR Systems	✓		• Add hard-pipe vent
	✓		• Portable fire pump to provide isolation condenser makeup
		✓	• Install new AFW pump or improve existing pump reliability
		✓	• Refill CST when using AFW
		✓	• Modification to align firewater pump to feed steam generator
Support Systems	✓	✓	• Procedures and portable fans for alternate room cooling upon loss of HVAC
	✓	✓	• Install temperature alarms in rooms to detect loss of HVAC
	✓	✓	• Revised procedures and training for losses of support systems
RCP Seal LOCAs		✓	• Evaluate or replace RCP seal material
		✓	• Add independent seal injection or charging pump for station blackout
		✓	• Supply RCP seals with alternate cooling
		✓	• Operator training on tripping pumps on loss of cooling
		✓	• Review HPSI dependency on CCW

The IPE submittals indicate that the CPI program provided a helpful checklist of potential containment related modifications which many licensees used during the conduct of their IPEs. Many licensees indicated that they had already incorporated the recommendations coming from the CPI program. This was especially true for licensees of BWR plants, many of which stated that they had hardened vents in place, had adopted Revision 4 of the BWROG Emergency Procedure Guidelines, and had provided for alternate water supplies for RPV injection and containment spray. In contrast, the licensees of PWR plants stated that no changes were called for by the CPI recommendations for their type of containments because local hydrogen accumulation was not found to be a problem, and backup power supplies for igniters (in ice condensers) would not significantly impact containment performance.

REFERENCES FOR CHAPTER 9

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

9.2 USNRC, "Individual Plant Examination: Submittal Guidance," NUREG-1335, Washington, D.C., August 1989.

9.3 Nuclear Management & Resources Council, "Severe Accident Issue Closure Guidelines," NUMARC 91-04, January 1992.

9.4 USNRC, "Implementation of Safety Goal Policy," SECY-89-102.

9.5 USNRC, "Integration Plan for Closure of Severe Accident Issues," SECY-88-147, May 25, 1988.

9.6 USNRC, "Mark I Containment Performance Improvement Program," SECY-89-017, January 23, 1989.

9.7 USNRC, "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities," Generic Letter 88-20, Supplement 1, August 29, 1989.

9.8 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.

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

PART 3

REACTOR AND CONTAINMENT DESIGN AND OPERATIONAL PERSPECTIVES

10. BACKGROUND ON OBTAINING REACTOR AND CONTAINMENT DESIGN PERSPECTIVES

Chapter 10 explains the approach chosen to obtain the perspectives provided in Chapter 11 and 12 (and summarized in Chapters 3 and 4). This chapter informs the reader of the key plant and containment characteristics, as well as the boundary conditions, assessments and assumptions used in the individual plant examination (IPE) modeling that can potentially affect the results reported in the IPEs. First, Section 10.1 outlines the approach used to obtain core damage frequency (CDF) and containment performance perspectives. (Chapter 13 discusses the approach used to obtain operational perspectives. The remainder of this section discusses the plant features and modeling characteristics that are most prevalent in probabilistic risk assessment (PRA) modeling, are addressed in the IPEs, and have potential significance to the IPE results. From Sections 10.2, 10.3, and 10.4 the reader should obtain an awareness of (1) the significant features of the different classes of plants and (2) the various boundary conditions, assessments, and assumptions that are used in the IPEs and why they can be important to the IPE results. With this knowledge, the reader can more easily understand the specific IPE perspectives and important insights discussed in subsequent chapters.

10.1 Approach for Obtaining Core Damage Frequency and Containment Performance Perspectives

The first set of perspectives obtained from the IPE submittals included those insights related to the quantification of the CDF for the plants, the classes of accident sequences with the greatest contribution to the CDF, and the factors driving the CDF. These factors include actual plant design, operational features and characteristics, and analytical modeling similarities or differences among the submittals, which also impact the results. As indicated earlier, because all of the licensees chose to use traditional PRA approaches to respond to Generic Letter (GL) 88-20 (Ref. 10.1) for the CDF portion of the analysis, all of the submittal summaries could be examined and compared on the basis of the three levels of insights identified above, which include CDF, accident classes, and factors most important to the likelihood of core damage.

In order to derive CDF perspectives among logical groupings of generally similar plants, all of the IPE submittals were first categorized by the type of plant covered by the submittal and either the plant vintage or the nuclear steam supply system vendor. Hence, all of the plants were first categorized as either boiling water reactors (BWRs) or pressurized water reactors (PWRs). The BWRs were then further grouped as to major model. Specifically, these groupings include BWR 1, 2, or 3 designs with isolation condensers (ICs) as a group; BWR 3 and 4 designs with reactor core isolation coolant (RCIC) injection as a group; and BWR 5 or 6 designs as the last group. The PWRs were put into three major groups by nuclear steam supply system vendor (i.e., Westinghouse (W), Combustion Engineering (CE), or Babcock & Wilcox (B&W)). The Westinghouse plants were further categorized on the basis of the number of primary coolant loops in the design (i.e., 2, 3, or 4-loop plants).

In addition to the CDF perspectives, containment performance perspectives obtained from the IPE submittals included those insights related to the containment failure modes, the releases associated with those failure modes, and the factors responsible for the types of containment failures and release sizes reported. These factors involve actual containment design characteristics and plant-specific hardware or operational features, as well as similarities and differences in assumptions and modeling techniques. GL 88-20 (Ref. 10.1) allowed the licensees considerable latitude in conducting the containment performance analysis portion of the IPE. While there is significant variability in the approaches chosen by the licensees, the essential information regarding containment failure modes, release type and size, and factors driving the analyses could be found in all submittals. Specifically, important parameters related to accident progression and containment performance, which could be obtained from almost all the submittals, were as follows:

- frequency and conditional probability of early containment failure (ECF) and bypass
- frequency and conditional probability of late containment failures
- magnitude of important source-term releases

To gain perspectives on similarities and differences in containment performance among the IPEs, the submittals were grouped according to the five containment types found in domestic nuclear plants (i.e., BWR Mark I, Mark II, and Mark III containments and PWR large-dry (including subatmospheric) and ice-condenser containments).

In deriving insights and perspectives, the reviews of the submittals focused on addressing the following objectives:

- Determine the nuclear power industry's assessment of the CDF potential and of the containment performance of operating nuclear power plants in the United States (U.S.)

- Determine the factors driving the CDF and the containment performance.

- Determine how similar or different the estimates are among and within plant groups and containment types.

- Determine the underlying causes for the similarities and differences found.

Figure 10.1 illustrates how plant and containment design, analytical boundary considerations, assessments, and assumptions provide inputs to the IPE model and, thus, the CDF results. To identify insights and perspectives with regard to addressing the above objectives, the review of the IPE submittals was carried out by reversing the analysis process shown in Figure 10.2. This was done by examining the results as reported in the IPEs, as well as the important plant and modeling characteristics driving the observed results. This examination process was carried out using four major steps as discussed below and illustrated in Figure 10.2.

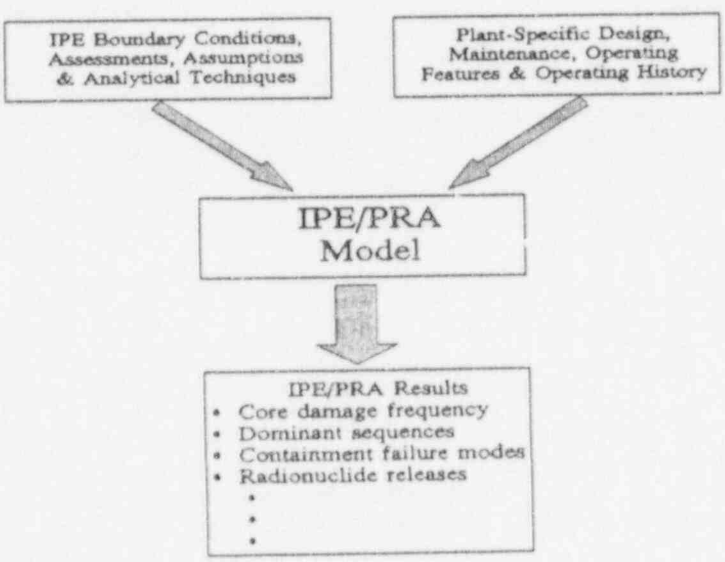


Figure 10.1 IPE analytical process flow diagram.

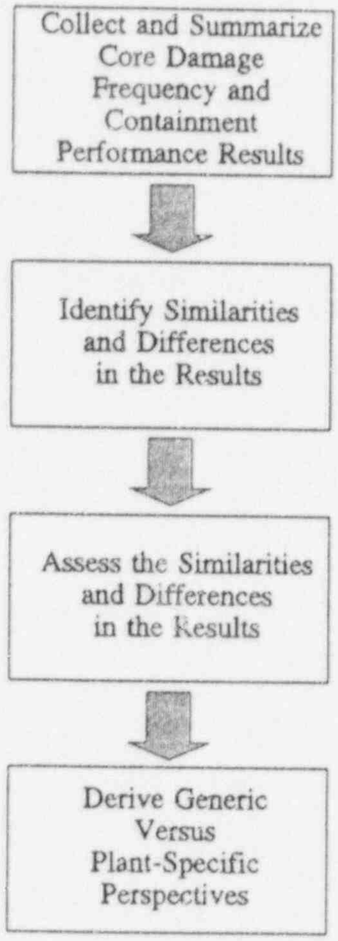


Figure 10.2 Approach for obtaining CDF and containment performance perspectives.

10.1.1 Step 1: Collect and Summarize CDF and Containment Performance Results

From the traditional PRA process, one quantitative outcome that provides a single overall comparable insight into the safety of nuclear power plants is each plant's CDF. Hence, the first step of the submittal examination process included the collection of the CDFs reported in the IPE submittals. It should be noted that the quantitative value reported in each submittal was sometimes a point estimate and sometimes the mean value derived from a data uncertainty analysis if one was performed.

Next, the major classes of accidents dominating the likelihood of core damage for each plant were summarized. In order to later be able to compare these insights from among all of the IPEs, a set of accident classes were defined and used to categorize the results of each individual IPE. This "standard" set of accident classes was defined on the basis of accident classes of general interest within the industry (e.g., loss of coolant type accidents) and also on how the IPEs generally reported the results using major "classes" of accidents. The resulting set of accident classes used to categorize the IPE results is shown in Table 10.1. In many cases, the IPE results were reported using these accident class definitions; in other cases, the reported IPE results had to be re-categorized to fit this standard set of accident classes. This collection and categorization process required examination of the reported dominant accident sequences and often the dominant accident sequence cut sets (each cut set is a "string" of failures that must occur to cause the identified accident sequence). On the basis of the characteristics and failures associated with each sequence or cut set, the IPE sequence-level results were categorized using the standard classes. The respective frequencies were modified, if necessary, to account for any adjustments caused by the re-categorization process.

Table 10.1 Summary sequences.

SBO -	Station Blackout
ATWS -	Anticipated Transient Without Scram
DHR -	Transients with Loss of Containment Heat Removal (BWRs only)
T -	Other Transients
LOCA -	Loss of Coolant Accidents
FLD -	Internal Flood Initiators
R -	Vessel Rupture
V -	Interfacing Systems LOCA (ISLOCA)
SGTR -	Steam Generator Tube Rupture (PWRs only)

In addition to CDF information, the traditional PRA process yields an assessment of containment performance. A crucial element of containment performance is whether the containment fails and at what time. Therefore, an important concern of the review process was to categorize the containment performance results in the submittals regarding early failure, late failure, and no containment failure. Early loss of containment integrity usually involves the most severe consequences and was, therefore, described in greater detail in the submittals. Therefore, the review established conditional probability and frequency of containment isolation failure and containment bypass, as well as actual early structural failure, and collected the results for each. Results for late failure and no containment failure were also collected.

Another important aspect of a containment performance analysis is the type and size of the release postulated to occur as a result of loss of containment integrity. Early releases of significant size can be expected to lead to the most severe consequences. Therefore, whenever possible as part of the review process, early release information was extracted from each submittal and releases of significant size were determined. For purposes of the review, a significant release was defined as one containing iodine (I) or cesium compounds equal or greater than 10% of core inventory.

10.1.2 Step 2: Identify Similarities and Differences in the Results

Following Step 1 for CDF perspectives the plant CDFs and associated accident class frequencies were investigated within each previously discussed plant grouping. For each grouping of plants, these investigations included comparisons summarized first by identifying the range in the individual plant CDF values and then calculating an average for the distribution of individual plant CDFs within that group. This average was calculated on the basis of the CDFs for each nuclear unit and not for each submittal. Thus, the reported CDF value for each unit at a dual-unit site was counted twice in deriving the average CDF, even though a single submittal covered both units. Comparison of the individual plant CDFs included determining the degree of similarity or difference among all of the CDFs, with respect to the average CDF as well as to each other. This comparison process was summarized both in tabular form and with plots that illustrate the degree of spread (or closeness) in the results and particularly whether any high- or low-"outliers" seem to exist relative to all other plants within each group. Note that the term "outliers" as used here does *not* necessarily mean that the plant is an outlier from the standpoint of overall plant safety or that the plant has a vulnerability that requires correction.

The same investigative process was carried out using the accident class level of results. Similar to the plant CDFs above, an average frequency for each accident class within each group of plants was calculated. Comparisons were then made and documented in both tabular and plot form to illustrate the degree of spread (or closeness) in the accident class frequencies and particularly whether any high- or low-"outliers" existed. Additionally, the percent contributions of each accident class to the total plant CDF were calculated for each individual IPE. Similarly, an average percent contribution was calculated for each plant group, and comparisons to this average value were performed. Examining these percent contribution results yielded added perspectives on the relative importance of each accident class for each IPE, again noting any outliers among the results.

Similarly, for containment performance perspectives, once the results regarding containment failure modes were collected, the values for the different failure modes were compared within the previously mentioned containment groupings. For each grouping, average values were calculated and these averages, as well as the individual plant values, were compared to provide perspectives on the range and, therefore, the variability and similarity among a group of plants. Again, the averages were based on the number of units, not the number of submittals. Therefore, if one submittal stated that a particular set of results applied to both units at the site, the results were included twice in the averaging process.

The early release values collected for each plant grouping were also compared with each other to gain insight into the range of such releases among the group. Significant early releases, as defined above, were then established for each plant, based on the submittal information and also compared. Not all early releases were significant, since the containment failure size and the degree of scrubbing (removal of radionuclides associated with various mechanisms) were calculated or assumed, to have a large influence on determining release size.

For both the failure mode and early release results, comparisons were summarized in both tabular and graphic formats. Results that deviated from average values within a class were particularly noted for scrutiny during the next step of the examination process.

10.1.3 Step 3: Assess the Similarities and Differences in the Results

The third step in the examination process was to assess the reasons for the similarities and differences observed in the IPE results from the previous two steps. For CDF perspectives, this assessment process involved identifying those factors driving the plant CDFs and the dominating accident classes for each individual IPE, paying particular attention to any "outliers" noted above. These driving factors are called the dominant contributors to the potential for core damage. These dominant contributors may take the form of key plant design or operational features (such as the number of high-pressure systems available at one plant as opposed to another), or they may be the types of analytical boundary conditions or modeling assumptions discussed in Section 10.4, below (such as the ability of systems to operate in beyond-design environmental conditions).

The identification of these dominant contributors consisted of simultaneously implementing two processes (1) thorough identification of the contributors on the basis of what was reported in the IPEs and (2) a systematic search for plant characteristics and modeling methods and assumptions known to be dominant factors in previously published PRAs. These possible plant characteristics and analytical variations are discussed in the sections below. In identifying these dominant contributors, determinations were made as to which system failures were involved in the various accident sequences, which specific equipment failures or outages were involved, and which human errors were most important to the results of each sequence. This simultaneous search identified those dominant contributors most important to the CDF results and to the most significant accident classes for each plant.

In order to identify the dominant factors driving containment performance, the accident progression analysis of individual submittals was reviewed to identify and categorize methods, data, boundary conditions, and assumptions used in the containment performance analysis. The following important analysis features which were investigated for each submittal:

- How was the containment performance analysis performed (i.e., large or simple containment event trees, etc.)?
- Which failure mechanisms were considered?
- How were severe accident phenomena handled (i.e., how was direct containment heating, shell melt-through, etc. considered)?
- What was assumed for containment strength?
- How were source terms obtained (from previous studies or plant-specific code runs)?
- Were there any plant unique containment features?

The actual accident mechanisms causing containment failure, and the phenomena involved in producing them, were investigated as well. The review considered whether these phenomena were addressed by established methods, such as those of NUREG-1150 (Ref. 10.2) for instance, or if novel procedures were used.

Methods employed to obtain source terms were also scrutinized to see if actual modular accident analysis program (MAAP) (Ref. 10.3), MELCOR (Ref. 10.4), or other calculations were made, and under what assumptions, or if source terms were obtained through analogy and comparison with existing results from previous analyses.

An evaluation was performed to determine the degree of variability or similarity in the dominant contributors results for each plant or containment grouping. The results of this evaluation yielded "clues" as to the possible reasons for the degree of similarity or differences in the results. This suggested why the results differ and why some plants may appear as "outliers" when compared to other plants within the same plant or containment group. Perspectives from this evaluation process are included in Chapters 11, 12, and 13 of Part 3 of this report.

10.1.4 Step 4: Derive Generic versus Plant-Specific Perspectives

Throughout the examination outlined in the first three steps (above), attention was constantly given to understanding the degree of variability or similarity in the results of the IPEs. Both similarities and differences in results among containment types were investigated to see if generic trends could be discerned. The following questions were important in understanding the generic implications of the results of the IPEs:

- Do all BWR 3s and BWR 4s conclude that their core damage potential is dominated by the same major type of accident and similar dominant contributors?
- Do all PWRs conclude that reactor coolant pump seal LOCAs are a dominant contributor to CDF in their designs?
- Was shell melt-through a significant contributor in all Mark I containments?
- Did all submittals for plants with ice condenser containments assume a high-reliability for their igniters?
- Are there unique plant-specific issues that cause plants to appear as outliers when compared with other plants in the same plant group?

The ranges of results obtained were compared with each other to determine if the differences were in line with previous experience and to establish the reasons for variation. In other words, this comparison sought to determine whether previous generic conclusions regarding certain plant or containment types were borne out by the results of the IPE submittals.

Insights into the generic implications of the results or the uniqueness of the nuclear power plants in the U.S. are among the most important products of this work. For each plant grouping, the IPE results and dominant factors were examined and generic trends, if any, were identified. Close examinations of outlier plant results were performed to determine if any noteworthy plant-specific perspectives could be gained from the results for these plants. The findings of the IPE examination process, and indications of the general or unique nature of the results for each plant and containment grouping, are discussed in the remaining sections of this chapter.

10.2 Plant Characteristics

This section discusses the differences between each of the BWR and PWR types, with regard to the design, operation, and interfaces that can affect the CDF. Important containment characteristics are discussed in Section 10.3. Differences in the primary power conversion, reactivity control, emergency core cooling, decay heat removal, and support systems that can impact the IPE results are presented. Specific plant characteristics impacting the IPE results are identified for each reactor type in Chapter 11.

10.2.1 BWR Plant Characteristics

General Electric (GE) supplies six different types of BWRs, known as BWR 1 through BWR 6, which encompass all operating BWRs in the U.S. These reactor types represent design vintages, which have evolved over time, starting with the BWR 1, which was introduced in 1955, and ending with the BWR 6, which was introduced in 1972. In general, these reactor types evolved with progressively refined design features and increased maximum power outputs. A summary of important design features for each BWR vintage that can impact the IPE results is presented in Table 10.2.

Table 10.2 BWR plant characteristics.[1]

Function/system	BWR 1	BWR 2	BWR 3	BWR 4	BWR 5	BWR 6	Comments
Number of units/multi-unit sites	1/0	2/0	7/2	19[2]/6	4/1	4/0	[2] Only 17 included in this study
RCS/PCS							
Turbine bypass capacity	100%	40%	15% to 105%	25% or 105%	25%	10% or 35%	
Number of recirculation loops/total number of jet pumps	2/0	5/0	2/20	2/16 or 20[3]	2/20	2/20 or 24[3]	[3] Typically 20 jet pumps
Number of feedwater pumps	2	3	2 or 3[4]	2[4] or 3	2 or 3[4]	2 or 3[4]	[4] Typical number
Type of feedwater pump	Motor-driven (M)	Turbine-driven (T) and/or M	M	T or M[5]	T and/or M[6]	T and/or M[6]	[5] 2 plants have M [6] 1 plant has 1 M & 2 T
Reactivity control							
SLCS	SLCS is a two-train system, which is either manually initiated (most plants) or auto-actuated. Some plants use enriched boron (one pump for success); others inject lower boron concentrations with two pumps.						
RCS overpressure protection							
Number of safety, relief, & Safety/Relief Valves	6/4[7]/0	16/5 or 6[7]/0	0, 2, or 8/ 0 or 4[8]/ 1, 3, or 6	0, 2, or 3/ 0/4 to 16	0/0/ 17 or 18	0/0/ 16 to 20	[7] power actuated relief valves (PARVS) [8] Includes PARVS
Coolant injection							
feedwater coolant injection (FWCI)		1 plant	1 plant				
high-pressure coolant injection (HPCI) or high-pressure core spray (HPCS)			HPCI[9]	HPCI	HPCS	HPCS	[9] Except for 1 plant
RCIC			Some[10]	All	All	All	[10] 2 plants have ICs; 3 have RCIC
low-pressure core spray (LPCS) (loops/total number of pumps)	2/2	2/4	2/2	2/2[11] or 2/4	1/1	1/1	[11] Typical number

Table 10.2 BWR plant characteristics.[1]

Function/system	BWR 1	BWR 2	BWR 3	BWR 4	BWR 5	BWR 6	Comments
low-pressure coolant injection (LPCI) (loops/total number of pumps)			2/4[12]	2/4[12]	3/3[12]	3/3[12]	[12] Mode of DHR system
Alternate injection systems	Plant-specific. Alternate injection systems typically include an enhanced CRDHS, condensate, service-water, and firewater.						
Decay heat removal							
IC	1	2 or 4	0 or 1[13]				[13] 2 plants have ICs; 3 have RCIC
SDC (loops/total number of pumps)	2/2[14]	3/3[14]	3/3[14], 2/2[14], or 2/4[15]	2/4[15]	2/2[15]	2/2[15]	[14] Single-mode SDC system) [15] Multi-mode RHR
CSS/SPC (loops/total number of pumps)	1/2[16]	2/4 or 4/4	2/4[17] or 2/4[18]	2/4[18]	2/2[18]	2/2[18]	[16] Mode of core spray [17] Mode of LPCI [18] Multi-mode RHR
Support systems	Support system configurations are plant-specific.						
[1]See text for definition of acronyms.							

Although the nuclear steam supply systems (NSSSs) used in each BWR vintage are nearly identical, there is some variation in the plant characteristics, including differences in the emergency operating procedures (EOPs). For example, the elevation of the emergency core cooling system (ECCS) pump suction in the suppression pool can vary, impacting the net positive suction head (NPSH) of the pumps and their continued operation under accident conditions. Protective trip setpoints that can be reached during an accident, such as the high-turbine exhaust pressure trip for high-pressure coolant injection (HPCI) and RCIC systems, can also vary, resulting in differences in the period these systems are available. In addition to these NSSS variations, there is considerably more variation in the design and operation of the balance-of-plant (BOP) systems. Variations in the design of support systems such as electrical power; cooling water systems; and heating, ventilation, and air conditioning (HVAC) systems can result in considerable differences in plant responses under accident conditions. The variation in BWR characteristics, both within and across BWR vintages, is discussed subsequently according to the major functions typically modeled in PRAs.

10.2.1.1 BWR Primary and Power Conversion System

The BWR primary system, or reactor coolant system (RCS), comprises the reactor vessel, core, internal structures, and two to five external recirculation loops. (The earliest BWR 1s used natural circulation.) The recirculation loops contain the pumps that force coolant flow through the reactor vessel. All BWRs currently operating in the U.S. have external recirculation loops.

Big Rock Point, the only BWR 1 currently operating in the U.S., uses two external recirculation pump loops, taking suction on a steam drum and discharging to the lower plenum of the reactor vessel. In BWR 2s, the recirculation loops contain motor-driven pumps, which drive all of the recirculation water through the core. In BWR 3s through BWR 6s, the core is physically separated from the recirculation loops with the only communication path through jet pumps. Core flow is provided by the combined action of motor-driven pumps in two external recirculation loops and jet pumps which enhance the amount of coolant provided to the core.

The jet pumps provide about two-thirds of the core flow rate. Because the core only communicates with the recirculation loops through the jet pumps, the jet pumps also serve as standpipes that ensure two-thirds coverage of the core following a recirculation line LOCA with successful ECCS operation. This feature allows for mitigation of large recirculation line breaks with emergency coolant injection systems in addition to core spray systems; by contrast, non-jet pump plants cannot be reflooded and require mitigation by a core spray system (or alternate system providing water through the core spray sparger) that sprays above the core.

The primary heat transfer system in a BWR consists of two fluid system loops. The primary heat transfer loop is comprised of the RCS and the power conversion system (PCS). Steam is generated in the reactor core and is supplied to the turbine generator to generate electricity. The main turbine generator exhausts to the main condenser, which transfers heat to the secondary water cooling loop. The condensed steam is returned to the reactor vessel by the main condensate and feedwater systems. The secondary cooling loop consists of the circulating water system, which rejects plant waste heat to the ultimate heat sink. Immediately following a reactor scram, decay heat is generally transported to the condenser through a turbine bypass path and/or to the suppression pool via relief valves or dual-function safety/relief valves (SRVs).

The PCS configuration can be important to IPE results in several areas. First, some BWRs have turbine-driven feedwater pumps, some have motor-driven feedwater pumps, and others have both. The type of feedwater pumps varies within each BWR type, since the PCS is not part of the NSSS. Turbine-driven feedwater pumps are powered by steam from the NSSS and thus trip when the NSSS supply is isolated upon a main-steam isolation valve (MSIV) closure that initiates or follows a reactor scram. Coolant injection must then be supplied by RCIC or other high-pressure injection systems (e.g., HPCI). However, for plants with motor-driven feedwater pumps, coolant injection can be provided for transients resulting in MSIV closure, as long as makeup water is provided to the condenser hotwell to compensate for the coolant lost to the containment through SRVs. Second, the turbine bypass capacity in BWRs ranges from 10% to 105%, with most plants having capacities in the 25% to 40% range. The turbine bypass capacity can be important in ATWS scenarios with boron injection failure where only reduction of the vessel coolant level is available to control power (see the following section). The higher the capacity, the more likely level control can result in reaching a stable power level without resulting in MSIV closure (as a result of a low-low vessel level signal) and a subsequent loss of turbine-driven feedwater and the condenser.

10.2.1.2 Reactivity Control Systems

BWR reactivity control is performed by three independent systems, which are used under different circumstances. These systems are the (1) reactor recirculation flow control system, (2) control rod drive hydraulic system (CRDHS), and (3) standby liquid control system (SLCS).

Recirculation flow rate directly affects the density of water in the reactor core, which in turn impacts the reactor power level in a BWR. During normal power operation, recirculation flow is controlled by the reactor recirculation flow control system. This system is not modeled in the IPEs; however, an important ATWS mitigating factor, tripping the recirculation pumps, is typically modeled. Typically, tripping the pumps decreases the power level to approximately 40% by increasing the voiding in the reactor core. For some BWRs, this power level is within the turbine bypass capacity. In addition, most BWRs have incorporated special recirculation pump trip (RPT) logic that functions during ATWS situations. Further power reduction to within the turbine bypass capacity of most BWRs can be achieved by decreasing the water level in the vessel, which increases the voiding in the core.

The CRDHS provides reactivity control for both long- and short-term reactivity changes and is used for rapid shutdown (e.g., reactor trip or scram). In all BWRs, the CRDHS consists of bottom-entry control rods that are individually controlled by hydraulic control units (HCUs) located outside the drywell. Directional control valves permit high-pressure hydraulic fluid (water) to enter on one side of the hydraulic piston, while simultaneously opening an exhaust path on the other side of the piston. Scram is accomplished by opening the scram inlet and outlet valves and deenergizing both scram pilot valves in each HCU to allow rapid insertion of all control rods. Alternatively, scram can be implemented by energizing either of the two backup scram pilot valves in the air supply path to the HCUs. Signals are provided to the scram valves by sensors and logic designed to respond to a wide variety of upset conditions. The scram valves and protective sensors and logic makeup what is referred to as the reactor protection system (RPS).

In all BWRs, an additional set of pilot valves exists in the scram valve air supply to provide backup scram capability. These valves are actuated in response to an ATWS by the alternate rod insertion (ARI) system; the actuation logic for the ARI system is independent from the RPS, but is tied to the ATWS-related recirculation pump trip logic. Failure of the ARI system in addition to the RPS system is required for an ATWS scenario to exist.

The SLCS system is also typically modeled, so such features as the type of initiation (automatic or manual) and plant-specific testing requirements can influence the reliability of this system as well. The SLCS is comprised of two trains of high-pressure, low-capacity pumps used to inject a concentrated boron solution into the reactor vessel. This provides a redundant and independent means to reach and maintain subcriticality in the event that an insufficient number of control rods can be inserted into the core to accomplish shutdown in the normal manner. Most BWRs have a manually initiated SLCS; however, a couple of plants have only automatically initiated systems. The actuation logic for automatic SLCS initiation can be tied to the logic used for the ARI system and for tripping the recirculation pumps.

PRAs generally do not model the RPS but instead treat the system as a whole with its overall reliability determined from an engineering analysis (Ref. 10.5). The degree of dependence between the ARI system, the RPT logic, and the remainder of the RPS (which logically provides the scram signals to the CRDHS, is often reviewed in a PRA (if not modeled directly) and can affect the overall combined reliability of both ARI and RPT. Additionally, the arrangement and level of redundancy in the RPT logic and the associated pump trip breakers can vary among plants and hence affect the failure probability of RPT.

10.2.1.3 RCS Overpressure Protection

Depending on the type, BWRs use various valve combinations to provide RCS overpressure protection and pressure relief, and to perform the automatic depressurization system (ADS) function that is part of the ECCS (see Section 10.2.1.4). The Big Rock Point BWR 1 and the various BWR 2s provide overpressure protection using mechanical safety valves that open on high-RCS pressure. In these plants, the ADS function is accomplished by PARVs that are controlled by a solenoid pilot valve. Most BWR 3s have safety valves and one or more safety/relief valves capable of both lifting mechanically upon high-pressure and being actuated at lower pressures to perform either pressure relief or the ADS function. However, some BWR 3s also have PARVs for performing the ADS function. All BWR 4s use safety/relief valves to provide overpressure protection, pressure relief, and depressurization. About half of the BWR 4s also have safety valves to accomplish the overpressure protection function. All BWR 5s and BWR 6s use only safety/relief valves to accomplish overpressure protection, pressure relief, and depressurization. BWR 3s through BWR 5s use a three-stage safety/relief valve. The BWR 6s use a safety/relief valve that is actuated by an external pneumatic piston.

The number of valves and their sizes dictate the success criteria for successful depressurization under various conditions (e.g., small LOCA, transients, ATWS events), as well as how many valves will be demanded to cycle under various scenarios. Additionally, the probability that the valves will successfully open when required, as well as the probability that a valve will stick in the open position depend on the valve type. This is because the different valves have a demonstrable difference in reliability. This variability can potentially impact the frequency of accident sequences requiring SRV operation.

The type of containment can influence the availability of SRVs during a loss of containment heat removal scenario, since valve opening requires that a pressure differential exist across the valve air supply and exhaust (the containment) and, thus, is affected by the containment pressure. In Mark I and II containments, containment pressure can exceed 100 psig and result in closure of SRVs, because of a loss of the required pressure differential. However, Mark III containments fail at pressures significantly less than 100 psig and, thus, closure of SRVs from high-containment pressures is not an issue.

10.2.1.4 Coolant Injection

Non-safety coolant injection into the vessel is provided by the feedwater/condensate systems. As indicated in Section 10.2.1.1, continued operation of feedwater following a reactor scram depends upon the type of transient and feedwater system design. When feedwater is not available, high-pressure coolant injection can be provided in BWR 3s through BWR 6s by the RCIC system (earlier BWRs do not have a RCIC system). The RCIC system uses a single steam turbine-driven pump that is supplied with steam from one of the main steam lines and exhausts to the suppression pool. The RCIC system initially takes water from the condensate storage tank and injects into the reactor vessel via a main feedwater (MFW) line in most BWR 3s (Millstone 1 and Dresden 2&3 do not have RCIC systems) and all BWR 4s. In BWR 5s and BWR 6s, the RCIC system initially injects from the condensate storage tank through either a reactor vessel head spray nozzle or through a feedwater line. In all BWRs, when the condensate storage tank is depleted, RCIC pump suction is aligned to the suppression pool. In some BWRs, the RCIC also will realign suction from the condensate storage system (CST) to the suppression pool upon a high-suppression pool level signal. At some plants, procedures requires the operators to defeat the switchover to the suppression pool upon a high-suppression pool level in order to prevent negative impacts on pump operability as a result of high-pool temperatures.

The ECCS injects makeup water into the RCS in the event of a LOCA or transient with loss of feedwater and RCIC (if available), and recirculates water through the core following the event in order to provide long-term core cooling. The ECCS typically comprises a number of integrated subsystems, including high-pressure injection or core spray, low-pressure injection (LPI) and/or core spray, and ADS for depressurizing the RCS (required for injection from the low-pressure systems). The high-pressure injection or core spray subsystem is designed to respond to small LOCAs and transients when feedwater is unavailable, while the low-pressure subsystem(s) is (are) designed to respond to large LOCAs.

For the Nine Mile Point (NMP)1 BWR 2 and the Millstone 1 BWR 3, the high-pressure ECCS subsystem is the FWCI system. The FWCI system uses the main condensate and feedwater systems (described in Section 10.2.1.1) to provide RCS makeup. It is different from other motor-driven feedwater systems in that the power sources are emergency buses powered either by diesel generators or gas turbine generators. Oyster Creek (a BWR 2) is the only plant that does not have a high-pressure ECCS.

For the other BWR 3s and all BWR 4s, the high-pressure ECCS subsystem is the HPCI system, which uses a single steam turbine-driven pump that is supplied with steam from one of the main steam lines and exhausts to the suppression pool. Initially, the HPCI system takes water from the condensate storage tank and injects it into the reactor vessel. When the suppression pool level is high or when the condensate storage tank is empty, pump suction is aligned to the suppression pool. At some plants, procedures require the operators to defeat the switchover to the suppression pool upon a high-suppression pool level in order to prevent negative impacts on pump operability as a result of high-pool temperatures. The HPCI system can provide makeup at RCS pressures from 1150 to 150 psig. Below 150 psig, operation is not possible because of poor steam conditions for the steam-driven pump and, thus, the system is automatically tripped.

In BWR 5s and BWR 6s, the high-pressure ECCS subsystem is the HPCS system, which uses a single motor-driven pump that initially takes water from the condensate storage tank and injects it into the reactor vessel via a spray sparger that is located above the core. Similar to the HPCI system, when the suppression pool level is high or the condensate storage tank is depleted, the pump suction is aligned to the suppression pool. The high-suppression pool level switchover is bypassed by the operator at some plants to prevent negative impacts on pump operability from high-suppression pool temperatures.

All BWRs have a LPCS system that injects water via spargers located above the core (the system is called a core spray system in BWR 1 through BWR 4s and is called an LPCS system for BWR 5s and BWR 6s). The spargers are supplied by motor-driven pumps that draw from the suppression pool. In BWR 1s through BWR 4s, the core spray system is comprised of two redundant trains with two redundant spargers. The number of pumps in a core spray train can vary from one to two as indicated in Table 10.2. BWR 5s to BWR 6s have a single-train LPCS, supplemented by the HPCS system, each with one pump and an independent sparger.

BWR 3s to BWR 6s also have a LPCI system to provide core flooding capability in the event of a large LOCA. BWR 1s and BWR 2s do not have an LPCI system. (Core flooding during a large recirculation line LOCA is not possible in these plants since they do not have jet pumps.) In early BWR 3s (Millstone 1 and Dresden 2&3), the LPCI system provides injection into the vessel and can be aligned for containment spray and suppression pool cooling. In later BWR 3s and all BWR 4s through BWR 6s, LPCI is an operating mode of the residual heat removal (RHR) system (which also performs containment spray, suppression pool cooling, and shutdown cooling functions). BWR 3s and BWR 4s have two LPCI trains, each with two pumps that inject via a recirculation loop (some BWR 4s inject directly inside the reactor vessel shroud), while BWR 5s and BWR 6s have three LPCI trains, each with one pump that injects directly into the reactor vessel shroud (only two of these trains can perform the RHR functions in addition to the LPCI function). During a large recirculation line LOCA, portions of LPCI flow is directly lost through the break for BWR 3s and most BWR 4s, whereas the LPCI injection inside the shroud in BWR 5s and BWR 6s (and a few late model BWR 4s) ensures that water passes through the core before being lost through the break.

Although the ECCSs for the various BWR vintages are all similar in basic design, there are some differences that can impact the system operability. For example, the elevation of the ECCS pump suction in the suppression pool is a variable in determining the NPSH of the pump and, thus, can impact when the pump will fail during an accident that results in adverse suppression pool conditions. Some ECCS pumps have external seal cooling to allow for operation with high-suppression pool temperatures, while other self-cooled pumps experience seal failure under the same conditions. Some systems (such as the RCIC and HPCI) also have protective trip logic with setpoints that can vary from plant to plant within a particular BWR vintage. Examples of these protective trips include a high-turbine exhaust back pressure trip and area temperature trips that are indicative of a steam leak in the system. Variability in these trip setpoints can result in variability in the time the system trips during accident sequences where these conditions would be expected to occur. These variations in trip times can be important in determining the probability of various recovery actions (e.g., recovering offsite power) and whether alternative injection systems, such as the CRDHS, can be used. Variations in support system requirements and configurations for the coolant injection components also influence the system availabilities (see Section 10.2.1.6 for further discussion). Finally, some adverse containment condition impacts (e.g., high-suppression pool temperature) on RCIC, HPCI, and HPCS can be delayed as long as the pump suction is aligned to the condensate storage tank. However, the condensate storage tank can eventually be depleted at different times for each plant (allowing for differences in accident recovery potential) dependent upon the condensate storage tank volume. These examples indicate that even though the ECCSs for the various BWR vintages may appear to be quite similar, subtle differences do exist in designs or arrangements that can impact the systems operations under accident conditions.

All BWRs currently operating in the U.S. have an ADS function consisting of primary relief valves (see Section 10.2.1.3) that were designed to automatically open to depressurize the RCS. The ADS is used to allow the low-pressure subsystem to provide core cooling when the high-pressure ECCS subsystem fails to perform adequately. Procedural changes employed at most plants direct the operator to inhibit this automatic function and to manually perform vessel depressurization when required by opening relief valves or by other means such as the turbine bypass.

In addition to the ECCS, most BWRs have the capability to use other systems to provide coolant injection to the vessel. The availability of these alternative injection systems is highly plant specific and includes such systems as service water cross-tied to inject into the RHR system, CRDHS, and firewater. All of the systems provide coolant from external sources. Most of them are not safety-grade. Some can only be successfully used after decay heat in the vessel has decreased to match the capacity (e.g., CRDHS) or when sufficient time is available to manually align the system (typical for firewater and also for service water cross-tie at some plants). Some of the systems (such as service water cross-tie) can immediately be aligned from the control room at some plants for successful coolant injection.

The containment design can influence the operability of coolant injection systems in several ways. For instance, the ultimate failure pressure of the containment (typically two to three times the design pressure), if high enough, can result in the forced closure of open relief valves under high-pressure conditions in the containment. This impacts the ability to maintain a depressurized reactor vessel to allow LPI pumps to inject into the vessel. As discussed in Section 10.2.1.3, this is a characteristic of Mark I and II containments but not Mark IIIs. Specific suppression pool designs and relative elevations of ECCS pump suction piping relative to the pool water level can impact the NPSH of the pumps under extreme high-pool temperature and low-pool level conditions. In fact, some BWR 6s have a suppression pool makeup system that must operate under LOCA conditions to maintain pump NPSH and also to prevent uncovery of the drywell-to-wetwell vents. The uncovery of the vents can result in suppression pool bypass resulting in containment overpressurization. Finally, the containment design determines the likely location where containment failure will occur. The location of containment failure can impact the continued operation of coolant injection systems through either direct effects (e.g., rupturing the coolant injection piping or failing the source of water such as the suppression pool) or by harsh environments in the reactor building that can fail important components such as electrical switchgear.

In total, the redundancy and diversity of the design of the ECCS, the specific support system needs to the ECCS (power, cooling, instrument air (IA), HVAC, etc.), flow and pressure capabilities of these systems, as well as the existence of any alternative injection systems which may be manually aligned, all have a significant impact on the success criteria for performing emergency injection/core cooling under a spectrum of LOCA and transient events. This in turn directly impacts the probability of failing to supply adequate injection/cooling under a variety of challenges to the plant.

10.2.1.5 Decay Heat Removal

In all BWRs, the same heat transfer loop used for normal power operation (see Section 10.2.1.1), consisting of the RCS and the PCS, is used for normal shutdown at high-RCS pressure. The main turbine is tripped and bypassed and the steam, condensate, and feedwater systems operate at a greatly reduced flow-rate. Variability in the PCS design, support system requirements, and protective trip setpoints can impact the use of this preferred system for decay heat removal under accident conditions.

If the PCS is unavailable, normal shutdown cooling is provided by other means, depending on the BWR type. In the Big Rock Point BWR 1, the BWR 2s, and early BWR 3s, back-up high-pressure cooling is provided by operation of an IC that also provides high-pressure decay heat removal. An IC is a simple condensing heat exchanger; on its primary side, it receives steam from the reactor vessel and returns the condensate to the reactor vessel via closed-loop, natural circulation. Secondary water is boiled in the IC and secondary steam is vented to the atmosphere. Makeup is provided to the secondary side by various systems including the condensate transfer system and the firewater system, which can function during a station blackout. Any significant loss of RCS inventory will defeat the use of the ICs.

High-pressure cooling in other BWRs is provided when steam is relieved to the suppression pool through relief valves and containment heat removal is initiated. Containment heat removal capability in BWRs typically includes suppression pool cooling (SPC) and the containment spray system (CSS). For the Big Rock Point BWR 1, both sump cooling (this plant has a dry containment) and containment spray are uniquely provided as separate operating modes of the LPCS system. The BWR 2s have a separate containment spray system, which also cools the suppression pool and can be aligned for suppression pool cooling. Early BWR 3s had a containment spray/suppression pool cooling capability as an operating mode of the LPCI system. All other BWR 3s through BWR 6s have a multi-mode RHR system that includes these functions. These systems all include heat exchangers that transfer heat to an ultimate heat sink through one or more intermediate cooling water systems.

Normally, the RHR system provides post-shutdown (low-RCS pressure) cooling. The Big Rock Point BWR 1, the BWR 2s, and some BWR 3s have dedicated shutdown cooling (SDC) systems, separate from the low-pressure ECCS subsystems, which take a suction on the reactor vessel and return coolant directly to the reactor vessel. For most BWR 3s and all BWR 4s through BWR 6s, the multi-mode RHR systems also provide post-shutdown cooling in addition to ECCS (i.e., LPCI) and containment cooling functions discussed above. Except for the BWR 2 and some BWR 3 plants, the SDC and RHR systems for most BWRs consists of two trains with one or two pumps and heat exchangers per train (see Table 10.2).

All BWRs with Mark I containments will have installed a hard-pipe vent that can be used to relieve pressure in either the drywell or wetwell area of the containment. The pressure at which the system is initiated is plant-specific and is generally a function of the size of the vent path. At some plants, venting can have an adverse impact on ECCS pump NPSH and can result in pump failure.

As with the ECCS, the level of redundancy and diversity of these systems, their support system needs, and their flow and pressure capacities dictate the success criteria for achieving decay heat removal for each plant. This in turn directly affects the reliability of this function under different challenges to the plant (particularly the degree of common cause failure affecting multiple modes of heat removal).

10.2.1.6 Support Systems

The support systems required by the coolant injection, decay heat removal, and other accident mitigating systems typically include electrical power, cooling water, and HVAC systems. The designs of these systems vary from plant to plant and can significantly impact accident mitigating system availability.

The onsite electric power system in all nuclear power plants consists of two parts, including (1) the non-Class-1E system, supplying non-safety loads, and (2) the Class 1E system supplying safety systems. Normally, onsite electric power is supplied from the output of the main generator and/or the offsite grid. Diesel generators provide backup alternating current (AC) power for the Class 1E portion of the system and batteries provide standby direct current (DC) power.

Loss of normal offsite power can cause an automatic shift to an alternative source of offsite power (available at some plant sites) and starts the standby diesel generator(s). If both sources of offsite power are unavailable, the non-Class-1E and Class 1E portions of the onsite electric power system are separated by opening various circuit breakers and the diesel generators are aligned to supply the Class 1E systems. The diesel generator control systems interface with a load-sequencing system that adds selected loads in prescribed sequences at the proper times. In addition to DC power, the diesel generators rely on a number of other support systems for operation (e.g., cooling water and HVAC).

The number of diesel generators at each plant is variable, ranging from one at Big Rock Point to four at various BWR 4 plants. BWR 5 and BWR 6 plants all have three diesel generators with one of them normally aligned to the HPCS system. (The HPCS division of AC power can be cross-tied at some plants to one of the other electrical divisions during station blackout conditions.) Some plants (such as Millstone 1) also have gas turbine generators available, while others have black-start diesel generators. (These diesel generators can start and operate without any outside support systems.) At multi-unit sites, one or two shared diesel generators are typically available in addition to dedicated diesels generators. However, some multi-unit sites (e.g., Brunswick 1&2) do not have any shared diesel generators and one (Susquehanna 1&2) has four diesel generators, all of which are shared between both units. Cross-tieing emergency buses between multi-unit plants is possible at some locations, and cross-tieing divisions of power at a single unit plant is also possible.

All BWRs have DC buses available to provide power to both safety and non-safety grade equipment. Normally, the DC power is provided from the AC buses through inverters and is backed-up by banks of batteries that are also charged from the AC buses. The number of battery banks at each plant is also variable, ranging from one to four. All plants (except Big Rock Point) have at least two 125-V battery banks and some also have one or two 250-V battery banks. The battery depletion time is an important factor in determining the response to a station blackout scenario. The battery depletion time is significantly impacted by the availability of load shedding procedures, which indicate a prescribed order to shed unnecessary loads during the station blackout period.

Typically, direct cooling for BWR components (such as pumps and diesel generators) is required to prevent failure during operation. This cooling is provided by one of several cooling water systems at a plant. The number and arrangement of cooling water systems is highly variable among the plants and BWR vintages. However, most plants have a reactor building closed cooling water (RBCCW) system that is used to cool safety and non-safety related loads and a turbine building closed cooling water (TBCCW) system that cools non-safety loads. Both systems are closed-loop designs and reject heat to an ultimate heat sink (cooling towers or a natural body of water) through the intermediate service water system(s). The service water system(s) also can be an open or closed loop (or combined) design. In some plants, the safety and non-safety loads are cooled by the same service water system, with non-safety loads isolated under accident conditions. In other BWRs, a standby service water system that only operates under accident conditions cools the safety loads, while a normal service water (NSW) system cools non-safety loads. At some multi-unit sites, the cooling water systems can be cross-tied to serve both units.

Room cooling is also required for some mitigating components and is provided by a variety of HVAC systems. Typically, room cooling is required for ECCS and service water pumps, diesel generators, electrical switchgear, and the control room. The HVAC systems can be once-through or can be recirculation systems that have cooling coils cooled by one of a number of diverse cooling water systems.

As identified above, the support system requirements for systems like ECCS and decay heat removal directly affect the overall reliability in the design and operating features of these architect-engineeredsystems. In addition, support systems (such as IA) can also impact other support systems and, thus, can indirectly affect accident mitigating system reliability. The resulting variability in the reliability of core and containment cooling can be significant. In fact, it has been found in past PRAs, that the support system features often dominate the estimated core damage frequency and the specific equipment failures or human errors most important to the core damage potential.

10.2.2 PWR Plant Characteristics

The three types of commercial PWRs found in the U.S. are supplied by Westinghouse (W), Combustion Engineering (CE), and Babcock & Wilcox (B&W). In all PWRs, the heat generated in the reactor core is transported to the secondary coolant system by the RCS via the external primary coolant loops with steam generators. Control and removal of heat from the reactor and conversion of this heat into usable electrical power requires a broad spectrum of operating, auxiliary, and safety systems, which are briefly described in the sections that follow.

Also discussed, are the differences among the various PWR types regarding the design, operation, and interfaces of these systems that can have a significant bearing on IPE results. In general, these differences occur in the primary, reactivity control, coolant injection, decay heat removal, and support systems. A summary of the design differences across the PWR types is presented in Table 10.3.

Table 10.3 PWR plant characteristics.[1]

Function/system	B&W	CE	W 2-Loop	W 3-Loop	W 4-Loop	Comments
Number of units/multi-unit sites	7/1	15/4	6/2	13/5	32/13	
RCS/PCS						
Number of loops/total number of pumps	2/4	2/4[2]	2/2	3/3	4/4	[2] One plant has 3 loops/pumps
Type of steam generator	Once-through	U-tube	U-tube	U-tube	U-tube	

Table 10.3 PWR plant characteristics.[1]

Function/system	B&W	CE	W 2-Loop	W 3-Loop	W 4-Loop	Comments
Reactivity control						
CVCS (number of pumps/type[3])	1, 2, or 3/C (3 typical)	3/PD (typical) or 3/C & 1/PD	3/PD	2 or 3/C or 3/PD (3/C typical)	2/C & 1/PD, 2 or 3/C, or 3/PD (2/C & 1/PD typical)	[3] C - Centrifugal PD - Positive Displacement
RCS overpressure protection						
Number of PORVs	1	0[4] or 2	2	2[5] or 3	1, 2[5], or 3	[4] System 80 plants [5] Typical number
Coolant injection						
high-pressure safety injection (HPSI) (number pumps)	2 or 3[6] (3 typical)	2 or 3[7] (3 typical)	2 or 3 (2 typical)	2 or 3[6] (3 typical)	2 or 3	[6] Most plants use charging pumps [7] One plant uses charging pumps
LPSI	2[8]	2[8]	2[8]	2[9]	2[9]	[8] Mode of RHR [9] Mode of RHR in some plants, separate system in others
high-pressure recirculation (HPR) piggyback off LPI?	yes	no 9-units yes 1-unit	yes	no 3-units yes 10-units	no 3-units yes 29-units	
Number of accumulators	2	4[10]	2	3	4	[10] One plant has 3
Decay heat removal						
Number/type[11] auxiliary feedwater pumps	1 or 2/M & 1/T, 2/M, or 2/T (1/M & 1/T typical)	1 or 2/M & 1/T (typical) or 2/T	1, 2, or 4/M & 1/T	1 or 2/M & 1/T (2/M & 1/T typical) or 2/T	1, 2 or 3/M & 1/T (2/M & 1/T typical) or 2/T	[11] M - Motor-Driven T - Turbine-Driven
RHR (number pumps)	2	2	2	2	2	
Support systems	Support system configurations are plant-specific.					
[1] See text for definition of acronyms.						

10.2.2.1 PWR Primary and Power Conversion System

The PWR primary system, or RCS, consists of the reactor vessel and, depending on the vendor, two to four external primary coolant loops. Each external primary coolant loop is equipped with a steam generator, one flow path from the vessel to the steam generator (hot leg), and one or two coolant flow paths from the steam generator back to the vessel (cold legs). The Westinghouse PWRs have 2-, 3-, and 4-loop configurations. A Westinghouse primary loop consists of a U-tube steam generator; a single vertical, centrifugal reactor coolant pump; and connecting loop piping. CE PWRs all have 2-loop configurations, with the exception of Palisades, which is a 3-loop plant. A CE primary loop consists of a hot leg that enters the bottom of a U-tube generator and two cold legs that return flow to the reactor vessel, with one reactor coolant pump in each cold leg. A primary loop in a B&W plant consists of a hot leg connected to the top of a once-through steam generator and two cold legs that exit the bottom of the steam generator and return flow to the reactor vessel, with one reactor coolant pump in each cold leg.

In all PWRs, the pressure in the primary system is controlled by a pressurizer connected to one primary loop hot leg. The pressurizer uses electric heaters to increase the pressure and spray to reduce the pressure. During normal operation, RCS coolant inventory is inferred from the pressurizer water level, and is controlled by the chemical and volume control system (CVCS).

During power operation, the normal heat transfer path consists of three fluid system loops, including the RCS, the PCS, and the tertiary coolant loop. This same flow path can remove decay heat following a normal plant shutdown. (The RCS is described above.) The PCS removes heat from the RCS by boiling water in the steam generators. The type and size of the steam generators vary among the PWR designs. The Westinghouse and CE PWRs use U-tube steam generators with large water capacities on the secondary side. The B&W PWRs use once-through steam generators that have a relatively small secondary side inventory that results in a more rapid boil-off of the steam generator secondary following a loss of all feedwater. The inventory of water on the secondary side of the steam generators is important because it affects the time to core uncovery (the RCS inventory boils off after loss of the steam generator secondary cooling) and, thus, the time available to recover before core damage occurs.

Generally, the PCS and the variations in design are only important in a PRA with respect to initiating a transient requiring mitigation. However, the impacts of RCS design differences on a PRA can be more substantial. Two major impacts are the steam generator size (previously discussed) and the reactor coolant pump seal design. Loss of seal cooling to the reactor coolant pumps (RCPs) is an important accident in PWRs resulting in a small LOCA.

Reactor coolant pump seals are of variable design as is cooling water system alignment. For example, at some plants, the same cooling system cools the RCP seals and the ECCS pump bearings or seals. Thus, loss of this cooling water system results in a seal LOCA with failure of the ECCS. ECCS actuation or containment isolation signals at some plants can also result in RCP seal cooling isolation requiring a recovery action to prevent subsequent failure.

10.2.2.2 Reactivity Control Systems

Reactivity control is provided by the control rod system and the CVCS. The control rod system provides short-term control and rapid shutdown. Automatic reactor trip is initiated by the RPS by opening the circuit breakers supplying power to the control rod system. Although the configuration of the RPS circuitry can vary, in general, the system acts to deenergize the drive mechanisms, which allows the control rods to fall via gravity into the reactor core. During normal power operation, the CVCS continuously compensates for long-term reactivity changes by adjusting the boron concentration in the primary coolant. If the control rods fail to insert following an abnormal occurrence, the CVCS alone can take the reactor subcritical by significantly increasing the boron concentration. Boron injection is required even if rod insertion is successful since the control rods by themselves are not sufficient to reach a cold shutdown condition. The water injected by the ECCS during a LOCA is thus borated.

PRAs typically do not model the RPS, but like the BWRs, often use a lumped-model approach with reliability based on engineering evaluation. However, to the extent that the number of scram relays and their design configurations are modeled or otherwise considered, some differences may be estimated in the failure-to-scram probability.

10.2.2.3 RCS Overpressure Protection

RCS overpressure protection is accomplished by power-operated relief valves (PORVs) and/or safety valves mounted on the pressurizer. Most PWRs have PORVs, which can be controlled to open at lower pressures to reduce the demands on the mechanically lifted (upon high-pressure) safety valves. The PORVs and safety valves discharge to a quench tank inside the containment. When the quench tank is full, protective disks burst to allow primary coolant to flow from the quench tank into the containment sump.

Feed-and-bleed cooling can be used in PWRs as a means of removing decay heat (see Section 10.2.2.5). Many PWRs with HPSI pumps with low-shutoff heads require opening PORVs to reduce the RCS pressure for feed-and-bleed. The size of the PORVs and, thus, the number required to reduce RCS pressure can vary from plant to plant. Also, some plants have operated in the past with their PORV block valves closed to prevent minor leakage past the PORVs. This can impose an additional step in the depressurization process (open the block valves), which can affect the overall reliability of feed-and-bleed and/or primary system depressurization, as well as the ability of the primary system to withstand the pressure pulse from an ATWS event since the PORV pathways may be isolated. These factors are generally included in a PRA and can influence the resulting estimate of core damage. Note that opening PORVs for feed-and-bleed operation is not required at plants with HPSI systems that have shut-off heads above safety valve setpoints.

10.2.2.4 Coolant Injection

The ECCS provides long-term core cooling by injecting makeup water into the RCS in the event of a LOCA and recirculating water through the core following the event. In all PWRs, the ECCS includes pressurized safety injection tanks (also referred to as accumulators) and HPSI and low-pressure safety injection (LPSI) pumps. In Westinghouse PWRs, these pumps initially inject into the cold legs, then into the hot legs while the safety injection tanks inject into the cold legs. In CE plants, all three subsystems inject into the colds legs. In B&W PWRs, the HPSI pump injects into the cold legs, while LPSI pump and the safety injection tanks inject directly into the reactor vessel. The set point at which the passive safety-injection tank injects into the vessel varies from plant to plant.

During a large LOCA, the ECCS rapidly injects borated water to the RCS to shut down the reactor and provide core cooling. The RCS rapidly depressurizes and makeup is initially provided by the safety injection accumulators. Both HPSI and LPSI pumps deliver makeup to the reactor vessel from the refueling water storage tanks (RWSTs). ECCS injection to the vessel must also be switched during a large cold leg break from cold leg injection to hot leg injection to prevent boron precipitation which can result in core flow blockage and fuel damage. Not all PRAs model the need for switching to hot leg injection.

Following a small LOCA, the RCS may slowly depressurize or remain at normal operating pressure. In all Westinghouse 2-loop plants, some 3-loop and 4-loop plants, some B&W plants, and all CE 2-loop plants, the limited HPSI pump shutoff head is insufficient to overcome normal RCS operating pressure. Therefore, the RCS must depressurize before HPSI pumps can provide makeup. Depressurization of the RCS is accomplished by opening the PORVs. In some PWRs, both the RCS and the steam generator secondaries can be depressurized during a small break LOCA to reduce RCS pressure low enough to allow injection with LPSI if HPSI fails. The remaining Westinghouse 3-loop plants, CE's Maine Yankee plant, and the majority of the B&W plants use centrifugal CVCS pumps (see Section 10.2.2.2) to perform the HPSI function and can provide makeup to the RCS against full system pressure. Coolant injection with LPSI following depressurization of the RCS can also be used in some plants.

Once the RWST makeup water is exhausted, the ECCS is switched to a recirculation alignment where the LPSI pumps take suction from the containment sump, which contains water released from the RCS through the break. Following a large LOCA, low-RCS pressure allows the LPSI pumps to provide makeup to the RCS without the operation of the HPSI pumps. In the event of a small LOCA, high-RCS pressure can preclude LPSI pump injection, and in most PWRs, the HPSI pumps cannot be aligned to take suction directly from the containment sump. In these plants, the HPSI pumps must operate in tandem with the LPSI pumps. The LPSI pumps take suction from the containment sump and deliver the water to the HPSI pump suction, which in turn injects water into the RCS. In contrast, many CE plants only use high-pressure ECCS pumps for the recirculation mode of injection. The switchover to recirculation is a manual action in some plants, while other plants have a semi-automatic or fully automatic switchover. Since the capacity of the RWST varies among the plants, the time available to accomplish manual actions during switchover also varies.

The containment design, particularly the containment heat removal, can impact the operability of ECCS in the recirculation mode. The impact of insufficient containment heat removal leading to high-sump water temperatures and low-water levels on pump operability is plant-specific.

The redundancy and diversity of the design of the ECCS, the specific support system needs to the ECCS (power, cooling, IA, HVAC, etc.), flow and pressure capabilities of these systems, as well as variations in system setpoints, all have a significant impact on the success criteria for performing emergency injection/core cooling under a spectrum of LOCA and transient events. This in turn directly impacts the probability of failing to supply adequate injection/cooling under a variety of challenges to the plant.

10.2.2.5 Decay Heat Removal

In all PWRs, the same heat transfer loop used for normal power operation (see Section 10.2.2.1), consisting of the RCS, the PCS, and the circulating water system, is used for normal shutdown (at high-RCS pressure), with one exception. The main turbine is tripped and bypassed and the steam, condensate, and feedwater systems operate at a greatly reduced flow-rate.

If the steam and power conversion system is not available, heat can be removed from the RCS by the auxiliary feed water (AFW) system (referred to as emergency feed water (EFW) in B&W plants) and the secondary steam relief system (SSRS). This involves transferring heat from the reactor core to the steam generators using forced circulation or natural circulation when the reactor coolant pumps are not available. The AFW draws water from the condensate storage tank or another source and supplies it to the steam generators, where it is boiled and vented to the atmosphere via atmospheric dump valves in the SSRS. Typically, most PWRs have both motor-driven and steam-driven AFW pumps. However, a few plants only have turbine-driven pumps. The number and flow capacity of the AFW pumps varies among the plants and, thus, the number required to remove heat following a transient can be variable. If all feedwater is lost to the steam generators, the condensate system can be used for cooling by depressurizing the secondary side of the steam generators by opening atmospheric dump valves. This capability is dependent upon the size of the atmospheric dump valves and the capacity of the condensate system, which are plant-specific features.

Most PWRs can use the high-pressure ECCS pumps to implement feed-and-bleed cooling, which is a post-transient decay heat removal method. Certain CE PWRs that do not have PORVs, including ANO 2, Palo Verde 1,2&3; San Onofre 2&3; and Waterford 3, are not capable of feed-and-bleed cooling. As mentioned in Section 10.2.2.3, feed-and-bleed cooling is similar to a small LOCA mitigation, but with intentional use of the pressurizer PORVs, rather than the occurrence of a break. This is implemented by aligning a high-pressure pump to provide makeup to the RCS and by modulating the PORVs to control RCS cooldown rate. Feed-and-bleed operation will overfill the pressurizer quench tank and the tank rupture disks will burst and vent the tank to the containment. The containment cooling systems (see Section 10.3.2) transport the heat to the ultimate heat sink.

In most Westinghouse and B&W plants, heat exchangers in the low-pressure ECCS recirculation systems (see Section 10.2.2.4) provide decay heat removal from water collected in the containment sump before it is reinjected into the RCS. If a plant does not have heat exchangers in the ECCS recirculation system, there are typically heat exchangers in the containment spray recirculation system.

For post-shutdown (low RCS pressure) core cooling, a multi-mode RHR system (which also provides LPSI as part of the ECCS) transfers RCS hot leg coolant to the RHR heat exchangers. Heat is transferred from the RHR system to the component cooling water (CCW) system, which forms the secondary cooling loop, then to a tertiary loop that rejects heat to the ultimate heat sink.

As with the ECCS, the level of redundancy and diversity of these systems, their support system needs, operational characteristics (such as protective trip setpoints), and flow and pressure capacities dictate the success criteria for achieving decay heat removal for each plant. This in turn directly affects the reliability of this function under different challenges to the plant (particularly the degree of common-cause failure affecting multiple modes of heat removal).

10.2.2.6 Support Systems

The support systems required by the coolant injection, decay heat removal, and other accident mitigating systems typically include electrical power, cooling water systems, and HVAC. The designs of these systems vary from plant to plant and can significantly impact mitigating system availability.

The onsite electric power system in all PWRs consists of two parts, including (1) the non-Class-1E system supplying non-safety loads, and (2) the Class 1E system supplying safety systems. Normally, onsite electric power is supplied from the output of the main generator and/or the offsite grid. Diesel generators provide backup AC power for the Class 1E portion of the system and batteries provide standby DC power. (Oconee is unique in that it does not have diesel generators, but uses an upstream turbine plant for power.) Loss of normal offsite power typically causes an automatic shift to the alternative source of offsite power and starts the respective standby diesel generator(s). If both sources of offsite power are unavailable, the non-Class-1E and Class 1E portions of the onsite electric power system are separated by opening the circuit breaker and the diesel generators are aligned to supply the Class 1E systems. The diesel generator control systems interface with a load-sequencing system that adds selected loads in prescribed sequences at the proper times. In addition to DC power for starting purposes, the diesel generators typically rely on a number of support systems for operation.

The number of diesel generators at each plant is typically two or three. Some plants (such as Salem 1&2 also have gas turbine generators available, while others (e.g., Robinson 2) have a dedicated shutdown diesel generator. At multi-unit sites, one or three shared diesel generators are typically available in addition to dedicated diesels generators. However, most multi-unit sites do not have any shared diesel generators and some (e.g., Calvert Cliffs 1&2) share all of the diesel generators between units.

All PWRs have DC buses available to provide power to both safety and non-safety grade equipment. Normally, the DC power is provided from the AC buses through inverters and is backed up by banks of batteries that are also charged from the AC buses. The number of batteries at each plant is also variable, ranging from one to four. All plants have at least two 125-V batteries and a few have 250-V batteries.

Typically, direct cooling for PWR components (such as pumps and diesel generators) is required to prevent failure during operation. This cooling is provided by one of several cooling water systems at a plant. The number and arrangement of cooling water systems is highly variable among the plants. However, most plants have a CCW system that is used to cool safety and non-safety related loads. This system is closed loop and rejects heat to an ultimate heat sink (cooling towers or a natural body of water) through the intermediate service water system(s). The service water system(s) also can be open or closed loop (or combined) design. Cross-tieing of cooling water systems is possible at some multi-unit sites.

Room cooling is also required for some mitigating components and is provided by a variety of HVAC systems. Room cooling is often required for ECCS and service water pumps, diesel generators, electrical switchgear, and the control room. The HVAC systems can be once-through or can be recirculation systems that have cooling coils cooled by a fluid system.

As identified above, the support system requirements for systems like ECCS and DHR directly affect the overall reliability in the design and operating features of these architect-engineered systems. The corresponding variability in the reliability of core and containment cooling can be significant. In fact, past PRAs have revealed that the support system features often dominate the estimated core damage frequency and the specific equipment failures or human errors are most important to the core damage potential.

10.3 Containment Characteristics

This section discusses the differences in the containments used for BWR and PWR reactors, focusing on how these differences can influence the accident progression analysis performed in the IPEs. The principal emphasis is on containment design variations, but differences in operation and interfaces with the plant systems important for the core damage frequency analysis are also presented. Differences in containment structural features, vapor suppression systems, containment heat removal, combustible gas control, containment venting, and containment bypass and isolation issues, that are important for understanding the IPE results are discussed. Additional plant-specific containment features that impact the IPE results for a particular plant are identified in Chapter 12, where the IPE accident progression results for the different containment types are considered.

10.3.1 BWR Containment Characteristics

Of the two types of commercial, power producing reactors used in the U.S., (i.e., BWRs and PWRs), BWRs operate at a lower pressure (approximately half that of PWRs) and generally have smaller containments. In addition to their structural strength, all BWR containments, except that of the Big Rock Point, rely on water pools to promptly condense steam to prevent overpressure. The Big Rock Point BWR 1 plant is housed in a large, dry, spherical steel containment which is functionally similar to the type of containment used for most PWR plants. These pressure suppression containments all consist of (1) a "drywell" which encloses the reactor vessel, (2) a pressure suppression chamber called "wetwell" containing a large amount of water (suppression pool), (3) a vent system connecting the drywell and the suppression pool, (4) containment isolation valves, (5) containment cooling systems, and (6) other service equipment.

The primary containment system is designed to (1) condense the steam released during a postulated LOCA and provide a temporary heat sink, (2) limit the release of radionuclides in an accident, and (3) provide a source of water for the ECCS. If a failure of the primary system pressure boundary occurs inside the pressure suppression containment, the drywell is pressurized and its atmosphere is routed via a vent system to discharge points beneath the water surface of the suppression pool. In this manner, steam is condensed in the pool and any radionuclides are scrubbed as they pass through the pool. The geometry and arrangement of the drywell, wetwell, and vent system differs depending on the containment type.

The three pressure suppression containments used for currently operating BWR reactors are (1) the Mark I type used for BWR 2, BWR 3, and most BWR 4 reactors; (2) the Mark II type used for late BWR 4 and BWR 5 reactors; and (3) the Mark III type used for BWR 6 reactors. Figure 10.3 compares the general arrangements of the Mark I, II, and III containments. Table 10.4 summarizes the important BWR containment characteristics. Again, with the exception of Big Rock Point, all U.S. BWRs have a secondary containment, which surrounds all or most of the primary containment structure.

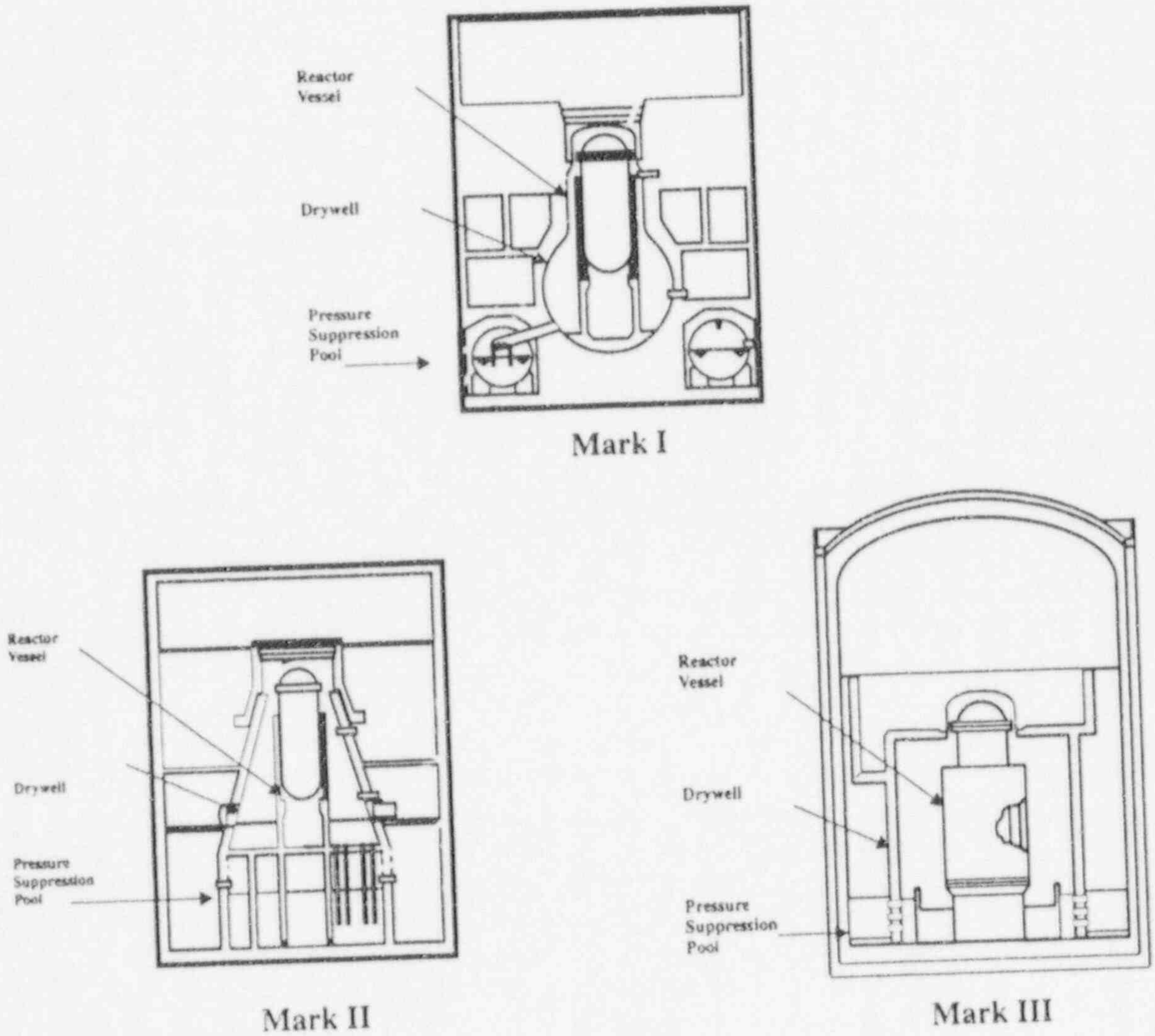


Figure 10.3 BWR pressure suppression containments.

Table 10.4 BWR containment characteristics.

Characteristic	Containment type		
	Mark I	Mark II	Mark III
Number of units	24	8	4
Reactor thermal power (MWt)	1593 – 3293	3293 – 3323	2894 – 3833
Containment free volume (ft^3) Drywell Wetwell Total	 110,000 – 180,000 90,000 – 140,000 200,000 – 320,000	 200,000 – 310,000 140,000 – 190,000 340,000 – 500,000	 250,000 – 280,000 1,165,000 – 1,550,000 1,440,000 – 1,800,000
Cont. volume to thermal power ratio (ft^3/MWt)	90 – 160	100 – 150	440 – 620
Containment strength Containment design pressure (psig) Median containment failure pressure (psig) used in IPE	 56 – 62 98 – 190	 45 – 55 140 – 191	 15 56 – 94
Containment construction	22 steel 2 concrete	1 steel 7 concrete	2 steel 2 concrete
Vapor pressure suppression system	Vent header with vertical vents (downcomers) DW/WW vacuum breakers	Vertical vents (downcomers) DW/WW vacuum breakers	Horizontal vents and SPMU[1] DW/WW vacuum breakers[2]
Containment heat removal system[3]	RHR[1] system in SPC[1] or DWS[1] mode	RHR system in SPC or DWS mode	RHR system in SPC or DWS mode[4]
Combustion gas control	Inerted by N2	Inerted by N2	Igniter System
Containment venting for pressure control	Hardened vent pipe requested by CPI[1]	Hardened vent pipe not requested by CPI	Hardened vent pipe not requested by CPI

Notes:
[1] RHR - Residual Heat Removal; SPC - Suppression Pool Cooling; DWS - Drywell (or Containment) Spray System; SPMU Suppression Pool Makeup; CPI - Containment Performance Improvement
[2] River Bend does not have an SPMU system or DW/WW vacuum breakers
[3] There is also a fan cooler system for CHR during normal plant operation. It is not a safety system and is usually not credited in the PRA.
[4] River Bend does not have a containment spray system but has two safety-related containment unit coolers.

10.3.1.1 Containment Structure

The most common BWR containments are the Mark I type, which is used at 24 BWR facilities in the U.S. For these containments, the drywell is typically a steel pressure vessel supported in concrete, with a spherical lower section and a cylindrical upper section (light bulb shape). The suppression chamber also is typically a steel pressure vessel but it takes the shape of a torus located below the drywell and encircling it. This construction is typical of all but two Mark I units (Brunswick 1&2) which use a steel-lined, reinforced concrete drywell and suppression chamber instead of steel vessels.

Table 10.4 shows the range of containment design pressures for the Mark I containments. Because of the conservatism in design and construction, these containments are unlikely to fail when the design pressure is reached. A containment failure pressure is estimated in the IPEs to predict the containment failure probability under pressure load. Table 10.4 shows the range of the median values of containment failure pressures used in the IPEs. The failure pressure is usually about 2 to 3 times above the containment design pressure.

The values presented in Table 10.4 for containment failure pressure are those at normal containment temperature. For Mark I containments, the drywell temperature can be very high during a severe accident. As a result, the containment pressure capability as a function of containment temperature is an important consideration. Usually, a pressure-temperature containment performance plot is presented in the IPE submittal to show containment pressure capabilities at various temperatures. Although the median containment failure pressure may be greater than 100 psig at normal temperature, containment failure may occur at a much lower pressure at high-temperature. At high-temperature (e.g., 900° F), containment failure may be caused by the large upward and radial thermal growth of the containment. For example, containment failure may occur if some of the numerous small and large penetrations bind on the biological shield wall and fail, or if the radial growth of the containment causes the seismic stabilizers to punch through the upper portion of the drywell. In addition, sealing material may completely degrade at high-temperature.

Typical containment volumes are also indicated in Table 10.4. The Mark I containments have the smallest free air volume of the three BWR pressure suppression designs, with only about one-sixth the free air volume of a large-dry containment.

There are eight BWR units with Mark II containment designs in the U.S. In a typical Mark II containment, the two parts of the primary containment (the drywell and the wetwell) comprise a structurally integrated concrete pressure vessel lined with welded steel plate and provided with a steel domed head for closure at the top of the drywell. Seven Mark II units (NMP2, Limerick 1&2) and Susquehanna 1&2, use reinforced concrete, while two units (LaSalle 1&2) use post-tensioned concrete. Instead of a concrete structure, one Mark II containment
Washington Nuclear Plant 2 (WNP 2) has, instead of a concrete structure, a free-standing steel primary containment, surrounded by a reinforced concrete structure providing support and biological shielding.As shown in Table 10.4, the containment free volume is greater for Mark IIs than for Mark Is. However, the containment volume to thermal power ratio is similar for both containment types. Design and failure pressure ranges for the Mark II containments are shown in Table 10.4.

Four commercial BWRs operating in the U.S. use the Mark III containment design. For two of the Mark III designs, the primary containment is a steel reinforced concrete structure, consisting of a vertical cylinder, a hemispherical dome, and a flat base. The concrete provides both structural strength and biological shielding. A thin welded steel liner plate is used on the inside surface to form a leakage barrier. Two other plants use a free standing steel primary containment made with steel plate. The structure consists of a vertical cylinder with an ellipsoidal head and a flat bottom steel liner plate. The entire structure is anchored to the concrete basemat. While providing both structural strength and a leakage barrier, the free-standing steel primary containment offers little biological shielding. This is provided by the reinforced concrete secondary containment.

The main internal structures are, for the most part, common to all Mark IIIs. The drywell is a cylindrical reinforced concrete structure with a flat roof slab. A circular opening in the roof slab is covered with a removable steel head to allow access for refueling. Enclosed within the drywell are the reactor vessel and a large portion of the reactor coolant pressure boundary including the coolant recirculation loops and associated pumps. This seismic Category I structure was designed to contain LOCA pressure transients and direct the resulting air-steam mixtures into the suppression pool. The drywell also provides support for the upper containment pool, as well as radiation shielding and protection for the containment from pipe whip, missiles, and jet impingement. The containment volume outside the drywell consists of the upper dome and the lower wetwell. It includes the suppression pool, the annulus formed by the drywell outer wall and the primary containment inner wall, and the upper containment pool plus the volume above it.

As shown in Table 10.4, the total free volume for a Mark III containment is significantly greater than that for a Mark I or Mark II containment. The containment volume to thermal power ratio for Mark IIIs is about four times that for Mark Is or Mark IIs, while, as Table 10.4 also shows, the containment design pressure as well as the estimated failure pressure are significantly lower than those of Mark Is and Mark IIs.

Surrounding and, in the case of Mark I and Mark II containments, completely enclosing the primary containment is the secondary containment, which includes the reactor building, the reactor building heating and ventilation system, and the standby gas treatment system (SGTS). The secondary containment is designed to provide a controlled, filtered, and elevated release (through a stack) of the reactor building atmosphere. The reactor building can play a role in severe accident progression since it provides additional radionuclide retention from natural processes such as aerosol deposition. The SGTS is designed to provide a filtered release of the reactor building atmosphere at an elevated release point. Although the SGTS was not designed for a severe accident and may not have the capacity to handle the release from such an accident, a judicious use of the system may help to mitigate radionuclide release.

10.3.1.2 Vapor Pressure Suppression

The geometry and special features of the vapor suppression system depend very much on the containment type. The components of the vapor suppression system are the vent system connecting the drywell and wetwell, the vacuum breakers between the drywell and wetwell, and, in some cases, the suppression pool makeup (SPMU) system. These features are discussed below.

As Figure 10.3 indicates, in a Mark I containment, the vent system consists of a series of main vents emanating from the outside drywell wall, about two feet above the drywell floor. The main vents enter the toroidal wetwell structure and connect with a manifold system, or vent header, which distributes the flow to numerous pairs of vertical pipes, called downcomers. These downcomers have their exit below the surface of the suppression pool. In the "over/under" Mark II configuration of the drywell and wetwell, the two are simply connected by vertical pipes that have openings uniformly distributed, radially and circumferentially, around the drywell floor, and exit below the surface of the cylindrical suppression pool. As shown in Figure 10.3, the suppression pool encircles the drywell in the Mark III design. In this design, the downcomers have been replaced with three rows of horizontal vents through the drywell wall, and a weir wall has been added to achieve the separation of drywell and wetwell.

Vacuum breakers are provided between the drywell and wetwell (or the containment) to limit the buildup of a negative pressure differential between the drywell and the wetwell (i.e., drywell pressure lower than wetwell pressure). In the Mark I and Mark II designs vacuum breakers are provided mainly to protect the drywell integrity (i.e., to prevent structural failure of the drywell as a result of external pressure). In the Mark III design, a negative pressure differential can cause an overflow of the weir wall with suppression pool water flowing into the drywell. This will reduce the suppression pool level and flood the drywell cavity. With the reduced suppression pool level a clearing of the top row of horizontal vents is possible (i.e., a reverse vent clearing) with subsequent gas flow from the containment to the drywell. The vacuum breakers in the Mark III design are used to equalize the pressure between the containment and the drywell, thereby preventing the above scenario.

Vacuum breakers are provided in all BWR pressure suppression containments, except in the River Bend Mark III containment. River Bend relies on reverse vent clearing to eliminate the negative pressure differential between the drywell and the containment.

A failure of the vacuum breakers in the open position may result in suppression pool bypass and eventual containment failure. This failure mode is considered in the accident progression analysis of some of the IPEs. Suppression pool bypass also means that the ability to scrub radionuclides in the pool is lost or severely impaired (depending on the amount of bypass).

The Mark III design incorporates an upper containment pool and an SPMU system, which provides a means of rapidly replenishing the suppression pool via a gravity feed from the upper pool to ensure that there is an adequate water volume in the suppression pool to keep the horizontal vents covered under all circumstances. Failure of this system may cause suppression pool bypass and/or insufficient NPSH for the pumps taking suction from the suppression pool. River Bend is the only Mark III design that does not incorporate SMPU via an upper pool dump. In this plant the upper pool dump is not required because the suppression pool water inventory is sufficient.

Excessive suppression pool temperature in any of the pressure suppression containments can lead to large loads from unstable steam condensation. This failure mode is considered in some IPEs. The containment is assumed to fail when substantial power is being produced in the core and discharged into the pool at a temperature exceeding 260°F.

This is because of the concern raised by the following suppression pool issues:

- condensation phenomena
- temperature profile at the quencher device
- limitation of calculated models
- vacuum breaker performance with cycling drywell sprays
- containment structural capability under hydrodynamic loads
- cycling pressure effects
- elevated pool water levels affecting hydrodynamic loads

As noted previously, besides containment failure, a saturated pool may cause a failure of the pumps that take suction from the pool.

10.3.1.3 Other Containment Features

Particular characteristics of the different types of BWR suppression containments can play a significant role in determining the containment's robustness or vulnerability in the face of severe accident phenomena.

For instance, for the layout and construction of the Mark I design, a failure mode has been postulated for the possibility of melt-through if the containment shell comes in direct contact with core debris. Because the reactor pedestal and drywell floor in this design are at the same level, and because openings exist between the pedestal region and the floor, core debris exiting the failed reactor vessel, even at low-pressure, could, in theory, flow across the drywell floor and contact the steel containment shell. If the hot debris contacts the drywell shell, two failure modes may occur:

(1) The combined effects of elevated containment pressure and local heating of the steel shell may result in a creep rupture.

(2) If hot enough, the debris may melt-through the steel shell.

These two modes are usually referred to collectively as "drywell shell melt-through" or "shell melt-through." The probability of shell melt-through is known to depend on the condition of the core debris (i.e., physical state, composition, release rate from the reactor vessel, etc.) as it relocates to the reactor pedestal floor, the configuration of the reactor pedestal region (e.g., the sump volume and the doorway through the pedestal wall), and containment conditions (e.g., presence of water on the drywell floor). Plant-specific features can determine the importance of this failure mode for individual Mark I containments. For example, as noted above, the Brunswick Mark I containment uses concrete and not steel and, therefore, the IPE for this plant did not consider shell melt-through a threat. Another IPE submittal stated that, because of the plant's large sump volume and relatively small core volume, shell melt-through will not occur.

Drywell shell melt-through is not a significant failure mode for Mark II containments because of geometry differences and because all but one are of concrete construction. For the Mark II containment that uses a steel shell, the containment failure modes considered in the IPE include those that would be induced by the impact of hot core debris on the containment shell during high-pressure melt ejection (HPME). However, there are certain unique Mark II features that are important for particular accident scenarios.

The design of the region inside the reactor pedestal of a Mark II containment may have a significant effect on the progression of a severe accident after the debris is discharged onto the drywell floor. The design features that are most important to accident progression are the relative elevation of the in-pedestal floor to the drywell floor and the existence of downcomers inside the pedestal region. In general, the BWR 5 plants have a recessed in-pedestal region (reactor cavity) and the BWR 4 plants have a flat in-pedestal floor at approximately the same elevation as the ex-pedestal drywell floor. Among the domestic Mark II plants, only NMP2 has downcomers inside the pedestal region.

After a vessel failure and discharge of core debris, a recessed cavity would confine the core debris in the cavity. Extensive core-concrete interaction is expected to occur because the potential for corium cooling is minimal. On the other hand, a shallow reactor cavity would allow the corium to spread out through the personnel pathway onto the drywell floor. A portion of the corium could enter the first row of downcomer pipes. The remainder would be cooled by heat losses to the containment atmosphere and the drywell floor and by the drywell spray if it is operational.

For a plant that has downcomers in the pedestal region, corium released from the vessel would rapidly enter the suppression pool. This design may eliminate the problems associated with core-concrete interaction, if the corium is primarily in a liquid phase and the vessel is not pressurized, but this increases the potential for a severe and damaging fuel-coolant interaction (FCI).

A steam explosion as corium flows down the downcomer pipes into the suppression pool, or the thermal attack of the downcomers by corium, could also fail the downcomer pipes and cause a suppression pool bypass. Such a bypass could have a significant impact on containment integrity and, therefore, radionuclide release.

In addition to the downcomer pipes, drain tubes located in the drywell floor could fail as a result of corium attack. This would result in a suppression pool bypass and FCI when the corium falls into the suppression pool through the failed drain tubes.

In the Mark III design, because the drywell is completely enclosed by the primary containment, 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 the risk is most affected by containment failures in which both the drywell and the containment 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.

Among Mark III designs, there is variation in the reactor pedestal region. The design of this region is unique in River Bend among Mark IIIs in that the access door is water tight and kept closed while the plant is operating. Therefore, the probability of water accumulating in the pedestal region before vessel failure is much lower in River Bend than in other Mark IIIs.

10.3.1.4 Containment Heat Removal

While the suppression pool of BWR pressure suppression containments provides a short-term heat sink, in the long term, containment heat removal is usually accomplished either by directly cooling the suppression pool or by cooling the containment spray water recirculated from the suppression pool. The SPMU system of Mark III containments can increase the suppression pool volume and extend its heat capacity.

With the exception of some early Mark I containments, containment/suppression pool cooling is one of the operating modes of the multi-mode RHR system, also used to provide shutdown cooling in one of its other modes as discussed in Section 10.2. The BWR 2 Mark Is have a single-mode containment spray system and no suppression pool cooling system. In two early BWR 3 plants (Millstone 1, and Dresden 2&3) containment spray and pool cooling is an operating mode of the LPCI system. In Big Rock Point (a BWR 1), containment spray is provided as an operating mode of the LPCS system. All of these systems incorporate heat exchangers to transfer the containment heat to an ultimate heat sink via intermediate cooling systems. The RHR systems for most BWRs have two separate loops with two pumps and heat exchangers per loop. Heat can of course be removed directly from the core as in the shutdown cooling mode of the RHR system.

In most Mark I plants, high-pressure shutdown cooling is provided by the RCIC system. However, an IC (or emergency condenser) is used for high-pressure shutdown cooling in BWR 2 plants (Oyster Creek and NMP1) and early BWR 3 plants (Dresden and Millstone). In one IPE, a failure mode related to the IC was considered. According to the NMP1 IFE, the tubes of the emergency condenser, if not isolated, may fail as a result of high-temperature during core melt progression. This would result in a tube failure outside the containment. This failure mode is considered in the IPE as a bypass failure similar to that of steam generator tube rupture (SGTR) in PWR plants.

While many BWRs have drywell fan coolers that provide cooling during normal operation via a closed loop ventilation system, these are generally not credited in the IPEs for containment heat removal since they are not safety-grade systems. The exception is the Mark III containment of River Bend, which does not have a containment spray system but has two safety-grade containment unit coolers.

Containment venting is another means of containment heat removal invoked in some BWR IPEs. Containment venting is discussed in Subsection 10.3.1.6, below.

It should be noted that the drywell spray can also play a significant role in severe accident conditions because of its ability to (1) remove radionuclides from the containment atmosphere and (2) provide water to the corium on the drywell floor to mitigate core-concrete interaction (CCI) and reduce the probability of drywell shell melt-through.

10.3.1.5 Combustible Gas Control

Because of their relatively small volumes (especially Mark I and Mark II containments) and the significant amount of zirconium in BWR cores, BWR pressure suppression containments are susceptible to high-hydrogen concentrations in a post-accident environment. The potential for combustion events can occur in all phases of accident progression. Before vessel breach, hydrogen is released to the containment through the safety relief valve tailpipes into the suppression pool. Hydrogen can also be released directly to the containment during a loss-of coolant-accident. At the time of vessel breach, hydrogen is produced by the rapid oxidation of metal that accompanies energetic events such as ex-vessel steam explosion (EVSE) and direct containment heating. The ejection of hot core debris from the vessel also provides numerous ignition sources. Late in the accident, core concrete interaction produces both hydrogen and carbon monoxide, both of which are combustible. Hydrogen recombiners, intended to deal with design-basis hydrogen concentrations, are ineffective for handling severe accident hydrogen levels.

As a consequence, Mark I and Mark II containments operate with a nitrogen inerted drywell atmosphere. The drywell inerting system is capable of producing high-nitrogen flow rates before reactor startup and maintaining low-flow rates to keep oxygen concentration below 4% during normal operation. Because they operate with an inerted atmosphere, hydrogen combustion events are considered very unlikely in the IPE submittals of plants with Mark I and Mark II containments.

Because of their relatively larger volume, Mark III containments are not inerted but rely on glow plug igniters to burn off accumulating hydrogen during a severe accident and, thus, avoid energetic hydrogen events. However, as reflected in the IPEs, hydrogen combustion presents an important challenge to containment integrity for Mark IIIs because of the significant amount of zirconium in the reactor cores, the low-containment failure pressures, and the use of the pressure-suppression systems (which condense steam released from the vessel and thereby allow flammable mixtures of hydrogen and air to form). Although hydrogen ignition systems (HIS) are installed in the Mark III containments to burn the hydrogen at low-concentrations, they are not available during station blackout and may not be effective when there is a rapid increase of hydrogen concentration in the containment such as when dispersed core material is ejected at high-pressure.

10.3.1.6 Containment Venting

Containment venting has been recognized as an important accident management tool for BWR Mark I containments. It is used to prevent uncontrolled containment failure by providing a controlled release of the containment atmosphere if the containment pressure equals or exceeds a specified limit (the primary containment pressure limit, PCPL). After core damage, venting via a path through the suppression pool will provide considerable radionuclide scrubbing and a reduced release. Important issues for containment venting are the containment venting pressure, the areas and flow capacities of the vent paths, and the structural capability of the vent paths. Mark I containments are requested to have hardened vent paths that can be expected to survive under severe accident conditions.

Besides containment pressure control, venting is also used for combustible gas control. Combustible gas venting is directed by the EOPs when the containment reaches a combustible condition (i.e., the containment is deinerted and sufficient amount of hydrogen exists in the containment). Since Mark I containments are usually inerted by nitrogen, the likelihood of needing combustible gas venting is generally not significant.

For Mark Is, containment venting may also be used in the containment flooding process. Containment flooding is called for in the EOPs when the reactor pressure vessel (RPV) level cannot be restored and steam cooling is insufficient to cool the core. Containment flooding is accomplished by pumping water from external water sources into the containment to raise the containment water level above the top of active fuel (TAF). Containment flooding requires the use of a drywell vent or venting through the RPV because torus vents are not available during and after containment flooding. Since drywell venting does not have the benefit of suppression pool scrubbing and venting through the RPV, this results in a release bypassing the containment, and the venting used during containment flooding may result in very severe releases.

For Mark IIs, a hardened vent system is not required, but venting could still be useful for mitigating a severe accident.

A hardened vent system is also not required for Mark IIIs. However, venting (sometimes through the SGTS or directly into the turbine building) is discussed and plays a significant role in some Mark III IPEs.

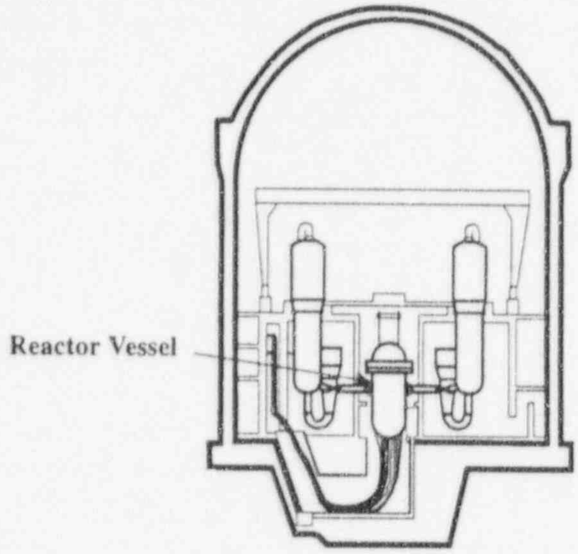
10.3.1.7 Containment Bypass and Isolation

Following an accident, the containment isolation system activates valves to close off certain lines that penetrate the containment primary boundary. Loss of containment isolation during a severe accident may have consequences as severe as a large structural failure. Past history shows that isolation failures have occurred under normal operating conditions and, therefore, must be taken into account when discussing severe accidents. Containment isolation failure can result from inadvertent pre-existing openings in the containment boundary or from the failure of valves used to isolate the major process lines and other boundary penetrations. These valves together with the associated sensors and power supplies comprise the containment isolation system. In BWRs with Mark I and Mark II containments, the atmosphere inerting systems may alert the operator to an open access hatch or other inadvertent opening in the containment boundary by showing higher than normal nitrogen flows. However, nitrogen monitoring may not be performed on a continuous basis. Most BWRs have eight groups of containment isolation valves. Isolation failure of the BWR containments was found to be small or insignificant in the IPEs.

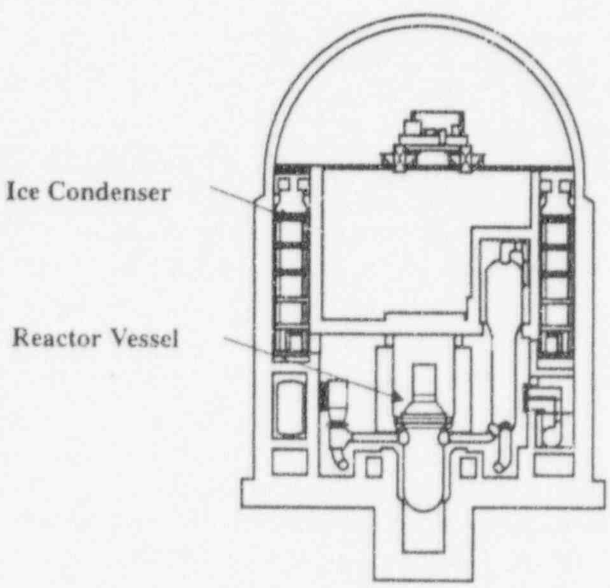
Failure of the barriers between the high-pressure reactor coolant system and connected low-pressure systems, with some components outside of primary containment, represents another way the containment function can be bypassed in both PWRs and BWRs. Although such ISLOCA sequences have been found to be relatively low-frequency events, they may lead to potentially high-radiological releases because these events provide a direct path for release of radionuclides to the atmosphere. In BWRs the breach of the low-pressure system outside of the primary containment will occur in the reactor building. Bypass events were also found to be relatively unimportant in most of the BWR IPEs.

10.3.2 PWR Containment Characteristics

PWRs have primary systems that normally operate at very high-pressures. These reactors use three containment types, including large-dry, subatmospheric, and ice condenser containments. Figure 10.4 compares the general arrangements of the large dry and ice condenser containments. A summary of the important containment characteristics for the three containment types are presented in Table 10.5.



Large, Dry Containment



Ice Condenser Containment

Figure 10.4 PWR large dry and ice condenser containments.

Table 10.5 PWR containment characteristics.

Number of Units	Containment type		
	Large-dry	Subatmospheric	Ice condenser
	57	7	9
Containment size and reactor power reactor thermal power (MWt) Containment free volume (ft^3) Cont. volume to thermal power ratio (ft^3/MWt)	 1500 - 3800 1,000,000 - 3,300,000 630 - 1200	 2441 - 3411 1,700,000 - 2,300,000 600 - 740	 3411 1,200,000 - 1,300,000 350 - 380
Containment strength Containment design pressure (psig) Median containment failure pressure (psig) Used in IPE	 41 - 61 90 - 190	 45 - 60 120 - 130	 11 - 30 36 - 95
Containment construction	7 steel 50 concrete	7 concrete	7 steel 2 concrete
Vapor pressure suppression system	No	No	Ice condenser and recirculation fans
Containment heat removal system	Containment spray* fan coolers	Containment spray* and fan coolers	Containment spray*
Combustion gas control	Hydrogen recombiner (for design-basis)	Hydrogen recombiner (for design-basis)	Hydrogen igniters
* Recirculation spray, taking suction from the containment sump.			

As shown in the table, most PWRs have large-dry containments. The subatmospheric containment also has a large-dry containment but with sub-atmospheric pressure inside. These large-dry containments rely on structural strength and large internal volume to maintain containment integrity during an accident. In order to structurally fail these containments early in an accident sequence, they must be subjected to very severe and rapid pressure loads. Such loads can be produced in the absence of containment heat removal systems and if direct containment heating occurs. If the primary system is at low-pressure and the containment heat removal systems are operating, the likelihood of ECF is much lower.

Nine PWR units have ice-condenser containments. These are smaller in volume than typical large-dry containments (see Table 10.5 for containment free volume and containment volume to thermal power ratio) and are equipped with ice beds to condense steam during an accident.

Structural failure caused by long-term pressure and temperature buildup or penetration of the containment basemat by core debris are both possibilities for late containment failure in PWR containments. The likelihood of these failure modes depends on the individual containment type and the absence or presence of decay heat removal systems. In some large-dry containments, even with decay heat removal systems inoperable, structural failure may never occur.

10.3.2.1 Containment Structure

The three types of construction techniques that have been used for currently existing PWR dry containments include (1) reinforced concrete, (2) prestressed concrete, and (3) steel.

A reinforced concrete containment has three basic structural elements, including the basemat, cylinder, and dome. Reinforcing bars are placed in all three elements. The containment accommodates the design-basis loads via the reinforced concrete and through the net free volume of the containment. Many reinforced concrete containments have a steel liner attached to, and supported by, the concrete. The liner primarily functions as a gas-tight membrane and also transmits loads to the concrete. All subatmospheric containments are constructed of reinforced concrete with a steel liner. Subatmospheric containments are found at seven Westinghouse PWRs (six three-loop plants and one four-loop plant).

In more recent plants, the reinforced concrete design has been replaced, to a large extent, by fully prestressed containments. In this design, the reactor containment is in the shape of a cylinder with a shallow domed roof and a flat foundation slab. The cylindrical portion is prestressed by a post-tensioning system consisting of horizontal and vertical tendons. The dome has a three-way post-tensioning system. The foundation slab is conventionally reinforced with high-strength reinforcing steel. The entire structure is lined with a 1/4-inch welded steel plate to provide vapor tightness. A prestressed concrete containment requires less net free volume for a given blowdown load. The external force applied by the tendons allows a higher internal pressure. (Zion is a representative plant for this category.)

Most steel containments use a steel plate interior structure enclosed by a separate biological shield concrete building. (The only exceptions is San Onofre 1 which lacks the concrete shield building.) The concrete shield structure is not designed for high-internal pressure but serves to protect the steel shell from extreme environmental effects. The internal pressure of the containment is carried by the structural strength of the steel plating. (A typical steel shell design is Davis Besse 1.) The steel containment may be susceptible to direct contact with core debris. This is an important failure mode for one PWR with a large-dry steel containment.

The ice condenser containment design incorporates a large, passive heat sink, in the form of the ice condenser system, to absorb accidental energy release from the primary coolant system of the reactor. The large heat sink permits a containment designed for lower pressure and smaller volume than would be allowable in a "dry" or non-pressure suppression containment (see Table 10.5).

The nuclear reactor system exclusively employed in the ice condenser plants is the Westinghouse four-loop NSSS with a thermal output of about 3400 Mwt. Seven of the nine ice-condenser units feature a cylindrical steel containment surrounded by a concrete secondary containment. The remaining two, the D.C. Cook plants, feature reinforced concrete containments with steel liners, and lack secondary containments. The internal volume of each ice-condenser containment is approximately 1,200,000 ft^3, and each has a diameter of approximately 115 ft. and a height of approximately 155 ft.

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. The design pressure of ice condenser containments vary from a low of 12 psig to 30 psig. The failure pressures used in the IPEs have median values varying from 36 psig to 95 psig (Table 10.5).

As the table indicates, the 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 the other types. These containments rely on the pressure suppression capability of the ice condenser feature to prevent overpressure.

10.3.2.2 Vapor Pressure Suppression

The only PWR containments using a vapor suppression system are the ice condenser containments.

The ice condenser system consists of a partial annulus extending 300° along the perimeter of the primary containment. Typical parameters are 95 ft. high and 13 ft. wide, holding approximately 2,300,000 lbs of borated ice in perforated metal baskets maintained at 15°F. Channels around and through the ice baskets allow free flow of steam and gases. Insulated, spring-loaded doors are located at the lower end of the ice condenser at the lower plenum. The door panels are provided with tension spring mechanisms that produce a small closing torque equivalent to providing approximately one pound per-square-foot pressure drop at the inlet port. Intermediate-deck doors are provided at the upper end of the ice condenser. These are normally closed under the action of gravity but open under slight pressurization to permit upward flow. Both the lower-plenum and intermediate-deck doors are designed to close to block flow from the upper to the lower compartment, although both may fail open after a sudden pressurization event. The top-deck doors are comparatively lightweight, and are expected to open early and remain open. Steam and gases released into the lower compartment from a break in the primary coolant system enter the ice condenser through the inlet doors, are directed up by turning vanes, and pass through the ice baskets where steam and aerosols are removed by condensation and deposition. As long as ice is available in the ice chamber, the ice condenser has been found to be effective in reducing the pressure spikes associated with the release of steam into the containment as well as in removing radionuclides from the containment atmosphere.

10.3.2.3 Other Containment Features

An important feature of a PWR dry containment, which can significantly influence the progression of a severe accident, is the configuration of the reactor cavity located below the reactor vessel. There is a large variation in reactor cavity design among PWR dry containment plants. This is largely because the cavity plays no role in design-basis accidents. However, under severe accident conditions, the reactor cavity could strongly affect the challenges imposed on the containment. The cavity's size, geometry and outlets into the containment regions could affect the interactions of corium/water and corium/concrete, and these, in turn, could affect the subsequent containment pressurization and basemat erosion. For example, the presence or absence of sumps or curbs around access ports to the cavity region would determine whether the cavity would be flooded during a particular accident sequence, thereby influencing whether the core debris could be cooled. The outlet flow paths from the cavity can significantly impact the amount of material that might be ejected into other containment regions by a high-pressure release from the reactor vessel.

There are containment failure modes that are unique to particular PWR plants because of some plant-unique features. For example, for Palisades, there is the potential for the core debris to relocate to the engineered safeguards rooms. The unique plant feature that contributes to this early failure mode is the location of the engineered safety features (ESFs) sump. The failure mode postulates the flow of molten core debris from the reactor cavity into the ESF sump and subsequently into the ESF recirculation piping. In the Palisades IPE submittal, the debris is assumed to eventually melt through the pipe wall and enter the auxiliary building. The maximum failure area for this mode is presumed to be twice the area of an ESF recirculation pipe (there are two pipes), resulting in a large containment failure area.

10.3.2.4 Containment Heat Removal

The high-pressures and temperatures during an accident in a PWR dry containment may be reduced by two containment heat removal systems, including the containment water sprays and the atmospheric fan coolers. In some designs, both systems are ESFs and are designed to operate during a LOCA assuming a single component failure. In other designs, only the sprays are an ESF system. The containment heat removal is accomplished by heat exchangers in the containment spray system and containment fan coolers. Typically, the sprays and fan cooler systems are sized to accommodate energies associated with the reactor decay heat and the sensible and latent heat of the primary system coolant.

The CSS of a PWR large-dry plant like Zion has three independent 100% capacity subsystems with no common headers. A single active or passive failure in any of these subsystems will not affect the operation of either of the other two subsystems. Of the three containment spray pumps, two are motor-driven and the third is diesel-driven. All three pumps take suction from the RWST. When spray is required during the recirculation phase of the accident, two of the three spray subsystems can be supplied with water from the containment sump via the RHR pumps. Therefore, spray pump operation is not necessary during the recirculation phase.

The reactor containment fan cooler (RCFC) system is designed to filter, cool and dehumidify the reactor containment environment during both normal and abnormal conditions. It is a recirculation system.

Typically in the subatmospheric plants, the sprays, not the fan coolers, are an ESF system. A subatmospheric containment, like Surry, typically has two spray systems, including an injection spray system (that draws from the RWST) and a recirculation spray system. When the RWST has been emptied in this plant, the injection spray system is secured and the recirculation spray system is started.

The ice condenser system described above provides for passive, although limited, containment heat removal in ice condenser containments. The other containment heat removal system in these containments is the CSS, which functions similar to other PWR spray systems by removing heat in the recirculation mode. The CSS at Sequoyah consists of two 100% capacity-pump trains. Each train includes a centrifugal pump, a heat exchanger, a minimum flow circulation line, and associated piping and valves. In the injection mode, each train draws water from the RWST at the rate of 4750 gpm. Upon depletion of water in the RWST, the CSS is shifted to the recirculation mode through a combination of operator and automatic actions. In the recirculation mode the CSS draws water from the containment sump and cools the water by passing it through the CSS heat exchangers. The residual heat removal system (RHRS) pump trains are similar to CSS pump trains with the exception that each RHRS train can either provide safety injection to the reactor vessel or flow to the RHRS sprays.

Direct decay heat removal from the core in PWRs, as done during shut down cooling, may also be possible during an accident. This is discussed in Section 10.2.

10.3.2.5 Combustible Gas Control

All PWR dry containments are equipped with combustible gas control systems (CGCSs) to maintain the post-design basis accident hydrogen buildup at a level below the flammability limit. The system contains four elements:

(1) A hydrogen sampling system alerts the plant operator to the hydrogen concentration in the containment.

(2) A hydrogen/air mixing system minimizes the formation of locally high-hydrogen concentrations.

(3) Hydrogen recombiners heats gases drawn from the containment to high-temperatures (to combine hydrogen with oxygen) and returns the gases back to the containment.

(4) A containment purge system allows venting of the containment atmosphere to the outside environment.

However, these systems are designed to accommodate hydrogen accumulation for design-basis events. The systems are not designed for the hydrogen generation that might accompany a core meltdown accident. PWR dry containments are not required to have the intentional ignition systems required for the ice condenser plants, as discussed below. However, as part of the U.S. Nuclear Regulatory Commission's (NRC's) Containment Performance Improvement (CPI) program, licensees with large-dry containments were requested to perform as part of their IPE an

> "evaluation of containment and equipment vulnerabilities to hydrogen combustion (local and global). This would include consideration of gaseous pathways between the cavity and upper compartment volume to confirm adequate communication to promote natural circulation and recombination of combustible gases in the reactor cavity."

The NRC rulemaking resulting from the Three Mile Island (TMI) 2 accident required PWR ice condenser and BWR Mark III containments to be installed with hydrogen control systems capable of accommodating amounts of hydrogen equivalent to that generated in the oxidation of 75% of the clad without loss of containment integrity as a result of this rulemaking. A distributed igniter system (DIS) was installed in ice-condenser containments to burn hydrogen before it can accumulate to hazardous levels.

The DIS seeks to mitigate combustion pressurization challenges to containment integrity by deliberate ignition of hydrogen at low-concentrations. The DIS at Sequoyah consists of 68 Tayco thermal igniters distributed throughout the containment and deployed in two separate groups, each with its own independent and separate power supplies and controls. The igniters operate at 120 VAC. A separate train of 480 VAC power is provided for each group of igniters and is backed by automatic loading onto the diesel generators upon loss of off-site power (LOOP). Of the 68 igniters, 22 are deployed in the lower compartment, 16 in the upper plenum of the ice condenser, 4 in the upper compartment dome, 8 at intermediate elevations in the upper compartment, 2 above the two air return fans, and the remaining 16 in the dead-ended regions of the containment. The igniters are manually actuated by the operator from the control room.

To aid the function of the DIS system, air return fan systems (ARFSs) are installed at all ice condenser containments. The ARFS maintains the circulation of the containment atmosphere through the ice condenser, and ensures that the local hydrogen concentration in the containment, especially in the dead-ended regions, does not reach excessively high-levels. The ARFS, therefore, supplements the functions of the DIS. The ARFS at Sequoyah consists of two axial-flow fans that return air from the upper compartment to the lower compartment and reduce the post-accident stratification of hydrogen in stagnant areas. The fans push air and gases to the lower compartment and maintain forced circulation of the containment atmosphere through the ice condenser. Both fans are actuated by a high-containment pressure (>2.81 psig) signal after a delay of 10 minutes. The ARFS is an engineered safety system and operates on AC power.

Finally, the ice condenser CSS can influence hydrogen combustion and can have beneficial effects, such as promoting circulation and mixing of the containment atmosphere, particularly when operated in conjunction with the DIS.

Because of the significantly smaller amount of zirconium in the core of a PWR than a BWR, it appears from the IPEs that the hydrogen combustion problem is less severe for ice condenser containments than for BWR Mark III containments.

10.3.2.6 Containment Bypass and Isolation

In PWRs, it is difficult to isolation problems, other than those identified by the isolation valve status indicators in the control room. Loss of vacuum in PWRs with subatmospheric containments would be one indication of inadequate isolation. Some PWRs use an enclosure building that is maintained at less than atmospheric pressure. A pressure increase in this enclosure building would again be evidence of an isolation problem. However, the majority of PWRs have neither a subatmospheric containment nor an enclosure building. Nonetheless, the probability of isolation failure found in the IPEs is usually small. A large probability of isolation failure is most likely attributable to the lack of operator actions to locally or remotely close the isolation valves if no containment isolation signal is provided or in a station blackout. This usually involves a small leak area and, therefore, does not significantly contribute to radionuclide release.

ISLOCA scenarios are also important for PWRs. Failure of the barriers between the high-pressure reactor coolant system and connected low-pressure systems, with some components outside containment, is a way the containment function can be bypassed. An important failure site of the PWR primary system during a severe accident is in the steam generator tubes. Although SGTR has many of the characteristics of a small LOCA, it is unique in that it is also a potential containment bypass LOCA, releasing radionuclides in the primary reactor coolant into the secondary-side of steam generators. This could provide several potential paths for the release of radionuclides to the environment outside the containment (e.g., via the main steamline, turbine, turbine bypass, condenser, condenser exhaust, steam generator atmospheric relief or safety valves, and the steam generator blowdown line).

An SGTR could be the initiator of a severe accident or could be induced under severe accident conditions arising from another initiator. The most important cause for induced SGTR is high-RCS temperature.

Temperature-induced SGTR occurs if one or more steam generator tubes have a creep rupture caused by the flow of high-temperature hot gases from the core when the RCS is at system pressure. Since a hot leg or surge line break is more likely than an induced SGTR when such a high-temperature condition exists, induced SGTR usually has a low-probability of occurrence. However, temperature-induced SGTR may be significant under certain conditions. According to some IPE submittals, the procedural guidance requires that the operators restart the RCPs when inadequate core cooling conditions are indicated. This restart clears the RCP seals and establishes a natural circulation path, which results in increased steam generator tube heating and the potential for an induced SGTR. Secondary side depressurization, also included in the procedures for restoration of heat removal, can increase the pressure differential across the tubes and, thus, may further increase the potential for failure.

Containment bypass, especially resulting from SGTR is significant in a number of PWR IPEs and, in some cases, dominates the probability of early release.

10.4 IPE Boundary Conditions, Assessments, and Assumptions

This section addresses the potentially more important boundary conditions, assessments, and assumptions that have been identified during the review of the IPEs and discusses how variability in these modeling issues impacts the IPE analyses and results. This summary is provided under two major headings, those that impact the CDF portion of the analysis, and those that impact the severe accident progression and radionuclide release portion of the analysis. The specific impacts of many of these issues on individual IPE results are more apparent in subsequent chapters in this report. Since the IPE submittals are *summaries* of the work performed for the IPEs, review of the submittals alone cannot be expected to identify the full variability of assumptions, analysis judgments, or other factors having potential effects on the IPE results. Nevertheless, variations within the IPEs have been noted during the review of the submittals.

10.4.1 Core Damage Frequency Boundary Conditions, Assessments, and Assumptions

This section identifies those analytical boundary conditions, assessments, and assumptions that can have a potentially significant impact on the CDF results and were noted to have some variability among the IPEs. Since all of the licensees chose to use PRA approaches in responding to GL 88-20 (Ref. 10.1) for the CDF portion of the analysis, these issues are discussed in the context of the PRA core damage analysis tasks that are most affected. This includes identification of initiating events (IEs), accident sequence analysis, systems analysis, data analysis, human reliability analysis (HRA), and the quantification task.

The first task of the overall modeling process involves identifying initiating events (IEs) (i.e., challenges to normal plant operation) that require successful mitigation to prevent core damage. As a part of this task, these events are grouped into initiating event classes whereby all of the individual initiators within a class have similar characteristics and require the same overall plant response.

For each initiating event class, event trees are developed in the accident sequence analysis task. These event trees graphically depict the possible sequences of events that could occur during the plant's response to each initiating event class. These trees delineate the possible combinations of functional and/or system successes and failures that lead to either successful mitigation of the initiator or core damage. Determining the success criteria to avoid core damage is a very important part of the accident sequence analysis task.

As defined by the event tree structures, the systems analysis task involves modeling the failure modes of systems that are functionally important to preventing core damage. This modeling process, which is usually done with fault trees, defines the combinations of equipment failures, equipment outages (such as for test or maintenance), and human errors that cause failure of the systems to perform the desired functions.

The data analysis task involves determining the equipment reliability data and initiator frequencies used to derive the quantitative results of the IPE. As part of this task, plant maintenance and other operating records are evaluated to derive plant-specific equipment failure rates and the frequencies of the initiating events (IEs). Where insufficient plant experience exists, failure rates and initiator frequencies based on industry-wide "generic" databases are used as input to the database used in the IPE.

The HRA task contributes to the modeling portions of both the accident sequence analysis and systems analysis tasks for inclusion of human errors that are potentially important to the sequences of events and system failures included in the overall IPE model. Additionally, the HRA task involves quantifying these human errors included in the analysis. HRA is a special area of analysis requiring unique skills to determine the types and likelihoods of human errors germane to the sequences of events that could result in core damage.

Finally, quantification involves combining all of the information from the previous tasks to calculate the CDF for individual accident sequences, as well as the total plant CDF.

A summary of key boundary conditions, analysis assessments, and assumptions obtained from the IPE submittals is provided in Table 10.6. The following paragraphs provide further discussion on these analysis features and how they can affect the task areas described above. As noted earlier, where appropriate, the specific impacts of many of these features on the IPE results are further discussed in subsequent sections of this report.

Table 10.6 Potentially important boundary conditions, assessments, and assumptions affecting individual IPE results.

Task area	Issue(s)
IEs	Exclusion of some initiators Grouping of initiators
Accident sequence analysis	Definition of core damage Success criteria to prevent core damage
Systems analysis	Completeness of failure modes modeling Operability of equipment in abnormal environments
Data analysis	Treatment/quantification of common cause Some individual data values
HRA	Completeness of human events modeling including amount of recovery considered Details/judgments affecting the quantification of human error rates
Quantification	Truncation (relatively minor effect)

10.4.1.1 Initiating Events

Typically, in this task area, the most important characteristics of the analysis that are subject to potentially wide variability and that can also have an impact on the CDF include (1) assumptions or assessments made to exclude initiators from the analysis and (2) the grouping of individual initiators into initiating event classes. Excluding certain initiators obviously affects the boundaries of the analysis by defining the completeness of the analysis scope. The way the initiators are grouped impacts the resolution of the analysis.

In particular, the IPEs vary in the assumptions and related assessments used to determine which support system initiators should be included in the IPE. Support system initiators involve losses of systems or equipment that support the operability of the accident prevention systems (e.g., systems that flood the core) in the plant. These support system initiators include the loss of electrical buses, service water, HVAC for room cooling, and other systems that disrupt the normal operation of the plant and require a mitigating response. Some submittals often justifiably exclude certain support system initiators (e.g., loss of control room HVAC or loss of IA) on the basis of plant-specific design features and analyses of those features. These analyses typically find that (1) there is no significant dependency on the lost system, (2) the resulting effects on the plant are similar to some other initiator that is being modeled, (3) the time required for the effect of the initiator to develop an adverse condition is sufficiently long to "guarantee" recovery (thereby eliminating the initiator), or (4) the expected frequency of the initiator is so low that it does not warrant further analysis. Conversely, some licensees include events as initiators (e.g., the loss of certain electrical buses) even though they might not actually be initiators, on the assumption that the event requires mitigation rather than spending analysis time to determine if the event could be legitimately eliminated as an initiator. For instance, some licensees consider the loss of control room HVAC as an initiator or analyze interfacing LOCAs while others do not. Therefore, the reader should recognize and be sensitive to the fact that any differences in the IPE results as to the most important initiators contributing to core damage may be partially determined by this portion of the analysis.

The grouping and "definition" of each initiating event class also vary among the IPEs. For instance, some licensees combine steam line and feedwater line breaks into a single initiator even though one is an overcooling transient while the other is an undercooling transient. Some licensees use very broadly defined LOCA sizes while others use a more refined set of LOCA sizes that allows examination of potentially subtle differences in the required plant response. Some licensees grouped some initiators in classes with more stringent assumptions such as categorizing the loss of one feedwater pump in with those initiators associated with a total loss of feedwater. Such groupings are typically defined on the basis of the plant design, past PRAs of similar plants, and analysis judgment. Variability is also evident in how licensees categorize their initiator-accident sequence combinations. For example, transient-induced LOCAs are sometimes broadly classed as transients while other times they are classed as LOCAs. This latter point is more of a reporting difference. If handled consistently within an IPE, it does not provide cause to question the results but does make it more difficult to compare IPEs. Again, the reader should be sensitive to the fact that the differences in the IPE results, and particularly how they are reported, can be impacted by this grouping process.

10.4.1.2 Accident Sequence Analysis

Two particularly important characteristics where variability exists among the IPEs and which can potentially impact the results for this task area include (1) the definition of core damage and (2) the success criteria to prevent core damage. These are interrelated in that both set the limits on what constitutes successful avoidance of core damage, or conversely, what must fail to result in core damage.

The criteria used to define core damage vary among the IPEs and in many cases, no specific definition is provided in the submittals. Typical definitions for successful avoidance of core damage in PWRs include (1) long-term core exit temperatures less than 1200°F, (2) peak cladding temperature less than 2250°F, or (3) no "sustained" core uncovery. Typical definitions for BWRs include (1) fuel temperatures less than 4040°F, (2) the core is more than 2/3 covered, or (3) the collapsed water level must be more than 2 feet above the bottom of the active fuel. The definition of core damage can be important since it is used to determine whether a particular system/action successfully mitigates an accident (i.e., it impacts the success criteria). In a practical sense, the definition is not critical to defining whether or not most systems can successfully prevent core damage. However, there are certain systems/actions (such as those involving smaller flow rates), where, depending on the core damage definition, credit for preventing core damage may or may not be given in the analysis. This is particularly true in cases where it is expected that portions of the fuel will be uncovered for a period of time before significant reflood occurs. For example, the time available for operator actions can be affected when feed-and-bleed must be started following loss of all secondary cooling in a PWR, thereby, affecting the amount of flow required and impacting the human error probability (HEP) to fail to start feed-and-bleed. This in turn affects the probability of success or failure of these actions and hence the CDF.

Various core damage criteria lead to the broader issue of success criteria definition in general. Some licensees tend to be very pessimistic with regard to the equipment needed to prevent core damage, such as using the bounding final safety analysis report (FSAR) criteria. Other licensees deviate significantly from these design-basis definitions and use best-estimate, thermal-hydraulic analyses along with judgment for defining more realistic criteria. There are numerous instances of variability with regard to success criteria definitions found in the IPEs, as illustrated by the following examples:

- whether some PWRs can depressurize in response to small LOCAs and transients when all high-pressure cooling is lost
- whether a switch to hot-leg injection is required in PWRs late in an accident to avoid boron precipitation
- the number of relief valves required for primary depressurization in BWRs
- the effectiveness of and time required to use CRDHS-enhanced flow as the only means of core cooling in BWRs.

While much of the variability in the definition of success criteria among the IPE submittals is indeed justified on the basis of actual plant design differences, the extent of the justification is not always clear and sometimes is partially attributable to modeling assumptions. This variability in success criteria can likely have one of the more significant impacts on the results when compared with any of the other issues discussed in this section. Hence, the reader should be aware that variability in the success criteria of seemingly similar plants can have a significant impact on the CDF and the overall results of the IPE.

10.4.1.3 Systems Analysis

There are two general categories of boundary conditions, assessments, and assumptions in this task area which can be important to the IPE results. These are (1) the failure modes and dependent failures included in system modeling, and (2) assessments/assumptions made regarding the operability of equipment. The first addresses completeness issues regarding the definition of all "credible" failure modes of each system, while the second defines whether equipment is credited as functioning in environments that are beyond the design basis for the equipment.

The failure modes issue could generally not be reviewed in detail to determine the extent of variability, if any, in the IPEs. This is because the licensees were not requested to include the failure modeling (typically fault trees) of each system in the submittals. Possible variations that might be expected among the IPEs on the basis of differences seen in prior PRAs include (1) whether or not spurious faults were modeled, (2) to what level of detail common cause failure modes were modeled (i.e., which components were included for common cause failure modeling and whether common cause was only considered for components within the same system or also across systems), and (3) whether a systematic search for subtle system dependencies was carried out (e.g., RCIC failure associated with steam leak detection actuation upon loss of HVAC, indicating a subtle dependency on room cooling). Since it is apparent that the licensees generally modeled system failures in the same way (i.e., with failures to start and run equipment, equipment unavailability as a result of test and maintenance, common cause failures, and operator-induced failures) any variability in the modeling of failure modes may not be as critical to the results as the issue involving equipment operability.

Equipment operability during beyond design-basis conditions is treated differently among the IPEs, possibly because of plant-specific design features as well as assumption differences. For example, different licensees either credit or do not credit operability of switchgear with total loss of HVAC. This can be a result of plant-specific room sizes and, hence, the heatup rate of the equipment as well as other factors or assumptions. With station blackout being a prevalent contributor in many IPEs, the operability of DC-powered equipment with loss of HVAC, and the assessment of battery life without AC charging, are prime examples of important issues in the modeling of DC equipment operability during loss of all AC. Some licensees credit battery life under these conditions for only 1 to 2 hours, while others take credit for as long as or greater than 15 hours, on the basis of battery designs and other operational aspects. Such a conclusion can have a significant impact on a plant's coping capability under a prolonged station blackout and, hence, on the importance of blackout accident sequences to the total CDF. Similarly, some PWR licensees assume that a turbine-driven AFW pump can be operated following battery power depletion in blackout, while others do not credit AFW operation under this same condition. As another illustration, some BWR licensees assume rising containment temperatures and pressures (and eventual failure of containment) will adversely affect operability of pumps using the suppression pool for suction because of decreasing NPSH, while others conclude their pumps can still function successfully under the same condition as a result of specific design characteristics of their pumps. These operability issues, probably more than the fault tree modeling issue, appear to be assessed differently among many IPEs and can strongly affect which sequences are dominant as well as the overall CDF. As with other issues previously discussed, the reader should be sensitive to the fact that variability in system modeling, and particularly with regard to equipment operability, can cause distinct differences in the IPE results.

10.4.1.4 Data Analysis

As would be expected, the numerical results and, to some degree, the ranking of important sequences and contributory equipment failures, is dependent on the failure data used in each IPE. Variability in the common cause failure data appears to be large among the IPEs and, thus, affects the degree to which common cause failure plays a role in the IPE results. For example, some licensees reviewed "generic" common cause events and data and justified exclusion of some events as "not credible" at their plants. This lowers the corresponding common cause failure probabilities and affects how important common cause failures are to the results of the IPE. In addition, this approach might (although not as likely) affect the relative ranking of dominant accident sequences.

Limited review of individual data values in the IPEs also identified probabilities for the same equipment failure modes ranging by well over an order of magnitude in some cases. This degree of variability, sometimes attributable to differences in component failure experience among the plants, can affect the relative importance of specific equipment failures and their contribution to the potential for core damage.

It is important to note that differences in the databases used in the IPEs can particularly affect the relative importance of equipment failure contributions to core damage, and therefore are largely responsible for the observed differences in the IPE results.

10.4.1.5 Human Reliability Analysis

The probability and treatment of human error in accident scenarios can be a critical element in determining the overall and sequence-specific CDFs, ranking dominant sequences, and identifying human actions that are most risk significant. For example, some licensees dismiss routine maintenance and calibration errors (such as failure to properly realign a system after testing or maintenance) on the basis of insignificant failure probabilities or that the failure associated with such events is contained in the system unavailability data. Other licensees explicitly analyze all such events (although even then, some licensees apparently dismiss miscalibration events, while others include them). Particularly for accident response events (such as failure to depressurize following loss of all high-pressure injection or failure to initiate feed-and-bleed), a significant difference in human error rates (often as much as two orders of magnitude) is evident among IPEs modeling similar events. The justification for these differences is sometimes, but not always, evident in the submittals. These different human error rates could, for instance, be attributed to actual differences in plant procedures and training, assumptions regarding the treatment of factors affecting human performance, or the degree to which dependency was or was not considered among multiple but related human errors (e.g., the reactor operator fails to initiate standby liquid control *and* the shift supervisor also fails to correct this error).

In the PRA process, accident scenarios are typically quantified by also applying one or more "non-recovery" events to the sequence of possible failure events that cause the scenario of interest. These non-recovery events are used to model the failure of human actions that would enable the accident scenario to be recovered and prevent core damage. For example, in a station blackout scenario, failure to recover offsite power accounts for an additional action that, if it successfully occurred, would prevent the scenario from proceeding to core damage. The failure to perform such an action is an additional probability applied to determine the overall scenario probability. Hence, the degree that human recovery is applied to each accident scenario in the sequence modeling process can significantly affect the importance and numerical results of each accident sequence. Some licensees appear to limit the extent to which recovery was applied, either on the basis of a minimum combined HEP to carry out multiple recovery actions, or based on the number of multiple actions credited (e.g., two actions-recovery of MFW *and* recovery of auxiliary feedwater). Some submittals appear to be more liberal in the number of actions credited and/or the degree to which beyond-EOP actions are credited. For example, some licensees credit refill of water storage tanks during accident scenarios while others do not.

On the basis of the identified differences and important results of past PRAs, the variability in the number of accident response and recovery actions credited and the corresponding failure probabilities are likely to be much more important to the results than differences in the routine action human error modeling. Hence, the reader's understanding of the degree of similarity (or differences) in the IPE results discussed in subsequent chapters should take into account the extent to which accident-related human error rates or credited recovery actions are highlighted and discussed later in this report. Because of the valid perception that HRA is a very important part of the IPE process, and that differences in HRA are likely because the state-of-the-art in human error modeling is not as mature as other parts of the IPE model, Chapter 13 of this report is dedicated to this subject area and addresses specific findings gleaned from reviewing the IPE submittals.

10.4.1.6 Quantification

All of the above task products come together in the quantification task of the IPE to yield the numerical results of the analysis. Different computer codes and some variation in the steps followed to perform the CDF quantification exist. Nevertheless, use of different codes or variations in the detailed steps for performing the quantification are not sources for the variability in the IPE results. However, use of different truncation limits during the quantification process can have some effect on the results. Truncation (usually based on low-probability) is a standard means to simplify the quantification process and make it less time intensive. During the quantification process, numerical results for ways to damage the core that are probabilistically below a prescribed threshold value, are screened out and not considered in the final overall IPE results since they are probabilistically insignificant. For instance, if during the quantification process, a combination of equipment failures to yield core damage must occur that collectively has a probability of 1E-10, such a combination may be dropped from the overall results and hence this 1E-10 contribution is not reflected in the plant CDF. By itself, any one such combination is insignificant. However, if 1000 such combinations are "screened out," a combined effect of 1E-7 is missing from the reported CDF value.

The degree of variability among the IPEs on this issue could not always be determined, since quantification thresholds were not often reported. Nonetheless, differences in this area should not be as significant as many of the other issues discussed above. (This presumption is based on past PRA findings and studies regarding the use of different threshold values.) Hence, while different truncation limits may have some effect, this issue is not likely to be a key factor as to why the results vary (or are similar) among the IPEs.

Related to the subject of quantification, differences were identified in the number and complexity of sensitivity analyses reported in the submittals to provide insights on the sensitivity of the quantified results to changes in boundary conditions, assessments, and key assumptions. The reported results of such sensitivity studies are helpful in understanding how "sensitive" the results are to many of the issues that have been addressed in this section of the report. Familiarity with an individual IPE's sensitivity analyses can give to the reader an additional perspective or "feeling" for the robustness of each IPE's results and the implications regarding the findings discussed in this report.

10.4.1.7 Flooding Analysis

As a special topic area and included in Table 10.6 for completeness, GL 88-20 requested that licensees conduct an internal flooding analysis to identify the potential importance of floods internal to the plant (e.g., because of a breached service water line). Flooding analysis involves all of the same task areas discussed above and is subject to the same variability issues and the potential effects that have already been described. In general, the degree of variability is even greater for the flooding analyses since there tends to be considerably more uncertainty and hence a greater reliance on judgment and analysis assumptions. Some example issues unique to flooding analyses are (1) determination as to whether a pipe will guillotine rupture or only crack thereby affecting the estimated flow rate and water volume from the breach, (2) whether the force behind a flooded door will cause the door to yield or whether the door will remain intact and only allow leakage past it, and (3) whether the spray from a flood source will cause failure of electrical equipment or whether the equipment remains operable. Numerous decisions such as these are incorporated throughout the flooding analyses, on the basis of both plant-specific design differences and modeling assumptions.

With just a few exceptions, most of the IPEs conclude that internal flooding is a relatively minor contributor to the overall potential for core damage; thus, the importance of the above-described variabilities does not appear to be significant. However, and particularly for those plants that do find flooding to be a significant issue, the reader should be sensitive to the fact that variations such as those illustrated above, can significantly impact the flooding analysis results and, thus, affect the relative importance of internal floods to the plant CDF.

10.4.2 Severe Accident Progression Boundary Conditions, Assessments and Assumptions

This section identifies those boundary conditions, assessments, and assumptions that can potentially impact accident progression and, thus, the containment performance assessment, before, during, and after the start of core damage. These issues are then discussed in the context of the tasks performed as part of an accident progression analysis (refer to Table 10.7). Typically, these tasks include defining an appropriate interface between the core damage analysis and the subsequent accident progression analysis, performing the accident progression analysis, and estimating radionuclide releases for the various containment failure modes.

Table 10.7 Potentially important boundary conditions, assessments, and assumptions affecting the accident progression analysis in IPEs.

Task area	Issues
Interface between the CDF analysis and the accident progression analysis	• Characteristics used to define PDS groups • Operability of mitigating systems
Accident progression analysis	• Completeness of containment failure modes • Structural capacity of containment • Completeness of the containment event tree • Basis for quantification of the event tree
Radionuclide release	• Selection of representative source term groups • Basis for source terms

Generally, the coupling of the core damage frequency analysis to the accident progression analysis is done through the use of plant damage states. These plant damage states define the attributes of the accident sequences (e.g., LOCA with failure of ECCS injection) and the status of those plant features (e.g., containment sprays) that influence accident progression after core damage. The intent is that all core damage accident sequences within a particular plant damage state can be treated as a group for the purpose of assessing accident progression, containment response, and radionuclide release.

The accident progression analysis consists of several steps as indicated in Table 10.7. One step is to determine challenges to containment integrity for the accident sequences identified in the core damage frequency analysis. Some of these challenges (such as high-pressures and temperatures) may be the result of physical processes; others could be caused by failure of systems such as isolation valves. For each challenge, the ability of the containment and its systems to contain the challenge has to be determined. Most of the utilities use some form of containment event trees (CETs) to organize the accident progression analysis. CETs are, in general, developed for each plant damage state. These event trees describe the possible sequence of events that could occur before, during, and after core damage. The CETs should include all important challenges to containment integrity and determine the response of containment systems to these challenges. The objective is to identify all possible ways in which radioactivity might be released to the environment. Quantification of the CETs then determines the probability of various releases (containment bypass and failure modes) conditional on the various plant damage states.

For each release path identified in the CETs, the characteristics and quantities of radionuclides released to the environment have to be estimated. Typically, the large number of radionuclide releases are binned into a smaller number of representative radionuclide groups (sometimes called source terms). The final product is a listing of the representative radionuclide groups and their associated frequencies of occurrence.

A summary of potentially important boundary conditions, assessments, and assumptions found in the IPE submittals is provided in Table 10.7. The following sections further discuss how these issues can influence the task areas described above. In a number of cases, the impact of these analysis characteristics on the IPE results are further discussed in subsequent chapters of this report.

10.4.2.1 Interface between the CDF Analysis and the Accident Progression Analysis

It is necessary to ensure that the core damage accident sequences are appropriately treated in the subsequent accident progression analysis. NUREG-1335 (Ref. 10.6) provided some guidance on how to ensure that adequate coupling exists between the core damage frequency analysis and the accident progression analysis. The concept of using plant damage states (PDSs) to provide the needed coupling was suggested in NUREG-1335 and most utilities use this approach. However, the attributes used to define the PDS groups vary considerably among the IPE submittals.

Several IPEs use a relatively small number of attributes to define the PDS groups, thereby producing a small number of event trees. While this approach has the advantage of simplicity, potentially important failure modes can be missed if the grouping is too broad. Other submittals use a relatively large number of attributes to define a correspondingly larger number of PDS groups. In some IPEs, plant damage states are not explicitly defined and each accident sequence is individually processed through the accident progression analysis task. The object is to identify those attributes of the various accident sequences that influence accident progression and to ensure that they are correctly incorporated into the subsequent analyses. Some of the more important attributes identified in the submittals are discussed in the following paragraphs.

One of the most important attributes identified in the IPEs is whether the containment is isolated during an accident. Isolation failures are classed in the IPEs as pre-existing or postulated to occur at the time of the accident. Pre-existing isolation failures can be detected in containments that have a controlled atmosphere (PWR subatmospheric containments and BWR Mark I and Mark II containments) and, therefore, are not important in IPEs for these containment designs. However, pre-existing isolation failures contribute to the frequency of early failure in several IPEs for PWR large-dry containments. Failure to isolate the containment at the time of the accident contributes to the frequency of early failure for all containment designs. The impact of this failure mode is quite small for BWR Mark I and Mark II containments; however, it can be important for other containment designs. In fact, it is the dominant contributor to the early failure frequency at Beaver Valley (PWR with a subatmospheric containment). In this case, isolation failure is assumed to occur for SBO sequences caused by loss of the emergency switchgear ventilation. This is an example of where information from the CDF analysis can significantly influence the subsequent accident progression analysis. In addition, potential operator actions to manually isolate the containment for these SBO sequences are not modeled in the CET.

For those accident sequences in which containment isolation fails, the size of the opening has to be determined and the flow path to the environment identified to estimate the magnitude of radionuclides release. Other accident characteristics, found important in some IPEs, include the availability of containment heat removal and the spray systems. Operation of the containment heat removal system will keep the containment pressure low and, thus, reduce the driving force for leakage through the isolation failure (i.e., lower source term). Sprays significantly reduce the aerosols in the containment atmosphere and are also found in the IPEs to reduce the quantity of radionuclides released.

Another important consideration in the IPEs is the identification of those accident sequences that can bypass containment (i.e., ISLOCA and steam generator tube rupture). These accidents are, in general, important contributors to the frequency of early loss of containment integrity for PWR IPEs. ISLOCA and steam generator tube rupture are IEs that were identified and quantified in the core damage frequency analysis. Information that has to be transferred to the subsequent accident progression analysis for these accidents includes the flow path (dimension and configuration of piping, size of break, etc.) from the damaged core to the environment, timing of the accident progression, and whether or not the flow path is submerged. This information is used in the IPE submittals to determine the quantity and characteristics of radionuclides released to the environment. Other important information relates to potential operator actions that could be taken to mitigate these accidents. If assumptions are made regarding human performance in the core damage frequency analysis, these assumptions should be consistent with any actions modeled in the accident progression analysis.

For sequences in which the containment is isolated and not bypassed, the accident progression has to be analyzed to determine if the containment eventually fails An important consideration is the thermal-hydraulic conditions in the reactor coolant system. Therefore, transient events are separated in the IPEs from LOCAs, and then subdivided into various break sizes. The thermal-hydraulic conditions can significantly affect accident progression (as discussed below in Section 10.4.2.2) and mission times that provide the time frame for possible recovery actions. The IPEs find that if coolant injection can be restored before large-scale core meltdown and vessel failure (as occurred during the TMI accident), then the containment remains intact. However, not all IPEs model this recovery action.

Other important considerations in the IPEs relate to the availability of containment heat removal or vapor suppression (refer to Section 10.3). It is necessary to determine the status of these systems for each of the accidents identified in the CDF analysis. Some IPEs develop detailed dependency tables relating the various systems for core damage prevention and mitigation for each of the accident sequences. If a system is unavailable in the core damage frequency analysis, it is also modeled as unavailable in the accident progression analysis. In other IPEs, the coupling between the CDF analysis and accident progression analysis appears to be rather weak in relation to the information in the submittal.

In some IPEs, power status is an important attribute that should be consistently treated throughout the analysis tasks. If all power is lost (station blackout accident), then active containment systems (such as sprays, fan coolers, hydrogen control systems, etc.) will not be available. In some IPEs, power recovery is explicitly modeled and its impact on the subsequent accident progression assessed. However, power recovery is not consistently modeled in all IPEs. It is not clear in some submittals how loss of power and the potential for power recovery are treated.

The potential for flooding the region below the reactor vessel (cavity or pedestal region) is an important consideration in the IPEs. The availability of water in this region can significantly influence the accident progression analysis (refer to Section 10.4.2.2) and, therefore, has to be determined for accidents identified in the CDF analysis. In some submittals, the status of cavity flooding is explicitly treated in the plant damage states; in others, it is implied from the status of other systems. For example, if all of the water in the refueling water storage tank is injected into the containment of some PWRs, the cavity would be flooded.

All of the above attributes are found to be important for coupling the core damage accident frequency analysis to the accident progression analysis, as described in the following section.

10.4.2.2 Accident Progression Analysis

This analysis task consists of several steps as shown in Table 10.7. An important first step is to identify a list of potential containment failure modes or ways in which the containment might be bypassed. NUREG-1335 (Ref. 10.6) provided a list of failure modes (reproduced in Table 10.8), which was intended only as a starting point for the IPE analysis.

Table 10.8 Potential containment failure modes identified in NUREG-1335.

Potential PWR containment failure modes	Large-dry	Subatmospheric	Ice condenser
Containment bypass			
Interfacing-system LOCA	Yes	Yes	Yes
Failure to isolate containment	Yes	Yes	Yes
Steam generator tube rupture	Yes	Yes	Yes
ECFs			
Overpressurization with high-temperature because of			
noncondensible gases and steam	Yes	Yes	Yes
combustion processes	Yes	Yes	Yes
direct containment heating	Yes	Yes	Yes
Missiles or pressure loads because of steam explosions	Yes	Yes	Yes
Melt-through because of			
direct contact between core debris and containment	No	No	Yes
Vessel thrust force because of blowdown at high-pressure	Yes	Yes	Yes
Late containment failures			
Overpressurization with high-temperature because of			
noncondensible gases and steam	Yes	Yes	Yes
combustion processes	Yes	Yes	No
Melt-through because of basemat penetration by core debris	Yes	Yes	Yes
Vessel structural support failure because of core debris erosion	No	No	No
Potential BWR containment failure modes	**Mark I**	**Mark II**	**Mark III**
Containment bypass			
Interfacing-system LOCA	Yes	Yes	Yes
Failure to isolate containment	Yes	Yes	Yes
ECFs			
Overpressurization with high-temperature because of			
noncondensible gases and steam	Yes	Yes	Yes
combustion processes	No	No	Yes
direct containment heating	Yes	Yes	Yes
Missiles or pressure loads because of steam explosions	Yes	Yes	Yes
Melt-through because of			
direct contact between core debris and containment	Yes	No	No
Vessel thrust force because of blowdown at high-pressure	No	No	No
Late containment failures			
Overpressurization with high-temperature because of			
noncondensible gases and steam	Yes	Yes	Yes
combustion processes	No	No	No
Melt-through because of basemat penetration by core debris	Yes	Yes	Yes
Vessel structural support failure because of core debris erosion	Yes	Yes	Yes

Some IPEs identify failure modes that are not given in the tables. In other IPEs, the relative importance of the failure modes in Table 10.8 to the various containment types also change as a result of the plant examinations. An example of how the plant examinations changed the information in these tables relates to loss of containment integrity as a result of direct contact with core debris. This was not considered to be an important failure mode for large-volume containments at the time NUREG-1335 was published. However, the IPE for Palisades (a PWR with a large-volume containment) identified this failure mode as an important contributor to the probability of ECF.

Another important step is to determine the structural capability of the containment. The utilities were given the option of carrying out plant-specific calculations or using calculations performed for other containments of similar design. Both approaches are adopted in the various IPE submittals. Some IPEs used a relatively simple approach to determine the ultimate capability of the containment. For example, a single failure pressure is usually determined on the basis of some calculated yield limit. If the pressure is calculated to exceed this failure limit, the containment is assumed to fail. Other IPEs adopted a more elaborate approach in which distributions were developed for the probability of a containment failure that is conditional on pressure. In some cases, the effect of elevated temperature was incorporated into the distributions. For these submittals, the overlap of the calculated containment pressure and the failure distribution determined the probability of failure, and the distributions were usually derived from calculations and engineering judgment. The shape of the assumed distribution is very important. The probability assigned to containment failure at lower pressures is found to be a critical assumption in several IPEs. In one IPE, the shape of the assumed failure distribution results in a conditional probability of ECF that is significantly higher than for other containments of similar design.

Most of the utilities use containment event trees to organize the accident progression portion of the IPEs. However, significant variability exists in the scope and size of the trees in the individual IPEs. Some IPEs contain rather detailed CETs with supporting calculations using computer codes such as MAAP. Other IPEs use the results of previous studies (such as NUREG-1150) as the basis for the analysis. Most IPEs divide the CET into various time frames (as suggested in NUREG-1335), such as events before and during core melt, and at the time of vessel failure, and events related to long-term core debris disposition. The potential for recovery actions during the various time frames considered in the CET is modeled in some IPEs but not in others. Isolation failures and containment bypass sequences are usually identified in the PDS structure and treated separately in the CETs. Most CETs capture the major containment threats (identified in Table 10.8), although the level of detailed used in the quantification process varies considerably between submittals. Modeling assumptions are made that significantly influence the predicted mode and timing of containment failure. Some of the modeling assumptions can be important for all reactor and containment types, whereas other issues are important for only one containment design. Table 10.8 indicates phenomena that have been found to be important challenges to containment integrity in past studies. These failure modes are discussed below in terms of early and late containment challenges.

Early Containment Challenges

In some IPEs for PWR plants, the potential for failing the steam generator tubes because of high-pressures and temperatures during core meltdown exists. If this occurs, a potential path exists for radionuclide release that bypasses containment. This is found to be significant early failure mode in several PWR IPEs. This failure mode is classified as induced SGTR to distinguish it from SGTR as an initiating event.

If the reactor coolant system remains at high-pressure during core meltdown then at the time molten core materials are released from the reactor vessel they could be dispersed and directly heat and chemically react with the containment atmosphere. An additional complication is that hydrogen combustion could also occur at the same time adding to containment pressurization. The phenomena associated with direct containment heating (DCH) were uncertain and the knowledge base limited at the time GL 88-20 (Ref 10.1) was issued. Therefore, the utilities were given the option of not addressing DCH in their IPE. If DCH was included, its impact on the probability of ECF is quite significant for some IPEs but insignificant for others. In some cases, differences in the predictions are caused by plant features (e.g., a retentive cavity configuration) in other cases the differences are driven by differences in modeling assumptions. For example, a failure in the hot leg is predicted in some IPEs, which causes the reactor coolant system to depressurize and, thus, minimizes the potential for high-pressure core melt ejection at the time of reactor pressure vessel failure.

The likelihood that the in-vessel steam explosion will be of sufficient magnitude to generate a missile which fails the containment is generally considered to be low. For those IPEs that consider this potential failure mode, the probability of containment failure is also determined to be low.

During a core meltdown accident in a BWR with a Mark I containment, molten core materials could spread across the floor of the drywell and contact the steel containment liner. If this happens, it is possible that the hot core debris could melt through the shell causing ECF. The phenomena associated with this failure mode were uncertain at the time GL 88-20 was issued. Therefore, utilities were given the option of not including it in their accident progression analysis. If shell melt-through is included in the IPE, then its impact on the probability of early drywell failure could be significant. In some IPEs, this failure mode results in a relatively high-probability of early failure. However, in other IPEs the failure mode is mitigated by plant-specific features (e.g., large sumps or the presence of curbs).

Combustion can threaten the integrity of some containments early or late during a core meltdown accident progression. However, combustion is not a concern in IPEs for BWRs with Mark I or Mark II containments because the atmospheres of these containments are inerted during operation. Also the probability of a H_2 combustion event of sufficient magnitude to fail a large-dry or subatmospheric containment is generally found to be relatively low in the IPEs for these containment designs.

Combustion is potentially important for BWRs with Mark III containments and PWRs with ice condenser containments. Both containment types incorporate ignition systems (glow plugs) designed to deliberately burn H_2 at low-concentrations. The ignition systems are intended to prevent combustion events (large burns or detonations) that might fail containment. The effectiveness of these ignition systems can, therefore, significantly influence accident progression for these containment designs. For some accident sequences (e.g., station blackout) the ignition systems will not be operating and if power is restored a damaging combustion event was postulated in some IPEs. The IPEs indicate whether hydrogen combustion is a significant threat in individual analyses is partly driven by modeling assumptions but also by plant features such as the much larger quantity of zircalloy (and hence hydrogen generation) in the BWR cores versus the PWR cores.

Late Containment Challenges

If the containment heat removal systems fail, an important late containment failure mode in many IPEs is overpressurization failure. If the cavity or pedestal is flooded, the main driving force for pressurization is steam. If the cavity or pedestal is dry, the driving force comes from gases released from core-concrete interactions. The gases released from core-concrete interactions are predicted in the IPEs to be at much higher temperatures than the steam released from a flooded cavity. Consequently, in some IPEs, the impact of high-temperatures as well as high-pressures is considered when estimating the containment retaining capability.

Another important late containment failure mode in some IPEs is basemat melt-through. This is usually only modeled to occur for accident sequences in which the cavity is dry. In addition, the thickness of the concrete basemat determines whether the containment fails as a result of the core debris penetrating the basemat or by overpressurization caused by gases released during core-concrete interactions.

For those containments in which the reactor vessel is supported by a pedestal, the potential for the core debris to cause structural failure of the pedestal wall is identified in some IPEs. The failure of the wall can cause the reactor vessel to move which in some designs can fail penetrations through the containment wall.

10.4.2.3 Radionuclide Release

When the various radionuclide release paths are identified the timing, magnitude, and characteristics of the radionuclide releases have to be determined. Those attributes of the release path that are found important in the IPEs include the size and flow path of the opening in containment, the operability of sprays, and whether or not parts of the release path are flooded. For ISLOCA sequences in some PWR IPEs, significant aerosol retention is predicted in the piping leading from the primary system to the release point. In addition, in some IPEs, the release point is submerged so that additional retention of the aerosols is predicted.

For BWR IPEs, whether or not the radionuclides pass through the suppression pool has a strong effect on the quantity of radioactivity eventually released to the environment. Any paths that bypass the suppression are, therefore, found to be significant in BWR submittals. These paths include failure of vacuum breakers and penetrations.

Operation of the spray system is found to reduce the quantity of aerosol in the IPEs. Thus, for those accident sequences in which the spray systems were operating the quantity of radionuclides released are relatively low.

The utilities were given the option to perform plant-specific source-term calculations or to use existing calculations for similar release paths and containment designs. A large number of utilities use the MAAP code (Ref. 10.3) to calculate source terms for a few "representative" accident sequences. These representative source terms are then used to represent a range of accident progression paths. While this approach is generally reasonable, there are a few cases in which accident sequences with potentially large source terms were binned into a representative source term with relatively low-release fractions.

REFERENCES FOR CHAPTER 10

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10.5 USNRC, "Anticipated Transient Without Scram for Light-Water Reactors Staff Report," NUREG-0460, Vols. 1 and 2, April 1978.

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11. CORE DAMAGE FREQUENCY PERSPECTIVES

Chapter 3 of Part 1 summarizes the key perspectives regarding plant-specific features and assumptions that play a significant role in core damage frequency (CDF). This chapter provides a more in-depth discussion of these perspectives addressing (on a reactor class basis) in further detail the dominant contributors for each accident class, the range of CDFs, factors causing the range, and the significant plant improvements.

This chapter presents the perspectives on the factors that play a significant role in determining the CDF reported in the Individual Plant Examination (IPE) submittals. The key design and operational features that affect the CDF and the impact and influence of methods and assumptions on the CDF results are provided for different reactor classes. The perspectives within each reactor class were obtained by analyzing the contributions to the different accident classes defined in Table 11.1. The results of the IPEs were also reviewed to identify perspectives that are generally applicable to boiling water reactors (BWRs) and pressurized water reactors (PWRs). These general perspectives are presented first followed by more detailed discussions for each reactor class.

11.1 General CDF Perspectives

In many ways, the IPE results are consistent with the results of previous NRC and industry risk studies. The CDFs reported in the IPE submittals are shown in Figure 11.1[11.1].

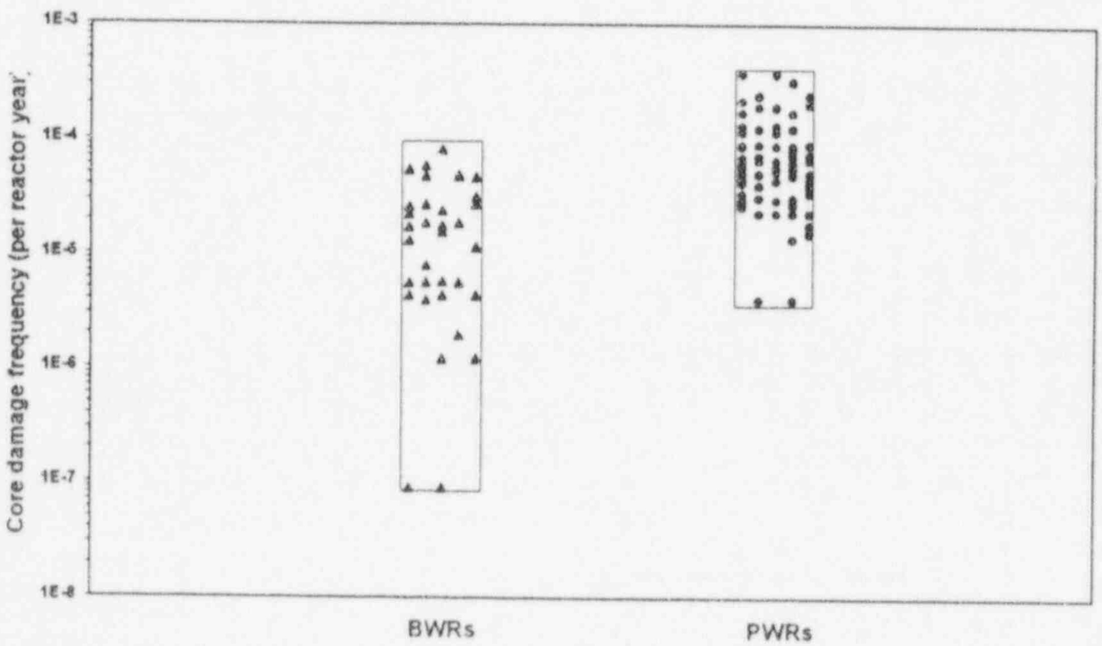


Figure 11.1 BWR and PWR CDFs.

[11.1] 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.

Table 11.1 Definition of core damage accident classes.

Accident class	Accident class definition
Transients	— *transient events that disrupt the normal conditions in the plant requiring a reactor trip with the need for core heat removal. The transient initiators include those events related to the balance-of-plant (e.g., turbine trip, loss of feedwater) and those events associated with plant support systems (e.g., loss of service water (SW), 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 — Transient event followed by immediate loss of all coolant injection systems resulting in core damage and potentially containment failure. (2) Transients with loss of DHR — Transient event followed by initial success of coolant injection system and immediate failure of DHR 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.
SBO	Transient events that strictly involve an initial loss of off-site power (LOSP) followed by a failure of emergency on-site 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 transients without scram (ATWS)	Transient events followed by a failure to terminate the nuclear chain reaction by failing to insert the control rods.
LOCAs	— *events that disrupt the normal conditions in the plant because 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 SORV are included in the transient accident classes).
Interfacing System LOCAs (ISLOCAs)	LOCAs in systems that interface with the primary system (including 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 because of submergence or spraying of required components with water.*

The IPEs indicate that the plant CDF is determined by a collection of 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 and the dominant failures contributing to that accident class vary considerably among the plants (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 result in failures of multiple front-line systems (e.g., SBO sequences tend to be important contributors for both BWR and PWR plant groups). The support system designs and dependencies of front-line systems on support systems vary considerably among the plants, which explains much of the variability in the IPE results. This variability was the motivation for the IPE program as noted in the Severe Accident Policy Statement.

Consistent with previous risk studies, the CDFs reported in the IPE submittals are lower than average for the BWR plants than for the PWR plants, as shown in Figure 11.1. Although both the BWR and PWR results are strongly affected by the plant-specific support system considerations discussed above, there are a few key differences between the plant types that cause 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 and can depressurize more easily than PWRs by using low-pressure injection systems (LPISs). This results in a lower average contribution from LOCAs for BWRs.

For transients, most PWRs can remove decay heat either through the steam generators or by using primary system feed-and-bleed. However, BWRs only remove decay heat directly from the primary system through a process analogous to feed-and-bleed, involving coolant injection and subsequent steaming either to the main condenser or to 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 require coolant injection to maintain the reactor coolant system inventory. Transient-induced LOCAs are not a significant problem for most BWRs because the normal means of decay heat removal (DHR) always requires coolant injection, and as noted above, BWRs have more injection systems available than PWRs.

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

The results for some of the individual plants vary from the general trends noted above for some plants. As shown in Figure 11.1, there is considerable variability in CDFs within the BWR and PWR plant groups that results in considerable overlap between the CDFs of the PWR and BWR plants. That is, the CDFs for many BWR plants are higher than the CDFs for many PWR plants. The specific reasons driving the differences in results among the plants (including the significantly lower CDFs for the two outlier plants shown in Figure 11.1) are discussed in Sections 11.2 and 11.3. Discussions of the factors that have the largest influence on the bulk of plants, as well as the plants with the highest and lowest CDFs are included. The variability is driven by a combination of factors including plant design differences (primarily in support systems such as cooling water, electrical power, ventilation, and air systems), variability in modeling assumptions (i.e., whether the models accounted for alternate accident mitigating systems), and differences in data values (e.g., human error probabilities) used in quantifying the models.

A summary of the key observations regarding the importance and variability of each accident sequence is provided in Table 11.2. Further details are provided in Sections 11.2 and 11.3.

Table 11.2 Overview of key IPE observations for LWRs.

Accident class	Key observations
Transients	Important contributor for most plants because of reliance on support systems whose failure can defeat redundancy in front-line systems Both plant-specific design differences and IPE modeling assumptions contribute to variations in results. Major factors are: • capability to use alternate injection systems for BWRs • capability to use feed-and-bleed cooling and susceptibility to reactor circulation pump (RCP) seal LOCAs for PWRs
SBOs	Significant contributor for most plants, with variables driven by: • number of emergency AC power sources • alternate off-site power sources • battery life • availability of firewater as injection sources for BWRs • susceptibility to reactor coolant pump seal LOCAs for PWRs
LOCAs	LOCAs are significant contributors for many PWRs BWRs generally have lower LOCA CDFs than PWRs • BWRs have more injection systems • BWRs can depressurize more readily to use low-pressure systems
Internal Floods	Small contributor for most plants, but significant for some because of plant-specific designs Largest contributors involve water system breaks that fail multiple mitigating systems (directly or through flooding effects)
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; PWR variability mostly driven by plant operating characteristics and IPE modeling assumptions
Bypass Sequences	ISLOCAs are small contributor to plant CDF for BWRs and PWRs because of low frequency of initiator Steam generator tube rupture normally a small contributor to CDF for PWRs because of opportunities for operator to isolate break and terminate accident

11.2 BWR Perspectives

Perspectives were obtained for three different BWR reactor classes that were differentiated by the vintage of the design. As indicated in Table 11.3, early BWRs with isolation condensers were placed in one group, BWRs with reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems were placed in a second group, and later BWR models with a high-pressure core spray (HPCS) system were placed in a third group. The total CDFs for all operating BWRs in each of the above groups are shown in Figure 11.2. With the exception of a few outliers, the total CDF for most BWRs falls within an order of magnitude range. The variability in the results is attributed to a combination of factors including: (1) plant design differences especially in support systems such as electrical power, cooling water, ventilation, and instrument air systems, (2) modeling assumptions, and

(3) differences in data values including human error probabilities. The largest variation logically exists in the BWR 3/4 group which is the group with the largest number of plants and where variability in plant design and modeling assumptions resulted in several plants with CDFs below the remaining BWRs and one plant (2 units) considerably below the others. This outlier will be discussed in Section 11.2.2.

Table 11.3 Summary of BWR plant classes and associated nuclear power plants.

Class	IPE submittals
BWR 1/2/3	• Big Rock Point • Dresden 2&3 • Millstone 1 • Nine Mile Point 1 • Oyster Creek
	These plants generally have separate shutdown cooling and containment spray systems and a multi-loop core spray (CS) system. An isolation condenser is utilized for these plants with the exception of Big Rock Point which is housed in a large dry containment.
BWR 3/4	• Browns Ferry 2 • Brunswick 1&2 • Cooper • Duane Arnold • Fermi 2 • Fitzpatrick • Hatch 1 • Hatch 2 • Hope Creek • Limerick 1&2 • Monticello • Peach Bottom 2&3 • Pilgrim 1 • Quad Cities 1&2 • Susquehanna 1&2 • Vermont Yankee
	These plant are designed with two independent high-pressure injection systems (HPIS), RCIC and HPCI. The associated pumps are each powered by a steam-driven turbine. These plants also have a have multi-loop CS system and a multi-mode residual heat removal (RHR) system that can be aligned for low-pressure coolant injection (LPCI), shutdown cooling, suppression pool cooling (SPC) and containment spray function.
BWR 5/6	• Clinton • Grand Gulf 1 • LaSalle 1&2 • Nine Mile Point 2 • Perry 1 • River Bend • Washington Nuclear Power Unit 2
	These plants utilize an 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 have a single train low-pressure CS system and also have a multi-mode RHR system similar to the system design in the BWR 3/4 group.

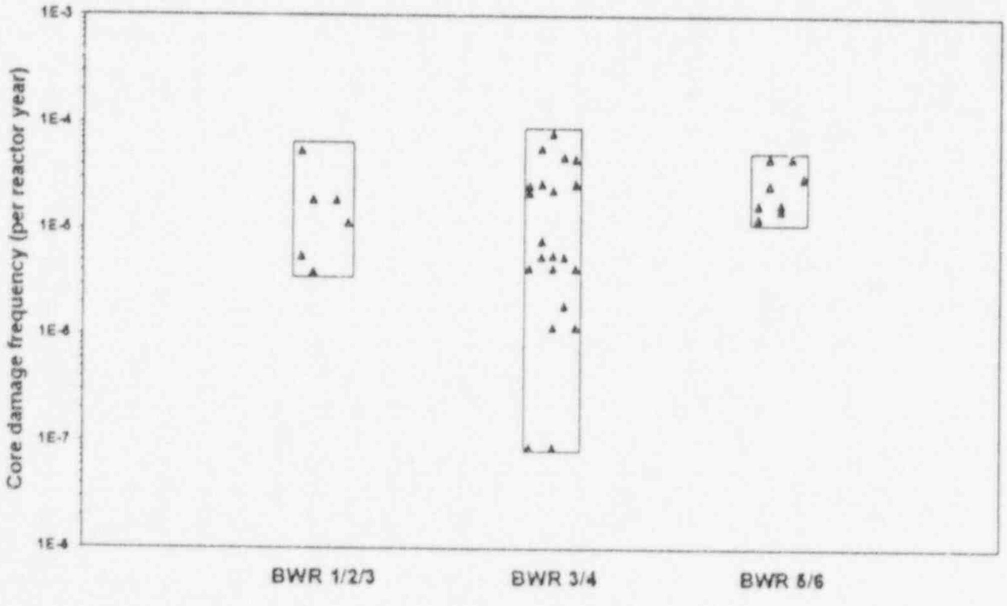


Figure 11.2 BWR plant group CDFs.

A summary of the importance of the various accident classes to the BWR CDFs and the factors influencing the results is provided in Table 11.4.

Table 11.4 Summary of CDF perspectives for BWRs.

Accident importance	Important design features, operator Actions, and model assumptions	Important plant improvements
SBO Accidents		
Important for most BWRs, regardless of plant group	Availability of AC-independent systems (e.g., HPCI system, diesel driven firewater system RCIC interface with suppression pool) Turbine bypass and isolation condenser capacity Battery life DC dependency for diesel generator startup SW system design and heating, ventilating, and air 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 (alternate 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 systems dependencies on support systems defeating redundancy Availability and redundancy of injection systems (e.g., control rod drive (CRD), motor driven feedwater pumps, service cross-tie to RHR, fire water system) Failure to depressurize influenced by operator direction to inhibit the automatic depressurization system (ADS)	Procedural and hardware enhancements to use alternate 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 emergency core cooling system rooms

Table 11.4 Summary of CDF perspectives for BWRs.

Accident importance	Important design features, operator Actions, and model assumptions	Important plant improvements
Transients with loss of 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 NPSH problems with emergency core cooling systems on suppression pool Availability of injection system located outside containment and reactor building Capability of ECCS to pump saturated water	Improved operator training Increased reliability of equipment (i.e., hardware modifications: replace pump motors with air-cooled motors) Use of alternate systems or alignment for coolant injection (i.e., align LPCI pump to condensate storage tank (CST)) Increase availability of injection systems (replenishment of CST, increase exhaust pressure trip setpoint on RCIC turbine) Revise isolation logic for plant SW and instrument air Provide control room temperature indicator for rooms containing SW pumps
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 (MSIV) open, to control vessel level, and/or to maintain pressure control Use of alternate means of injecting boron Availability of HPCS to mitigate	Improved operator training Installation of automatic inhibit of ADS Installation of an alternate boron injection capability
LOCAs		
Relatively unimportant at all but one of the BWR plants	High redundancy and diversity in coolant injection systems	Hardware modification: pipe whip constraints, replace torus suction strainers to reduce probability of clogging Expand environmental qualification program
Interfacing systems LOCAs		
Not important for BWR plants	Compartmentalization and separation of equipment	None identified

Table 11.4 Summary of CDF perspectives for BWRs.

Accident importance	Important design features, operator Actions, and model assumptions	Important plant improvements
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

RCIC and HPCI systems were placed in a second group, and later BWR models with a HPCS system were placed in a third group. The total CDFs for all operating BWRs in each of the above groups are shown in Figure 11.2. With the exception of a few outliers, the total CDF for most BWRs falls within an order of magnitude range. The variability in the results is attributed to a combination of factors including (1) plant design differences especially in support systems such as electrical power, cooling water, ventilation, and instrument air systems, (2) modeling assumptions, and (3) differences in data values including human error probabilities. The largest variation logically exists in the BWR 3/4 group which is the group with the largest number of plants and where variability in plant design and modeling assumptions resulted in several plants with CDFs below the remaining BWRs and one plant (2 units) considerably below the others. This outlier will be discussed in Section 11.2.2.

A large variability exists for each BWR group in the contributions of the different accident classes to the total plant CDF. However, licensees in all three BWR groups generally found that the following three types of accidents are the major contributors to the total plant CDF: (1) SBOs, (2) transients with loss of coolant injection, and (3) transients with loss of DHR. These three accident categories involve accident initiators and/or subsequent system failures that defeat the redundancy in systems available to mitigate potential accidents. 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 fail 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 HPCS system. Most of the accident sequences contributing to the transients with loss of coolant injection category involve failure of HPISs such as feedwater, RCIC, and HPCI (or HPCS) with a subsequent failure to depressurize the plant for injection by LPIS. 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 instrument air) that impact many of the available accident mitigating systems contribute to the importance of this accident category and also to the transient with loss of DHR category. In all loss of DHR sequences involving transient or other initiators, redundancy in mitigating systems can be lost because 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, there are a few BWRs that 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.

Important factors that impact the CDF contributions from these accident categories are discussed for each BWR plant group in Sections 11.2.1 through 11.2.3. 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, it was 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 the 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 and partially account for the higher SBO CDFs for this group. However, it should be noted that some of the licensees with isolation condenser plants generally ignored the potential for recirculation pump seal failures that effectively defeat the use of the isolation condensers. Finally, the BWR 5/6 plants had lower contributions on the average from sequences involving loss of HPISs, coupled with failure to depressurize the vessel for LPI, than BWR 3/4 since the HPCS system in the BWR 5/6 plants tends to be more reliable than the HPCI system in the BWR 3/4 plants.

11.2.1 CDF Perspectives for BWR 1/2/3 Reactors

As indicated in Table 11.5, there are six BWRs with isolation condensers grouped in the BWR 1/2/3 category. The plants in this group have more diversity in their design than the later BWRs that are more standardized. Big Rock Point is the only unit in this group that is a BWR 1 and has several unique design characteristics that influence the CDF calculated in the plant submittal. Two of the plants (Oyster Creek and Nine Mile Point 1) are BWR 2s while the other three (Dresden 2&3 and Millstone 1) are early BWR 3s that have isolation condensers (later BWRs replaced the isolation condensers with a RCIC system). Big Rock Point is housed in a dry containment; the other five plants in this group are housed within Mark I containments.

Table 11.5 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	

11.2.1.1 Summary of Results and Perspectives for BWR 1/2/3 Reactors

The total CDFs for the plants in the BWR 1/2/3 group are shown in Figure 11.2. The licensees in each design calculated similar CDFs with Big Rock Point calculating the highest value followed by the BWR 3s and BWR 2s. This trend can be mostly attributed to differences in plant design although modeling assumptions also influence the results. The calculated CDFs range from 4E-6 per reactor-year (ry) to 5E-5/ry with an average CDF for this group of plants of 2E-5/ry. Figure 11.3 provides the CDFs for each of the accident classes considered in this study as calculated by each licensee in the BWR 1/2/3 group. As indicated in Figure 11.3, the CDFs in most of the accident classes exhibit the same order of magnitude spread as is present in the total plant CDFs. However, the spread in the LOCA and transients with loss of DHR accident classes is more pronounced. As indicated in Table 11.6, the variation in the accident class CDFs is attributable to a combination of plant design differences and modeling assumptions. For all of the submittals in this group, the total CDF at each plant was dominated by one accident class. Big Rock Point is dominated by LOCAs, Nine Mile Point 1, Oyster Creek, and Millstone 1 are dominated by SBO sequences, and Dresden 2&3 are dominated by transients with loss of DHR. Although some licensees report significant contributions from ATWS and transients with loss of injection, these accident classes contribute to the overall plant CDFs to a lesser extent. None of the licensees report significant contributions from internal flooding or ISLOCA sequences.

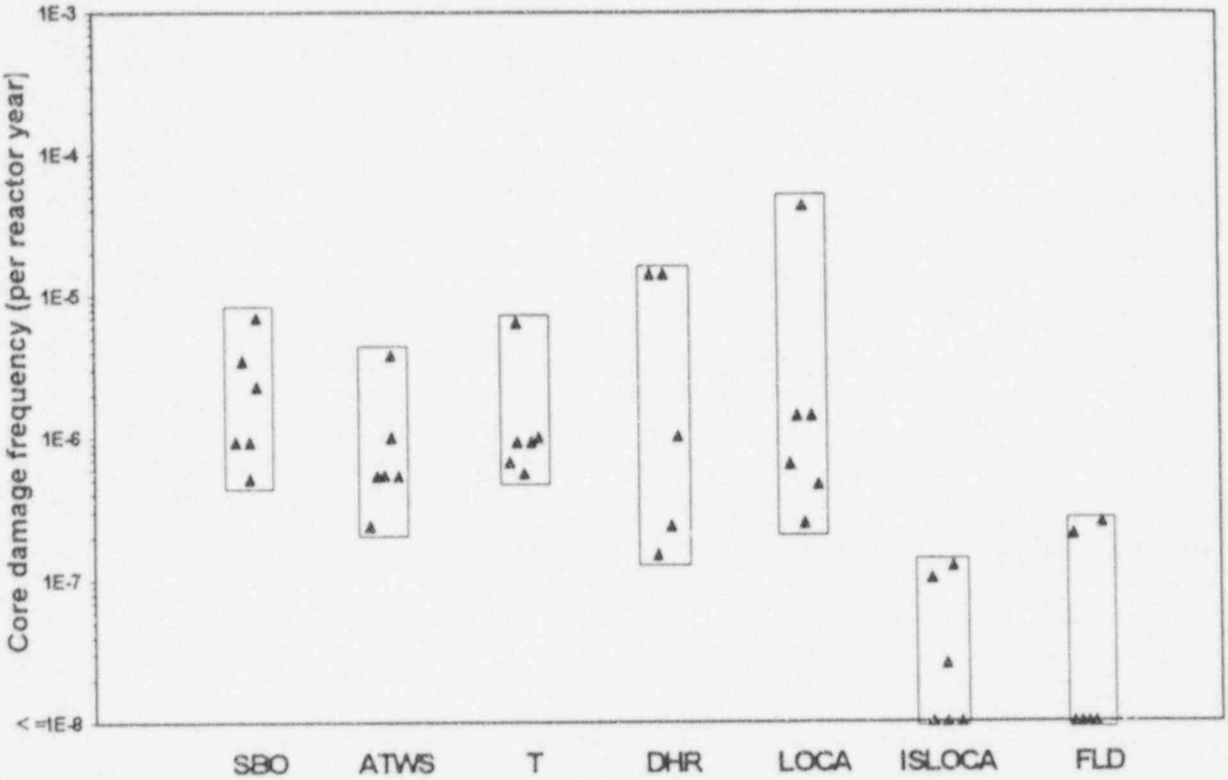


Figure 11.3 BWR 1/2/3 accident class CDFs.

Unique to this group of BWRs is the availability of an isolation condenser system that removes decay heat from the vessel. The ability of the isolation condensers to mitigate accidents is defeated by events that result in loss of primary reactor coolant. These include SORVs, LOCAs including pump seal LOCAs, and failure of the scram discharge volume to isolate. All of the plants modeled the impact of SORVs on the isolation condensers. However, only one of the licensees in this group (Nine Mile Point 1) explicitly modeled pump seal LOCAs in their submittals (others either dismiss them or do not address them in their submittals). Recirculation pump seal LOCAs can defeat the successful utilization of the isolation condensers. The other licensees either assumed that the amount of leakage was minimal, and thus, would not impact isolation condenser operability, or performed calculations that indicated that other factors such as battery depletion would fail the isolation condenser before failure because of a loss of inventory through the pump seals. The pump seal LOCAs are not found to be an important contributor to core damage at Nine Mile Point 1. Oyster Creek was the only plant to model the failure of a scram discharge volume to isolate and found it not to be important to the CDF. Other licensees dismissed this accident scenario while others did not address it.

Table 11.6 Summary of BWR 1/2/3 plant group CDF perspectives.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
	SBO	
Dominant contributors for Nine Mile Point 1, Millstone 1, and Oyster Creek. Minor contributor for remaining plants.	Availability of AC-independent HPCI system (reduces importance of SBO at Dresden 2&3) Relatively large turbine bypass and isolation condenser capacity reduces the SBO contribution at Big Rock Point DC dependency for transferring power following scram (important contributor at Oyster Creek) AC power reliability (number of diesel generators, cross-tie capability between buses and units, diverse AC power sources) Battery life	SBO CDFs range from 5E-7/ry to 7.0E-6/ry. Average CDF is 3E-6/ry. Contribution to total plant CDF ranges from less than 5% to 65%.
	Transients with loss of injection accidents	
Relatively unimportant at most BWRs 1/2/3 plants	Diversity in available coolant injection systems Failure of support systems dominant contributor since it defeats redundancy of coolant injection systems Availability of alternate injection systems (e.g., CRD, motor-driven feedwater pumps, fire water system) Failure to depressurize influenced by operator direction to inhibit the ADS	Transient CDFs range from 6E-7/ry to 7E-6/ry. Average CDF is 2E-6/ry. Contribution to total plant CDF ranges from 5% to 10%.
	Transients with loss of DHR accidents	
Dominant contributors for Dresden 2&3. Minor contributor for remaining plants.	Dominant contributor at Dresden 2&3 because of limited credit given for DHR systems (e.g., torus sprays) compared to other plants 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 at Big Rock Point only)	DHR CDFs range from 2E-7/ry to 1E-5/ry. Average CDF is 5E-6/ry. Contribution to total plant CDF ranges from less than 5% to 75%. Big Rock Point DHR contribution included in transient with loss of injection CDF.

Table 11.6 Summary of BWR 1/2/3 plant group CDF perspectives.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
	ATWS accidents	
Relatively unimportant for all plants in the BWR 1/2/3 group	Operator failure to initiate SLC in timely manner, maintain MSIVs open, control vessel level, and/or maintain pressure control Turbine bypass capacity Boron injection system design	ATWS CDFs range from 2E-7/ry to 4E-6/ry. Average CDF is 1E-6/ry. Contribution to total plant CDF ranges from less than 5% to 10%.
	LOCAs	
Dominant contributor at Big Rock Point. Minor contributor at other plants.	Failure to switchover ECCS from injection to recirculation mode at Big Rock Point requires eventual termination of ECCS to prevent overstressing of containment High redundancy and diversity in coolant injection systems	LOCA CDFs range from 3E-7/ry to 4E-5/ry. Average CDF is 8E-6/ry. Contribution to total plant CDF ranges from 5% to 80%.
	ISLOCAs	
Not important for BWR 1/2/3 group	Harsh environments induced by the ISLOCA Compartmentalization and separation of equipment	ISLOCA CDFs range from 4E-10/ry to 1E-7/ry. Average CDF is 6E-8/ry. Contribution to total plant CDF ranges from negligible to less than 5%.
	Internal flood accidents	
Relatively unimportant at most BWR 1/2/3 plants	Plant layout: separation of mitigating system components and compartmentalization	Internal flood CDFs range from negligible to 3E-7/ry. Average CDF is 8E-8/ry. Contribution to total plant CDF ranges from negligible to 5%.

11.2.1.2 SBO Accident Sequences

SBO accidents involve an initial LOSP followed by failure of the emergency on-site AC power sources (e.g., the emergency diesel generators (EDGs)). The failure of the AC power sources results in failure of multiple systems, leaving only the isolation condensers and, at some plants, steam-driven HPCI and/or diesel driven firewater systems available for mitigating this type of accident. The ability of the isolation condensers to mitigate an SBO accident (or any other transient) is defeated by the occurrence of an SORV (a dominant SBO scenario for this group) or by any loss of primary reactor coolant including pump seal LOCAs.

SBO is an important accident class for this plant group and is the dominant accident class for three plants. The variability of SBO accident class contributions for the BWR 1/2/3 plants primarily is a result of plant design features such as the availability of a HPCI system, the turbine bypass capacity, the ability to cross-tie to buses at sister units, DC bus loading, and the ability to inject coolant into the vessel using a diesel firewater pump. For example, the availability of a HPCI system helps reduce the impact of SORVs during short-term SBO sequences. Modeling assumptions and differences in data also contribute, to some extent, to the variability in the results.

Differences in AC power source availability contribute to the variability in the SBO-related CDFs for the BWR 1/2/3 group. SBO sequences involve the failure of on-site AC power sources (primarily diesel generators but also can include gas turbine generators) in addition to an LOSP. The dominant failure modes of diesel generators identified in the submittals include failure to start or run, test and maintenance outages, and common cause failure. Some of the licensees also identify support system faults such as loss of diesel generator room cooling or pump failures in the cooling water systems cooling the diesels or diesel rooms that contribute to diesel generator unavailability. All of the plants have two emergency on-site AC power sources with half of the plants configured with two diesel generators. Millstone 1 only has one diesel generator but also has an air-cooled gas turbine generator. This diversity in emergency power sources removes the potential for common cause failures. However, since the gas-turbine generator is more unreliable than the plants diesel generator, and both have relatively high maintenance unavailabilities, the emergency power configuration at Millstone 1 is an important reason for the higher CDF calculated for this plant.

Dresden 2&3 each have only one diesel generator and share a swing diesel. Failure of all three diesel generators must occur to result in a dual-unit SBO but failure of only two diesel generators (the diesel dedicated to the unit and the swing diesel) will result in a single unit SBO if off-site power is lost for only that unit. Also of importance for Dresden 2&3, is the fact that credit is given for cross-tieing unit buses during a single-unit LOSP. The modeling of the bus cross-tie during single-unit SBO sequences helps result in SBO-related CDFs less than most of the single unit plants in the group. Since the frequency of a dual-unit LOSP at Dresden 2&3 is also less than the LOSP frequencies used by the other plants in the group, the contribution from dual unit SBOs is also less than the SBO-related CDFs at most of the other plants.

SBO is not important in the Big Rock Point submittal because of a number of factors. The site has two separate incoming transmission lines that can supply required loads. Big Rock Point has a 100% turbine bypass capacity that allows the plant to withstand an LOSP by opening the turbine bypass valves and continuing to turn the turbine, producing enough power to provide the house loads, including systems required for power operations. Thus, Big Rock Point can withstand a total LOSP and continue operating without a reactor scram or turbine trip. Even if the load rejection was unsuccessful and a reactor trip occurred, the large isolation condenser capacity provides up to 6 hours of cooling before makeup is required (provided by diesel fire water pump during a SBO). The other plants in the group require that makeup be provided to the isolation condenser secondaries in approximately 20 to 30 minutes.

Loss of vessel inventory contributes significantly to the SBO contribution at some of the plants in this BWR 1/2/3 group. SBO sequences resulting in core damage in this BWR plant group must involve failure of the isolation condensers and coolant injection systems that are not dependent on AC power (e.g., a steam-driven HPCI system or a diesel-driven firewater train). Specific isolation condenser faults are noted in at least two of the submittals, such as condenser valve failures to open, failures of the makeup supply water to the condenser, failures in the condenser control logic (i.e., miscalibration of the vessel pressure switch used for condenser actuation), and operator failure to operate the condenser and its makeup supply particularly without DC power. As mentioned, some submittals specifically identify a SORV as a contributing factor to shortening the time to core damage by making the isolation condenser ineffective. Safety relief valve (SRV) operation will generally occur before isolation condenser actuation (except for Big Rock Point as discussed below) and thus provides an opportunity for a SORV to occur that requires coolant injection to maintain the vessel inventory. Most plants in this group do not have coolant injection systems that are independent of AC power and that can be used in such a scenario. Thus, short-term SBO sequences involving SORVs are important contributors except at those plants that have a steam-driven HPCI system (Dresden 2&3) or credit low-pressure injection (LPI) early from a diesel-driven firewater pump (Nine Mile Point 1) for mitigating the SORV. One licensee (Millstone 1) models an operator action to initiate the isolation condenser before a SRV demand will occur, and thus, preclude an SORV. The dominant SBO sequences for Millstone 1 include failure of this operator action which is assigned a failure probability of [illegible]. Modeling of this human error effectively results in lower SORV sequence frequencies compared to plants that do not include it in their model. Because of the unique design of the Big Rock Point plant, SRV actuation will not occur for several hours during a SBO, thus, allowing sufficient time for isolation condenser alignment before the potential for a SORV would occur. Therefore, SORVs are not contributors to the SBO CDF at Big Rock Point.

Only one submittal (Nine Mile Point 1) explicitly models a recirculation pump seal failure following loss of pump seal cooling during a SBO. As in the case of SORVs, pump seal LOCAs require vessel makeup, and thus, render an isolation condenser ineffective as the sole mitigating system in these scenarios. Some of the other licensees address and dismiss pump seal failure during a SBO as resulting in a significant loss of vessel inventory. For example, the Millstone 1 submittal states that the inventory lost from a maximum seal LOCA in both recirculation pumps will result in core uncovery at approximately 12 hours which is comparable to the battery depletion time. Since the battery depletion will fail the isolation condenser at essentially the same time as the seal LOCA, and also will prevent depressurizing the vessel for firewater injection, it is concluded that the pump seal LOCA does not affect the outcome of SBO sequences at Millstone 1. It should be noted that Nine Mile Point 1 has five recirculation pumps compared to two at Millstone 1, and thus, the impact of seal failure is expected to be more significant for Nine Mile Point 1. However, the Oyster Creek submittal which also has five recirculation pumps assumes that the amount of coolant lost through the seals is insignificant and does not contribute to core damage. Similarly, the Dresden 2&3 submittal assumes that cooling from the isolation condenser will result in minimal leakage that does not impact continued cooling by the isolation condenser. No discussion of recirculation pump seal LOCAs can be found in the other submittals. Finally, it should be noted that the contribution from pump seal LOCAs during a SBO is found to be minimal in the Nine Mile Point 1 submittal.

The Oyster Creek licensee is the only submittal to identify a scenario that involves loss of vessel inventory during a SBO through a scram discharge volume that failed to isolate. However, the contribution from this scenario is negligible compared to SORV scenarios. Some other licensees also discuss and dismiss this scenario, and yet others do not address it.

Loss of DC power is important at several plants in this group. DC power failures (typically battery failures) are important contributors to SBO at several plants. Immediate failures of DC power are typically found to contribute to the loss of injection from the HPCI system (at Dresden 2&3 only), loss of the isolation condenser or its makeup supply, or the inability to operate SRVs (and hence, keep the reactor depressurized for low-pressure firewater injection). The most significant short-term DC failure is identified in the Oyster Creek submittal. A large portion of the SBO contribution (approximately 25% of the total plant CDF) reported in the Oyster Creek submittal involves a transient initiator with independent failure of both emergency batteries on demand (common cause failure of the batteries was not modeled). Any plant trip at Oyster Creek concurrently involves transfer of AC power from the auxiliary transformers to the startup transformers, requiring DC power to perform the transfer. Hence, any initiator followed by failure of these DC batteries results in a loss of normal AC power with a subsequent loss of the main condenser and aligning the isolation condenser for operation (Oyster Creek does not have a HPCI system). The diesel generators can start and be connected to the 4160V bus without these DC buses since control power is provided by the associated diesel generator batteries. However, without DC power, the emergency loads can not be connected to the diesel generators. The vessel also can not be depressurized for firewater injection without DC power, and core damage occurs. The Millstone submittal identified similar sequences with lower frequencies (approximately a factor of 5 lower) apparently caused by data differences.

All of the plants have similar long-term SBO contributors involving battery depletion. Battery depletion times typically vary between 2 hours and 12 hours, depending on the estimated capability of the plant's batteries and whether or not DC bus load shedding is successfully performed by the plant staff. Battery depletion results in loss of the isolation condensers because of closure of secondary side makeup valves, and it also fails LPCI from diesel-driven firewater systems (credited in all of the submittals for long-term sequences) since DC power is required by the SRVs used to depressurize the plant.

11.2.1.3 ATWS Accident Sequences

ATWS sequences involve a failure to shutdown the nuclear chain reaction in the reactor by the insertion of control rods. If the control rods fail to insert, the power level remains much higher than that from decay heat loads. If the power conversion system (PCS) is lost, then most of the power is dumped into the containment (i.e., the suppression pool for all plants except Big Rock Point) which then overheats possibly leading to containment failure. Mitigation of ATWS scenarios involves injection of boron into the core using the SLC system to shutdown the chain reaction and several operator actions to control vessel level and pressure.

ATWS sequences are not dominant contributors to the CDF for this plant group. Sequences involving failure to initiate SLC injection, failure to inhibit the ADS, failure to control vessel level, and failure to trip the recirculation pumps are the most important ATWS sequences. Plant characteristics such as the turbine bypass capacity impact the modeling of certain ATWS mitigating features such as vessel level control and recirculation pump trip (RPT). Human errors are also identified as impacting the CDF in all of the submittals with a significant variability in the assigned values.

In general, dominant contributors to ATWS CDF for the BWR 1/2/3 group involve transient initiators with failure to initiate SLC. The dominant contributors to the ATWS accident class among the BWR 1/2/3 isolation condenser designs are quite similar for all of the plants in this group, except for Dresden 2&3. The dominant ATWS sequences for these plants involve transient initiators with failure to initiate SLC. The Oyster Creek and Nine Mile Point 1 submittals also identify sequences involving failure to inhibit ADS and control vessel level (results in boron flushing) as important sequences. The important contributors to these sequences tend to be human errors involving failure to initiate SLC in a timely manner, failure to maintain the MSIVs open, failure of proper water level control including lowering level and avoiding low-pressure system injection, and failure of proper pressure control such as inhibiting ADS. Variability in these human error probabilities results in some variability in the individual ATWS sequence frequencies. The Millstone 1 submittal also identifies SLC hardware faults as important, such as failure of the explosive valves to operate (independently or common cause).

The Dresden 2&3 submittal is different in its findings in that failure to trip the recirculation pumps is identified as a significant contributor. Failure to trip the recirculation pumps results in vessel pressure above the relief capacity of the SRVs leading to a breach in the primary system, followed by eventual containment failure and subsequent failure of injection. The dominant sequences involve an ATWS with the main condenser unavailable in which the automatic RPT signal fails. No credit is given for a manual RPT as insufficient time is assumed to be available to trip the pumps before the vessel overpressurizes and fails. Other licensees credit manual tripping of the pumps. The contribution from RPT failure is not as significant for Millstone 1 since it has a 100% turbine bypass capacity and only requires an RPT if the main condenser is not available.

Big Rock Point also has a 100% turbine bypass capacity that will prevent a high reactor pressure signal from initiating an automatic RPT. However, if the recirculation pumps are not tripped manually (this action was not credited in the submittal), a vessel level swell causes a rejection of condensate to the CST that in turn causes a feedwater pump trip. Without feedwater, the vessel level drops within minutes potentially resulting in vessel depressurization if ADS is not inhibited. Thus the large turbine bypass capacity at Big Rock Point does not impact the ATWS results.

Big Rock Point has a fast-acting liquid poison injection system that utilizes nitrogen pressure for injection, rather than pumps. Shutdown occurs within one minute under natural circulation core flows. Initiation of the liquid poison system in a timely manner permits reactor shutdown before vessel depressurization and restoration of vessel level with high-pressure systems (feedwater or the CRD system which has a high capacity at this plant). Even with this unique liquid poison system and a 100% turbine bypass capacity, the ATWS-related CDF for this plant is the highest in the BWR 1/2/3 group. The ATWS contribution is dominated by sequences involving loss of the condenser and feedwater and failure to initiate SLC. The vessel level decreases in these scenarios and LPI must be provided. The power level with LPI is above the isolation condenser capacity, and thus, decay heat is transmitted to the containment. Operating procedures dictate that the operator must secure injection during an ATWS when the containment pressure reaches 10 psig. This results in core damage.

11.2.1.4 Transient Accident Sequences

For all of the plants in the BWR 1/2/3 group, with the exception of Big Rock Point, this class of accident involves transients with loss of all coolant injection systems only. Transients involving loss of DHR and subsequent loss of injection systems are included in the DHR accident class. Because of the unique design of Big Rock Point which is similar to that of PWRs, the contributions to these two transient categories can not be differentiated and are discussed below. Loss of injection sequences for the BWR 1/2/3 group of plants involve failure of the isolation condensers and all available coolant injection systems.

Transient sequences are relatively minor contributors to the CDF for BWR 1/2/3s contributing between 5% to 20% of the total CDF. The dominant contributors in the transient accident class are quite similar for all of the plants in this group, primarily because of the fact that the plants in the group all have motor-driven feedwater pumps. Thus, the dominant sequences tend to involve accident initiators that fail feedwater such as loss of feedwater itself or support system initiating events (e.g., loss of AC or DC power, instrument air, or cooling water systems). Whether the dominant transient sequences involve loss of injection with the vessel at high-pressure or low-pressure is dependent somewhat on the injection systems available at each plant and which of these systems fail upon the loss of a particular support system.

Transients with loss of injection generally involve support system initiating events. The dominant transient sequences identified in the Dresden 2&3, Millstone 1, and Nine Mile Point 1 submittals are sequences involving support system initiating events that fail some or all of the available coolant injection systems. Millstone 1 and Nine Mile Point 1 both have feedwater coolant injection (FWCI) alignments with the feedwater pumps powered off emergency buses and a CS system. Millstone 1 also has a LPCI system and the Nine Mile Point 1 submittal takes credit for LPI from both the condensate and raw water systems. Because of these additional LPISs and the limited HPCI capability at these two plants, the dominant transient sequences for these two plants involve loss of HPCI coupled with failure to depressurize the vessel for LPCI. The dominant transient sequence in the Nine Mile Point 1 submittal involves a loss of instrument air initiator that results in the loss of the main condenser through closure of the MSIVs and requires the operator to take manual control of feedwater. The operator fails to control feedwater and the vessel water level rises to the isolation condenser steam line resulting in isolation condenser isolation. The

CRD is not credited for coolant injection (unless the isolation condenser functions for 1 hour) and the operator fails to depressurize the vessel for low-pressure system injection. It is not clear from the other submittals for this plant group if failure to control the vessel water level is considered a failure mode of the isolation condenser as it is in the Nine Mile Point 1 submittal.

In the Millstone 1 submittal, dominate failure contributors include load rejections with loss of normal power and LOSP initiators with the failure to recover feedwater, failure to provide makeup to the isolation condenser (diesel fire pump failure and failure to align the city water supply for makeup), and operator failure to depressurize the vessel. The submittal also identified similar contributors involving failure of the isolation condenser to actuate because of pressure sensor miscalibration.

Dresden 2&3 are the only plants in the BWR 1/2/3 group that have an isolation condenser and feedwater, HPCI, CS, and LPCI systems capable of coolant injection. As a result, the dominant transient sequences identified in the IPE involve initiators that fail many of these systems. For these plants, the dominant sequence involves an LOSP that fails feedwater (i.e., the isolation condenser operates but the operator fails to provide makeup) HPCI fails because of pump problems, and the operator fails to depressurize the vessel for LPI. Another dominant sequence involves a loss of DC power as an initiator, which apparently results in loss of feedwater, the isolation condenser, HPCI, and loss of one emergency AC division. The vessel is depressurized, but the other emergency AC division fails resulting in loss of CS and LPCI.

At Oyster Creek, the transient accident class is dominated by transient-induced LOCAs. The dominant transient sequences identified in the Oyster Creek submittal are all sequences involving initiators that fail feedwater and generally involve a SORV. With the SORV, CRD can not provide sufficient makeup to maintain the reactor vessel level nor can the isolation condenser be successfully utilized for mitigation. Since Oyster Creek does not have a HPCI system, the vessel must be depressurized for LPI. However, the only available LPI system, CS fails (condensate is failed by the initiator and Oyster Creek does not have a LPCI system) resulting in core damage. No other licensee identifies SORV sequences as being as significant most likely because of the availability of additional injection systems at those plants capable of mitigating this type of scenario.

The Oyster Creek submittal also identifies a unique set of sequences involving a loss of feedwater initiator and a failure of a scram discharge volume to isolate that results in loss of vessel inventory and a flood in the reactor building that fails one train of CS. The vessel must be depressurized to makeup the lost inventory and either the operator fails to depressurize the vessel or the other train of CS fails resulting in core damage. The other BWR submittals do not consider failure of a scram discharge volume to isolate.

Big Rock Point reports the highest transient-related CDF out of the BWR 1/2/3 group. The low-power output at Big Rock Point results in unique responses to transient sequences. For example, the actuation pressure setpoint for the isolation condenser is below the SRV setpoint. Thus, if the isolation condenser functions, there is no demand for SRVs and no potential for SORVs (SORVs can only occur with either immediate failure of the isolation condenser or failure to provide isolation condenser makeup). For transient sequences, feedwater, CRD, and the isolation condenser must fail before a demand for LPI occurs. A single CRD pump is sufficient to makeup for decay heat losses immediately following a reactor scram at Big Rock Point (two pumps in the enhanced flow mode are generally required at other plants immediately following a reactor scram). However, CRD is assumed failed for sequences in which the SRVs cycle since the CRD pumps are located inside containment and may fail in the resulting steam environment (Big Rock Point does not have a suppression pool). Because of these unique plant designs and characteristics, boil-off of vessel water to the ADS setpoint requires two or more hours even if the HPCI and isolation condenser systems immediately fail. If the isolation condenser functions, but secondary makeup fails, then boil-off to the ADS setpoint will take over eight hours.

The dominant transient sequences at Big Rock Point involve a loss of an instrument air initiator which fails the main condenser, the normal makeup to the emergency condenser (makeup must be provided by the firewater system in these scenarios), and the makeup to the condenser hotwell (required for long-term injection using the feedwater system). Pessimistically, ignored in the analysis, however, is the additional use of an alternate shutdown portable pump that can be used to maintain either the emergency condenser inventory or provide makeup to the hotwell. The dominant sequences also involve failure of high-pressure injection (HPI), in addition to the isolation condensers, and either successful vessel depressurization and failure of LPISs or failure to depressurize the vessel. Recirculation of the low-pressure coolant is not modeled for transients since the amount of coolant added to the containment sump is limited to that required to match decay heat levels.

11.2.1.5 DHR Accident Sequences

Transient sequences with loss of DHR involve accidents where coolant injection succeeds but containment heat removal fails. In this situation, 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 because of either high suppression pool temperatures, leading to loss of adequate NPSH for emergency coolant pumps, or adverse environments created in the containment or reactor building following containment venting or failure. As indicated above, the Big Rock Point submittal does not report any contributions to CDF from transients with loss of DHR. Decay heat is removed at Big Rock Point through either the main condenser or the emergency condensers. If both of these systems are lost, coolant makeup is required and DHR occurs by recirculating coolant in the sump through heat exchangers. However, for transients, the amount of coolant and heat added to the containment is minimal and recirculation is assumed not to be required.

Loss of DHR sequences are the dominant sequences reported in the Dresden 2&3 submittal. The other plants in this group reported small contributions from this accident class. The variability in the results is primarily due to the variable credit given for DHR systems. The dominant sequences in this accident class involve support system failures which fail one or more DHR systems.

Transients with loss of DHR are the dominant accident class reported in the Dresden 2&3 submittal because of the limited credit given for DHR systems. The DHR systems modeled in the Dresden 2&3 submittal are the isolation condensers and the SPC alignment of the LPCI system. The LPCI system can also be aligned for torus and drywell sprays but these alignments are not credited for prevention of core damage in the analysis (but are credited after core damage to prevent containment failure). The main condenser is not credited as a DHR system because many unspecified events will result in its unavailability. Dresden 2&3 also have a separate shutdown cooling system that also is not credited in the submittal since it is believed not to be a major DHR system during upset conditions. Containment venting is not credited for preventing core damage since it is not initiated until after adequate pump NPSH is lost resulting in the loss of available coolant injection systems. As a result, loss of DHR sequences are the dominant contributors to core damage in the Dresden 2&3 submittal. The two dominant DHR sequences involve a loss of a DC power initiator that fails feedwater and the main condenser, fails one division of AC power (and thus, one division of SPC and CS), and fails the isolation condenser. Additional valve failures in the other SPC division result in a complete loss of DHR. The available injection systems all take suction from the suppression pool and fail because of a loss of adequate pump NPSH when the suppression pool temperature increases. Loss of reactor water makeup leads to core uncovery and core damage.

Taking credit for multiple DHR systems results in lower DHR CDF contributions at other plants in the BWR 1/2/3 group. The other three plants in the BWR 1/2/3 group all take credit for additional DHR capabilities including containment venting, shutdown cooling, and torus or drywell sprays. As a result, the DHR contribution from these plants is significantly lower than for Dresden 2&3 and is dominated by initiators or conditions that fail multiple DHR systems. The dominant DHR sequence identified in the Oyster Creek submittal involves a loss of feedwater initiator with an SORV. The SORV transfers heat to the suppression pool, and thus, eliminates the use of the isolation condenser and shutdown cooling systems as heat removal systems. The operator fails to initiate the containment spray system and fails to vent the containment. Injection fails upon failure of the containment. SORV sequences are also identified as being important in the Millstone 1 submittal. One sequence involves an LOSP initiator with failure of the gas turbine generator that fails one division of power and DHR. Additional bus failures result in the loss of all other DHR trains. Another sequence involves a loss of SW initiator with an SORV. Without SW the containment spray system can not cool the containment. Cooling by the shutdown cooling system is failed when the emergency SW system and venting fail when it is not initiated by the operator. A similar loss of SW sequence, not involving an SORV, is also identified in the Millstone 1 submittal. In this sequence, the isolation condenser fails to initiate because of vessel pressure sensor miscalibration.

In the Nine Mile Point 1 submittal, the dominant DHR sequence also involves a complete loss of cooling water systems through failure of the lake intake structure that feeds the various systems. This failure also results in loss of makeup to the isolation condenser since the firewater system also draws from this intake structure. A loss of instrument air initiator also results in a significant DHR contribution as it causes an MSIV closure and a subsequent high vessel level that isolates the isolation condenser. Instrument air is also required to align torus cooling, shutdown cooling, and venting. Finally, LOSP DHR sequences (no details are provided) are also important in the Nine Mile Point 1 submittal.

11.2.1.6 LOCA Accident Sequences

LOCAs involve primary coolant system pipe breaks of various sizes and locations that require emergency coolant injection with systems such as HPCI (Dresden 2&3 only), feedwater, CS, LPCI, or alternate cooling systems such as the raw water and firewater systems at Nine Mile Point 1. Also included in the LOCA accident class are inadvertent open relief valve (IORV), transients that result in plant behavior similar to a small LOCA. Excluded from the LOCA accident class are ISLOCAs, vessel ruptures, and transient initiators with subsequent SORVs (included in the transients with loss of injection and loss of DHR accident classes).

With the exception of Big Rock Point, LOCAs are not important contributors to the CDF for this class of BWRs. LOCAs are important at Big Rock Point because of the unique design of the CS system and a pessimistic modeling assumption resulting in termination of coolant injection and subsequent core damage. The variation in LOCA contributions from the other plants in this group is primarily a result of differences in available coolant injection and DHR systems. Support system failures, plant-specific features, and modeling assumptions all helped determine what the dominant LOCA contributors are for the plants in this group.

LOCAs are the dominant contributors to core damage at Big Rock Point. In the Big Rock Point submittal, LOCAs are the dominant accident class contributing 80% of the CDF. Following the initiation of a LOCA at Big Rock Point, coolant is injected during the injection phase with either CS or feedwater (successful for only small breaks) and must eventually be switched to the recirculation phase. However, following failure of the recirculation phase, procedures require continued injection from outside the containment until the containment water level reaches a prescribed level (an unlimited supply is available since the source of injection water is Lake Michigan). At this time, injection is terminated to prevent over-stressing the containment shell. A gradual depletion of water in the reactor occurs after termination of injection resulting in core uncovery and fuel damage. The time at which core damage occurs can be several days after initiation of the LOCA allowing for restoration of the recirculation system (located outside of containment). However, recirculation system repair activities are pessimistically ignored in the submittal analysis.

Significant contributions from LOCAs involving loss of injection systems are also reported by the Big Rock Point licensee. The calculated CDFs appear to be significantly higher than at the other plants. A major factor for this is the design of the CS system which has two valves in each train that must open for successful operation (other plants generally only have one valve that must open). In addition, plant-specific operating experience yield an important contribution from failure of the ADS valves to open during small break LOCAs following loss of HPI systems.

For the majority of BWR 1/2/3s, failure of DHR is a dominant contributor to LOCA sequences. Several of the BWR 1/2/3 plant submittals identify LOCA sequences with failure to remove decay heat as being important contributors to this class of accident. These types of sequences are particularly important in the Dresden 2&3 submittal, because of the limited DHR capability that is modeled, and partially account for the higher LOCA contributions reported in the submittal. The Oyster Creek licensee identifies a small LOCA initiator with loss of a DC bus as an important LOCA contributor. The loss of the DC bus fails one division of AC power (and thus, one division of containment spray), feedwater, and venting. Hardware failures in the other containment spray division result in failure of the remaining DHR capability (the isolation condenser and shutdown cooling system are ineffective for a LOCA). Containment failure results in the loss of injection from CS and subsequent core damage. The Millstone 1 licensee also identifies a small LOCA sequence as being important within the LOCA accident class. In this sequence, a common cause failure of emergency SW results in the loss of containment spray, and the operator fails to vent the containment leading to containment failure and loss of available injection systems. The Dresden 2&3 licensees identify several medium LOCA sequences involving either failure of the operator to initiate SPC or failure of the SPC because of plugging or common cause valve failures. Vessel injection by the CS or LPCI systems is lost as a result of the loss of adequate NPSH (feedwater and condensate can not mitigate a medium LOCA). An IORV initiator is also identified as resulting in an important LOCA sequence at Dresden 2&3. In this sequence, feedwater is lost upon a high level trip when the operator fails to control flow. The isolation condenser can not be used to mitigate an IORV initiator and the operator also fails to initiate SPC.

Other important contributors to LOCA sequences involve loss of coolant injection. The BWR 1/2/3 plant submittals also identify LOCA sequences involving loss of injection as important contributors. The Oyster Creek submittal dominant LOCA sequence involves a large LOCA with failure of both divisions of CS. The Oyster Creek licensee also identifies a unique LOCA sequence involving overpressurization of the reactor water cleanup (RWCU) with relief to the reactor building emergency drain tank. The drain tank overfills releasing water into the reactor building and flooding one division of CS. The other CS division fails resulting in core damage (feedwater can not mitigate this LOCA since it is effectively a large break LOCA).

The Millstone 1 licensee identifies several small LOCA and IORV sequences in which electrical bus failures or relay failures that result in loss of power occur providing a common failure mechanism for all available injection systems (e.g., feedwater coolant injection, CS, and LPCI). The Millstone 1 licensee also identifies a small LOCA sequence with a failure of vapor suppression as an important LOCA contributor.

Both the Nine Mile Point 1 and Dresden 2&3 licensees identify medium LOCAs with failure of injection as important LOCA contributors. For Nine Mile Point 1, operation of a turbine-driven feedwater pump depressurizes the vessel along with LOCA but is not capable of maintaining the vessel level. The low reactor vessel pressure permissives required to open the CS valves fail, and there is insufficient time to align the raw water system for injection. For Dresden 2&3, the feedwater pumps are all motor-driven and can not provide sufficient injection or help depressurize the plant. The HPCI system has sufficient capacity to initially cool the core and also depressurize the plant but it fails. The operator must manually depressurize the vessel for LPI but fails to do so in the important sequences.

11.2.1.7 Internal Flood Accident Sequences

Internal flooding events involve rupture of water lines or other components or operator errors that result in a release of water that can directly result in the failure of required mitigating systems (e.g., through loss of cooling) and/or fail other mitigating systems because of submergence or spraying of required components with water. Excluded from this category is any failure in a system that interfaces with the reactor coolant system. The contribution from those types of failures are included in the ISLOCA accident class.

Internal flooding accidents are not identified as important contributors to core damage by any of the submittals in this group. In fact, only three of the six BWR 1/2/3 licensees report any CDF values for internal flood induced core damage accidents. None of these licensees find internal flood scenarios that lead directly to core damage. All require additional failures, unrelated to the flooding event, to proceed to core damage.

Internal flooding events are minor contributors to the BWR 1/2/3 group. The Millstone 1 licensee utilizes the internal flooding analysis reported in the Millstone 1 probabilistic risk assessment (PRA) performed in 1986-1987 and submitted to the NRC. The flooding analysis considers submergence and spray effects, loss of the system in which the flood originates, and propagation of flood water from one area to another. The dominant flooding events identified consist of internal floods originating from failures in the fire protection system or fuel pool cooling system leading to a reactor transient with no flood induced failures. The dominant sequence involves failure of feedwater, isolation of the condenser as a result of auto-actuation or makeup failures, and failure of the operator to provide makeup to the core. Internal flooding events that also result in loss of safe shutdown equipment are less significant and generally involve flood sources that flooded the LPCI/CS pump rooms.

The Oyster Creek flooding analysis considers submergence and spray effects, flood propagation, and the availability of drains. The dominant flood scenario identified in the Oyster Creek submittal consists of a feedwater line break in the turbine building failing both condensate and feedwater. Following the break, two emergency relief valves fail to reclose, followed by failure of both trains of CS to inject for core cooling. Note that the submittal does not identify any other failures the feedwater line break (e.g., flood) has on other plant systems, other than feedwater and condensate. Other internal flooding scenarios identified in the Oyster Creek submittal involve a SW failure in the reactor building (fails the core and containment spray systems) coincident with an LOSP and an SORV, and a circulating water failure in the turbine building (fails instrument air, condensate, and feedwater) coincident with a SORV and failure of CS.

The Big Rock Point submittal identifies only one flooding location of any significance. The screenhouse contains support systems for the operation of the main condenser, feedwater, makeup to the emergency condenser, and CS. However, sufficient systems remain available to prevent a loss of reactor inventory or failure of DHR failure even if all equipment in the screenhouse is lost from the flood. The domestic water system can be used to supply makeup to the isolation condenser and cool the air compressors (air is required for isolation condenser makeup).

11.2.1.8 ISLOCA Accident Sequences

An ISLOCA occurs when valves that normally isolate the reactor coolant system from low-pressure systems (e.g., CS) fail, resulting in backflow from the reactor coolant system through the low-pressure piping. If the low-pressure piping or other components (e.g., seals, relief valves, or flanges) can not withstand the resulting pressurization, then a LOCA results. If the breach occurs in a portion of the piping that is outside containment, then a LOCA that bypasses containment results, with effluent from the LOCA being discharged into the reactor or auxiliary building. The adverse environments created in these buildings can fail required mitigating systems. Furthermore, the loss of coolant from the containment can require coolant injection from external sources.

ISLOCAs are minor contributors to the BWR 1/2/3 group. None of the BWRs in this group report a significant contribution from ISLOCAs. These scenarios are typically low frequency events, because they involve multiple valve failures (typically check valves and motor-operated valves in series) followed by piping or other component failures resulting from overpressurization and subsequent failure of available mitigating systems. However, ISLOCAs can be important to risk because the containment bypass leads to larger fission product releases.

Variation in the ISLOCA contributions appear to be caused by a combination of plant design and modeling assumptions. Specifically, the impact of harsh environments in the reactor building created by the ISLOCA that induce failure of mitigating systems appears to be important. Modeling of this impact is somewhat variable and is affected by the separation of systems and the degree of compartmentalization in the plant. The highest reported CDFs generally occur for licensees which explicitly stated that they modeled harsh environment impacts. Additional insights could not be identified because of the generally poor documentation of this accident class in the IPE submittals.

11.2.2 CDF Perspectives for BWR 3/4 Reactors

Twenty one units (15 IPE submittals) make up the BWR 3/4 plant group with RCIC systems. A list of the plants in this plant group is provided in Table 11.7. All of the units are housed in Mark I containments, except for Limerick 1&2 and Susquehanna 1&2, which are in Mark II containments.

Table 11.7 Plants (per IPE submittal) in 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

11.2.2.1 Summary of Results and Perspectives for BWR 3/4 Reactors

The total CDFs for the plants in the BWR 3/4 group are shown in Figure 11.2. With the exception of a few outliers, the total plant CDFs for this group vary by an order of magnitude. The average CDF for the group is 2E-5/ry, with the reported CDFs ranging from 9E-8/ry to 8E-5/ry . Figure 11.4 also shows the CDFs for each of the accident classes considered in this study as calculated by all of the BWR 3/4 plants reviewed.

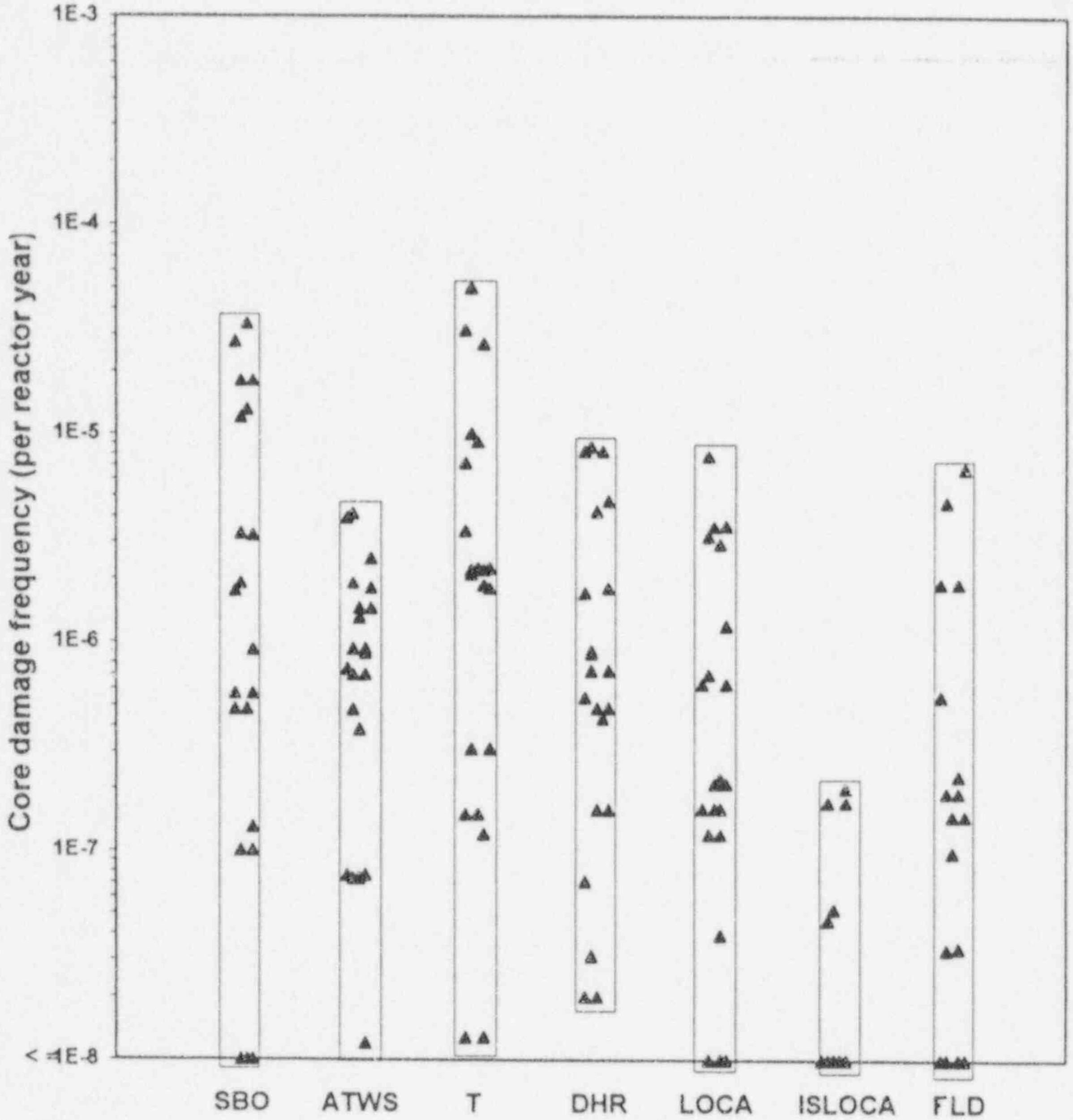


Figure 11.4 BWR 3/4 accident class CDFs.

As indicated in Figure 11.4, the CDFs for each accident class also varied substantially. However, a review of the accident class contributions indicates that multiple submittals in the BWR 3/4 group identify SBO, ATWS, transients with loss of injection, and transients with loss of DHR as dominant contributors to core damage. In addition, several licensees calculate significant contributions from internal flooding events and LOCAs. On average for the group, the total plant CDFs are driven equally by SBO and transients with loss of HPI systems with failure to depressurize the vessel for LPCI and (to a lesser extent) by loss of DHR sequences. This is not surprising since these are accident classes or sequences where a relatively few number of systems must fail to result in core damage. Lesser contributions occur, on average, from accident classes involving low frequency initiating events such as ATWS, LOCAs, and internal flooding events. All of the licensees in the BWR 3/4 group calculate negligible contributions from ISLOCAs. Important design features, operator actions, and modeling assumptions impacting the results for this plant group are listed in Table 11.8.

Table 11.8 Summary of BWR 3/4 plant group CDF perspectives.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
	SBO accidents	
Important for most BWRs in this plant group	Availability of AC-independent systems (e.g., HPCI system, diesel-driven firewater system, RCIC interface with suppression pool) Battery life (shorter battery life typically results in higher CDF) 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)	SBO CDFs range from negligible to 3E-5/ry. Average CDF is 7E-6/ry. Contribution to total plant CDF ranges from 0% to 90%.
	Transients with loss of injection accidents	
Important for most BWR 3/4 plants	Injection systems dependencies on support systems defeating redundancy Availability of alternate injection systems (e.g., CRD, motor-driven feedwater pumps, service cross-tie to RHR, fire water system) Failure to depressurize influenced by operator direction to inhibit the ADS	Transient CDFs range from 1E-8/ry to 5E-5/ry. Average CDF is 7E-6/ry. Contribution to total plant CDF ranges from less than 5% to 85%.
	Transients with loss of DHR accidents	
Important for many plants in this BWR plant group	Availability of injection system located outside containment and reactor building and ability to operate following containment failure Loss of NPSH for ECCS pumps when suppression pool temperatures increase Limited analysis to support success criteria - no credit for DHR system (e.g., venting) Dependency of DHR systems on support systems	DHR CDFs range from negligible to 9E-6/ry. Average CDF is 2E-6/ry. Contribution to total plant CDF ranges from 0% to 30%.
	ATWS accidents	
Significant contributors for a few BWR 3/4s	Operator failure to initiate SLC in timely manner, maintain MSIVs open, control vessel level, and/or maintain pressure control Use of alternate means of injecting boron Turbine bypass capacity	ATWS CDFs range from 1E-8/ry to 4E-6/ry. Average CDF is 1E-6/ry. Contribution to total plant CDF ranges from less than 5% to 80%.

Table 11.8 Summary of BWR 3/4 plant group CDF perspectives.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
LOCAs		
Not important for BWR 3/4s	High redundancy and diversity in coolant injection systems Depressurization requirement for LPCI during medium LOCAs	LOCA CDFs range from 8E-10/ry to 8E-6/ry. Average CDF is 1E-6/ry. Contribution to total plant CDF ranges from less than 5% to 20%.
ISLOCAs		
Not important for BWR 3/4s	Compartmentalization and separation of equipment Harsh environments induced by ISLOCA	ISLOCA CDFs range from negligible to 2E-7/ry. Average CDF is 2E-8/ry. Contribution to total plant CDF is negligible.
Internal flood accidents		
Relatively unimportant at most BWR 3/4s	Plant layout: separation of mitigating system components and compartmentalization	Internal flood CDFs range from negligible to 7E-6/ry. Average CDF is 8E-7/ry. Contribution to total plant CDF ranges from 0% to 25%.

11.2.2.2 SBO Accident Sequences

SBO sequences involve an LOSP, followed by a loss of all on-site AC power provided by EDGs, and in some cases, by gas turbine generators. Only steam-driven systems such HPCI and RCIC and diesel-driven firewater trains are available for coolant injection during an SBO. However, operation of these systems is limited because of the loss of support systems such as HVAC and DC power. HVAC is lost immediately, while DC power will eventually fail during an SBO unless some mitigating actions are performed.

SBO is an important accident class for this plant group and is the dominant accident class for some plants. The variability of SBO accident class contributions for the BWR 3/4 plants is primarily because of plant design characteristics, such as the availability of more than one off-site AC power source, the number of diesel generators, the battery depletion time, whether load shedding is modeled, and the ability to inject coolant from a diesel-driven firewater pump. In general, having more diesel generators resulted in lower SBO-related CDFs, except where some plant-specific feature (i.e., the cooling water system alignment at Hope Creek) or assumption defeated the redundancy. In addition, a lower SBO contribution is more certain to occur when more of these factors are credited in the IPE. Modeling assumptions are also found to impact the SBO contribution. The most notable are the treatment of common cause failure of the diesel generators and the failure of HPCI and RCIC, resulting from a loss of pump room cooling or high suppression pool temperatures. Variation in component failure data, the LOSP frequency, and the LOSP recovery probabilities also contribute, to some degree, to the variation in the BWR 3/4 SBO contributions.

Both short-term and long-term SBO sequences are important SBO contributors for BWR 3/4s. The important SBO sequences for the BWR 3/4 plants include short-term and long-term sequences. Short-term SBO sequences result in core damage within approximately 1 hour or less while long-term SBO sequences result in core damage at longer periods ranging from 3 hours to greater than 15 hours. By definition, SBO sequences involve failure of diesel generators, and thus, diesel faults and faults in their supporting equipment (particularly SW for cooling; but also DC power and diesel output breakers), as well as failure to recover AC power are dominant contributors to the SBO-related CDF. Generally, the higher the number of EDGs and other on-site AC power sources (e.g., gas turbine generators) at a plant, the lower the SBO contribution (see Table 11.9). However, the cooling water system alignment required for diesel generator cooling can impact the SBO frequency. For example, Hope Creek has four diesel generators but calculates the highest SBO CDF. This is because of the fact that the assumed cooling water system success criteria results in failure of all four diesel generators if two cooling water pumps fail or the two diesel generators supplying power to these pumps fail. (Preliminary calculations performed by the utility have shown that, with operator intervention, failure of four cooling water pumps is required to fail all four diesel generators).

Table 11.9 SBO characteristics for BWR 3/4 plants.

Plant	LOSP frequency (/ry)	No. of EDGs	Battery depletion time (hrs)	SBO CDF (/ry)	% of total CDF	Short-term SBO frequency (/ry)	Long-term SBO frequency (/ry)	Comments
Browns Ferry 2	0.044	4	4	1E-05	25%	≥ 1E-05*	≥ 1E-06*	*Based on top 100 sequences, comprising 55% of CDF
Brunswick 1&2	0.074	2/unit	2	2E-05	60%	≥ 2E-06*	≥ 2E-06*	*Based on top 4 sequences, comprising approx. 95% of CDF.
Cooper	0.035	2	4	3E-05	35%	1E-05	2E-05	
Duane Arnold	0.117	2	6-12	2E-06	25%	Unknown*	≥ 6E-07*	*Based on top 5 sequences, comprising only 45% of CDF.
Fermi 2	0.012	4, 1 gas turbine	4	1E-07	<5%	≥ 6E-09*	≥ 1E-07*	*Based on top 100 sequences.
Fitzpatrick	0.057	4	8	2E-06	90%	≥ 4E-07*	≥ 1E-06*	*Based on sequences comprising only 90% of CDF
Hatch 1	0.022	2	2.5	3E-06	15%	≥ 6E-07*	≥ 2E-06*	*Based on top 100 sequences, comprising 60% of CDF
Hatch 2	0.022	2	2.5	3E-06	15%	≥ 4E-07*	≥ 2E-06*	*Based on top 100 sequences, comprising 55% of CDF

Table 11.9 SBO characteristics for BWR 3/4 plants.

Plant	LOSP frequency (/ry)	No. of EDGs	Battery depletion time (hrs)	SBO CDF (/ry)	% of total CDF	Short-term SBO frequency (/ry)	Long-term SBO frequency (/ry)	Comments
Hope Creek	0.034	4	4	3E-05	75%	Negligible	3E-05	
Limerick 1&2	0.059	4/unit	8	1E-07	<5%	Negligible	1E-07	
Monticello	0.079	2	4	1E-05	45%	≥ 1E-06*	≥ 1E-05*	*Based on reported sequences (<100% of CDF)
Peach Bottom	0.059	4	8	5E-07	10%	Unknown*	≥ 3E-07*	*Based on sequences comprising 85% of CDF
Pilgrim 1	0.142	3	8 without load shedding, 13-14 with load shedding	Negligible	0%	Negligible	Negligible	
Quad Cities 1&2	0.032 (1), 0.016 (2)	1/unit, 1 shared	8	6E-07	50%	2E-08	6E-07	(1) single unit frequency, (2) dual unit frequency
Susquehanna 1&2	0.0056	4 shared	4	4E-11*	<5%	Unknown	Unknown	*Adjusted to yearly frequency.
Vermont Yankee	0.1	2	8	9E-07	20%	Unknown	Unknown	

Short-term SBO sequences resulting in core damage also must involve failure of the HPCI and RCIC systems. For early core damage scenarios in the BWR 3/4 IPEs, HPCI and RCIC fail early typically because of turbine failure to start, test and maintenance outages, as well as HPCI/RCIC support faults such as early battery failure or, in two cases, because of a loss of pump room cooling that results during an SBO. For the delayed or late core damage SBO scenarios, loss of DC control power required by the HPCI and RCIC systems occurs as a result of battery depletion and is consistently found as a dominant reason for eventual core damage in the IPEs. Battery depletion times range from 2 hours to 14 hours depending on battery capability and proper load shedding (or whether or not load shedding is even credited in the analysis). In a few IPEs, HPCI and RCIC fail in the long-term before battery depletion caused by pump room cooling faults and/or failure to bypass steam-leak detection trips that occur when other HVAC systems stop during an SBO. Long-term failure of HPCI and RCIC during an SBO involving failure to prevent switching of the pump suction from the CST to the suppression pool (high suppression pool temperatures result in failure of the pumps seals) are also identified. Plant results do vary as to the importance of SORVs as a contributor to these sequences, since only some of the licensees in this group identify this type of failure as contributing to shortening the timing of the scenario, and hence, adding to the CDF potential for SBO scenarios.

Most of the licensees identify short-term blackout sequences with failure of HPCI and RCIC caused by hardware failures. As indicated in Table 11.9, the total short-term SBO CDFs range from 1E-5/ry for Cooper, down to a negligible level. Part of this large spread appears to result from plant-specific factors or assumptions. For example, half of the Cooper short-term SBO CDF is attributable to the fact that the Cooper HPCI system is assumed to fail immediately upon loss of pump room cooling. Generally, loss of HPCI or RCIC pump room cooling is modeled as failing the systems in the other IPEs only after the pump rooms heat up substantially (requiring several hours, and thus, is a long-term concern) and can be mitigated by opening doors to establish natural circulation cooling.

Additional factors influencing the spread in the short-term SBO contribution include the use of firewater as an alternate injection source in the short-term as credited in the Brunswick IPE and the variability in the SW system designs and its impact on diesel generator cooling. Firewater is also generally credited only in the long-term, since it takes time for manually aligning for injection and even then can only inject into the vessel if DC power is available to depressurize the vessel. Finally, the variation in component failure probabilities, the LOSP frequency, and the mission time used in quantifying the events also will account for a portion of the spread in the short-term SBO sequences.

All of the BWR 3/4 plants with significant SBO contributions identify long-term SBO sequences as contributors to the plant CDF. The BWR 3/4 group long-term SBO contributions range from a negligible level for Pilgrim to 3E-5/ry for Hope Creek. The most common long-term SBO sequence involves failure of HPCI and RCIC caused by battery depletion. Some plant-specific factors impacting the spread in the long-term SBO include the number of diesel generators, the battery depletion time (and thus, the time available to recover off-site power), and the availability of firewater for injection. The submittals with low, long-term SBO sequence contributions all have more than 2 EDGs, have battery depletion times greater than 4 hours, credit use of firewater, or meet more than one of these conditions. In addition, the ability to bypass failure modes of HPCI and RCIC by performing actions such as bypassing high room temperature trip signals, switching suction from the suppression pool to the CST to avoid seal failure, and arranging for alternate pump room cooling also reduce the long-term SBO contribution, especially for units with long battery depletion times. Most of the submittals with the higher values have two diesel generators, short battery depletion times (2 to 4 hours), and thus, short off-site power recovery times, and do not take credit for firewater. Cooper, Hope Creek, and Monticello have 4-hour battery depletion times and effectively, two EDGs, resulting in the highest long-term SBO contributions in the BWR 3/4 group. Variation in component failure data, the LOSP frequency, and the LOSP recovery probabilities also contribute to the spread in the long-term SBO sequence frequencies.

Negligible SBO contributions result at some BWR 3/4s because of unique plant features that reduce the frequency of an SBO. Three BWR 3/4 licensees (Pilgrim and Susquehanna 1&2) report negligible SBO contributions. The negligible SBO contribution in the Pilgrim IPE is attributed to several factors. First, Pilgrim has a completely separate off-site power source in addition to the normal grid connection. This source of AC power can provide power to both emergency AC power divisions, and thus, the potential for a loss of all off-site power is significantly reduced. Pilgrim has also installed an "SBO diesel generator" which can be manually loaded to either emergency AC division, and thus, supplements the normal EDGs. In addition, plant-specific data indicates a high reliability for all three EDGs. These factors all reduce the potential for entering an SBO condition. The potential for battery depletion before AC power is recovered is relatively low at Pilgrim since the installed DC batteries will operate for 8 hours without charging or load shedding and for 13 to 15 hours with load shedding. Finally, Pilgrim has installed a spool piece to the LPCI system that allows manually aligning firewater for injection apparently even in the short-term. The low SBO contribution for Susquehanna 1&2 is primarily because of the fact that common cause failure of the EDGs or emergency SW pumps is not determined to be important, resulting in an extremely low SBO frequency. Additional factors include injection from firewater and the use of a mobile generator for charging the batteries, and thus, preventing loss of HPCI and RCIC and closure of SRVs (required for vessel depressurization and injection from the firewater system).

11.2.2.3 ATWS Accident Sequences

ATWS sequences involve a failure to shutdown the nuclear chain reaction in the reactor by the insertion of control rods. With failure of the rods to insert, the power level remains much higher than that from decay heat loads. If the PCS is lost, then most of the power is dumped into the suppression pool, which then overheats (the power level exceeds the capacity of SPC), possibly leading to containment failure. Mitigation of ATWS scenarios involves injection of boron into the core using the SLC system to shutdown the chain reaction and several operator actions to control vessel level and pressure. One important ATWS issue is whether depressurization followed by LPI will be successful or lead to damaging power oscillations.

ATWS sequences are not dominant for this plant class as a whole, but are important for some units. Sequences involving failure of SLC injection, failure to inhibit ADS, failure to control vessel level and failure to trip the recirculation pumps are generally the most important sequences. Plant characteristics such as the turbine bypass capacity are utilized in some IPEs to credit level control during an unisolated ATWS as a means of reaching a stable operating condition. However, the modeling assumptions made concerning level control and other ATWS-related operations are variable, indicating a lack of consensus on the ATWS sequence modeling greater than that for other accidents modeled in the IPEs. Component data differences are not significant in most IPEs, with the notable exceptions identified in the Susquehanna 1&2 IPE, where a lower reactor protection system (RPS) failure probability is used when evaluating a full ATWS scenario. Human errors during ATWS scenarios are also identified as impacting the CDF in most of the IPEs with a large variability in assigned probabilities.

Modeling assumptions, plant characteristics, and operator actions account for the variability in the ATWS CDF within the BWR 3/4 group. Examples of each are provided below.

Modeling differences causing variability in ATWS contributions are best illustrated by the consideration of sequences involving failure to inject boron into the core. Most of the BWR 3/4 licensees identify sequences involving SLC failure as dominant contributors to the ATWS CDF, since failure to inject boron is generally assumed to result in core damage in most of the IPEs. In addition to SLC hardware failures and failure to initiate SLC, the SLC system being out for test or maintenance, operator failure to restore SLC properly after testing, and failure of a battery charger in support of the SLC system are identified as SLC failure modes. Monticello and Fitzpatrick licensees also take credit for alternative boron injection given SLC failure. Taking credit for this alternate boron injection helps reduce the contribution from SLC failure for both units. Failure to initiate SLC is important for most of the BWR 3/4 plants. However, the contribution from this human error is variable because of a large variability in the assigned probabilities. Three units, Hope Creek and Limerick 1&2, have auto-actuated SLC systems, and thus, have no significant contribution from SLC manual actuation errors.

The degree to which RPT affects the ATWS contribution is influenced by both modeling differences and plant variability. RPT is required to reduce core power from 100% to approximately 40% during an ATWS. Most of the units in the BWR 3/4 group assume RPT failure will result in core damage since the resulting power is greater than the turbine bypass capacity and will result in excess heat rejected to the containment resulting in containment failure. Some of the licensees in the group, such as Cooper, identify failure of RPT as a dominant contributor. The RPT success criteria (i.e., credit given for the main breaker trips in addition to the field breakers that are signaled to trip during an ATWS) is an important factor in determining the importance of this type of ATWS sequence. For example, the Cooper submittal does not take credit for the main breaker tripping and assumes that both pumps are required to trip. Other licensees assume only one pump has to trip and take credit for the main breakers. RPT failure is not as critical for Vermont Yankee, since it has a 105% turbine bypass capacity that allows for full power to be transported to the main condenser as long as the MSIVs remain open.

Additional required operator actions during an ATWS are found to be important and include inhibiting ADS from automatically depressurizing the plant, controlling coolant injection flow to reduce core power and prevent power oscillations, and prevent flushing injected boron from the core. The probabilities assigned to the failure to perform these actions are obviously important in determining the impact on the ATWS CDF, as are modeling assumptions. For example, failure of the operator to inhibit ADS is assumed to result in core damage in several of the group IPEs and is an important ATWS contributor in the Cooper IPE since uncontrolled LPCI from the ECCS results in flushing injected boron from the core. However, several licensees in this group, including Pilgrim and Brunswick 1&2, assume failure to inhibit ADS will not lead to core damage if the operator controls coolant injection flow. Most licensees model level reduction as a means of reducing core power only in conjunction with operation of SLC. However, some licensees (e.g., Hatch 1&2) take credit for level reduction during un-isolated ATWS scenarios by itself as a means of reaching a stable power condition within the turbine bypass capacity of the unit. Failure to control increasing the vessel level after SLC successfully injects the required amount of boron into the core is important in some submittals (e.g., Limerick 1&2) since it is assumed to result in flushing boron from the core. Boron flushing as a result of failure of SRVs to reclose is also indicated as a failure mode of interest contributing to the ATWS CDF in the Pilgrim IPE.

Plants with low ATWS CDFs credit systems or actions not generally included in other IPEs. Fitzpatrick, Quad Cities 1&2, and Susquehanna 1&2 licensees calculate the lowest ATWS frequencies of all the BWR 3/4 plants with ATWS CDFs less than 1E-7/ry. The Fitzpatrick submittal lists the frequency of sequences involving failure of RPT, failure to inhibit ADS, and failure of the operator to initiate SLC as all less than 1E-8/ry. The Fitzpatrick submittal takes credit for use of RCIC as a means of level control following SLC injection. Credit is also taken for alternate boron injection using the CRD system when SLC fails and the operator controls level at the top-of-active fuel (TAF). The Quad Cities 1&2 submittal also takes credit for injection from RCIC and also from a separate motor-driven safe shutdown injection system which reduces the contribution from loss of injection sequences. The dominant ATWS sequences for Quad Cities 1&2 involve failure to initiate SLC but are several orders of magnitude below similar sequences at other units because of a relatively low human error probability for SLC initiation and to credit given for level control, by itself, as a means of reducing core power to a safe level. In the Susquehanna 1&2 submittal, the operator is assumed to always initiate SLC when required. In addition, manual rod insertion is credited since it can be accomplished before containment failure occurs. Another significant factor in the Susquehanna 1&2 ATWS modeling is the partitioning of the ATWS frequency into partial and complete ATWS scenarios. Partial ATWS scenarios involve failure to insert half the control rods and are less demanding than full ATWS scenarios.

11.2.2.4 Transient Accident Sequences

This class of accident involves transients with loss of all coolant injection systems only. Transients involving loss of DHR and subsequent loss of injection systems are included in the DHR accident class. Loss of injection sequences can involve failure of HPI systems such as feedwater, HPCI, and RCIC with either a failure to depressurize the vessel for subsequent injection with low-pressure systems or a successful vessel depressurization, and failure of the low-pressure systems which include LPCI, low-pressure core spray (LPCS), and condensate.

Overall, transients with loss of injection have the highest CDF contribution of all the accident classes for BWR 3/4s. This accident class is dominated by sequences involving failure of all HPI systems and failure to depressurize the vessel. Transient initiating events that fail feedwater and support system initiators that can fail feedwater and other HPI systems are found to be important. A loss of a DC bus initiator is particularly important at some units because of its impact on HPI and ADS valves. The importance of loss of HPI sequences is impacted by plant characteristics such as the type of feedwater pumps available (i.e., motor- vs steam-driven), the credit given for use of CRD in the enhanced flow mode, and the availability of other HPI systems. Modeling assumptions such as not preventing ADS auto-actuation or allowing for feedwater recovery can also significantly impact the CDF from this class of accident. Operator error probabilities for failure to manually depressurize the vessel are widely variable in the submittals and significantly impact the results. Transient sequences with loss of HPIS and LPIS with successful vessel depressurization are of lesser importance than transient sequences with loss of only HPI systems since multiple LPI systems must fail in addition to the high-pressure systems. Support system initiators that fail multiple systems are identified as important systems to this accident class.

Transients with loss of HPI and failure to depressurize the vessel for LPI dominate this accident group for BWR 3/4s. These accidents are dominated by initiators, such as LOSP and MSIV closure that result in the loss of feedwater. MSIV closure initiators are not as important for units with motor-driven feedwater pumps (e.g., Pilgrim and Vermont Yankee), since MSIV closure do not fail feedwater. Where the loss of feedwater is the initiating event, most of the licensees take credit for restoring feedwater. Where feedwater is not failed by the initiating event, two licensees identify operator failure to manually control feedwater to prevent high-level trip as a dominant contributor. The Monticello submittal identifies specific feedwater faults as important contributors, such as failure of the feedwater control panel and common-mode failure of feedwater injection check valves which fail feedwater, HPCI, RCIC, and maximized CRD flow, all of which inject into the feedwater header.

Given loss of feedwater, other HPI systems such as HPCI, RCIC, or other systems such as CRD must operate to provide coolant injection. The failure modes of HPCI and RCIC and the credit given for other HPIS influence the loss of HPI transient CDF contributions in the submittals. The licensees generally agree that early failures of HPCI/RCIC are dominated by pump failures to start or run, test and maintenance unavailability, or injection valve hardware faults. Loss of DC buses is also an important contributor for some units since their failure impacts HPCI, RCIC, and the ability to ADS. The Hatch 1&2 submittal identifies a contribution associated with an initiating event involving a loss of control room cooling that results in loss of feedwater and a spurious trip of HPCI. Some licensees are also significantly impacted by other failure modes of these systems. The Cooper submittal identifies the failure of two SRVs to reclose as a dominant sequence since RCIC cannot successfully be used for injection (RCIC cannot make up the loss out the two SRVs) and additional SRVs must be opened to depressurize the vessel before low-pressure systems can inject into the vessel.

The availability of alternate HPIS generally reduces the contribution from loss of HPI sequences. Three BWR 3/4 licensees (Brunswick 1&2, Cooper, and Susquehanna 1&2) take credit for injection from CRD in the short-term on the basis of plant-specific, thermal-hydraulic calculations. As a result, the Brunswick 1&2 and Susquehanna 1&2 contributions are small and the Cooper contribution is dominated by SORV sequences for which RCIC and CRD cannot provide sufficient flow, but depressurization is assumed to be required for LPI. Most units indicate CRD can not provide sufficient flow in the short-term. The Quad Cities 1&2 submittal is unique in that it takes credit for injection from a special motor-driven safe shutdown injection pump. This system partially accounts for the relatively low contribution for those units.

The availability of motor-driven feedwater pumps also generally reduces the loss of HPI transient contribution. However, the Pilgrim licensee calculates the highest frequency for this type of sequence, despite having motor-driven feedwater pumps. Pilgrim's dominant loss of HPI transient sequences involve initiators that fail feedwater, hardware failures of HPCI and RCIC, and failure of the operator to depressurize the vessel. Monticello, Fermi, and Vermont Yankee also have motor-driven feedwater pumps and their loss of HPI transient contributions are dominated by initiators that fail feedwater, but the contribution from these types sequences are an order of magnitude lower than Pilgrim. The lower contributions for these units appear to be related to the credit given for feedwater recovery (Monticello), modeling of early depressurization for injection with condensate and auto-actuation of ADS (Fermi), and a lower operator error in failing to manually depressurize the vessel (Vermont Yankee—see below).

Given loss of all HPI, all the licensees generally agree that operator failure to depressurize the vessel for LPI is a dominant contributor to core damage for these accident types. Most of the submittals, except for Cooper and Fermi, assume that ADS is inhibited following the initiating event; thereby requiring operator intervention to depressurize via the SRVs if there is loss of HPI. None of the submittals give credit for alternate means of depressurization. As discussed in Chapter 13, the operator error probabilities for failure to depressurize range from approximately 1E-1 to 1E-4. This variability in the human error probability for failure to depressurize the vessel contributes to the spread in the sequence frequencies. Submittals with low human error probabilities tend to have low loss of HPI sequence frequencies. However, since this is not the only factor influencing the frequency of these types of sequences, some plants with low human error probabilities still have significant contributions. The two licensees that allow auto-actuation of ADS have lower failure-to-depressurize probabilities than the licensees where operator action is required to depressurize the vessel.

Loss of DC power is an important contributor at some plants, since it fails HPIS and ADS. The Duane Arnold, Fermi 2, Hatch 1&2, and Quad Cities 1&2 licensees identify loss of DC power as the dominant transient initiating event. The Duane Arnold licensee assumes that the loss of all DC power leads directly to core damage because there are no procedures in place to direct the operator on how to respond to this condition. The other licensees identify the loss of DC power as a dominant transient because this initiator fails feedwater, standby feedwater, HPCI, RCIC, the safe shutdown system at Quad Cities and prevents control of the SRVs from the control room for manual depressurization for LPI.

Transients with loss of both HPISs and LPISs with the vessel depressurized are smaller contributors than transients with loss of HPI and failure to depressurize the vessel. Transients with failure of all coolant injection systems (high- and low-pressure) with the vessel depressurized also contribute to the transient with loss of injection accident class for the BWR 3/4 plants. However, the contribution from these sequences is generally less than the transient sequences with loss of only HPI (previously discussed), since more systems must fail. Systems generally credited in the submittals include those high-pressure systems identified above in the previous discussion and low-pressure systems including condensate, LPCI, LPCS, and SW cross-tied to LPCI. The Pilgrim submittal also credits use of firewater in short-term loss of all injection scenarios.

Support system initiating events are important for loss of all injection transients since they fail multiple injections systems. LOSP initiators are important contributors at most of the units since they fail both feedwater and condensate. For the Browns Ferry submittal, the dominant transient events look almost like SBO scenarios in that they are dominated by LOSP initiators followed by failure of three of four EDGs. HPCI and RCIC tend to fail late because of battery depletion, followed by failure of the lone remaining train of LPI as a result of failure of the pump to start, testing and maintenance outages, and failure of the pump discharge check valve to open (all other trains are lost because of failure of three of four EDGs). Other initiators that fail feedwater (loss of feedwater and MSIV closure) are also important. Support system initiators, such as loss of instrument air and loss of a division AC bus that can fail multiple injection systems, are also important contributors to this class of accident in the Cooper submittal. Loss of DC buses initiators are important at several units including Hatch 1&2 where the loss of DC bus A causes a loss of the main condenser, prevents transferring AC power from the main generator to off-site power, and fails RCIC and one-half of the low vessel pressure permissive logic required to open the LPCI/LPCS injection valves. The Browns Ferry submittal also identifies the loss of DC power to two battery boards as leading to core damage because the loss fails HPCI/RCIC, and also fails LPI (LPCI and CS) because the loss of DC power fails the low-pressure permissives for the injection valves. Power to the battery boards is lost because of random maintenance on the batteries or chargers and/or as a result of battery charger demand faults.

Most of the important transients with loss of all injection sequences are short-term sequences where all coolant injection fails immediately because of a combination of the initiating event and hardware failures. However, a few of the important sequences involve long-term failure of some coolant injection systems. For example, the Fitzpatrick submittal identifies a late core damage scenario as being a dominant (although a small contributor to the overall CDF) transient event. For this sequence, HPCI initially operates providing reactor water makeup, but it fails late because of one SRV being stuck open, thereby depressurizing the vessel (HPCI turbine motive force is lost as a result of vessel depressurization). In addition to loss of HPCI, LPISs fail to provide makeup. Dominant failures of the LPISs identified by the Fitzpatrick submittal include miscalibration of reactor pressure transmitters failing LPI permissives, hardware failure of reactor pressure transmitters and/or control circuits for low-pressure permissives, and operator failure to locally open (at the valve) the LPI valves.

The Susquehanna 1&2 submittal does not consider common mode failure of the LPI system (LPCI, LPCS, and RHR SW cross-tied to LPCI) pumps or valves, and thus, predicts an extremely low loss of injection sequence contributions from these types of failures. The only common mode failure modeled in the Susquehanna submittal that influence loss of injection sequences is the low-pressure vessel permissives required to open the LPCI and LPCS injection valves. However, credit is given in the IPE for bypassing these permissive signals. These permissive signals are modeled in other IPEs but generally no credit is given for bypassing them to open the LPCI and LPCS valves.

11.2.2.5 DHR Accident Sequences

Transients with loss of DHR sequences are defined differently among the submittals; however, they generally involve a transient with a successful reactor trip, that fails to remove heat by venting or various means to the ultimate heat sink. RHR failures in various modes, such as SPC, are important contributors to these sequences. The DHR sequences result in adverse conditions inside the containment and in the reactor building following a containment failure that impacts continued operation of coolant injection systems.

Loss of DHR sequences are important in BWR 3/4s, because of the susceptibility of coolant injection systems to adverse containment conditions and harsh environments following containment failure. Variability in the importance of theses sequences for each unit is dependent on the assumed ability of ECCS pumps to continue operating under harsh containment conditions and the availability of alternate injection sources that supply water from sources outside the containment and their survivability following containment failure. Loss of cooling water system initiators can be important contributors to this accident class, since they impact both the DHR function and also cooling required by coolant injection systems.

Operator errors and cooling water system failures are found important in many DHR sequences. Dominant failures consistently found in DHR sequences include operator failure to restore the condenser as a heat sink and vent the containment (venting was not modeled in all submittals). Six of the BWR 3/4 licensees find loss of the ultimate heat sink (e.g., standby SW, river water, etc.) as an initiating event to be a dominant contributor to core damage for this accident category. Loss of the ultimate heat sink prevents the removal of heat from the containment and the core and also impacts continued operation of coolant injection systems.

Containment failure impacts on coolant injection systems are variable in the IPEs. Most of the licensees assume loss of coolant injection from systems taking suction from the suppression pool following containment failure because of a loss of NPSH, injection piping rupture, or harsh environments in the surrounding areas. However, there is some variability in these assumptions, and possibly in the pump design. For example, Pilgrim assumes loss of LPCI and CS pump NPSH as a result of high suppression pool temperatures. However, the Duane Arnold submittal assumes the pumps could pump saturated water. Loss of NPSH for pumps pulling suction off the suppression pool is also assumed to occur following venting in some of the submittals. HPCI and RCIC are also failed in most submittals as a result of high suppression pool temperature impacts on NPSH or pump seals. However, some licensees model switchover of the HPCI and RCIC pump suction to the CST which prevents these failure modes. RCIC failure will also occur because of a high turbine exhaust backpressure trip during a DHR scenario or following vessel depressurization to maintain containment heat-capacity temperature limits. Various probabilities are assigned to the availability of systems providing coolant injection from outside sources following containment failure. Harsh environment and piping failure probabilities for these systems following containment failure range from 0 to 1.0. In addition, the alternate injection systems credited in the submittals are also variable and include systems such as condensate, residual heat removal service water (RHRSW), CRD, and firewater.

The submittals with the highest DHR contribution are Brunswick 1&2 and Cooper. In the Brunswick submittal, containment failure is assumed to result in loss of ECCS and condensate through the flashing of the suppression pool and/or failure of injection system piping. No credit is given for firewater injection, the CRD system, and SW cross-tie in the Brunswick submittal. In the Cooper submittal, venting is only credited for the dominant DHR sequence involving a loss of SW. In this sequence, coolant injection is provided by low-pressure ECCS taking suction from the CST after venting is successful in maintaining the containment pressure below the SRV closure pressure. Coolant injection is stopped at 72 hours to prevent containment failure as a result of the high water level. For other sequences, venting is not credited and containment failure is assumed to result in failure of injection systems (including ECCS, CRD, condensate, and SW cross-tie).

Only four of the BWR 3/4 licensees (Monticello, Fitzpatrick, and Susquehanna 1&2) in this group calculate DHR accident contributions of 1E-7/ry or less. In the Monticello submittal, the success of venting is assumed to have no negative impact upon the NPSH for injection pumps taking suction from the suppression pool. In addition, continued operation of coolant injection systems following containment failure is assumed on the basis of the size and location of the expected failure. These two factors, plus the length of time available to recover DHR (two days before containment failure occurs at 103 psig) result in a small contribution from loss of DHR sequences in the Monticello submittal. Similarly, the Fitzpatrick licensee assumes that some coolant injection systems, including CRD, condensate, and HPCI (if suction is switched from the suppression pool to the CST), will survive containment failure with some probability that is dependent upon the containment failure mode. In the Susquehanna 1&2 submittal, credit is given for use of the RWCU as a DHR system. The RWCU has limited heat removal capability but eventually will be able to remove decay heat. No other licensee credits use of this system. Containment failure is assumed to result in failure of all systems inside the containment and in the reactor building. However, credit is given for firewater injection since this system is not in the reactor building. Use of firewater requires connecting a small portable generator to charge the DC batteries to maintain the SRVs open since AC power is located in the reactor building and is assumed lost upon containment failure. Finally, common cause failure of RHR components is not modeled resulting in a relatively low RHR failure probability.

11.2.2.6 LOCA Accident Sequences

LOCAs involve primary coolant system pipe breaks of various sizes that require emergency coolant injection with systems such as HPCI, RCIC, LPCI, LPCS, and other alternate injection systems that are plant-specific. Included in the LOCA accident class, where their contribution can be separated from other transients, are inadvertent open relief valves since the response of the plant is similar to a small LOCA initiator. Excluded from the LOCA accident class are ISLOCAs, vessel ruptures, and transients with subsequent SORVs (included in the transients with loss of injection and loss of DHR accident classes).

Generally most of the BWR 3/4 licensees find all sizes of LOCAs to be important contributors to the LOCA CDF. However, the LOCA CDFs at several units are dominated by medium LOCA. For example, the Hatch, Hope Creek, Quad Cities, and Pilgrim LOCA contributions are dominated by medium LOCAs primarily because of the low contribution from the other LOCA sizes. However, the dominance of the medium LOCA contribution in the Hatch 1&2 IPE is attributable to the assumption that failure of a sufficient number of SRVs to open during an MSIV closure transient results in a medium LOCA. No other licensee makes this assumption. Important medium LOCA sequences in all of the submittals include sequences where either HPCI is operating until vessel depressurization results in loss of steam flow or the HPCI pump fails combined with loss of LPI or failure to depressurize the vessel. Variability in these medium LOCA frequencies, and in fact, for all LOCA sequences is mostly a result of the availability or credit given for the alternate injection systems such as motor-driven feedwater (only available at six BWR 3/4 plants), condensate, and SW cross-tie.

None of the licensee LOCA CDFs are dominated by large- or small-break LOCAs. However, many report large LOCA sequences involving failure of LPCI and LPCS or stuck-open wetwell/drywell vacuum breakers as being important contributors to this accident class. Some small LOCA sequences involving failure of HPISs and failure to depressurize the vessel for LPI are important in the Monticello and Pilgrim submittals. These sequences appear to be important since RCIC is not credited in the Monticello submittal and feedwater (motor-driven) is not credited in the Pilgrim submittal. RCIC is credited in all other BWR 3/4 submittals for mitigating a small LOCA. Steam-driven feedwater is also credited for mitigating a small LOCA in all the other submittals except for Brunswick 1&2.

The highest LOCA contribution occurs in the Cooper submittal and is associated with inadvertent open relieve valve events. The dominant scenario involves a second SORV with subsequent failure of HPCI (RCIC cannot mitigate this accident) and failure to depressurize the vessel in time for LPI (two SRVs are not sufficient for vessel depressurization). The lowest LOCA CDF contributions for the BWR 3/4 plant group are calculated in the Fermi 2, Fitzpatrick, and Susquehanna 1&2 submittals. The Fermi 2 submittal only models small and large LOCAs, thus, precluding the dominant medium LOCA scenarios calculated in the other submittals. Small LOCAs are not important at Fermi since HPCI, RCIC, feedwater, condensate, LPCI, CS, and RHRSW are all credited as injection systems. The large LOCA contribution at Fermi 2 from loss of injection scenarios is similar to that at other units and is the dominant LOCA initiator. The Fitzpatrick licensee takes credit for use of enhanced CRD flow during medium LOCA sequences involving loss of HPCI after the vessel depressurizes, thus, reducing the importance of a generally dominant BWR 3/4 LOCA accident sequence. The use of condensate for a short period of time and standby service water (SSW) cross-tie is also credited for mitigating large break LOCAs and serves to reduce the importance of the dominant BWR 3/4 large break LOCA sequence for Fitzpatrick. The Susquehanna submittal does not model common cause failures in the LPCI, LPCS, and RHRSW systems. In addition, credit is given for bypassing the low vessel pressure permissive required to open the LPCI and LPCS injection valves (the dominant failure mode for these three systems combined in many submittals). Thus, the probability of failure of the LPCI systems is very low, resulting in small LOCA contributions. Small break LOCAs and inadvertent open relief valve contributions are further reduced because of the credit given for enhanced CRD flow.

LOCAs are not dominant contributors for this plant class. Most BWR 3/4s licensees find all LOCAs to be generally of equal importance, but some have total CDFs dominated by medium LOCA sequences involving HPCI failure and failure to depressurize the vessel for coolant injection from low-pressure systems. Differences in the credit given for alternate injection systems is the major parameter accounting for the variability in the LOCA results.

11.2.2.7 Internal Flood Accident Sequences

Internal flooding events involve rupture of coolant injection lines that directly result in failure of required mitigating systems and/or fail other mitigating systems because of the submergence or spraying of required components with water.

Internal flooding is not important for most of the units in this class. The contributing sequences generally involve SW system breaks since they impact equipment through both loss of cooling and through flood impacts. No internal flood initiator is identified that will completely fail all systems required to mitigate a flood-induced transient without additional independent failures.

SW breaks in the reactor or turbine building are common flood events reported in the BWR 3/4 submittals. The majority of the identified flood scenarios involve either a SW line break in the reactor building or a flood in the turbine building. SW or other cooling water system-related floods are particularly important since they can directly impact mitigating systems through loss of required cooling to the components and by submergence or spraying of other components. The resulting scenarios vary in the mitigating systems impacted and result in either loss of injection or loss of DHR sequences. The most important factor in determining the importance of flooding is the plant layout. Separation and compartmentalization reduce the impact of flood initiators. Additional factors influencing the results are the frequency assigned to the initiators and whether or not spray effects were modeled.

The largest flood impact is for the Monticello plant, where SW or feedwater pipe breaks fail all HPI. Core damage occurs with failure to depressurize for low vessel injection. The Browns Ferry flooding contribution is also substantial and involves turbine building flooding that directly impacts cooling water systems, instrument air, condensate, and 250V DC power. These failures result in loss of the PCS, loss of HPCI and RCIC, and failure of initiation and permissive signals for CS and LPCI.

11.2.2.8 ISLOCA Accident Sequences

An ISLOCA occurs when valves that normally isolate the reactor coolant system from low-pressure systems (e.g., CS) fail, resulting in backflow from the reactor coolant system through the low-pressure piping. If the low-pressure piping or other components (e.g., seals, relief valves, or flanges) can not withstand the resulting pressurization, then a LOCA results. If the breach occurs in a portion of the piping that is outside containment, then a LOCA that bypasses containment results, with effluent from the LOCA being discharged into the reactor or auxiliary building. The adverse environments created in these buildings can fail required mitigating systems. Furthermore, the loss of coolant from the containment can require coolant injection from external sources.

None of the BWRs in this BWR 3/4 group report a significant contribution from ISLOCAs. These scenarios are typically low frequency events, because they involve multiple valve failures (e.g., check valves and motor-operated valves in series) followed by piping or other component failures as a result of overpressurization and subsequent failure of available mitigating systems. However, ISLOCAs can be important to risk because the containment bypass leads to larger fission product releases.

Variation in the ISLOCA contributions appear to be attributable to a combination of plant design and modeling assumptions. Specifically, the impact of harsh environments in the reactor building created by the ISLOCA that induce failure of mitigating systems appears to be important. Modeling of this impact is somewhat variable and is affected by the separation of systems and the degree of compartmentalization in the plant. The highest reported CDFs generally occur for licensees which explicitly stated they modeled harsh environment impacts.

Several additional modeling variations in the treatment of ISLOCAs were noted which likely impact the results. The locations considered for ISLOCAs generally include RHR and CS lines. However, ISLOCAs at other locations (e.g., in RCIC or HPCI systems) vary because of the design pressure of the system piping. In addition, the potential for overpressurizing piping is reduced or eliminated at some plants by preventing interfacing valve stroking during power operation. At Limerick, the licensee screened out ISLOCAs on the basis of a high/low-pressure interface valve study that looked for single valves, that if opened by a spurious fault, would let high-pressure water into a low-pressure pipe. Questionable valves were identified and provided with administrative controls to prevent them from spuriously opening. Additionally, online testing of interfacing valves is performed during shutdown.

Modeling of ISLOCA mitigation also varies. On the one hand, the licensee for Fermi 2 assumes that any ISLOCA results in core damage while on the other hand, the licensee for Brunswick stopped at pipe failure arguing that mitigation of the resulting ISLOCA results in a negligible CDF. However, most licensees model ISLOCA mitigation to some degree. Two key actions that are treated variably in the IPEs is the isolation of the ISLOCA (either early or late) and depressurization of the vessel to reduce the resulting loss of coolant outside the containment.

11.2.3 CDF Perspectives for BWR 5/6 Reactors

As indicated in Table 11.10, there are eight plants in the BWR 5/6 group. This group is different from the BWR 3/4 group primarily by the replacement of the HPCI system with a single-train HPCS system. Four of the plants (LaSalle Units 1&2, Nine Mile Point 2, and Washington Nuclear Power Unit 2) are BWR 5 Mark II designs and the other four (Clinton, Grand Gulf Unit 1, Perry 1, and River Bend) are BWR 6 Mark III designs.

Table 11.10 Plants 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	

11.2.3.1 Summary of Results and Perspectives for BWR 5/6 Reactors

The total CDFs for the plants in the BWR 5/6 group are shown in Figure 11.2. As indicated in the figure, there are no outliers in the group with the CDFs all within a factor of three of each other. The average CDF for the group is 2E-5/ry, with the reported CDFs ranging from 1E-5/ry to 4E-5/ry. Figure 11.5 shows the CDFs for each of the accident classes considered in this study as calculated in the BWR 5/6 group licensee submittals. As indicated in the figure, the CDFs for each accident class varies substantially more than the total CDF. The variability in the accident class CDFs is primarily a result of a combination of plant design differences and modeling assumptions. A review of the accident class contributions indicates that multiple licensees in this group identify SBO, transients with loss of injection, and transients with loss of DHR as important contributors to core damage. These accident classes are also the dominant contributors to the group as a whole. For one licensee, Perry, ATWS is the dominant accident class. In addition, several licensees calculated significant contributions from LOCAs and internal flooding events. None of the licensees report significant contributions from ISLOCA. The important design features, operator actions, and modeling assumptions impacting the IPE results are summarized in Table 11.11.

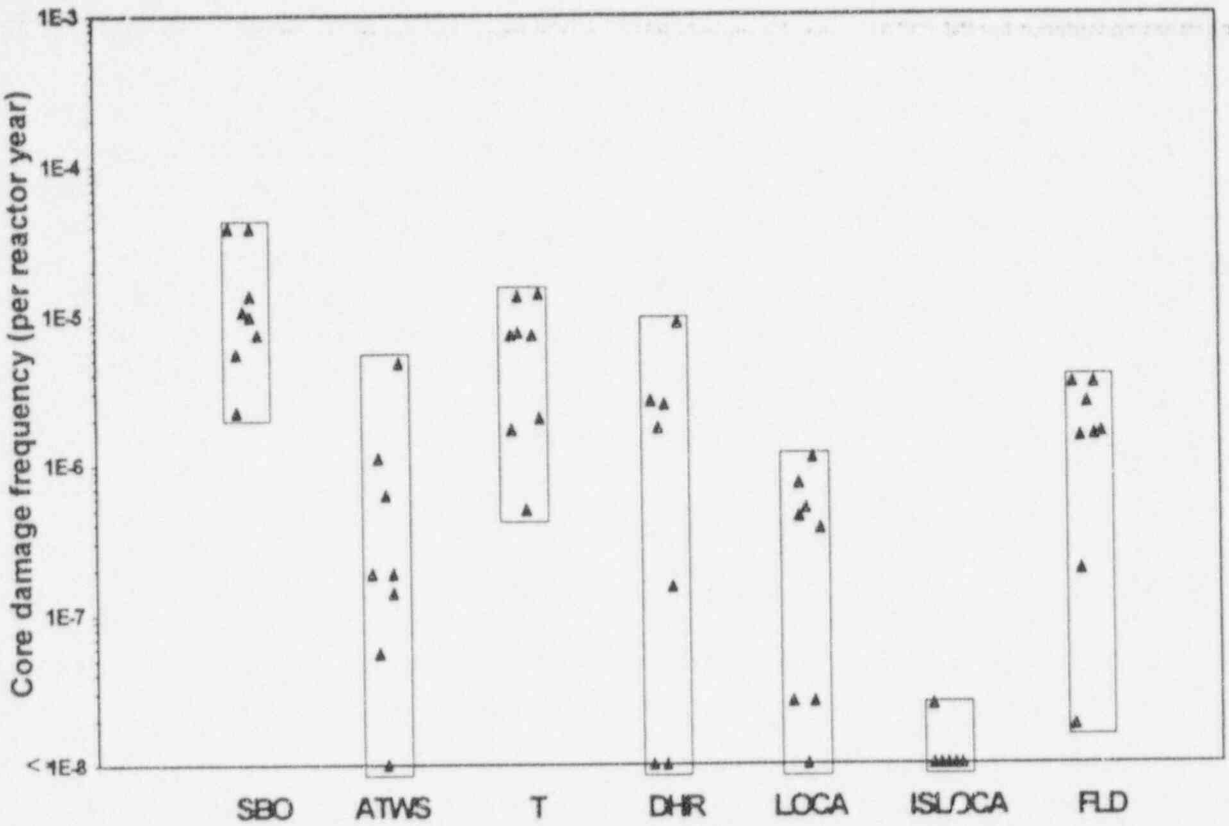


Figure 11.5 BWR 5/6 accident class CDFs.

Table 11.11 Summary of BWR 5/6 plant group CDF perspectives.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
	SBO accidents	
Important for all BWR 5/6 plants	Availability of AC-independent systems (e.g., 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)	SBO CDFs range from 2E-6/ry to 2E-5/ry. Average CDF is 1E-5/ry. Contribution to total plant CDF ranges from 15% to 90%.

Table 11.11 Summary of BWR 5/6 plant group CDF perspectives.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
	Transients with loss of injection accidents	
Important for most BWR 5/6 plants	Injection systems dependencies on support systems, defeating redundancy Availability of alternate injection systems (e.g., CRD, motor-driven feedwater pumps, service cross-tie to RHR, fire water system) Failure to depressurize influenced by operator direction to inhibit the ADS	Transient CDFs range from 5E-7/ry to 1E-5/ry. Average CDF is 5E-6/ry. Contribution to total plant CDF ranges from less than 5% to 50%.
	Transients with loss of DHR accidents	
Important for many BWR 5/6 plants	Availability of injection system located outside containment and reactor building and ability to operate following containment failure Capability of ECCS to pump saturated water Dependency of DHR systems on support systems	DHR CDFs range from negligible to 9E-6/ry. Average CDF is 3E-6/ry. Contribution to total plant CDF ranges from negligible to 30%.
	ATWS accidents	
Relatively unimportant for most BWR 5/6 plants	Operator failure to initiate SLC in timely manner, maintain MSIVs open, control vessel level, and/or maintain pressure control Use of HPCS for coolant injection	ATWS CDFs range from negligible to 5E-6/ry. Average CDF is 9E-7/ry. Contribution to total plant CDF ranges from negligible to 35%.
	LOCAs	
Not important for all BWR 5/6 plants	High redundancy and diversity in coolant injection systems Inclusion/exclusion of IORV in LOCA category	LOCA CDFs range from negligible to 1E-6/ry. Average CDF is 4E-7/ry. Contribution to total plant CDF ranges from negligible to less than 5%.
	ISLOCAs	
Not important for BWR 5/6 plants	Compartmentalization and separation of equipment Harsh environments induced by ISLOCA	ISLOCA CDFs range from negligible to 3E-8/ry. Average CDF is 3E-8/ry. Contribution to total plant CDF is negligible for all plants.

Table 11.11 Summary of BWR 5/6 plant group CDF perspectives.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
Internal flood accidents		
Significant for a few BWR 5/6 plants	Plant layout: separation of mitigating system components and compartmentalization	Internal flood CDFs range from 2E-8/ry to 3E-6/ry. Average CDF is 2E-6/ry. Contribution to total plant CDF ranges from negligible to 15%.

11.2.3.2 SBO Accident Sequences

An SBO is defined by all licensees in this group as an LOSP coupled with a loss of both Divisions 1 and 2 of emergency on-site power. The failure of these AC power sources results in loss of multiple mitigating systems, leaving only the HPCS system and AC-independent systems such as the steam-driven RCIC and diesel-driven firewater systems 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. Note that two licensees, Grand Gulf and Perry, model cross-tieing the HPCS diesel generator to either the Division 1 or 2 emergency buses. This cross-tieing arrangement allows for powering an RHR train to provide both coolant injection and DHR and helps reduce the SBO contributions for Grand Gulf and Perry.

SBO is an important accident class for all the plants in the group and is the dominant accident class for the group as a whole. Most of the licensees identify both short-term sequences (defined as sequences resulting in core damage within 1 to 2 hours of the initiating event) and long-term sequences (core damage occurs after 2 hours) as contributors to core damage. Some variability in the contributions from both types of sequences is present. This variability is primarily because of plant design differences and modeling assumptions related to the SW system that cool the diesel generators, the RCIC system failure modes, and the use of firewater for injection. Differences in the data used to quantify the models also contribute to the variability to some extent.

SW system failures are a major contributor to failure of the diesel generators. Several of the licensees in the BWR 5/6 group identify in their submittals SW system failures that result in failure of all three EDGs. Some plants, such as Clinton and Grand Gulf, have separate SSW systems that serve only emergency loads. Other plants have SSW systems that share piping with the normal service water (NSW) system, require that nonsafety loads be isolated during accidents, or do not have a separate division dedicated to the HPCS system and its diesel generator. Some impacts from the failure of these SW systems that are identified in several of the submittals as being major contributors to SBO are discussed below.

The River Bend licensee reports a significant contribution to failure of all three electrical division diesel generators from SSW failures. The SSW system at River Bend shares piping with the NSW system, requires isolation of non-safety related loads, and does not have a separate division dedicated to the HPCS system and its diesel generator. Important SSW failures include locking the pumps out during a periodic NSW test or during transfer from one operating NSW pump to a standby pump and failure to restore a manual SSW valve after test or maintenance of the system. These additional SSW failures result in a higher short-term SBO contribution for River Bend. The NSW system at River Bend has since been modified such that the SSW pump lockout is no longer required.

Nine Mile Point 2 does not have a separate SSW train for HPCS and its diesel generator. Cooling for the HPCS diesel generator is provided by either the Division 1 or 2 SSW train. Thus, failure of both Division 1 and 2 diesel generators results in failure of the HPCS diesel generator. However, despite this HPCS dependency upon Division 1 and 2 power, the dominant SBO sequence at Nine Mile Point 2 has a frequency that is below that of an identical sequence for Clinton, which has a totally independent SSW train serving the HPCS system. Most of this discrepancy appears to be attributable to the higher initiating event frequency and LOSP non-recovery probabilities used in the Clinton submittal.

SSW pump room ventilation system failures are important contributors to SBO in the Grand Gulf submittal. Grand Gulf has a SSW system consisting of three separate trains each supplying cooling to one of the three EDGs. However, the divisional separation of the SSW system is defeated in certain circumstances by the fact that the HPCS-related SSW pump is located in the same room as the SSW pump serving Division 1 loads. The ventilation system for this SSW pump room contains two redundant trains with power provided by Division 1 and 2 buses. Failure of the ventilation fans and dampers in this room is identified in the submittal as failing both SSW pumps, but only if the Division 1 SSW pump is operating. HVAC operation is only required if the larger Division 1 SSW pump is operating and is not required if only the smaller HPCS pump is operating. Thus, failure of the Division 1 SSW pump room ventilation can result in failure of both the Division 1 and the HPCS EDGs within approximately 2.5 hours.

Different RCIC failure modes are identified as being important in the submittals. Variation in the failure modes for the RCIC system, the timing of these failure rates, and the potential recovery of some trip signals modeled in the submittals account for some differences in the reported SBO CDFs for the BWR 5/6 plants. How much of this variation is the result of plant design or operational differences versus modeling assumptions is not clear. Some of the RCIC failure modes modeled in the submittals include failure caused by a high suppression pool temperature, high turbine exhaust backpressure, steam tunnel temperature trips, loss of pump room cooling, and battery depletion which varied from 1 to 8 hours. Some of the contributions of these failure modes on SBO identified in the submittals are highlighted below.

Early RCIC failures cited in the submittals generally involve hardware failures or maintenance unavailabilities. However, both River Bend and Perry also have a significant early RCIC failure contribution from failure to override a steam-tunnel temperature trip signal. The reason this failure mode is not identified in the other submittals is unknown. However, there is some variation in the setpoints for such trips which will result in differences in the time available to bypass such a trip if proceduralized (whether all of the licensees have procedures to perform such a bypass can not be ascertained from all of the submittals). Some of the licensees, including Perry, model bypassing the steam tunnel temperature and other expected RCIC trips in their SBO evaluations. The early occurrence of a steam tunnel temperature trip signal in the River Bend and Perry submittals apparently does not allow for much, if any, recovery. A similar early failure mode of RCIC involving the loss of RCIC pump room cooling is identified in the Washington Nuclear Power Unit 2 submittal. Generally, loss of pump room cooling is treated as a longer term failure mode in the other submittals (occurring at variable times) which can be prevented by opening doors. The early occurrence of a trip signal from loss of RCIC pump room cooling in the Washington Nuclear Power Unit 2 limits the ability to provide such alternate room cooling. It should also be noted that the presence of a high-steam tunnel or pump room temperature trip signal during an SBO has no impact on the system operation if the isolation valve or logic is AC powered as it is at Grand Gulf and LaSalle.

The other modeling assumption that impacted the Perry ATWS CDF is the fact that the licensee did not credit the use of HPCS for coolant injection. The Perry ATWS model is based upon Revision 4 of the BWR Owners Group Emergency Procedure Guidelines (EPGs) which does not direct the operation of coolant injection systems that inject inside the shroud until all other injection sources have failed. The use of HPCS during an ATWS is a concern since spraying cold water into the core will introduce positive reactivity leading to a power increase. Most licensees credit the use of HPCS during an ATWS. Those licensees that credit HPCS have a significantly lower ATWS-related CDF than those plants that do not (includes Perry). Note that injection of cold LPCI water inside the shroud is also a concern for this group of plants. Grand Gulf addresses this issue in their procedures by requiring that LPCI be injected through the shutdown cooling valves into the recirculation line during an ATWS. However, the licensee credits injection through this LPCI path as well as the normal injection path which the other licensees credit. Most of the licensees also credit use of LPCS during an ATWS.

11.2.3.4 Transient Accident Sequences

This class of accidents involves transients with loss of all coolant injection systems only. Transients involving loss of DHR and subsequent loss of injection systems are included in the DHR accident class. Loss of injection sequences involve failure of the PCS, HPISs such as feedwater, HPCS, and RCIC, and either a failure to depressurize the vessel for subsequent injection with low-pressure systems or successful vessel depressurization and failure of the low-pressure systems which include LPCI, LPCS, and condensate.

Transient sequences with loss of injection are important contributors to the CDF for several plants and to the plant group as a whole. No single factor can be identified that accounts for the variability in the CDFs calculated for this accident class. However, the dominant accident sequences include a substantial contribution from loss of support systems which fail many coolant injection systems including alternate systems that are credited in some of the IPEs. The impact of particular support systems failures is dependent on the plant design and to some extent modeling assumptions concerning the need for some support systems for coolant injection system operation.

Transients with loss of HPI and failure to depressurize the vessel for LPI are important contributors to this accident group for BWR 5/6s. Because of the availability of a motor-driven pump for HPI at many of the plants, initiating events that fail feedwater and other mitigating systems (especially loss of support system initiating events) are important contributors to the transient accident class. The impact of these initiating events on mitigating systems can be either direct or through other support systems. For example, Nine Mile Point 2 has one of the highest loss-of-injection CDFs because of a loss of the Division 1 emergency AC power initiator, which results in isolation of the reactor and turbine building cooling water systems. The subsequent loss of cooling to the condenser, feedwater, and turbine generator equipment requires an immediate shutdown by the operators. HPCS and RCIC fail immediately, resulting in a loss of HPI followed by operator failure to depressurize the plant. A similar impact and contribution occurs from an LOSP initiator.

The Clinton submittal also reports one of the highest transient with loss of injection accident class CDFs for the BWR 5/6 group. Transient sequences with loss of HPI sequences are the most important sequences within this accident class for Clinton. Although Clinton has both steam-driven and motor-driven feedwater pumps, no credit is given for the continued operation of the normally operating steam-driven pumps and little credit is given for properly aligning the motor-driven pump for injection. Thus, the dominant loss of HPI sequences with failure to depressurize the vessel involves general PCS transients since such transients have relatively high frequencies. Additional contributions from LOSP and loss of normal feedwater initiating events also occur at Clinton since both affect the availability of feedwater. An LOSP initiating event is also the dominant initiating event contributing to loss of HPI accidents at Perry.

Grand Gulf and River Bend are the only licensees in the BWR 5/6 plant group that credit the use of CRD as a HPI source in the short term (i.e., immediately after feedwater, HPCI, and RCIC fail). The use of CRD in the short term is successful only if vessel depressurization does not occur since depressurization is assumed to result in CRD pump runout and subsequent failure. LOSP is the most important initiating event leading to a loss of HPI at Grand Gulf since it fails the PCS and impacts the availability of AC power to HPCS and CRD (Grand Gulf does not have motor-driven feedwater pumps). Other initiating event contributors to this accident class at Grand Gulf include a loss of PCS, loss of either Division 1 or 2 AC power, SW, either Division 1 or 2 DC power, and feedwater. Most of these initiating events result in failure of the PCS and/or some HPCI systems. The dominant accident sequence of this type at River Bend occurs as a result of the unique NSW/SSW arrangement present at the plant. The dominant sequence involves a loss of an NSW initiator that fails feedwater followed by failure of HVAC for the standby switchgear room, either directly or because of failure of SSW cooling to the chillers (common cause SSW valve failures are important). The failure of the standby switchgear room cooling and subsequent failure of the switchgear result in loss of HPCS, RCIC (through room cooling), and ADS. Credit is given for providing alternate switchgear cooling by opening room doors.

Modeling of ADS inhibit during transient events impacts the importance of loss of HPI sequences. Most of the licensees in the BWR 5/6 group assume that the operator will inhibit ADS during normal transients and therefore, must manually depressurize the vessel for LPCI. In fact, an operator error to depressurize the vessel is the dominant failure mode of the ADS for these plants. The operator error probabilities for failure to depressurize the vessel range from 3E-4 to 2E-3 and contribute to the variability in the CDFs for this accident class. However, two of the licensees (LaSalle and Washington Nuclear Power Unit 2) in this plant group do not model ADS inhibit for these types of accidents. ADS is allowed to occur on a low vessel level signal. Since the failure probability for the automatic signal is significantly less than the human error rates used by the other licensees, the contribution from loss of HPI sequences is significantly less for LaSalle and Washington Nuclear Power Unit 2.

Transients with loss of both HPI and LPI with the vessel depressurized are also important for the BWR 5/6 plant group. However, the contribution from these sequences is generally less than the transient sequences with loss of only HPI (discussed previously) since LPCI systems must also fail. The low-pressure systems credited by all the licensees include LPCI, LPCS, and condensate. Most of the licensees, with the exceptions being Clinton and Perry, also credit SSW cross-tied to inject through the RHR system. However, Perry credits the use of firewater, condensate transfer, and suppression pool cleanup systems for coolant injection. As is the case with loss of HPI sequences, support system initiating events are important for loss-of-all-injection transients since they can fail multiple injection systems.

As mentioned previously, Nine Mile Point 2 has one of the highest loss of injection CDFs for this plant group. As with the loss of HPI sequences discussed above, the dominant loss of HPCI and LPCI sequences for this plant involves a loss of the Division 1 (or 2) AC bus initiator, which disables all safety systems dependent on Division 1 (2) AC power, and also fails condensate and feedwater. Following the loss of Division 1 (2) AC power, Division 2 (1) SW pump breakers open, which causes isolation of the reactor building and turbine building cooling water systems. Loss of the reactor building and turbine building cooling water systems is assumed by the licensee to result in a low flow trip of the lone remaining division of SW; therefore, requiring restart. Failure of the remaining division of SW to restart leads to loss of all injection and core damage. The dominant failure preventing the restart of the remaining division of SW is failure of DC power as a result of battery failure on demand. Failure of the battery also fails RCIC and prevents the start of the available low-pressure ECCS. If the tripped division of SW fails to restart because of pump hardware faults, then RCIC, LPCI, and LPCS are all assumed unavailable as a result of the loss of room cooling (assuming the operator fails to open pump room doors in a timely manner). Note that for

A summary of the importance of the various accident classes to the PWR CDFs and the factors driving variability in the results is provided in Table 11.13. Considerable variability exists for each PWR group 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: (1) transients, (2) LOCAs, and (3) SBO. These three accident classes involve accident initiators and/or subsequent system failures that defeat the redundancy in systems available to mitigate potential accidents. Lesser contributions are generally reported for ATWS, steam generator tube ruptures (SGTR), ISLOCAs, and internal flooding. However, a few PWRs do report significant contributions from these accident classes, and SGTR are found to be significant contributors for the Westinghouse 2-loop plants.

Table 11.13 Summary of CDF perspectives For PWRs.

Accident importance	Important design features, operator actions, and model assumptions	Important 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 EDGs 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 LPI system from common cause failure of pumps to start, common cause failure of valves, and LPI pump cooling failures Alternate actions to mitigate LOCA (e.g., depressurizing the RCS using the steam generator atmospheric dump valves (ADVs) when HPI fails during LOCA) Size of 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 ATWS rule Install ATWS mitigating system actuation circuitry (AMSAC) Add alternate scram button

Table 11.13 Summary of CDF perspectives For PWRs.

Accident importance	Important design features, operator actions, and model assumptions	Important 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/units Dependency of other plant systems on component cooling water (CCW) and/or SW systems Dependency on HVAC and Instrument Air 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/emergency feedwater (AFW/EFW)	Improved operator training and procedural modifications to reduce frequency of RCP seal LOCA Replacing manual crossover valves with motor operated valves Staggering HPI pump use during loss of SW events Providing alternate 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)
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 ISLOCA
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
Internal flood accidents		
Important for some PWRs, regardless of plant group	Plant layout: separation of mitigating system components and compartmentalization	Changes in plant layout Enhance procedures and training to respond to floods, including isolation of the flood source

Factors that have the largest influence on the CDF contributions from the most important accident classes are discussed for each PWR plant group in Sections 11.3.1 through 11.3.5. 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 are applicable to all of the PWR plant groups. Differences that tend to reflect design differences among the PWR plant groups are summarized below.

One of the most important factors affecting PWR CDFs is the 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 the 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. However, there is at least one plant in each group that has indicated a significant CDF contribution that involves RCP seal LOCAs. This lower susceptibility to RCP seal LOCAs in the B&W and CE plants tends to cause lower contributions from transient and SBO sequences for the B&W and CE plants relative to the Westinghouse plants.

Because the reported probability of RCP seal LOCAs is generally lower in the B&W and CE plants, they tend to show more benefit than Westinghouse plants from plant characteristics that improve the reliability of heat removal through the steam generators (e.g., reliable or redundant feedwater pumps, sustained source of water for feedwater, or longer battery life for control of AFW during SBO). The importance of these factors is less 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 HPI 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 feed-and-bleed 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 configuration for ECCS recirculation. Plants with a higher degree of automation in performing the switchover and plants that can achieve high-pressure recirculation (HPR) with fewer components operating tend to have lower failure rates resulting from the switchover to recirculation. For the plants with manual switchover, variability in the assessment of operator performance in performing the action is also important. The B&W plants require manual actions for ECCS switchover from injection to recirculation, and the HPI 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. However, the LOCA contribution is not much lower for the CE plants because of higher LOCA CDFs involving injection failures. Some Westinghouse plants require operator actions to perform the switchover while other plants have automatic switchover. For some Westinghouse plants, the high-pressure pumps draw directly from the sump during recirculation, while at other plants, the high-pressure pumps must be aligned to draw suction from the low-pressure pumps (which draw from the sump). This variability in Westinghouse designs contributes to the LOCA variability within the Westinghouse plants.

11.3.1 CDF Perspectives for B&W Reactors

As indicated in Table 11.14, the B&W plant group consists of five IPE submittals. Four of these submittals are for single-unit sites, and one is for a three-unit site with the same results reported for each unit. All of the plants in this group have large dry containments.

Table 11.14 Plants in B&W plant group.

ANO 1	Crystal River 3	Davis Besse
Oconee 1,2&3	TMI 1	

11.3.1.1 Summary of Results and Perspectives for B&W Reactors

The total CDFs for the plants in the B&W group and the CDFs for the accident classes considered in this study are shown in Figure 11.6 and 11.7, respectively. The CDFs for this plant group vary by less than a factor of five, but the variability in CDFs for the accident classes that contribute to the plant CDFs is larger. The average plant CDF for the group is 3E-5/ry, which is the lowest average CDF for the PWR plant groups. The accident class CDFs vary considerably among the B&W plants, but for most plants, transients, LOCAs, and SBOs are the largest contributors to plant CDF. This is not surprising since these are accident classes where a relatively few number of systems must fail to result in core damage. Internal floods are significant contributors (greater than 10%) for only the 3-unit plant, and ATWS, SGTR and ISLOCAs are insignificant contributors (less than 5%) for all of the B&W plants.

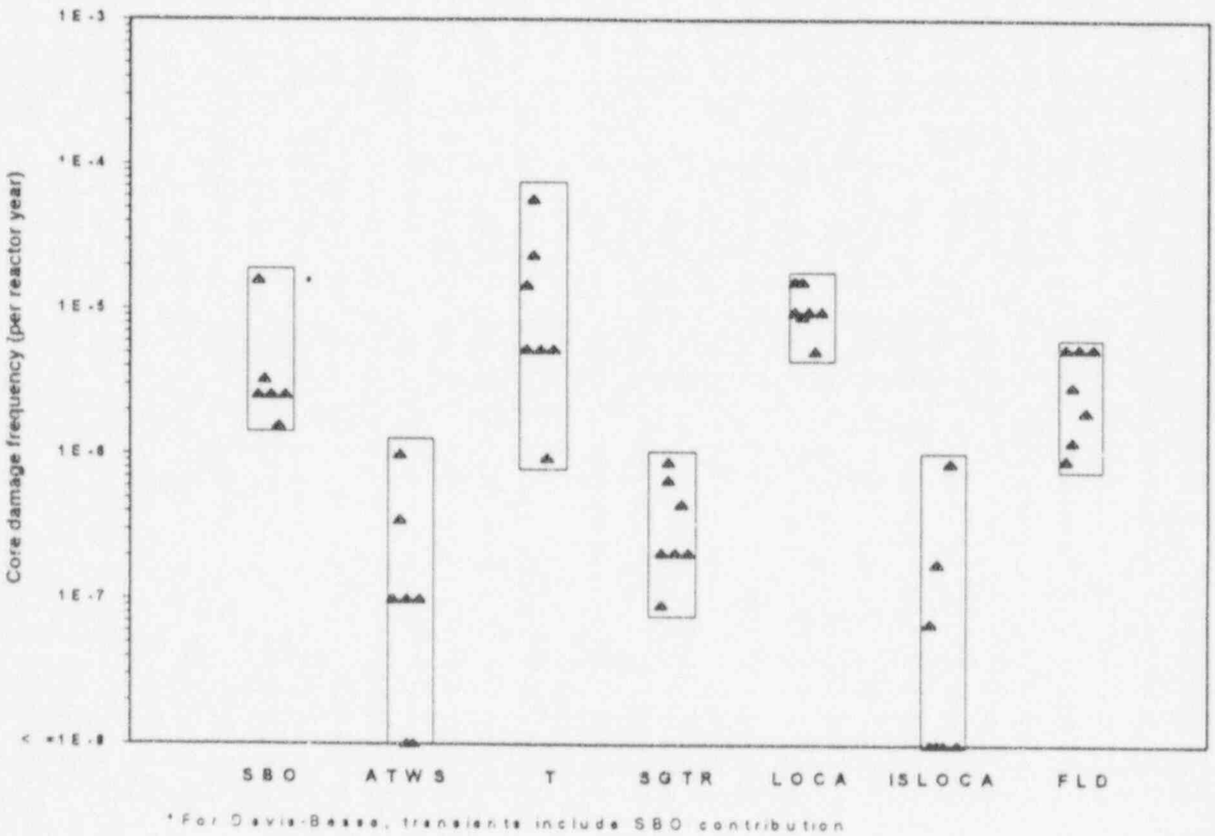


Figure 11.7 Sequence results for B&W plants.

Table 11.15 summarizes the perspectives obtained from examination of the B&W plants. The details that provide the basis for these perspectives are contained in the remainder of Section 11.3.1. The results for the accident classes are discussed, giving more details on the factors driving the CDFs, particularly for the plants with the highest and lowest CDFs. Design and operational factors along with differences in modeling assumptions are addressed.

Table 11.15 Key IPE observations for B&W plants.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
Transient accidents		
Important for nearly all of the 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	Transient CDFs range from 9E-7/ry to 6E-5/ry. Average CDF is 2E-5/ry. Contribution to total plant CDF ranges from 5% to 85%
SBO accidents		
Important for many of the B&W plants	Design of RCP seals, which affects susceptibility to RCP seal LOCAs Availability of alternate RCP seal cooling AC power reliability (number of diesel generators, cross-tie capabilities between units, diverse AC power sources) Battery life	SBO CDFs range from 2E-6/ry to 2E-5/ry. Average CDF is 5E-6/ry. Contribution to total plant CDF ranges from less than 5% to 35%
LOCAs		
Important for most of the 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 LPIS and modeling of common cause failures	LOCA CDFs range from 5E-6/ry to 2E-5/ry. Average CDF is 1E-5/ry. Contribution to total plant CDF ranges from 10% to 60%
Internal flood accidents		
Only important for one B&W site	Plant layout: separation of mitigating system components and compartmentalization	Flood CDFs range from 9E-7/ry to 6E-6/ry. Average CDF is 3E-6/ry. Contribution to total plant CDF ranges from less than 5% to 25%

Table 11.15 Key IPE observations for B&W plants.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
ATWS accidents		
Not important to CDF for any of B&W plants	Diverse scram system reduced ATWS contribution for all plants	ATWS CDFs range from negligible to 1E-6/ry. Average CDF is 2E-7/ry. Contribution to total plant CDF is less than less than 5% for all plants
ISLOCAs		
Not important for any of B&W plants	Compartmentalization and separation of equipment	ISLOCA CDFs range from negligible to 9E-7/ry. Average CDF is 1E-7/ry. Contribution to total plant CDF is less than negligible for all plants.
Steam generator tube rupture accidents		
Not dominant for any of B&W plants	Modeling of operator actions to isolate the rupture and provide long-term heat removal	SGTR CDFs range from 9E-8/ry to 9E-7/ry. Average CDF is 4E-7/ry. Contribution to total plant CDF ranges from negligible to less than 5%

11.3.1.2 SBO Perspectives

SBO occurs when a plant loses off-site AC power, and the backup AC power sources (almost always on-site diesel generators) also fail to function. Because so many safety systems rely on AC power either directly (for motive power) or indirectly (i.e., cooled by systems that rely on AC power), SBO causes most safety systems to be unavailable. Most SBO scenarios do have DC power available; however, through use of station batteries. For B&W plants, the loss of all AC power leaves turbine-driven AFW as the only available means for core cooling. Turbine-driven AFW can operate until the batteries deplete, leading to a loss of control for most plants. If AC power is not recovered soon after loss of control, core damage will follow. However, the Davis Besse submittal indicates that AFW could continue, even after battery depletion by manually controlling feedwater.

An additional complication for some B&W plants involves the potential for leakage from RCP seals. SBO results in loss of cooling for the RCP seals. The prolonged exposure to high temperatures gives the potential for seal failure, leading to a small LOCA through the pump seals. Three of the B&W sites use Byron Jackson RCPs, which are believed by industry to be less susceptible to RCP seal LOCAs than other designs. The remaining two sites use either Westinghouse or Bingham International pumps. The licensees for plants with Byron Jackson reactor coolant pumps only considered RCP seal LOCAs to be credible if the RCPs were left running when seal cooling was lost; therefore, RCP seal LOCAs were not found to be important for SBO sequences at these plants. IPEs for the remaining two sites (those with Bingham International or Westinghouse pumps) included the possibility of RCP seal LOCAs for SBO. If a seal LOCA occurs, injection systems are needed to provide makeup to the reactor coolant system. Since these are not normally available during an SBO, pump seal LOCAs will lead to core damage if off-site power is not restored in time.

SBO is an important contributor to plant CDF for about half of the B&W IPE submittals. Several factors are important for SBOs for the B&W plants, with differing combinations of these factors driving the results for the individual plants in this group. The factors that have the biggest impact on the results primarily represent actual plant characteristics, but modeling used in the IPEs is also important. The key factors are battery life (actual plant design as well as load shedding assumptions) and the availability of emergency AC power sources.

SBO sequences can be categorized as short-term, long-term, or involving RCP seal LOCA. On average, long-term SBO sequences are most frequent. To understand the reasons for differences in SBO CDFs, it is helpful to categorize SBOs as follows:

- short-term SBO - sequences in which turbine-driven AFW pump trains fail to operate and AC power is not recovered before the core is damaged
- long-term SBO - sequences in which turbine-driven AFW pump trains operate initially, but ultimately fail (normally because battery depletion causes loss of DC control power) before recovery of AC power
- SBO with RCP seal LOCA - sequences involving RCP seal LOCAs that are caused by loss of cooling to RCP seals with failure to recover AC power before the core is uncovered.

Short-term SBO sequences typically lead to core damage within about 1 hour for B&W plants, while long-term SBO sequences typically lead to core damage in more than 1 hour. SBO sequences with RCP seal LOCAs are typically long-term sequences.

The relative fractions of the SBO accident class that fall within these categories is not readily available from the IPE submittals for most plants. However, most submittals qualitatively indicate that short-term SBO sequences are not major contributors to the SBO CDF, because more failures are needed for the short-term case (i.e., turbine-driven AFW must fail). Since most of the B&W plants use Byron Jackson pumps with a lower susceptibility to RCP seal LOCAs, SBO sequences with RCP seal LOCAs are negligible contributors for most of the plants. The only plant with a significant relative contribution from RCP seal LOCAs for SBO is TMI 1, but TMI 1 has a low SBO CDF, so the absolute contribution from RCP seal LOCAs is small.

The SBO results are driven by a combination of factors involving both plant-specific features and IPE modeling characteristics. The biggest contributors to the variability are differences in battery life and the availability of emergency AC power sources. Some of the system and component failures that are dominant contributors to the SBO CDF are common across the plants, while others are highly plant specific. The failures that contributed most to SBO for the B&W plants can be grouped into the categories listed below:

- LOSP
- loss of backup AC power
- failure to recover AC power
- failure of turbine-driven AFW
- transient-induced LOCA

The system and component failures that contribute to each of the categories are discussed in the following paragraphs.

The initiator for SBO is, of course, loss of off-site AC power. Higher frequencies for loss of off-site AC power will lead to higher SBO CDFs. The frequencies used for each of the B&W plants are listed in Table 11.16. As can be seen, the frequencies vary by a factor of about 4, and the initiator frequencies do not correlate with the SBO CDFs (i.e., low initiator frequencies do not generally give low SBO CDFs). Thus, variability in this factor does not account for much of the variability in reported results.

Table 11.16 Key parameters affecting SBO CDF for B&W plants.

Plant	SBO CDF (1/ry)	No. EDGs Unit/SBO/x-tie[1]	Battery Life (hrs)	LOSP Initiator Frequency	Pump Seals[3]	No. TDAFW pumps[6]	Other Features
ANO 1	2E-5	2/1/N	2	.036	BJ	1	- SBO diesel generator available - Relatively high value for failure to restore off-site power
Crystal River 3	3E-6	2/0/N	4	.035[4]	BJ	1	
Davis Besse	~1E-5[2]	2/1/N	2	.035	BJ	2	- SBO diesel generator available
Oconee 1,2&3	3E-6	0/1/Y[5]	Un-known	.09	BI/W	1	- Safe shutdown facility diesel generator can power feedwater - Can cross-tie steam-driven AFW among units
TMI 1	2E-6	2/0/Y	6	.057	W	1	- Air-cooled diesel generators - Relatively low value for failure to restore off-site power

[1] The number of diesel generators dedicated to each unit, SBO or safe shutdown facility diesel generators, and cross connects between units (Y for cross connects, N for no cross connects) are noted.
[2] Estimated by examining results of importance studies.
[3] BJ = Byron Jackson pumps, BI = Bingham International pumps, W = Westinghouse pumps
[4] Does not include contribution from LOSP transformer," with frequency of 0.4/ry because it is readily recoverable
[5] Instead of diesel generators, emergency power is provided by connections to hydroelectric plant
[6] Turbine-driven AFW pumps.

Backup AC power is provided by two diesel generators at all of the B&W sites except Oconee 1, 2, 3. Oconee has a unique design with backup AC power from hydroelectric and combustion turbine generator plants. Failure of backup AC power is part of the SBO definition, and since diesel generators are used as the backup AC power source for nea y all of the B&W plants, diesel generator failures are dominant contributors to SBO CDF for most of the B&W plants. The most common reasons for diesel generator failures involve the failure of the diesel generator to start (multiple random failures or common cause failure), failures of the diesel generators to continue to run for a sufficient time interval after successfully starting, or failure of diesel generator cooling.

Alternate sources of emergency AC power are available at some of the plants. TMI 1 has the ability to cross tie emergency AC power to the Unit 2 diesel generator. Oconee has a diesel generator available from the safe shutdown facility that can provide feedwater to the steam generators. ANO 1 and Davis Besse have SBO diesel generators. These supplemental AC power sources lower the SBO contributions by providing redundant, independent power sources.

Although the probability of restoring off-site power is important to the SBO CDF and would, therefore, be desirable to compare among the plants, the necessary information is not readily available from most of the IPE submittals in a consistent form that could be compared. Table 11.16 does indicate the plants that generally used the highest and lowest values for the failure to recover off-site power.

Early failures of AFW lead to short-term SBO sequences while delayed failures lead to long-term SBO sequences. Early failures normally involve either pump failure or steam generator overfill. These failures have sufficiently low frequency to cause most plants to have a larger contribution to SBO CDF from long-term SBO sequences rather than these short-term sequences.

Late AFW failures are most commonly caused by the inability to control AFW when DC power is lost as a result of battery depletion. Thus, plants with a longer battery life have more time available to restore off-site power before core cooling is lost, and thus, have lower threat from SBO. Table 11.16 lists the battery life for the B&W plants. One licensee, Davis Besse, indicates in its IPE that AFW can be manually controlled after battery depletion, so that the battery life is not critical to successful plant shutdown.

The plants with the lowest and highest SBO CDFs are TMI 1 and ANO 1, respectively. Differences in SBO CDFs between the two plants are primarily attributable to differences in plant characteristics. TMI 1 has a longer battery life than ANO 1 (e.g., 6 hours vs. 2 hours), which gives more time for power recovery before core damage. In addition, the TMI 1 backup AC power is more reliable because the diesel generators are air-cooled and because an extra diesel generator is available through the cross-tie to TMI 2.

11.3.1.3 Transient Perspectives

Transient sequences involve events that cause the reactor to trip (initiators) followed by failure to bring the reactor to safe shutdown (excluding SBO sequences, which are treated separately). The transient accident class is a broad category, covering both general initiators (such as reactor trip or loss of main feedwater) as well as support-system initiators (e.g., loss of SW or AC/DC bus). After reactor trip, decay heat must be removed from the RCS. Normally, this would be provided by steam generator heat removal, with a fallback of primary system feed-and-bleed should secondary heat removal fail. If neither of these succeeds, the RCS inventory will boil off, leading to core damage. A second possible path to core damage for transients involves an induced LOCA during the transient (SORV or RCP seal leak) with failure to make up the RCS inventory. For both types of sequences (failure of DHR and transient-induced LOCA with failure of reactor coolant makeup), long-term operability must also be maintained (i.e., switchover from injection to recirculation must succeed), or core damage will result.

Transients are the largest contributor on average to plant CDF for the B&W plants, but there is considerable variability in individual plant results, with the range of transient CDFs spanning about one and a half orders of magnitude. There is wide variability as to the specific failures that lead to core damage for transients, but there is a significant contribution from support system failures (either as initiators or as failures subsequent to some other initiator). The results indicate that plant-specific dependencies are important to the results, but also, different IPE modeling affects the results. The results are mostly heavily driven by differences in HPI/charging design, RCP seal and seal cooling designs, and plant-specific configurations for system dependencies and cross-ties.

Transients cover a broad group of sequences, including both general and plant-specific initiators, but support system failures are generally important for transients. Some initiators occur from loss of support systems such as SW, CCW, HVAC, instrument air, or AC/DC buses. Because these support systems are needed for a large number of front line systems, their unavailability can simultaneously fail numerous front line systems, leaving few options for successful plant shutdown. The dependencies are very plant-specific. The remaining initiators are more general, such as reactor trip, turbine trip or loss of main feedwater. The general initiators typically have higher frequencies than the support system initiators, but more failures are needed (subsequent to the initiator) to result in core damage. LOSP is an exception in that it is usually a lower frequency event, but often has a higher potential to proceed to core damage because failures of one diesel generator often lead to unavailability of full trains of safety systems.

Table 11.17 lists the key contributors to transients for the B&W plants, and indicates that support system failures (particularly power related failures) are important for the B&W plants. Loss of AC or DC buses are dominant contributors to three of the sites, and LOSP or common cause failure of batteries are dominant contributors at the other two sites. SW failures are also important for two of the sites.

Table 11.17 Dominant contributors to B&W transients.

Plant name	Dominant contributors to transients	Plant/modeling characteristics
ANO 1	- Loss of AC bus leading to steam generator overfill, failure of HPI - Loss of SW and operator failure to trip RCPs which leads to small LOCA with HPI failed	- Condensate provides alternate cooling when EFW fails
Crystal River 3	- Loss of AC bus	- Relatively low equipment failure rates - Borated water storage tank (BWST) lasts for 12 hours for transients
Davis Besse	- Failure of all feedwater and operator failure to initiate feed-and-bleed cooling - Common cause failure of batteries	- Separate HPI and charging pumps - Operator action needed to initiate feed-and-bleed cooling
Oconee 1,2&3	- Loss of instrument air with failure of HPI in recirculation - LOSP	- Westinghouse and Bingham International pumps - Feedwater cross-connects among units - Safe shutdown facility can provide alternate steam generator makeup and RCP cooling - Relatively low setpoint for containment sprays which causes earlier switchover from BWST to sump
TMI 1	- Failure of HPI - Loss of SW - Loss of DC	- Westinghouse pumps with new o-rings - SBO diesel generator

For transient sequences with failure of heat removal through the steam generators and failure of primary feed-and-bleed, the dominant contributors vary widely, representing considerable variability in plant design and operation as well as variability in IPE modeling. By examining the results reported in the IPE submittals, it was found that there is wide variability as to the specific failures that lead to core damage for transients with failure to remove decay heat in the B&W plants. The results indicate that plant-specific dependencies are important to the results, but also different IPE modeling affects the results. Some of the observed contributors to failures of steam generator heat removal and failure of primary feed-and-bleed are discussed in the following paragraphs.

The dominant contributors for failure of steam generator heat removal that are listed in the IPE submittals include loss of motor-driven feedwater because of failures of AC buses or instrument air, common cause or random failures of AFW pumps, and loss of turbine-driven feedwater because of steam generator overfill.

The most common failures listed in the IPE submittals for failure to provide feed-and-bleed cooling if steam generator heat removal is lost are failure of injection pumps resulting from the same types of failures as listed above for steam generator heat removal, or operator failure to successfully perform the operation. The failures primarily occurred in the injection phase, rather than in recirculation.

For transient sequences with RCP seal LOCAs, the dominant contributors also widely vary, representing considerable variability in plant design and operation as well as variability in IPE modeling. The IPE submittals indicate a wide variability as to the specific failures that lead to core damage for transients with induced LOCAs (primarily RCP seal LOCAs) for the B&W plants. Most of the B&W plants use Byron Jackson pumps, which the industry considers less susceptible to RCP seal LOCAs than Westinghouse pumps. The only sequences that are considered to lead to RCP seal LOCAs in these plants involve loss of seal cooling coupled with operator failure to trip the pumps. However, this conclusion regarding RCP seal LOCAs is on the basis of a limited database, and so uncertainty remains regarding the likelihood of RCP seal LOCAs for these plants. Oconee 1,2&3 and TMI 1 use either Westinghouse or Bingham International pumps. The IPEs for these plants include the possibility of RCP seal LOCAs for sequences with loss of seal cooling, even if the pumps are tripped. Because of the variability in RCP designs, plant-specific dependencies are important to the results. Some of the observed contributors to RCP seal leakage and the subsequent failure to provide makeup are discussed in the following paragraphs.

The RCPs for the B&W plants can be cooled by either CCW or by the charging pumps. The charging pumps are cooled by either SW or CCW, with some plants having backup cooling sources. This variability does not have as much impact on the B&W transient CDFs as for the Westinghouse plants because of the reduced susceptibility to RCP seal LOCAs in the B&W IPEs. The dominant contributors to sequences involving RCP seal LOCAs include support systems failures (i.e., SW, CCW, instrument air, or DC power) and operator failure to trip the RCPs.

The results for the plants with the highest and lowest CDFs for transients are driven by both differences in actual plant characteristics and IPE modeling. Davis Besse has the highest transient CDF for this plant group and Crystal River 3 has the lowest. The higher transient CDF at Davis Besse is primarily attributable to the charging/HPI design. Most of the B&W plants have charging pumps that also provide HPI; therefore, a large flow of emergency core cooling can be provided even at RCS pressures up to the SRV setpoint. For these B&W plants, operator action to open the PORV is not necessary for feed-and-bleed cooling to succeed. Davis Besse has separate charging and HPI pumps with less capacity than the other plants. Therefore, in the Davis Besse IPE, opening of the PORVs is considered necessary for success of feed-and-bleed cooling. This increases the transient CDF for Davis Besse relative to the other B&W plants.

The low transient CDF from Crystal River 3 primarily reflects the use of relatively low failure frequencies in the IPE. The low failure frequencies are partially attributable to the use of more recent plant data (relative to the other IPEs) in Crystal River 3.

11.3.1.4 LOCA Perspectives

LOCA core damage sequences encompass any breaks in the reactor coolant system except SGTR. Normally, pressurizer relief valves that are stuck open or RCP seal leaks that initiate an accident are categorized as LOCAs in the IPEs. If either of these occur following another initiator (e.g., LOSP), they are normally categorized as an SBO or transient, as appropriate. After a LOCA is initiated, inventory makeup is needed to prevent the core from uncovering and proceeding to core damage. In addition, it is necessary to remove the decay heat from the reactor coolant system. Larger breaks can remove the decay heat through the break so that only inventory makeup is of concern. Smaller breaks exhaust less energy, and so supplemental cooling is needed. This is normally provided by steam generator cooling or by primary system feed-and-bleed. Injection to the reactor coolant system is initially from a source outside containment (i.e., RWST), but unless a refill is available, this source is ultimately depleted. Thereafter, the plant must switch from the injection mode to the recirculation mode, where water is drawn from the containment sump. Unless the injection mode is successful, however, there will not be sufficient water in the sump to provide reactor coolant system makeup during recirculation.

LOCAs are a significant contributor to plant CDF for most of the B&W plants, and the LOCA CDFs for the B&W plants do not vary much. The LOCAs are most often small LOCAs, and recirculation failures contribute more to CDF than do injection failures. Although the variability is not large, both actual plant characteristics and IPE modeling affect the variability in results for this plant group. The factors that have the greatest impact on the variation in LOCA CDFs among the plants are the size of the RWST and whether containment sprays will be actuated during small LOCAs and transients.

The most common type of LOCA is a small break LOCA with failure of recirculation, but two plants have large relative contributions from medium LOCAs. For this report, the following sizes of LOCAs are considered:

- Large - large enough to remove decay heat through the break, and also large enough to depressurize the system on its own, which allows injection from low-pressure systems (typically, greater than 4 inch diameter)

- Medium - the break is large enough to remove decay heat through the break, but not large enough to depressurize the system (typically, 2 to 4 inch diameter)

- Small - the break is not large enough to depressurize the system before core damage would occur and so the reactor coolant system must be depressurized by some other means if LPI is needed. The break does not remove sufficient energy to cool the RCS, so some other means is required to provide the energy removal (typically, 1/2 to 2-inch in diameter).

One IPE submittal, TMI 1, also reports results for very small LOCAs, but these very small LOCAs were collapsed into the more general categories listed above to provide a consistent basis for comparison of the IPE results.

The CDF contributions from small, medium, and large LOCAs are listed in Table 11.18 for the B&W plants. Small LOCAs are the dominant break size for most of the IPEs, with medium LOCAs relatively high at two of the plants. Recirculation failures are dominant for all of the B&W IPEs.

Table 11.18 LOCA contributors for B&W plants.[1,2]

Plant	CDF from large LOCAs (1/ry)[3]	CDF from medium LOCAs (1/ry)	CDF from small LOCAs (1/ry)
ANO 1	85E-7	--	2E-5
Crystal River 3	1E-7	2E-6	7E-6
Davis Besse	2E-6	2E-6	2E-6
Oconee 1,2&3	3E-6	7E-6	4E-7
TMI 1	3E-6	2E-6	8E-7

[1] Not including SGTR and ISLOCAs.
[2] Total LOCA CDF = large LOCA CDF + medium LOCA CDF + small LOCA CDF.
[3] Includes Vessel Rupture

The most common system and component failures for LOCAs are related to the switchover to recirculation. Some of the system and component failures that are dominant contributors to the LOCA CDF are common across the plants, while others are highly plant-specific. The failures that contribute most to LOCAs for the B&W plants can be grouped into the categories listed below. The system and component failures that contribute to each are discussed in the following paragraphs.

- LOCA initiator
- failure of injection
- failure of recirculation
- failures of alternate actions that could be used to mitigate the above failures

The initiator frequencies used in the IPEs for the small, medium and large break sizes for the B&W plants are listed in Table 11.19. The plants used generic data to quantify the LOCA initiator frequencies. TMI 1 subdivided small LOCAs into two categories, and the small LOCA initiator frequency listed in the table represents the sum of those individual initiator frequencies.

Table 11.19 Characteristics important for LOCAs for B&W plants.

Plant name	LOCA initiator frequency			Failure of recirculation switchover (small LOCA)	HPR draws suction from LPR[1]	Comments
	Large	Med.	Small			
ANO 1	1E-4	--	5E-3	3E-5	yes	Relatively High Spray Initiation Setpoint
Crystal River 3	5E-5	5E-4	2E-3	1E-3	yes	
Davis Besse	1E-4	3E-4	4E-3	5E-4/4E-3[2]	yes	Includes Strategy Using Steam Generator Depressurization and RHR. Can pull HPI directly from sump in recirculation
Oconee 1,2&3	7E-4	7E-4	4E-3	1E-2	yes	Relatively Low Spray Initiation Setpoint
TMI 1	1E-4	4E-4	6E-3	1E-4	yes	

[1] HPR (high-pressure recirculation), LPR (low-pressure recirculation)
[2] Lower value is for steam generator cooling available; higher value is without cooling

LOCAs with failure in recirculation are a larger contributor than LOCAs with injection failure for the B&W plants. This is reasonable because the use of recirculation is more complicated than injection and involves more components. The operators have to realign the systems such that the low-pressure pumps draw suction from the sump and align the high-pressure pumps to take suction from the low-pressure pumps. The dominant contributor found in the IPE submittals for recirculation failures is the operator's failure to successfully perform the switchover. Also important for some IPEs, are LPI equipment failures such as common cause or random failures of pumps or valves.

The setpoint for spray injection also affects the LOCA CDF for the B&W plants. Plants with spray actuation at lower containment pressures draw water from the RWST more rapidly. This shortens the time available for the switchover from the RWST to the containment sump, which decreases the probability of the operators successfully performing the manual switchover. For example, Crystal River 3 has a 30 psig set point for spray actuation while Oconee has a 10 psig set point. The higher spray injection set point preserves RWST inventory for emergency core cooling injection rather than containment spray injection for Crystal River 3. Similarly, some plants have larger

RWSTs so that the switchover to recirculation is delayed, which gives the operators more time to complete the necessary actions for the switchover to recirculation.

The plants with the highest and lowest LOCA CDFs have important design differences that account for the relatively small difference in results, but IPE modeling differences are also important. The Davis Besse IPE has the lowest CDF from LOCAs and the ANO 1 and TMI 1 IPEs have the highest. The lower LOCA contribution for Davis Besse is primarily attributable to modeling in the IPEs of alternate strategies for dealing with LOCAs. The Davis Besse IPE includes the possibility of depressurizing the steam generator secondaries, and thereby, depressurizing the reactor coolant system so that low-pressure systems can be used during a small LOCA. In addition, the high-pressure pumps can provide coolant to the reactor coolant system directly from the sumps if the PORVs are opened (avoiding the need for low-pressure pumps to operate). The higher LOCA contribution for ANO 1 reflects the coarse modeling of LOCAs in the ANO 1 IPE. The success criteria and timing for small LOCAs corresponds to the most limiting (largest) small LOCA, so the coarser grouping for ANO 1 results in a more pessimistic treatment of the small LOCA category. The TMI 1 LOCA CDF had a larger contribution from recirculation line failures than for the other B&W IPEs. It is unclear whether this is because of an actual plant design difference or IPE modeling.

11.3.1.5 ATWS Perspectives

An ATWS is an accident that is initiated by an event, such as loss of feedwater or turbine trip, followed by failure of the reactor to scram. The failure to scram can be caused by either failures of electrical components in the RPS or in the final control elements that de-energize the CRD mechanisms, or by mechanical failures involving failure of de-energized control rods to drop into the core. Partial scram failures are possible, involving insertion of a subset of the control rods; however, in most PRAs it is assumed that the scram failure is total and involves all of the rods. With failure of the rods to insert, power greatly in excess of decay heat loads, is still being generated and overpressurization of the RCS is possible in some cases.

ATWS is not a major contributor to the total CDF for any of the plants in this group. ATWS frequencies for this group range from negligible (below the reporting criteria) to 1E-6/ry and contribute less than 5% to the total CDF. All of the plants have installed a diverse scram system to comply with the ATWS rule, which increases the scram probability and decreases the frequency of ATWS sequences. The minor contribution from ATWS for the B&W plants primarily results from mechanical failures that prevent scram. Because the contribution to plant CDF is small, ATWS sequences will not be discussed further for B&W plants.

11.3.1.6 Internal Flood Perspectives

An internal flood sequence involves a release of water into a plant location such that a plant trip is induced and safety systems are compromised at the same time. For example, if a flood causes water to enter an area containing electrical switchgear, a plant trip can occur along with failure of all plant systems dependent upon that switchgear. Systems that are considered to be independent may all fail if they all contain equipment within the flooded location. The effects of internal flooding are highly plant-specific, depending on the layout of equipment within the plant and the relative isolation of rooms. Often, the systems most affected are support systems such as electric power and service water, which have plant-specific designs. Because of this diversity of design and layout, we expect that each plant will have different vulnerabilities to flooding and do not expect to draw many generic conclusions.

The highest CDF from internal flooding for the group is 6E-6/ry for Oconee 1,2&3 and the lowest CDF is 9E-7/ry for ANO 1. The average CDF from the reported values is 3E-6/ry. Internal floods contribute 25% to the plant CDF for Oconee 1,2&3. Turbine building floods resulting from failures in the SW system are the dominant contributor. The flood submerges main and EFW pumps because the drains are not large enough to prevent water buildup.

ANO 1 represents the lower end of the CDF range, with an internal flooding contribution of 9E-7/ry. The low contribution is attributed (in the IPE submittal) to widely separated equipment in the turbine building, with equipment that is important to preventing core damage located at the upper turbine building elevations where it is not susceptible to flooding. However, it is unclear how breaks in the SW system were treated, and this was found to be an important contributor at other plants.

Internal flooding is only an important contributor to plant CDF at one of the B&W plants, Oconee 1,2&3. The remaining B&W plants have internal flood contributions of less than 10%. The importance of internal flooding to plant CDF is highly plant specific, depending on the layout of water bearing lines and vessels, flood propagation, and the locations of equipment needed to provide shutdown cooling. Typically, important internal floods are those that affect important support systems such as SW and electric power. Newer plants sometimes reflect better design characteristics, generally better separation and tend to be less susceptible tc floods.

11.3.1.7 SGTR Perspectives

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. There are several actions that are normally taken to prevent the tube rupture initiator from developing into a core melt sequence. To mitigate the leak, the affected steam generator is normally isolated, and if this does not succeed, actions are taken to depressurize the primary system so that leakage is minimized. The primary system is depressurized either by aggressively cooling down through the unaffected steam generator, cooling down the primary system using the pressurizer sprays, or by depressurizing by initiating feed-and-bleed through the pressurizer PORVs. If the affected steam generator is isolated, long-term core heat removal can be established without inventory makeup, but if the leak is not isolated, inventory makeup must also be provided.

SGTR sequences are low contributors to plant CDF, but can be significant to risk because the releases bypass containment. The SGTR contribution to plant CDF is low because the frequency of the SGTR is relatively low (generally about 1E-2) and because there are more strategies available for coping with the rupture than for most LOCAs. Core damage can be prevented by isolating the ruptured steam generator and depressurizing the primary system so that the ruptured steam generator is not pressurized. In this case, RCS heat removal is necessary but RCS inventory makeup is not. If the steam generator is not isolated, core damage can still be prevented by providing a sustained injection source. In some of the IPE submittals, it is indicated that plant procedures direct the operators to depressurize the RCS to minimize leakage and provide more time for isolating the steam generator, or for refilling the BWST so that an injection source can be maintained. These actions further reduce the CDF from SGTRs.

The variability in CDFs from SGTR (about an order of magnitude) is not unreasonable because this accident class has a larger uncertainty than most other accident classes. This uncertainty results because there is considerable influence from operator actions, whose quantification is more uncertain than most equipment failures. Additionally, the sequence quantification is affected by the quantification of valve failures when passing water, which is also relatively uncertain. The B&W plant with the highest SGTR CDF is TMI 1. The largest contributors to the CDF from SGTRs for TMI 1 involve failure of HPI and operator failure to cooldown and isolate the ruptured steam generator. The plant with the lowest SGTR CDF is ANO 1. ANO 1 is dominated by SGTRs with failure to isolate the ruptured steam generator and failure to achieve cold shutdown before the BWST empties. All of the B&W IPE submittals reflect the same conclusion that SGTR is not a dominant contributor to plant CDF.

11.3.1.8 ISLOCA Perspectives

An ISLOCA occurs when valves that normally isolate the reactor coolant system from low-pressure systems (e.g., LPI) fail, resulting in backflow from the reactor coolant system through the low-pressure piping. If the low-pressure piping or other components (e.g., seals, relief valves, or flanges) can not withstand the resulting pressurization, then a LOCA results. If the breach occurs in a portion of the piping that is outside containment, then a LOCA that bypasses containment results, with effluent from the LOCA being discharged into the reactor or auxiliary building. Because it is not possible to recirculate the coolant through the reactor coolant system for this type of LOCA, coolant injection will eventually be lost (leading to core damage) unless some source of sustained makeup is available. These scenarios are typically low probability events, because they involve multiple valve failures (typically two check valves and one motor-operated valve in series). However, ISLOCAs can be important to risk because the containment bypass leads to larger fission product releases.

ISLOCAs are minor contributors to plant CDF in all of the IPE submittals but in some cases, there is a large enough contribution to be risk significant. Although there are important differences in plant characteristics that affect the results, the variability in results appears to be predominantly influenced by differences in modeling among the plants.

11.3.2 CDF Perspectives for CE Reactors

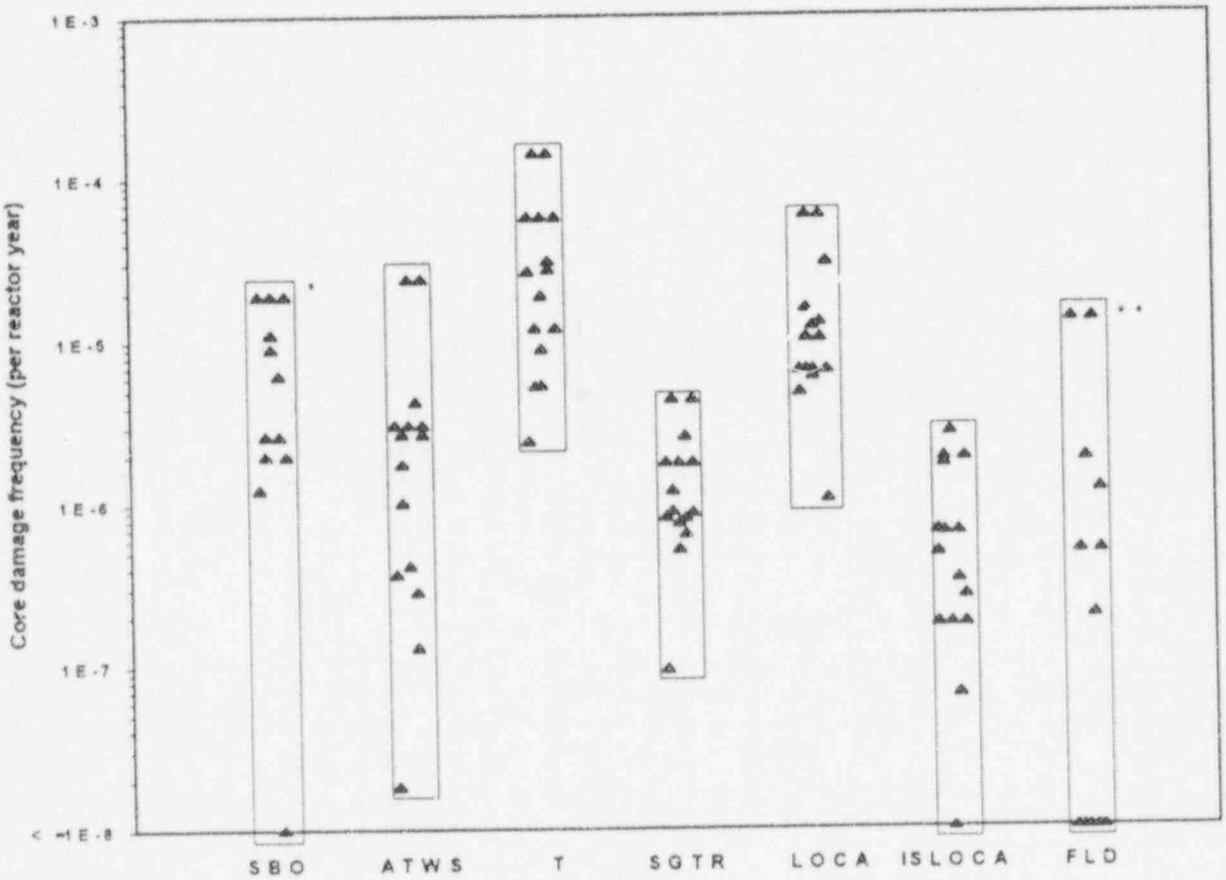
As indicated in Table 11.20, 15 plant units (10 IPE submittals) make up the CE plant group. Three of the submittals cover dual units (Calvert Cliffs, St. Lucie, San Onofre), one submittal covers three units (Palo Verde 1,2&3), and two submittals cover a single unit at a multi-unit site (ANO 2, Millstone 2). All of the plants have large dry containment designs and all utilize a two-loop reactor coolant system design except for Maine Yankee, which has three loops as well as unique reactor coolant loop isolation valves.

Table 11.20 Plants in the CE plant group.

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

11.3.2.1 Summary of Results and Perspectives for CE Reactors

The total CDFs for the plants in the CE plant group are shown in Figure 11.6, and include both the internal event contribution and the internal flooding contribution to the total plant CDF. All of the total plant CDFs for this group lie within the E-5/ry decade (1E-5/ry to 9E-5/ry) except for Calvert Cliffs 1&2 with total plant CDFs of 2E-4/ry for each unit. The average CDF for the group is 7 E-5/ry. Figure 11.8 shows the CDFs for each of the accident classes considered in this study where sufficient information exists in the submittals or in subsequent responses to staff questions to readily determine the accident class CDF contributions. Nearly all the CE plants identified the transient accident class as the most important contributor to total plant CDF. This accident class involves a transient condition (e.g., a turbine trip, loss of feedwater, reactor scram, etc.) and failure of either DHR or failure to replace reactor coolant inventory following a transient-induced LOCA. Many of the plants identified the LOCA class of accidents as an important contributor to CDF. This class of accidents involves LOCAs followed by failure to replace reactor coolant inventory either during the injection or recirculation phase of the accident. Some of the CE plants found SBO accidents to be important. This accident class is a special type of transient scenario in that it involves the total loss of all AC power and subsequent loss of either DHR or failure to replace reactor coolant inventory following an induced-LOCA condition (such as a RCP seal LOCA or a stuck-open primary system relief valve). Most all the CE plants found the other accident classes including ATWS, SGTR, ISLOCA, and internal flooding to be unimportant contributors to the total plant CDF. It should be noted however, that the SGTR and ISLOCA events can still be important to off-site consequences because these scenarios provide containment bypass pathways for radionuclide releases should the core be damaged.



*Fort Calhoun and Calvert Cliffs transients include SBO. **Maine Yankee deferred internal flood to IPEEE.

Figure 11.8 Sequence results for CE plants.

Table 11.21 summarizes the perspectives obtained from examination of the CE plants. The details that provide the basis for these perspectives are contained in the remainder of Section 11.3.2. The results for the accident classes are discussed, giving more details on the factors driving the CDFs, particularly for the plants with the highest and lowest CDFs. Design and operational factors along with differences in modeling assumptions are addressed.

Table 11.21 Key IPE observations for CE plants.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
	Transients accidents	
Important for nearly all of the CE plants	Degree of HPI and AFW dependence on support systems (e.g., SW, CCW, AC/DC bus, HVAC) Success criteria for feed-and-bleed (PORV capacity and redundancy, credit for alternate 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 (e.g., water supply size, alternate supplies, procedures, operator training)	Transient CDFs range from 2E-6/ry to 1E-4/ry. Average CDF is 4E-5/ry. Contribution to total plant CDF ranges from 15% to 80%
	SBO accidents	
Important for many of the 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	SBO CDFs range from negligible to 2E-5/ry. Average CDF is 8E-6/ry. Contribution to total plant CDF ranges from negligible to 35%
	LOCAs	
Important for many of the CE plants	Automatic switchover to ECCS recirculation LOCA initiating event frequency	LOCA CDFs range from 1E-6/ry to 6E-5/ry. Average CDF is 2E-5/ry. Contribution to total plant CDF ranges from 5% to 50%
	Internal flood accidents	
Important for two CE sites	Plant layout: separation of mitigating system components and compartmentalization	Flood CDFs range from negligible to 1E-5/ry. Average CDF is 2E-6/ry. Contribution to total plant CDF ranges from negligible to 15%

Table 11.21 Key IPE observations for CE plants.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
ATWS accidents		
Only important for one of the CE plants	Assessment of fraction of time that plant has unfavorable moderator .emperature coefficient	ATWS CDFs range from 2E-8/ry to 2E-5/ry. Average CDF is 5E-6/ry. Contribution to total plant CDF ranges from negligible to 10%
Steam generator tube rupture accidents		
Not dominant for any of CE plants	Modeling of operator actions to isolate the rupture and provide long-term heat removal	SGTR CDFs range from 9E-8/ry to 4E-6/ry. Average CDF is 2E-6/ry. Contribution to total plant CDF ranges from negligible to 5%
ISLOCAs		
Not important for any of CE plants	Compartmentalization and separation of equipment	ISLOCA CDFs range from negligible to 3E-6/ry. Average CDF is 8E-7/ry. Contribution to total plant CDF ranges from negligible to 10%

11.3.2.2 Transient Accident Sequences

Transient accident sequences involve events that cause the reactor to trip followed by failure to bring the reactor to safe shutdown, either by failure to remove decay heat or failure to replace reactor coolant inventory following an accident-induced LOCA such as a stuck-open primary system relief valve. Based on the accident class definitions used in this report, SBO and ATWS are excluded from this transient accident class. SBO and ATWS accidents are covered later as separate types of accident sequences. Transient accidents cover a broad range of initiators including both general initiators (e.g., reactor trip, turbine trip, loss of main feedwater) as well as support system initiators (e.g., loss of service water, loss of an AC/DC bus).

Transients are the most important accident class for nearly all the CE plants. The wide variability in the absolute CDFs for the plants and the dominant types of transient sequences; however, is driven by a complex combination of many plant features and modeling characteristics. Particularly, the variations in the dependence on support systems, the degree of redundancy in both the support systems and the mitigating heat removal systems for both the injection and recirculation phases of an accident, and the generally low susceptibility to RCP seal LOCAs dictate the transient results for the CE plants.

Transients are the most important accident class for nearly all the CE plants; however the absolute CDFs for individual plants vary over a wide range.

For 12 of the 15 plants in the CE plant group, the transient accident class is the largest contributing accident class to the total plant CDFs. For the remaining three plants, the transient accident class is still rather important since the lowest this class contributes to the total plant CDF is 15% for Waterford 3. Table 11.22 summarizes the transient CDF results for the CE plants and some key characteristics associated with the plant designs and the transient analyses for each plant. In general, transients are important contributors to CDF for the CE plants because they involve relatively high initiating event frequencies coupled with system failures (many times of support systems such as HVAC, SW, electrical buses, etc.) that often defeat the redundancy in systems available to mitigate potential accidents.

Transients cover a broad range of possible sequences for the CE plants, including both general and plant-specific initiators. Some initiators occur from loss of support systems (e.g., AC/DC buses, SW or CCW, instrument air, and loss of normal off-site power). Because these support systems are needed by most of the front-line systems (e.g., main feedwater, AFW, HPI/recirculation) used to mitigate the transient, support system unavailabilities can compromise front-line system redundancy, leaving few options for successful plant shutdown. Which support system initiators are important to each plant is dependent on plant-specific design details. The remaining initiators are more generally important to all the CE plants and include reactor scrams, turbine trips, and loss of elements of the PCS (e.g., main feedwater, condensate, condenser). The general initiator frequencies for the CE plants vary from approximately 5E-1/ry for Fort Calhoun 1 to as high as about 7/yr for Palo Verde 1,2&3. The support system initiators vary between the low E-4/yr to medium E-2/yr range (excluding LOSP which is discussed later under the SBO accident class), depending on the specific initiator and plant specific design details. Both support system and general initiators contribute to the dominant transient sequences for all the CE plants.

Important transient sequences for the CE plants are primarily four types. In order of general importance, these include (1) transients with early loss of all core cooling, (2) transients with late loss of core cooling, (3) transient-induced LOCAs with loss of late recirculation cooling, and (4) transient-induced LOCAs with loss of early injection. As is illustrated in Table 11.22, the four transient sequence types listed above dominate the transient contributions to the total plant CDF. The first of these involves transients with loss of all feedwater (FW in the table) including both main feedwater and AFW/EFW, thereby, losing all secondary heat removal from the steam generators. Subsequent failure of feed-and-bleed (F&B in the table) for those plants with the capability, or for those plants without this capability, leads to early core damage in typically 1-2 hours. Some of the plants credited capability to depressurize the steam generators and use condensate as a last resort to salvage core cooling. For the plants with feed-and-bleed capability, this early core damage scenario contributes between 15% and 65% of each plant's total plant CDF; corresponding to about 15% to 85% of the total transient contribution for each plant. For the plants without feed-and-bleed capability, this early core damage scenario contributes between 15% and 65% of each plant's total plant CDF; or typically over 90% of the total transient contribution for each plant.

Transients with late loss of core cooling usually occur as one of two scenarios. The first involves a transient with loss of all feedwater, success of feed-and-bleed, but late failure of HPR cooling. The second involves a transient with early success of AFW/EFW, but failure of long-term cooling because of depletion or other failures of outside water sources needed to maintain core cooling. This late core damage type of sequence contributes as high as 40% to the total plant CDF in the case of Fort Calhoun 1 (corresponding to about 60% of the total transient contribution to core damage).

Table 11.22 Transient characteristics for CE plants.

Plant	Transient CDF (/ry) - % of total CDF	Dominant sequence types	AFW/EFW pump config. (each plant)	HPI pumps need pump cooling?	Feed & bleed? - no. of PORVs for success?	HPR - auto or manual?	SG (steam generator) depressurized & core cooling considered?	HVAC failures generally considered?	RCP seal LOCA model
ANO 2	3E-5/ry 75%	T-FW-F&B (65%) T-FW-HPR (5%) T-LOCA-HPR (5%)	1 turbine 1 motor 1 non-safety pump	No	Yes 1 Low Temp Overpress Valves Path or 1 ECCS Vent Valve Path	Auto	No	No (based on analysis)	Unlikely if RCPs tripped within 30 min.
Calvert Cliffs 1&2	1E-4/ry 60% (includes SBO, not separated in submittal)	T-FW-F&B (40%) T-LOCA-HPI (10%) T-LOCA-HPR (10%)	2 turbine 2 motor (1 from Unit 2 cross-tie)	No (in final revised analysis) but does need room cooling	Yes (in general) 2 of 2 PORVs (small PORVs)	Auto	No (small single ADV per SG)	Yes	Potential for 200+gpm leak if RCPs not tripped in 45 min.
Fort Calhoun 1	9E-6/ry 65% (includes SBO, not separated in submittal)	T-FW-F&B (10%) T-LOCA-HPI (15%) T-longterm cooling (40%)	1 turbine 1 motor 1 diesel-driven	No	Yes 1 of 2 PORVs (plant has high capacity PORVs)	Auto	No	Yes	Unlikely if RCPs tripped within 30 min.

Table 11.22 Transient characteristics for CE plants.

Plant	Transient CDF (/ry) / % of total CDF	Dominant sequence types	AFW/EFW pump config. (each plant)	HPI pumps need pump cooling?	Feed & bleed? / no. of PORVs for success?	HPR - auto or manual?	SG (steam generator) depressurized & core cooling considered?	HVAC failures generally considered?	RCP seal LOCA model
St. Lucie 1	5E-6/ry 25%	T-FW-F&B (15%) T-LOCA-HPR (5%)	1 turbine 2 motor	Yes	Yes 2 of 2 PORVs	Auto	Yes	Yes (no HVAC initiators because of long recovery time)	Unlikely if RCPs tripped within 10 min.
St. Lucie 2	5E-6/ry 20%	T-FW-F&B (10%) T-FW-HPR (5%) T-LOCA-HPR (5%)	1 turbine 2 motor	Yes	Yes 1 of 2 PORVs	Auto	Yes	Yes (no HVAC initiators because of long recovery time)	Unlikely if RCPs tripped within 10 min. (has auto reactor trip if CCW lost for 10 min)
Millstone 2	3E-5/ry 75%	Not readily available in above format in submittal	1 turbine 2 motor	Yes (for small-small LOCAs and F&B)	Yes 2 of 2 PORVS	Auto	Yes (large ADVs)	Yes	Potential for 800+ gpm leak if RCPs not tripped
Palisades	2E-5/ry 40%	T-FW-F&B (10%) T-FW-HPR (20%)	1 turbine 2 motor	No	Yes 1 of 2 PORVs (block valves normally shut-must be opened)	Auto	Yes	Yes (screened out as an initiator)	No leak if cooling lost based on B-J pump/seal

Table 11.22 Transient characteristics for CE **plants.**

Plant	Transient CDF (/ry) - % of total CDF	Dominant sequence types	AFW/EFW pump config. (each plant)	HPI pumps need pump cooling?	Feed & bleed? - no. of PORVs for success?	HPR - auto or manual?	SG (steam generator) depressurized & core cooling considered?	HVAC failures generally considered?	RCP seal LOCA model
Palo Verde 1,2&3	6E-5/ry 65%	T-FW (65%)	1 turbine 1 motor 1 non-safety pump requiring manual start	Could not readily determine. (Appears to need room cooling?	No (no PORVs)	Auto (RWT valves to be manually shut but failure to do so does not fail HPR)	Yes	Yes	Appears unlikely if RCPs tripped
San Onofre 2&3	1E-5/ry 45%	T-FW (40%)	1 turbine 2 motor	Yes	No (no PORVs)	Auto	Yes	Yes	Unlikely if RCPs tripped within 30 min.
Waterford 3	2E-6/ry 14%	T-FW (15%)	1 turbine 2 motor 1 non-safety backup pump	Yes	No (no PORVs)	Auto	Yes	Yes	Unlikely if RCPs tripped within 30 min.
Maine Yankee	3E-5/ry 40%	Not readily quantifiable in the above format but appears mostly of T-FW-F&B type	1 turbine 2 motor	Yes	Yes 2 of 2 PORVs (failure =0.5 since not qualified for environment)	Auto	Yes	Yes	Unlikely (note: has high temp seals & plant has loop isolation valves)

Transient-induced LOCAs are typically caused by a stuck-open primary system relief valve or an RCP seal failure, particularly when seal cooling is lost as part of the transient. Failure of coolant makeup either during the injection or recirculation phase of the accident leads to early or late core damage, respectively. These transient-induced LOCA core damage sequences contribute up to 15% (early) and 10% (late) to the total plant CDFs for the CE plants.

The variability in the importance of transients (and the types of transient sequences) at the CE plants is driven by combinations of many factors; most significant of these appear to be the degree of support system dependence for systems used for heat removal, the degree of redundancy in heat removal systems, the general low susceptibility to RCP seal LOCAs, and long-term water resource capabilities. There is approximately a factor of 60 between the lowest and highest transient CDFs for the CE plants as already discussed above. This variability, and hence, the importance of transient accidents to the CE plants, is dependent on both design and operational characteristics for the plants as well as modeling characteristics that also influence the results. No single factor dominates the results (i.e., guarantees that the transient CDF will be high or low depending on the nature of any single factor). Combinations of factors are important, and those combinations vary from plant to plant. Review of the characteristics presented in Table 11.22, as well as other information from the CE plant submittals, provides key insights as to what plant design or analysis features drive the importance of transients for the CE plants.

As was discussed earlier, transients often consist of total or partial losses of support systems, either as initiators or as subsequent failures to general initiators. The key heat removal systems (i.e., main feedwater, condensate, AFW/EFW, and feed-and-bleed utilizing HPI/recirculation and PORV operation) are dependent on these support systems. The degree of this support and the level of redundancy in this support varies from plant to plant. Hence, some plants are more susceptible to partial or complete failures of some systems than other plants. This is illustrated in Table 11.22, for instance, by the variability in the need for high-pressure pump bearing and seal cooling during the injection phase. About one-half of the plants indicated such a dependency; one-half of the plants indicated this dependency does not exist at their plant. The variability in the need for HVAC in some areas of the plant, and hence, its effect as a contributor to the transient sequences, also varies among the plants. HPI room cooling faults were important in the Calvert Cliffs 1&2 submittal. Specific power bus loads can also be important as demonstrated by the significance of specific bus faults for most of the CE plants. The extent to which some of these dependencies are caused by plant design or modeling characteristics can not be readily determined solely on the basis of the submittals.

The degree of redundancy for removing heat in transient sequences varies from plant to plant as illustrated in Table 11.22. Particularly, whether feed-and-bleed is possible, the degree of redundancy for PORV operation for feed-and-bleed, variations in AFW/EFW redundancy, and the ability to depressurize the steam generators and use condensate for cooling, are major examples of this variability. The plants with the highest transient CDFs tend to (1) not have PORVs and so they can not feed-and-bleed, (2) have non-redundant features for success of feed-and-bleed, or (3) did not credit the ability to use steam generator depressurization and heat removal with condensate. For example, Calvert Cliffs 1&2 with the highest transient CDFs, require 2 of 2 PORVs (PORVs are small) and did not always credit feed-and-bleed for all types of events. Additionally, steam generator depressurization and condensate cooling is not credited at Calvert Cliffs 1&2 because of the small ADVs at this plant. The Palo Verde and San Onofre plants, among those with the highest transient CDFs, do not have PORVs and can not feed-and-bleed. Maine Yankee, Palisades, Millstone 2, and ANO 2 have less redundant or unusual bleed configurations. Plants with the lowest transient CDFs such as St. Lucie 1&2 have the combined redundancy of feed-and-bleed, among the most redundant AFW/EFW designs, and credited steam generator depressurization and condensate cooling. Waterford 3, the plant with the lowest transient CDF, has the greatest AFW redundancy and credited steam generator depressurization, although it has no PORVs for feed-and-bleed. Combinations of equipment independent and common cause hardware faults, failures in support systems, SW failures and sump valve faults affecting recirculation, and operator failures associated with initiating steam generator depressurization and condensate cooling are among the dominant contributors to failure of heat removal.

For all the CE plants, the susceptibility to RCP seal LOCAs is generally considered to be quite low on the basis of the typical Byron Jackson 4-stage seal design used in CE plants (or an equivalent) and the analyses, tests, and actual experience associated with these pumps. Hence, all the CE plants used optimistic models which are considered realistic for the CE plants. Nevertheless, some variability exists among the CE plant submittals as to the flow rate and probability of these LOCAs. For instance, one of the more relatively pessimistic RCP seal LOCA models was used by Calvert Cliffs 1&2 which have the highest transient CDFs of all the CE plants and among the highest relative contributions from transient-induced LOCAs. Overall, the generally low susceptibility to RCP seal LOCAs for the CE plants makes this failure and its demand on coolant system makeup unimportant.

The ability and actions associated with supplying long-term water to the suction for AFW/EFW is particularly significant for transients with no induced LOCA. Variability in water source size, alternate supplies, and the degree that the activity is proceduralized and trained all influence the relative importance of this characteristic.

11.3.2.3 LOCA Sequences

LOCAs involve a breach of the primary system as the initiating event, followed by failure of core cooling as a result of failing to provide adequate coolant makeup either during the injection or recirculation phase of the accident. On the basis of the accident class definitions used in this report, SGTRs and ISLOCAs are excluded from this LOCA class. They are special forms of LOCAs and are covered later as separate types of accident sequences. LOCAs cover a broad range of sizes; typically categorized as small, medium, or large LOCAs depending on the degree of primary system depressurization and the rate of coolant loss when the breach occurs.

LOCAs are generally the second most important accident class among all the CE plants and are even the dominant class of accident for a few plants. The range of LOCA CDFs for the CE plants is not large when compared to the variability found with other accident classes, with the exception of one plant. Small LOCAs are slightly more important than other size breaks, but there is no significant trend that either injection or recirculation failures are more important, regardless of the break size. Finally, many of the design and modeling characteristics important for transients are also important for LOCAs.

LOCAs are important for many of the CE plants, with absolute CDFs for all the plants (except one) within about a factor of 10. As shown in Table 11.23, all of the CE plants report total LOCA CDFs within the range of 5E-6/ry for ANO 2 to 6E-5/ry for Calvert Cliffs 1&2; or within a factor of 12 of each another. This is a tight clustering of values as compared to the range of transient CDFs discussed earlier and the range of SBO CDFs discussed later. The one plant exception that falls outside this range is Fort Calhoun 1 with a reported total LOCA CDF of 1E-6/ry, or about a factor of 5 below the ANO 2 value. The average CDF for this accident class across all plants is approximately 2E-5/ry. For nine of the fifteen plants, the LOCA CDFs represent a 20% or greater contribution to each plant's total plant CDF making this class of accident generally the second most important group of accident sequences for CE plants. In general, LOCAs are important contributors to CDF for many of the same reasons as the transients. Also, as the size of the LOCA gets larger and its initiating event frequency lowers, there is a corresponding drop in the level of redundancy available in the mitigating systems.

Table 11.23 LOCA results for CE plants.

Plant	Total LOCA CDF (/ry) --------------- % of total CDF	Smallest LOCA w/loss of injection CDF (/ry) --------------- % of total CDF	Smallest LOCA w/loss of recirculat'n CDF (/ry) --------------- % of total CDF	All other LOCAs w/loss of injection CDF (/ry) --------------- % of total CDF	All other LOCAs w/loss of recirculat'n CDF (/ry) --------------- % of total CDF
ANO 2	5E-6/ry 15%	>6E-7/ry <5%	>9E-7/ry <5%	>1E-6/ry <5%	>2E-6/ry 5%
Calvert Cliffs 1&2	6E-5/ry 25%	2E-5/ry 10%	9E-6/ry <5%	2E-5/ry 10%	7E-6/ry <5%
Fort Calhoun 1	1E-6/ry 10%	7E-7/ry 5%	1E-7/ry <5%	1E-7/ry <5%	1E-7/ry <5%
St. Lucie 1	1E-5/ry 55%	3E-6/ry 10%	4E-6/ry 20%	3E-6/ry 15%	2E-6/ry 10%
St. Lucie 2	1E-5/ry 50%	3E-6/ry 10%	5E-6/ry 20%	3E-6/ry 10%	3E-6/ry 10%
Millstone 2	6E-6/ry 20%	Combined inj. and recirc =	1 E-6/ry <5%	Combined inj. and recirc =	5E-6/ry 15%
Palisades	2E-5/ry 30%	8E-6/ry 15%	7E-6/ry 15%	negligible	5E-7/ry <5%
Palo Verde 1,2&3	7E-6/ry 10%	2E-6/ry <5%	2E-6/ry <5%	3E-7/ry negligible	3E-6/ry <5%
San Onofre 2&3	1E-5/ry 35%	10E-7/ry <5%	2E-6/ry 5%	3E-6/ry 10%	5E-6/ry 15%
Waterford 3	7E-6/ry 40%	4E-6/ry 20%	1E-6/ry 10%	10E-7/ry 5%	3E-7/ry <5%
Maine Yankee	3E-5/ry 40%	Combined inj. and recirc =	2E-5/ry; 25% (includes very,very small to small LOCAs	Combined inj. and recirc =	1E-5/ry 15%

Examination of possible reasons why the Fort Calhoun 1 submittal has a LOCA CDF that is considerably lower than the other plants reveals that while it shares the advantages of redundant cooling features like many of the other plants (including a diesel-driven AFW train that is independent of other cooling) and its HPI is not particularly susceptible to support system faults, one other significant reason applies. Review of the LOCA frequencies across all the CE plant submittals reveals that Fort Calhoun 1 used LOCA frequencies that are generally factors of 4 to 20 times lower than corresponding frequencies reported in the other CE submittals. It is not certain whether this observation is indicative of pessimistic values used by the other plants, or optimistic (i.e., low) values used in the Fort Calhoun 1 submittal.

While small LOCAs are somewhat more important than larger break sizes, there is no significant trend regarding the relative importance of injection or recirculation failures for any size break. Furthermore, different types of failures dominate either loss of injection or recirculation sequences. Table 11.23 summarizes the subsets of LOCA sizes and injection/recirculation failures making up the LOCA class of accidents for all the CE plants. In the table, the small LOCA category considers the smallest LOCA size covered in each submittal unless otherwise noted. Since many of the characteristics important to transients (discussed earlier) are also important for LOCAs (especially the smaller sizes), these characteristics are not presented again here in Table 11.23. For 11 of the 15 plants, small LOCA (by itself) CDFs makeup at least one-half of the total LOCA CDF for each plant. The remaining LOCA CDF contribution comes from the combined results of larger LOCA sizes. Hence, while small LOCAs tend to dominate for some plants, other medium to larger size LOCAs are also significant for these same plants and are even dominant for a few of the plants. There is no significant trend that injection or recirculation failures dominate the LOCA accident class, as both types of failures are important depending on the specific plant.

The dominant equipment failures and operator actions for the LOCA sequences vary among the plants and for the types of LOCAs. Recirculation failures tend to be driven by random valve failures and storage tank level instrumentation failures. Other failures include HPR cooling water and room cooling failures, and operator failure to switchover to hot leg injection to prevent boron precipitation. For LOCAs with failure of injection, similar valve and HPI cooling failures, as well as HPI common cause and independent pump failures tend to dominate. In some cases, operator failure to cooldown/depressurize so that low-pressure systems can be used is also a contributor.

11.3.2.4 SBO Accident Sequences

SBO sequences involve an LOSP, followed by a loss of all on-site AC power provided by EDGs. The failure of all AC power results in failure of all injection systems and failure of normal heat removal via feedwater and the steam-condenser system. Additionally, motor-driven AFW or EFW trains are also lost, leaving only the turbine-driven AFW/EFW available for cooling the core via the steam generators. Failure of all primary system injection capability results in no available means to makeup the loss of primary coolant through either an RCP seal failure or a stuck-open primary relief valve that may develop during such a scenario. Operation of the turbine-driven AFW/EFW is also ultimately limited by the loss of support systems such as HVAC for room cooling, air/nitrogen for maintaining valve operability in some cases, and particularly DC power (on the basis of battery life and load shedding) for instrumentation and control.

The SBO accident class is important for some of the CE plants. The variability in the importance of SBO is driven by combinations of plant features as well as to some degree, modeling differences among the submittals. The redundancy and diversity of AC power sources, including the ability to cross-tie buses within or between plants at a multi-unit site, is a key characteristic as to how important SBO is at each CE plant. The degree of dependence on diesel support systems, particularly cooling water and battery lifetimes including provisions for load shedding, are also key characteristics determining the importance of SBO at the CE plants. Either early or late SBO sequences can be the most significant, depending on the battery lifetimes as well as other features. Recovery modeling assumptions can also be an important factor in the calculated SBO CDFs. RCP seal LOCAs are not a factor in determining the importance of SBO for CE plants, on the basis of the RCP designs used at CE plants and testing results and experience associated with those designs.

SBO is important for some of the CE plants with absolute CDFs for individual plants covering a wide range. Table 11.24 summarizes the SBO CDF results for the CE plants and some key characteristics associated with the plant designs and the SBO analyses for each plant. In two submittals, for Calvert Cliffs 1&2 and for Fort Calhoun 1, the IPE results are not presented in a form that the SBO CDF contributions can be readily determined. Where the

SBO results are available, the SBO CDF is greater than 6E-6/ry for six plants represented by four submittals. For these six plants, the percentage contribution of SBO to the total plant CDF is approximately 15% or greater, and in one case, Waterford 3, represents about one-third of the total plant CDF. The remaining six plants have SBO CDFs less than 3E-6/ry, each less than a 10% contribution to the total plant CDF. For those plants with SBO results readily available, the average SBO CDF (counting each separate plant at a multi-unit site as a separate result) is approximately 8E-6/ry, with a range of SBO CDF values of 4E-7/ry for Millstone 2, to 2E-5/ry for each of the Palo Verde plants. Hence, there is a wide variation and somewhat uniform spread for the SBO CDF results among the CE plants.

Table 11.24 SBO characteristics for CE plants.

Plant	SBO CDF (/ry)	% of total CDF	LOSP freq'y (/ry)	No. of EDGs	Battery depletion time	RCP seal LOCA	Comment
ANO 2	1E-6	<5%	0.058	2	8 hrs	not a factor	-Bus crossties considered -Somewhat optimistic recovery
Calvert Cliffs 1&2	unavail.	unavail.	0.136	Note (a)	unavail.	low potential	
Fort Calhoun 1	unavail.	unavail.	0.2 (est.)	2 (radiator cooled)	8 hrs	not a factor	
St. Lucie 1	3E-6	10%	0.15	2 (radiator cooled)	8 hrs	not a factor	-Bus crossties considered
St. Lucie 2	3E-6	10%	0.15	2 (radiator cooled)	8 hrs	not a factor	-Bus crossties considered
Millstone 2	5E-7	<5%	0.091	2	8 hrs	not a factor	-Crosstie to Unit 1 credit -LOSP events <1/2 hr not considered
Palisades	9E-6	20%	0.03	2	6 hrs	not a factor	-AFW valves can remain open add'l. 6 hrs -Mostly early SBO
Palo Verde 1,2&3	2E-5	20%	0.078	2	2 hrs	not a factor for < 2 hrs	-87% late (3 hr) SBO -13% early (1 hr) SBO
San Onofre 2&3	2E-6	5%	0.11	2 (radiator cooled)	8 hrs	not a factor	-12% late (>8 hr) SBO -88% early SBO -Grid vs. weather recovery -Appears credit for manual AFWS oper.

Table 11.24 SBO characteristics for CE plants.

Plant	SBO CDF (/ry)	% of total CDF	LOSP freq'y (/ry)	No. of EDGs	Battery depletion time	RCP seal LOCA	Comment
Waterford 3	6E-6	35%	0.03	2	4 hrs	not a factor	-45% late (>4 hr) SBO -55% early SBO
Maine Yankee	1E-5 Note (b)	10-15% Note (b)	0.05	2 (backup cooling w/ firewater)	6-8 hrs	not a factor	-Fast transfer for 115kv degradation -Credited App.R self-cooled DG -Pessimistic recovery

Note (a) Newer design modeled with 2 DGs per unit & 1 non-safety swing DG; use to be 1 DG per unit with 1 swing DG

Note (b) Uncertain CDF and percentage; based on Technical Evaluation Report estimate and review of dominant scenarios

Both short-term (early) and long-term (late) SBO sequences are important, with a tendency that longer battery life lessens the relative importance of late SBO sequences for CE plants. Early SBO sequences involve the loss of all AC power with a failure-on-demand of the turbine-driven AFW/EFW such that core heat removal is lost at the onset of the accident. Without recovery of AC power, and hence, other means for heat removal in about 1 hour, core damage occurs. Failures-on-demand of the AFW/EFW were found to be driven by pump failures-to-start, failure of the steam admission valve to open (to supply steam to the pump turbine), and train maintenance outage contributions for the CE plants.

Late SBO sequences involve the loss of all AC power with initial success of the turbine-driven AFW/EFW so that core cooling via the steam generators is available. However, without eventual AC power restoration, continued control and operability of the turbine-driven AFW/EFW is compromised because of a loss of support systems. The CE plant submittals concluded that the eventual loss of DC power as a result of battery power being depleted, along with failures-to-run for the turbine-driven AFW/EFW, were the two most important late failure modes of this system. Once the turbine-driven AFW/EFW is failed, core heat removal is halted and eventually core damage occurs late (typically 2–3 hours after AFW/EFW failure) in the accident.

While most of the CE plant submittals did not provide information in a form to readily determine the relative importance of early and late SBO sequences, three submittals covering six plants provided such information that made this determination possible. Palo Verde 1,2&3 concluded that the SBO CDF for these plants was made up of approximately 90% late SBO sequences and 15% early SBO sequences. San Onofre 2&3 concluded that the SBO CDF for these plants was made up of 10% late SBO sequences and 90% early SBO sequences. Waterford 3 concluded its split to be almost 50% early and 50% late.

Examination of the characteristics presented in Table 11.24 for the six plants mentioned above, reveals the fact that Palo Verde 1,2&3 with the shortest battery life (2 hours), have the late SBO sequences as most significant. Waterford 3 with a medium battery life (4 hours) has early and late SBO sequences as approximately equally important. San Onofre 2&3 with the longest battery life (8 hours), has the early SBO sequences as most significant. This apparent correlation to battery life is explained by the fact that the longer battery life allows for a greater chance of long-term success of the turbine-driven AFW/EFW, hence, also providing a greater likelihood that AC power will eventually be restored so that the plant can be safely shutdown. Therefore, the CDF from late SBO sequences tends to be less than the CDF from early SBO sequences involving the immediate failure of AFW/EFW with little time to recover AC power to avoid core damage. In the case of the San Onofre 2&3 plants, this correlation is further supported by the fact that the submittal credited possible operation of AFW manually without control power, and accounted for differences in the likelihood of LOSP and the recovery of off-site power on the basis of grid versus weather interruptions. Both of these factors also tend to decrease the CDF from late SBO sequences relative to the CDF from early SBO sequences.

The above correlation is true only if the probability of an SBO-induced LOCA, caused by a stuck-open primary relief valve or a RCP seal LOCA (induced by loss of cooling to the pump seals), is small so that there is not an early demand for coolant injection, and hence, AC power restoration to makeup for the coolant loss. Such a demand would tend to yield higher early SBO CDFs. Limited discussion in the CE plant submittals concerning stuck-open primary relief valves indicates there is little demand for relief valve operation and/or a low probability of these valves sticking open. RCP seal LOCAs are also considered to be low probability events for the CE plants, as discussed later. Thus, SBO-induced LOCAs are low probability events allowing the above observations to be valid.

The variability in the importance of SBO at the CE plants is driven by combinations of factors; most significant of these appear to be the number of emergency AC power sources (including cross-tie capability within or between plants) and their dependencies, and battery power depletion times. There is approximately a factor of 50 between the lowest and highest SBO CDFs for the CE plants as already discussed above. This variability, and the importance of SBO accidents to the CE plants, is dependent primarily on design and operational characteristics for the plants although modeling nuances also influence the results. No single factor dominates the results (i.e., guarantees that the SBO CDF will be high or low depending on the nature of any single factor). Combinations of factors are important, and those combinations vary from plant to plant. Review of the characteristics presented in Table 11.24, as well as other information from the CE plant submittals, provides key insights to which plant design or analysis features drive the importance (or unimportance) of SBO for the CE plants.

Examination of the LOSP-initiating event frequencies reveals that the plants with the highest LOSP frequency values are among those with the lowest SBO CDFs. Conversely, plants with moderate to the lowest LOSP-initiating event frequencies are among those with the highest SBO CDFs. While the LOSP frequency certainly influences the probability of SBO and the frequency of SBO core damage, the grid reliability at each plant and the resulting LOSP frequency, by itself, is not a significant factor driving the importance of SBO.

For SBO to occur, the on-site EDGs must also fail. The CE plant submittals illustrate that the most common contributors to EDG failures are independent and common cause failures of the diesels to start or run, diesel generator cooling failures, diesel room cooling faults, fuel failures, and diesel output breaker faults. While the "as-analyzed" plant configurations in the CE plant submittals model two diesel generators for each plant, other analyses regarding plant improvements as well as other design and operational features associated with the diesels reveal features that tend to drive the importance of SBO. For instance, an analysis by the ANO 2 licensee to determine the benefit of adding a swing diesel between ANO 1 and ANO 2 estimates a 45% reduction in the total plant CDF and a further decrease in the SBO contribution. A Millstone 2 analysis shows that if the submittal had not taken credit for the bus cross-tie capability to Millstone 1, the SBO contribution to the Millstone 2 total plant CDF would have been 10% instead of the 1% in the existing analysis. The former diesel configuration at Calvert Cliffs 1&2 was one diesel dedicated per unit with one swing diesel between plants. The enhanced design credited in the submittal included the addition of two diesels at the site, one for each plant. This design difference was analyzed to lower the total plant CDFs by almost 20% with an associated drop in the SBO contribution. A Palo Verde 1,2&3 analysis shows that adding gas turbines at the multi-unit site will reduce the total plant CDF by almost 30% with an associated drop in the SBO contribution. Among the CE plants with SBO CDFs below 3E-6/ry, four of the six plants have EDGs that use self-cooled radiator designs, thereby eliminating the dependence on SW system operability for diesel cooling. At least four of the same six plants have the capability and credited bus cross-ties within and/or between plants which further increases the overall AC system availability and reliability and decreases the potential for SBO. Clearly, the collective set of these observations shows that the greater the degree of redundancy and diversity of the on-site AC power sources, the ability to utilize cross-ties between these sources, and the reduction of diesel generator support needs, all tend to lower the potential for SBO and reduce the importance of SBO sequences at the CE plants.

As has been previously discussed, battery depletion time (a function of both battery capability and load shedding during SBO) can significantly affect the importance of late SBO sequences and the overall importance of SBO. All six of the CE plants with SBO CDFs of less than 3E-6/ry credited 8-hour battery lifetimes in their submittals. The plants with the highest SBO CDFs generally credited battery lifetimes of 6 hours or less; Palo Verde 1,2&3 have the highest SBO CDFs and the shortest credited battery lifetimes (2 hours). Battery lifetime appears to strongly influence the importance of SBO among the CE plants.

During SBO, the operability of the turbine-driven AFW/EFW would be expected to also influence the importance of SBO sequences. However, differences in the failure databases for this equipment among the CE plants was found to not be significant. This is not a dominant factor in the variability in the importance of SBO sequences for the CE plants.

Review of the highest and lowest SBO CDF plants shows how these features, in combination, tend to drive the relative importance of SBO at the CE plants. The highest SBO CDFs are for Palo Verde 1,2&3 and Maine Yankee. Palo Verde 1,2&3 used moderately high LOSP frequencies and have the shortest battery lifetimes reported among the CE plants. Maine Yankee has a number of features that would suggest a low SBO CDF (e.g., a low LOSP frequency, a high battery lifetime, backup cooling for the diesels, among others). However, this submittal uniquely considers some dual bus initiator failures, has a dependency on switchgear room cooling, uses a 24-hour mission time for the diesel generators, as opposed to a more common 6 to 8 hours in other submittals, and uses a pessimistic power recovery model in that only one train of AC power is assumed recovered and subsequent equipment failures can still add to the SBO CDF. So the Maine Yankee SBO CDF is driven by mostly modeling characteristics as well as the switchgear room cooling dependency. The lowest SBO CDFs are for Millstone 2, ANO 2, and San Onofre 2&3. Millstone 2 has a fairly high LOSP frequency but did not treat LOSP events lasting less than one-half hour; a unique modeling difference from the other submittals. Millstone 2 has an 8-hour credited battery life, and considered cross-tie capability to Millstone 1. ANO 2 has a relatively low-to-moderate LOSP frequency, an 8-hour battery life, credited bus cross-ties, and additionally used a somewhat optimistic off-site power recovery model than the other CE plants. San Onofre 2&3, while using a high LOSP frequency, has an 8-hour battery life, radiator cooled diesels, treated grid versus weather-induced LOSP separately which allows for a somewhat optimistic (but also realistic) recovery model, and apparently credited possible operation of the turbine-driven AFW even without control power.

All of the CE plant submittals, with one possible exception, concluded that RCP seal LOCAs are not a factor in the determination of the SBO CDFs. The potential concern is that during an SBO, all cooling to the RCP seals is lost and there is the possibility of seal degradation and failure under primary system pressure and temperature conditions. Such a failure would result in a primary system LOCA requiring injection makeup which can not be provided until AC power is restored since all CE injection systems require AC power.

All the CE plants use Byron Jackson pumps with a multi-stage seal design including a fourth-stage vapor seal (or equivalent features). These pumps have been shown to be resistant to pump seal failure through testing, analysis, and actual experience; particularly when the pump is stopped as will occur automatically during an SBO (there is no power to the pumps). On this basis all but one of the CE submittals considered the chance that an RCP seal LOCA would occur during the time period of interest in SBO sequences, to be negligible. The Calvert Cliffs 1&2 submittal did consider a low potential for such a failure during SBO conditions. However, the calculated contribution of such a failure is not readily available from the submittal. In general, RCP seal LOCAs are not considered a factor in SBO sequence outcomes and are not treated in the CE submittals as part of the SBO accident analyses.

11.3.2.5 ATWS Accident Sequences

ATWS sequences involve a transient, followed by failure to shutdown the nuclear chain reaction by inserting the control rods. Power generation continues at levels far in excess of normal DHR. An ATWS sequence can be mitigated by primary system pressure control, boration, and heat removal. The failure to scram can be caused by electrical failures of components associated with the RPS or in the breakers that when opened, deenergize the control rods, or by mechanical faults associated with the CRD mechanisms or the control rods themselves. Partial scram failures are possible, but in most PRAs it is assumed that scram failure is total and involves all the rods. Given the redundancy of equipment associated with the scram function, common cause failure mechanisms must be the source of such assumed widespread failure.

Because of the low probability of failure to scram (typically judged to be a probability of low E-5) and by taking credit for the above-mentioned pressure control, boration, and heat removal capabilities, ATWS is a low contributor to the total CDF for all but one plant in the CE group. With that one multi-unit exception, the ATWS CDFs for the CE plants are below 5E-6/yr. Nine of the fifteen plants have ATWS CDFs between 1E-6/ry and 4E-6/ry and four of the plants have CDFs between 1E-7/ry and about 4E-7/ry. In all these cases, the ATWS CDF contribution to the total plant CDF is always below 10% for any plant. The one exception is the Calvert Cliffs 1&2 submittal which reports ATWS CDFs for both plants as 2E-5/ry, or about a factor of 6 above the next highest plant's reported value. Examination of the Calvert Cliffs 1&2 submittal reveals that this result is primarily attributable to an apparent design difference and pessimistic modeling. The Calvert Cliffs 1&2 submittal considers the fraction of time that the reactor cores at these units have an unfavorable moderator temperature coefficient (MTC), discussed below, to be about 40%, significantly above that normally seen in PWR PRAs. Whether this is a true design difference or a function of pessimistic analyses, can not be determined just from the submittal. Partially as a result of this consideration, the Calvert Cliffs 1&2 analyses did not model the mitigating features mentioned above at all. This submittal assumes that failure to scram leads to core damage, a pessimistic assumption. Even with this bounding approach, ATWS barely contributes 10% to the total plant CDFs for both plants.

ATWS sequences are not an important contributor to the total plant CDFs for the CE plants. All the plants found this contribution to be less than 10% (even for the pessimistic Calvert Cliffs 1&2 analyses). The dominating factors contributing to these sequences typically involve mechanical failure of the control rods followed by various equipment, human error, or an insufficiently negative MTC which cause failure of pressure control, boration, and/or heat removal. The results of the ATWS sequences are driven by both actual design characteristics, and a tendency to pessimistically model this otherwise complex type of scenario.

Mechanical failures of the rods followed by a variety of equipment failures, human errors, or an unfavorable moderator temperature coefficient dominate the ATWS sequences for the CE plants. In general because of the low contribution of ATWS sequences to the total plant CDFs for the CE plants, not much documentation of these sequences exists in the submittals. However, where available, it was evident that mechanical failures of the rods were generally judged to be more important than electrical failures causing failure to scram. The most important ATWS sequences also tended to individually involve equipment (valve and pump) failures causing loss of boration or heat removal (AFW/EFW), failure of a significant number of primary system relief valves to open thereby assuming an unmitigated LOCA developed, all primary system relief valves failing to reseat also causing an assumed unmitigatable situation, and operator errors associated with ensuring emergency boration and that the main turbine has tripped. These equipment and operator failures all lead to one or more failures of the pressure control, boration, and heat removal functions discussed above.

The MTC was also judged to be a significant factor in most of the CE submittals. This coefficient is a measure of the inherent power feedback caused by the temperature of the fuel in the reactor core (as temperature rises, power decreases). Typically during the early portion of a reactor core's life in a PWR, this temperature feedback may not be sufficiently negative to ensure that in an ATWS, the quick rising pressure in the primary system is halted (because the fuel temperatures are also rising quickly) to avoid overpressurization of the primary system. If this were to occur, PRAs typically assume an unmitigatable LOCA occurs and core damage will occur. Where information was available in the CE submittals, nearly all indicated a portion of the ATWS CDF also involved this aspect.

11.3.2.6 Internal Flooding Accident Sequences

Internal flooding events involve failure of water or steam sources within the plant, that result in a release of water or steam that can cause a transient-like scenario requiring mitigation. This failure can also directly fail the mitigating systems (such as if the failure is in a mitigating system itself) or indirectly fail other systems as a result of submergence or spraying of equipment.

Internal floods are not important to the CDF for the CE plants. Flooding effects are very plant-specific and depend on the specific design, layout, and isolation of rooms and equipment within the plant. The determination of the important flooding sequences is largely driven by these varying plant features, as well as the level of optimism used in the analyses.

Internal flooding accidents are not important for CE plants. All the CE plants reported internal flooding CDFs in the E-7/ry range or lower on the basis of screening methods, except for Waterford 3, Fort Calhoun 1, and Calvert Cliffs 1&2. These plants reported values in the 1E-6/ry to 2E-6/ry range (Waterford 3 and Fort Calhoun 1) and 2E-5/ry for the Calvert Cliffs 1&2 plants. Reasons for the higher plant results are discussed below, but generally, internal flooding is not a dominant contributor to CE plant CDFs.

The effects of internal flooding are very plant specific. In examining the results associated with the three highest plant internal flooding CDFs, it is demonstrated that the diversity of plant design and layout and the variability in isolation of plant rooms, significantly effect the internal flooding potential for serious accidents. The Waterford 3 submittal concludes that the only significant flooding scenario involves a turbine building flood which because of plant arrangement and layout, can fail off-site power buses and involve subsequent failure of EFW. The analysis is reported to contain pessimistic assumptions, and therefore, inflates the internal flood CDF. The Fort Calhoun 1 submittal reports a flooding CDF that while low, represents a 15% contribution to the total plant CDF. These floods are limited to the auxiliary building and involve floods caused by raw water, CCW, and AFW systems. As a result, plant improvements are being made to address these scenarios. The Calvert Cliffs 1&2 submittal shows its plant flooding contributions to be made up of multiple flood sources and locations, rather than a few dominant floods. In accumulating the frequencies of these nearly dozen small flooding contributions, the Calvert Cliffs 1&2 submittal arrives at an internal flooding CDF that is higher than the CDFs reported in the other CE plant submittals. This is different than the usual practice of screening individual flooding scenarios on the basis of their low frequency, and arriving at a negligible or non-quantified flooding contribution which is what appears to have been done with most of the other plants.

11.3.2.7 SGTR Accident Sequences

SGTR sequences involve leakage from the primary coolant to the secondary side of the plant through a ruptured or leaking steam generator tube. Failure to mitigate the leak (such as by depressurization or isolation of the steam generator) or failure to establish long-term core cooling results in core damage. This sequence class is generally a low CDF contributor, but can be a significant contributor to off-site consequences because the releases can bypass containment.

SGTR accidents are not important to the CDF for the CE plants. Their occurrence is largely driven by failures associated with equipment used for pressure relief and coolant makeup, as well as operator errors associated with isolating the faulted steam generator and ensuring long-term cooling.

SGTR accidents are not important to the frequency of core damage for CE plants. All of the CE plants reported SGTR CDFs of less than 5E-6/ry. This accident class contributes approximately 5% or less to any single total plant CDF. All of the plant CDFs fall within approximately one decade except for ANO 2 which has a SGTR CDF about a factor of 6 below the lowest of all the other plants. As such, while there is some spread in the results among the CE plants, the low absolute CDF values generally result from the low frequency of a significant rupture (typically the order of 1E-2/yr or less) and the redundant availability of heat removal equipment which can mitigate the event.

Both equipment failures and operator errors contribute to SGTR CDF. Because of the low contribution of SGTR sequences to the total plant CDFs for the CE plants, not much documentation of these sequences exists in the submittals. However, where available, it was determined that the contributors to SGTR accidents typically involve equipment failures associated with HPI/recirculation and the associated support systems used for coolant makeup, failure of pressure relief valves, and operator contributing failures involving the lack of providing long-term cooling and failure to isolate the faulted steam generator.

11.3.2.8 ISLOCA Sequences

An ISLOCA occurs when valves that normally isolate the reactor coolant system from low-pressure systems (e.g., LPI) fail, resulting in backflow from the reactor coolant system through the low-pressure piping. If this low-pressure piping or other components (e.g., seals, relief valves, flanges) can not withstand the resulting pressurization, then a LOCA results. If the breach occurs in a portion of the piping that is outside the containment, the effluent from the breach bypasses containment. Because it is not possible to recirculate the coolant through the reactor coolant system for this type LOCA, water used for injection necessary to makeup for the loss could be depleted. This sequence class is generally a low CDF contributor but can be a significant contributor to off-site consequences because the releases can bypass containment.

ISLOCAs are not important to the CDF for the CE plants. Their leading to core damage is largely driven by operator errors associated with failing to isolate the faulted pathway and ensure long-term cooling.

ISLOCAs are not important to the frequency of core damage for CE plants. All of the CE plants reported ISLOCA CDFs of less than 3E-6/ry. This accident class contributes approximately 10% or less to any single total plant CDF. All of the plant CDFs fall within the low E-7/ry to low E-6/ry range except for Millstone 2 which has a ISLOCA CDF about a factor of 3 below the lowest of all the other plants. While there is some spread in the results among the CE plants, the low absolute CDF values generally result from the low frequency of a significant rupture and the redundant availability of heat removal equipment which can mitigate the event.

Operator errors contribute to ISLOCA CDF. The most likely sources of such LOCAs involve the shutdown cooling line, RCP seal cooler lines, and the LPI lines. Because of the low contribution of ISLOCA sequences to the total plant CDFs for the CE plants, not much documentation of these sequences exists in the submittals. However, where available, it was determined that the contributors to ISLOCAs typically involve operator contributing failures associated with the lack of providing long-term cooling and failure to isolate the faulted pathway. The sources appearing in the CE plant submittals contributing most to the ISLOCA CDFs include the shutdown cooling line, the RCP seal cooler lines, and the LPI pathways.

11.3.3 CDF Perspectives for Westinghouse 2-loop Reactors

As indicated in Table 11.25, four PWR plants are grouped in the Westinghouse 2-loop category. Two of these plants are dual-unit sites, giving a total of six units in the group. All of the plants are housed in large dry containments.

Table 11.25 Plants in Westinghcuse 2-loop plant group.

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

11.3.3.1 Summary of Results and Perspectives for Westinghouse 2-loop Reactors

The total CDFs for the plants in the Westinghouse 2-loop group are shown in Figure 11.6. The CDFs for this plant group only vary within about a factor of two, with an average CDF of 8E-5/ry. The contribution of individual accident classes to plant CDF shows more variability, however, as shown in Figure 11.9. As indicated in the figure, the CDFs in most accident classes exhibit about an order of magnitude or smaller spread, with a larger variability in the CDFs for the ISLOCAs and internal floods. The variation in the accident class CDFs is due to a combination of plant design differences and modeling assumptions. Overall, the largest contributors are transients, LOCAs, SBOs, and SGTRs. The IPE submittals generally reported smaller contributions from ATWS, internal flooding, and ISLOCA sequences.

Table 11.26 summarizes the perspectives obtained from examination of the Westinghouse 2-loop plants. The details that provide the basis for these perspectives are contained in the remainder of Section 11.3.3. The results for the accident classes are discussed, giving more details on the factors driving the CDFs, particularly for the plants with the highest and lowest CDFs. Design and operational factors along with differences in modeling assumptions are addressed.

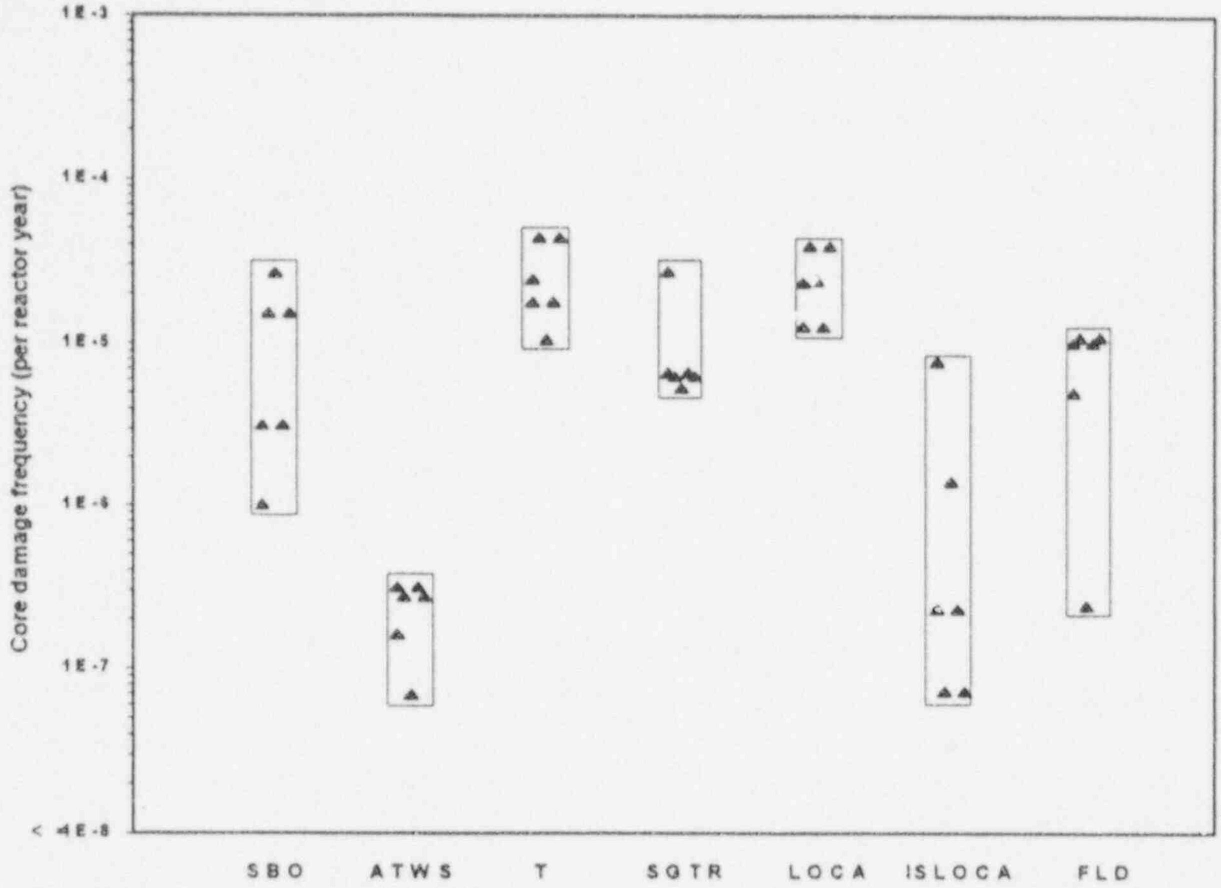


Figure 11.9 Sequence results for Westinghouse 2-loop plants.

Table 11.26 Key IPE observations for Westinghouse 2-loop plants.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
Transient accidents		
Important for all Westinghouse 2-loop plants	Degree of HPI and AFW dependence on SW and CCW RCP seal cooling is not dependent on SW or CCW, except one plant Modeling of RCP seal LOCA probability and size Dependence of PORVs on instrument air	Transient CDFs range from 1E-5/ry to 4E-5/ry. Average CDF is 3E-5/ry. Contribution to total plant CDF ranges from 15% to 40%
Internal flood accidents		
Important for some Westinghouse 2-loop sites	Plant layout: separation of mitigating system components and compartmentalization	Flood CDFs range from 2E-7/ry to 1E-5/ry. Average CDF is 8E-6/ry. Contribution to total plant CDF ranges from negligible to 20%

Table 11.26 Key IPE observations for Westinghouse 2-loop plants.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
SBO accidents		
Important for some Westinghouse 2-loop plants	AC power reliability (e.g., 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	SBO CDFs range from 1E-6/ry to 3E-5/ry. Average CDF is 1E-5/ry. Contribution to total plant CDF ranges from negligible to 40%
ATWS accidents		
Not dominant for Westinghouse 2-loop plants	Assessment of fraction of time that plant has unfavorable moderator temperature coefficient	ATWS CDFs range from 7E-8/ry to 3E-7/ry. Average CDF is 2E-7/ry. Contribution to total plant CDF is negligible for all plants
ISLOCAs		
Not important for Westinghouse 2-loop plants	Compartmentalization and separation of equipment	ISLOCA CDFs range from 7E-8/ry to 8E-6/ry. Average CDF is 2E-6/ry. Contribution to total plant CDF ranges from negligible to 10%
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 switchover) Whether containment sprays will be actuated (causing more rapid depletion of RWST) Assessment of human error probability for performing switchover to recirculation	LOCA CDFs range from 1E-5/ry to 4E-5/ry. Average CDF is 2E-5/ry. Contribution to total plant CDF ranges from 25% to 35%
SGTR accidents		
Important for one Westinghouse 2-loop plant	Modeling of operator actions to isolate the rupture and provide long-term heat removal	SGTR CDFs range from 5E-6/ry to 3E-5/ry. Average CDF is 1E-5/ry. Contribution to total plant CDF ranges from 5% to 30%

11.3.3.2 SBO Perspectives

SBO occurs when a plant loses off-site AC power, and the on-site backup AC power sources (diesel generators) also fail to function. Because so many safety systems rely on AC power either directly (for motive power) or indirectly (e.g., cooled by systems that rely on AC power), SBO causes most safety systems to be unavailable. Most SBO scenarios do have DC power available, however, through use of station batteries. For PWRs, the loss of all AC power normally leaves turbine-driven AFW as the only available means for DHR. Turbine-driven AFW can operate until the batteries deplete, potentially leading to a loss of control. If AC power is not recovered soon after loss of control, core damage will follow. For one IPE, Point Beach, AFW was assumed to continue even after battery depletion by manually controlling feedwater. However, the limited supply of water in the CST for feedwater injection often leads to core damage for such sequences.

An additional, important, complication for Westinghouse plants involves the potential for leakage from RCP seals. In the Westinghouse 2-loop plants, SBO results in loss of cooling for the RCP seals. The prolonged exposure to high temperatures can fail the seals, leading to a small LOCA through the pump seals. If a seal LOCA occurs, injection systems are needed to provide makeup to the RCS. Since these are not normally available during an SBO, pump seal LOCAs will lead to core damage if off-site power is not restored in time. Chapter 17 of this report provides further discussion of SBO sequences.

SBO is an important contributor (greater than 10%) to plant CDF for about half of the Westinghouse 2-loop submittals. Several factors are important for SBOs for the Westinghouse 2-loop plants, with differing combinations of these factors driving the results for the individual plants in this group. The factors that have the biggest impact on the results primarily represent actual plant characteristics (such as alternative arrangements for providing seal cooling), but modeling used in the IPEs is also important (such as the size and timing of seal leaks). The remainder of this section provides more details on these SBO perspectives.

SBO sequences can be categorized as short-term, long-term, or involving RCP seal LOCA. On average, long-term SBO sequences with RCP seal LOCAs are most frequent. To understand the reasons for differences in SBO CDFs, it is helpful to categorize SBOs as follows:

- <u>short-term SBO</u> - sequences in which turbine-driven AFW pump trains fail to operate and AC power is not recovered before the core is damaged

- <u>long-term SBO</u> - sequences in which turbine-driven AFW pump trains operate initially, but ultimately fail (normally because battery depletion causes loss of DC control power) before recovery of AC power

- <u>SBO with RCP seal LOCA</u> - sequences involving RCP seal LOCAs that are caused by loss of cooling to RCP seals with failure to recover AC power before the core is uncovered.

Short-term SBO sequences typically lead to core damage within about 2 hours, while long-term SBO sequences typically lead to core damage in more than 2 hours. SBO sequences with RCP seal LOCAs can be either short-term or long-term (depending on the model used for the timing and magnitude of a RCP seal LOCA), but they are typically long-term sequences.

The relative contribution of short-term and long-term SBO sequences varies among the Westinghouse 2-loop IPEs. Short-term SBO sequences are the larger contributors for Point Beach 1&2, long-term SBO sequences are the larger contributor for Prairie Island 1&2, and short-term and long-term sequences contribute about equally for Kewaunee.

The relative contribution of the short-term and long-term SBO sequences is not readily available from the Ginna IPE submittal. The plant that is dominated by long-term SBO, Prairie Island 1&2, is dominated by RCP seal LOCAs. The Point Beach 1&2 and Kewaunee IPE submittals both list a small contribution from RCP seal LOCAs.

The SBO results are driven by a combination of factors involving both plant-specific features and IPE modeling characteristics. Some of the system and component failures that are dominant contributors to the SBO CDF are common across the plants, while others are highly plant-specific. The failures that contribute most to SBO for the Westinghouse 2-loop plants can be grouped into the categories listed below:

- LOSP initiator
- loss of on-site AC power
- failure to recover AC power
- failure of turbine-driven AFW
- transient-induced LOCA

The system and component failures that contribute to each of the categories are discussed in the following paragraphs.

The initiator for SBO is, of course, LOSP. Higher frequencies for LOSP will lead to higher SBO CDFs. The frequencies used for each of the Westinghouse 2-loop plants are listed in Table 11.27. The initiator frequency for Ginna is about an order of magnitude lower than the other Westinghouse 2-loop plants, which contributes to the relatively low SBO contribution for Ginna.

Table 11.27 Key parameters affecting SBO CDF for Westinghouse 2-loop plants.

Plant	SBO CDF (1/ry)	No. EDGs unit/other/x-tie[1]	Battery life (hrs)	LOSP initiator frequency	Other features
Ginna	7E-6	2/2/N	Unknown	.0035	• technical support center and security diesel generators
Kewaunee	3E-5	2/1/N	8	.044	• technical support center diesel generator can provide RCP seal cooling
Point Beach 1&2	1E-5	0/2/Y	1	.06	• on-site gas turbine generator • turbine-driven AFW manually controlled after battery depletion • relatively small CST limits AFW to 4 hours in SBO
Prairie Island 1&2	3E-6	2/0/Y	2	.065	• large CST

[1] The number of diesel generators dedicated to each unit, other diesel generators such as shared or safety shutdown facility, and the presence of cross connects between units (Y for cross connects, N for no cross connects) are noted.

Failure of on-site AC power is also part of the SBO definition, and since diesel generators are used as the backup AC power source for all of the Westinghouse 2-loop plants, diesel generator failures are dominant contributors to SBO CDF for all of the Westinghouse 2-loop plants. The dominant contributors to diesel generator failures for the Westinghouse 2-loop plants involve the failure of the diesel generator to start (multiple random failures or common cause failure), failures of the diesel generators to continue to run for a sufficient time interval after successfully starting, diesel generator unavailability because of testing or maintenance, and failure of diesel generator cooling. The redundancy introduced by having additional diesel generators or cross-ties to diesel generators at a second unit at a site reduces the probability of failing all diesel generators, and so reduces the SBO CDF. This is particularly noticeable for Ginna and Prairie Island 1&2, which have relatively low SBO contributions, in part because of the availability of multiple, diverse diesel generators. The number of diesel generators available for each of the Westinghouse 2-loop plants is listed in Table 11.27.

Early failures of AFW lead to short-term SBO sequences while delayed failures lead to long-term SBO sequences. Dominant contributors to early failures include turbine-driven AFW pump failure to start or run, failure to cross-tie AFW between units, and pump unavailability associated with testing and maintenance.

Late AFW failures are most commonly caused by the inability to control AFW when DC power is lost as a result of battery depletion. Thus, plants with a longer battery life have more time available to restore off-site power before core cooling is lost, and have lower threat from SBO. Table 11.27 lists the battery life for the Westinghouse 2-loop plants. Manual control of AFW after battery depletion is considered for one plant, Point Beach 1&2, so that the battery life is not critical to successful plant shutdown. However, the CST capacity is relatively small, and therefore late AFW failures are dominated by the loss of a feedwater source when the tank supply is exhausted.

SBO sequences with stuck-open PORVs are not listed as dominant contributors to the SBO CDF in the Westinghouse 2-loop IPE submittals. RCP seal LOCAs are dominant contributors to the SBO CDF for one of the Westinghouse 2-loop IPEs. The size and timing of the leaks varies considerably among the submittals, but RCP seal LOCAs always increase the SBO CDF by reducing the time available for recovering off-site power before core damage would result. Three factors tend to reduce the SBO CDF that involves RCP seal LOCAs:

- using a seal LOCA model that gives lower leak rates and probabilities (e.g., Westinghouse seal LOCA model)
- having a backup source for pump seal cooling
- crediting installation of new, temperature-resistant o-rings

The Westinghouse 2-loop IPEs all use the Westinghouse seal LOCA model, which reduces the contribution from SBO sequences relative to that which would be calculated with other models, such as those that are based on NUREG-1150. However, the implementation of the seal LOCA model varies among the Westinghouse 2-loop plants, giving variability in the seal LOCA probability and leak rate. None of the IPEs reflect use of the new o-rings, which would reduce the SBO CDF. Two plants have alternate sources for RCP seal cooling during SBO. The technical support center diesel generators can be used to provide seal injection (which removes the seal LOCA concern).

Differences in results for the plants with the lowest and highest SBO CDFs are driven by differences in both plant design and IPE modeling characteristics. The plants with the lowest and highest SBO CDFs are Prairie Island 1&2 and Kewaunee, respectively. There are differences in plant characteristics that can partially explain the difference in SBO CDFs between the two plants, but some of the difference is also attributable to differences in IPE modeling.

The low SBO contribution for Prairie Island 1&2 primarily results from the availability of multiple, diverse diesel generators that can be cross-tied between units. The higher SBO contribution for Kewaunee (relative to the other Westinghouse 2-loop plants) is primarily caused by the specific implementation of the Westinghouse seal LOCA model in the IPE, which results in a relatively high probability of seal LOCAs.

11.3.3.3 Transient Perspectives

Transient sequences involve events that cause the reactor to trip (initiators) followed by failure to bring the reactor to safe shutdown (excluding SBO sequences, which are treated separately). The transient accident class is 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). After reactor trip, decay heat must be removed from the RCS. Normally, this would be provided by steam generator heat removal, with a fallback of primary system feed-and-bleed should secondary heat removal fail. If neither of these approaches succeeds, the RCS inventory will boil off, leading to core damage. A second possible path to core damage for transients involves an induced LOCA during the transient (SORV or RCP seal leak) with failure to make up the RCS inventory. For sequences involving feed-and-bleed cooling or an induced LOCA, both short-term and long-term coolant makeup (i.e., switchover from injection to recirculation) must be provided or core damage will result.

On average, transients are the largest contributor to plant CDF for the Westinghouse 2-loop plants, which is not unexpected since the transients encompass the most frequent initiators. The CDF from transients varies by less than a factor of five for the Westinghouse 2-loop plants. However, there is wide variability as to the specific failures that lead to core damage for transients. There is a significant contribution from support system failures (either as initiators or as failures subsequent to some other initiator) for all of the Westinghouse 2-loop plants. The results indicate that plant-specific dependencies are important to the results, however, different IPE modeling also affects the results. The results are mostly heavily driven by differences in SW and CCW dependencies, RCP seal leakage modeling in the IPEs, dependency on instrument air, and plant-specific configurations for system dependencies and cross-ties. The remainder of this section gives more details regarding these perspectives for transients.

Transients cover a broad group of sequences, including both general and plant-specific initiators. Some initiators occur from loss of support systems such as SW, CCW, HVAC, instrument air, or AC/DC buses. Because these support systems are needed for a large number of front line systems, their unavailability can simultaneously fail numerous front line systems, leaving few options for successful plant shutdown. The dependencies are very plant-specific. The remaining initiators are more general, such as reactor trip, turbine trip, or loss of main feedwater. The general initiators typically have higher frequencies than the support system initiators, but more failures are needed (subsequent to the initiator) to result in core damage. LOSP is an exception in that it is usually a lower frequency event, but often has a higher potential to proceed to core damage because failures of one diesel generator often lead to unavailability of full trains of safety systems. Table 11.28 lists the relative contribution of general transients and transients initiated by support system failures, and indicates that both types of transients are important for the Westinghouse 2-loop plants.

Table 11.28 Contributors to Westinghouse 2-loop transients.

Plant	CDF (1/ry)	
	General transients associated with balance of plant	Support system initiating events
Ginna	2E-6	2E-5
Kewaunee	8E-6	3E-6
Point Beach 1&2	3E-5	2E-5
Prairie Island 1&2	1E-5	7E-6

[1] Total Transient CDF = Generic Transients CDF + Specific Transients CDF. Transient CDF does not include SBO CDF and does not include ATWS CDF.
[2] CDF was estimated for these entries.

For transient sequences with failure of heat removal through the steam generators and failure of primary feed-and-bleed, the dominant contributors widely vary, representing considerable variability in plant design and operation as well as variability in IPE modeling. By examining the results reported in the IPE submittals, it was found that there is wide variability as to the specific failures that lead to core damage for transients with failure to remove decay heat in the Westinghouse 2-loop plants. The results indicate that plant-specific dependencies are important to the results, but also that different IPE modeling affects the results. Some of the observed contributors to failures of steam generator heat removal and failure of primary feed-and-bleed are discussed in the following paragraphs.

The dominant contributors to failure of steam generator heat removal that are listed in the IPE submittals include (1) loss of main feedwater because of LOSP, instrument air, or SW (2) random failures to run of the turbine-driven AFW pumps and (3) failure to establish long-term water supply for AFW.

The dominant contributors listed in the IPE submittals for failure to provide feed-and-bleed cooling if steam generator heat removal is lost are failure of injection pumps resulting from loss of instrument air or loss of SW, unavailability of PORVs because of loss of instrument air, and operator failure to successfully perform the feed-and-bleed operation. The failures primarily occur in the injection phase, rather than in recirculation.

Transient sequences with RCP seal LOCAs or SORVs are not dominant contributors to the transient CDF for most of the Westinghouse 2-loop plants. The IPE submittals do not list a large contribution from transients with induced LOCAs (RCP seal LOCAs or SORVs) for most of the Westinghouse 2-loop plants. However, transients with SORVs are important contributors for one plant, Ginna. Plant-specific dependencies are important to the results, but also different IPE modeling affects the results. Some of the observed contributors to transients with induced LOCAs and the subsequent failure to provide makeup are discussed in the following paragraphs.

The pump cooling configurations are similar for all of the Westinghouse 2-loop plants, with the RCPs cooled by either CCW or by the charging pumps. The charging pumps are air-cooled for this plant group, whereas other Westinghouse plants are normally cooled by either SW or CCW. This reduced dependency on CCW and SW for pump seal cooling reduces the contribution to the transient CDF from RCP seal LOCAs for the Westinghouse 2-loop plants relative to the other Westinghouse plant groups. The Prairie Island 1&2 charging pumps are indirectly dependent on SW because failure of SW results in a delayed failure of AC to the charging pumps. However, this dependence does not significantly increase the CDF from RCP seal LOCAs reported in the Prairie Island 1&2 IPE submittal. Because the RCPs in the Westinghouse 2-loop plants have less dependence on SW and CCW, the dominant contributors to transients with RCP seal LOCAs involve both loss of CCW and independent failure of the charging pumps.

The low CDF for transients involving RCP seal LOCAs is also attributable to the modeling of RCP seal leakage in the IPEs. The Westinghouse seal LOCA model is used in all of the IPEs for this plant group, and this model generally results in lower seal failure rates and leakage rates than other models used in IPEs for Westinghouse plants. However, the implementation of the Westinghouse model varies somewhat among the Westinghouse 2-loop plants and has a significant impact on the results. For example, a relatively low leak rate from the RCPs is used in the Prairie Island 1&2 IPE for cases with seal LOCAs, giving considerable time for recovery of failed equipment.

Although not generally the dominant contributor, sequences with SORVs are substantial contributors to the transient CDF for the Westinghouse 2-loop plants. The pressurizer PORVs rely on instrument air for all plants in this group, so failure of instrument air causes the PORVs to fail closed. Without the PORVs available to relieve the RCS pressurization, there are more demands on the SRVs, which can not be isolated if they stick open. Therefore, there is a higher probability of transient-induced LOCAs that cannot be isolated (relative to plants without this instrument air dependency). The Ginna IPE shows the greatest dependence on instrument air, with about a quarter of the plant's CDF initiating with failure of instrument air and leading to loss of the PORVs.

The results for the plants with the highest and lowest CDFs for transients are primarily driven by differences in plant characteristics. Point Beach 1&2 has the highest transient CDF for this plant group, and Kewaunee has the lowest transient CDF. The higher transient CDF for Point Beach 1&2 primarily reflects two design features that are different between the two plants, but modeling differences also appear to be important.

The first difference is the size of the CST. The Point Beach CST is modeled as supplying feedwater for 4 hours, while the Kewaunee feedwater supply is modeled as lasting 16 hours. The possibility of using backup feedwater sources is included in both IPEs but less time is available for Point Beach.

The second difference involves the dependence on SW. Point Beach requires SW for bearing cooling to both motor-driven AFW pumps. Only one motor-driven AFW pump depends on SW for Kewaunee, with SW needed for room cooling for that pump.

The difference in transient CDFs between the two plants also reflects modeling differences. The Point Beach 1&2 IPE uses a higher probability for operator failure to both provide a long-term source of water for AFW and establish feed-and-bleed cooling.

11.3.3.4 LOCA Perspectives

LOCA core damage sequences encompass any breaks in the RCS. Normally, stuck-open pressurizer relief valves or reactor coolant seal leaks that initiate an accident are categorized as LOCAs in the IPE submittals. If either of these occurs following another initiator (e.g., LOSP), they are normally categorized as an SBO or transient, as appropriate. After a LOCA is initiated, inventory makeup is needed to prevent the core from uncovering and proceeding to core damage. In addition, it is necessary to remove the decay heat from the RCS. For large breaks, decay heat can be removed through the break so that only inventory makeup is a concern. Smaller breaks exhaust less energy, and so supplemental cooling is needed. This is normally provided by steam generator cooling or by primary system feed-and-bleed. Injection to the RCS is initially from a source outside containment (e.g., RWST), but this source is ultimately depleted, and the plant must switch from this injection mode to the recirculation mode, in which water is drawn from the containment sump. Unless the injection mode is successful, however, there will not be sufficient water in the sump to provide RCS makeup during recirculation.

LOCAs are a significant contributor to plant CDF for the Westinghouse 2-loop plants. The LOCAs are most often small or medium LOCAs, and recirculation failures are more common than injection failures. Both actual plant characteristics and IPE modeling affect the results for this plant group. The factors that have the greatest impact on the variation in LOCA CDFs among the plants are (1) the size of the RWST, (2) whether the plant has the capability for refilling the RWST, (3) whether the plant has the capability for depressurizing the RCS through the steam generator secondaries for small LOCAs, and (4) whether credit was given for such actions in the IPEs when the capability exists at the plants.

The most common type of LOCA is a small LOCA with failure of recirculation, but some plants are instead dominated by medium or large LOCAs. For this report, the following sizes of LOCAs are considered:

- Large - large enough to remove decay heat through the break, and also large enough to depressurize the system on its own, which allows injection from low-pressure systems (typically, greater than a 6-inch diameter)

- Medium - the break is large enough to remove decay heat through the break, but not large enough to depressurize the system (typically, a 2- to 6-inch diameter)

- Small - the break is not large enough to depressurize the system before core damage would occur so the RCS must be depressurized by some other means if LPI is needed; the break does not remove sufficient energy to cool the RCS, so some other means is required to provide the energy removal (typically, a 1/2- to 2-inch diameter).

The CDF contributions from small, medium, and large LOCAs are listed in Table 11.29 for the Westinghouse 2-loop plants. Small LOCAs are the dominant break size for two of the IPEs, large LOCAs are dominant for one of the IPEs, and the remaining IPE has about equal LOCA contributions from the three LOCA size categories. Recirculation failures are dominant for all of the Westinghouse 2-loop plant IPEs.

The most common system and component failures for LOCAs are related to the switchover to recirculation, but failures during injection are also important. Some of the system and component failures that are dominant contributors to the LOCA CDF are common across the plants, while others are highly plant-specific. The failures that contribute most to LOCAs for the Westinghouse 2-loop plants can be grouped into the categories listed below. The system and component failures that contribute to each are discussed in the following paragraphs.

- LOCA initiator
- failure of injection
- failure of recirculation
- failures of alternate actions that could be used to mitigate the above failures

Table 11.29 LOCA contributors for Westinghouse 2-loop plants.[1]

Plant	CDF from large LOCAs (1/ry) [3]	CDF from medium LOCAs (1/ry)	CDF from small LOCAs (1/ry)
Ginna	3E-6	6E-6	1E-5
Kewaunee	2E-6	8E-6	1E-5
Point Beach 1&2	3E-5	1E-5	2E-6
Prairie Island 1&2	4E-6	5E-6	4E-6

[1] Not including SGTR and ISLOCAs
[2] Total LOCA CDF = Large LOCA CDF + Medium LOCA CDF + Small LOCA CDF
[3] Includes Vessel Rupture

The initiator frequencies for the small, medium, and large break sizes for the Westinghouse 2-loop plants are listed in Table 11.30. The plants generally use generic data to quantify the LOCA initiator frequencies. For plants that subdivide small LOCAs into two categories, the small LOCA initiator frequency represents the sum of the individual initiator frequencies.

Table 11.30 LOCA initiator frequencies for Westinghouse 2-loop plants.

Plant	LOCA initiator frequency		
	Large	Medium	Small
Ginna	2E-4	4E-4	1E-3
Kewaunee	5E-4	2E-4	5E-3
Point Beach 1&2	5E-4	1E-3	3E-3
Prairie Island 1&2	3E-4	8E-4	3E-3

LOCAs with failure in recirculation are larger contributors than LOCAs with injection failure. This is reasonable because the use of recirculation is normally more complicated than injection. For HPR in the Westinghouse 2-loop plants, the operators have to realign the systems such that the low-pressure pumps draw suction from the sump and align the high-pressure pumps to take suction from the low-pressure pumps. This switchover is manual for all of the Westinghouse 2-loop plants. The dominant contributors found in the IPE submittals for injection failures include random and common cause pump failures, and pump cooling failures (e.g., SW or CCW). Dominant contributors to recirculation failures in LOCAs that were found in the IPE submittals include pump failures, pump cooling failures, sump strainer plugging, and (most commonly) operator error in recirculation switchover.

Some licensees credited altern... actions that could be used to mitigate a LOCA when the above-mentioned standard approaches do not succeed. Particularly, some licensees took credit in the IPE for depressurizing the RCS using the steam generators relief valves when HPI fails during a LOCA.

The highest and lowest LOCA CDFs primarily reflect IPE modeling differences. The Prairie Island 1&2 IPE has the lowest CDF from LOCAs and the Point Beach 1&2 IPE has the highest. The design characteristics relevant to LOCAs are similar for both plants, and the difference in results primarily reflects a pessimistic value for the probability of the operator failing to successfully perform the switchover from injection to recirculation in large LOCAs.

11.3.3.5 ATWS Perspectives

An ATWS is an accident that is initiated by an event, such as loss of feedwater or turbine trip, followed by failure of the reactor to scram. The failure to scram can result either from failures of electrical components in the RPS or in the final control elements that de-energize the CRD mechanisms, or from mechanical failures involving failure of de-energized control rods to drop into the core. Partial scram failures are possible, involving insertion of a subset of the control rods; however, in most PRAs it is assumed that the scram failure is total and involves all of the rods. With failure of the rods to insert, power far in excess of decay heat loads is still being generated and overpressurization of the RCS is possible in some cases.

ATWS is not a major contributor to the total CDF for the plants in this group. ATWS frequencies for this group range from 7E-8/ry to 3E-7/ry and contribute negligibly to the total CDF. ATWS sequences are generally low contributors for all PWRs unless the plant has a specific weakness. This primarily involves plants that operate with the PORV block valves closed, which reduces the pressure relief capability. None of the Westinghouse 2-loop plants operates with PORV block valves closed, and therefore, ATWS is a small contributor for these plants.

11.3.3.6 Internal Flood Perspectives

An internal flood sequence involves a release of water into a plant location such that a plant trip is induced and safety systems are compromised at the same time. For example, if a flood causes water to enter an area containing electrical switchgear, a plant trip can occur along with failure of all plant systems dependent upon that switchgear. Systems that are considered to be independent may all fail, if they all contain equipment within the flooded location. The effects of internal flooding are highly plant-specific, depending on the layout of equipment within the plant and the relative isolation of rooms. Often, the systems most affected are support systems such as electric power and SW, which have plant-specific designs. Because of this diversity of design and layout, we expect that each plant will have different vulnerabilities to flooding and do not expect to draw many generic conclusions.

Internal flooding is an important contributor to the total CDF at some plants. The importance of internal flooding to the total CDF is highly plant-specific, depending on the layout of water bearing lines and vessels, flood propagation, and the locations of equipment needed to provide for shutdown cooling. Typically, important internal floods are those that affect important support systems, such as SW and electric power. Newer plants sometimes reflect better design characteristics, generally better separation and tend to be less susceptible to floods.

The contribution from flooding varies from negligible to about 20% of the CDF. The CDF from flooding ranges from 2E-7/ry to 1E-5/ry for the Westinghouse 2-loop plants. As indicated in the table, internal flooding is not a major contributor to the CDF for many plants, but for some plants it is an important contributor, comprising as much as about 20% of the total CDF.

The highest CDF from internal flooding for the group is 1E-5/ry at Point Beach 1&2 and Prairie Island 1&2. Two sequences comprise most of the flooding contribution for Point Beach 1&2. The first sequence is a SW pipe break in the auxiliary building that continues undetected for more than 8 hours, resulting in loss of all RCP seal cooling and loss of all injection to mitigate a seal LOCA. The second sequence is a circulating water expansion joint failure in the water intake facility that results in loss of SW and firewater. The largest contributor to flooding for Prairie Island 1&2 involves a flood in the AFW pump/instrument air compressor room that leads directly to core damage.

The lowest CDF from internal flooding for the group 2E-7/ry for Kewaunee. The dominant contributor is a sequence that involves failure of a SW expansion joint that floods a diesel generator room, failing a single train of several different systems.

11.3.3.7 SGTR Perspectives

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. There are several actions that are normally taken to prevent the tube rupture initiator from developing into a core melt sequence. To mitigate the leak, the affected steam generator is normally isolated, and if this does not succeed, actions are taken to depressurize the primary system so that leakage is minimized. The primary system is depressurized either by aggressively cooling down through the unaffected steam generator(s), cooling down the primary system using the pressurizer sprays, or by depressurizing by initiating feed-and-bleed through the pressurizer PORV(s). If the affected steam generator is isolated, long-term core heat removal can be established without inventory makeup, but if the leak is not isolated, inventory makeup must also be provided.

SGTR sequences are important contributors to plant CDF and risk for some of the Westinghouse 2-loop plants. SGTR sequences contribute more to plant CDF for the Westinghouse 2-loop plants than is typical of other PWRs. A relatively small SGTR contribution can be an important contribution to risk because the releases bypass containment. The individual plant results vary from a high of 3E-5/ry to a low of 5E-6/ry.

Variability in SGTR CDFs primarily reflects the effects of modeling uncertainties. The Westinghouse 2-loop plant with the highest SGTR CDF is Ginna, with a CDF of 3E-5/ry. The remaining Westinghouse 2-loop plants have SGTR CDFs in the range 5E-6/ry to 7E-6/ry. Modeling of SGTRs involves considerable uncertainty for operator actions to isolate the steam generators, and to cool down and depressurize the RCS. An additional uncertainty involves the probability of the steam generator relief valves reclosing after passing water through them. The variation in SGTR CDFs reflects differences in modeling these uncertain aspects of SGTR sequences.

11.3.3.8 ISLOCA Perspectives

An ISLOCA occurs when valves that normally isolate the RCS from low-pressure systems (e.g., LPI) fail, resulting in backflow from the RCS through the low-pressure piping. If the low-pressure piping or other components (e.g., seals, relief valves, or flanges) can not withstand the resulting pressurization, then a LOCA results. If the breach occurs in a portion of the piping that is outside containment, then a LOCA that bypasses containment results, with effluent from the LOCA being discharged into the reactor or auxiliary building. Because it is not possible to recirculate the coolant through the RCS for this type of LOCA, coolant injection will eventually be lost (leading to core damage) unless some source of sustained makeup is available. These scenarios are typically low probability events, because they involve multiple valve failures (typically two check valves and one motor-operated valve in series). However, ISLOCAs can be important to risk because the containment bypass leads to larger fission product releases.

ISLOCAs are minor contributors to plant CDF in all of the IPE submittals except Ginna, which has a 10% contribution from ISLOCAs. This relatively large contribution is important to risk because the releases bypass containment. The relatively high ISLOCA frequency for Ginna is attributable to the relatively large number of lines with the potential for an ISLOCA.

11.3.4 CDF Perspectives for Westinghouse 3-loop Reactors

As indicated in Table 11.31, the Westinghouse 3-loop group of plants consists of nine IPE submittals. Three of these submittals are for single-unit sites, four are for dual-unit sites with the same results reported for each unit, and two cover a dual unit site (Beaver Valley 1&2) with different analyses performed for each unit. Accounting for the dual unit sites, the Westinghouse 3-loop group comprises thirteen plant units. All of the plants in this group have either a large dry or subatmospheric containment.

Table 11.31 Plants in Westinghouse 3-loop plant 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

11.3.4.1 Summary of Results and Perspectives for Westinghouse 3-loop Reactors

The total CDFs for the plants in the Westinghouse 3-loop group are shown in Figure 11.6. The CDFs for this plant group show a moderate variability, with about a factor of 5 spread between the highest and lowest CDFs. The average CDF is 2E-4/ry, which is the highest average CDF for the PWR plant groups. The contribution of individual accident classes to plant CDF also shows considerable variability, as depicted in Figure 11.10. The variation in the accident class CDFs is attributable to a combination of plant design differences and modeling assumptions. Overall, the largest contributors are transients, LOCAs, and SBOs. The IPE submittals generally listed smaller contributions from ATWS, SGTRs, internal flooding, and ISLOCA sequences.

Table 11.32 summarizes the perspectives obtained from examination of the Westinghouse 3-loop plants. The details that provide the basis for these perspectives are contained in the remainder of Section 11.3.4. The results for the accident classes are discussed, giving more details on the factors driving the CDFs, particularly for the plants with the highest and lowest CDFs. Design and operational factors along with differences in modeling assumptions are addressed.

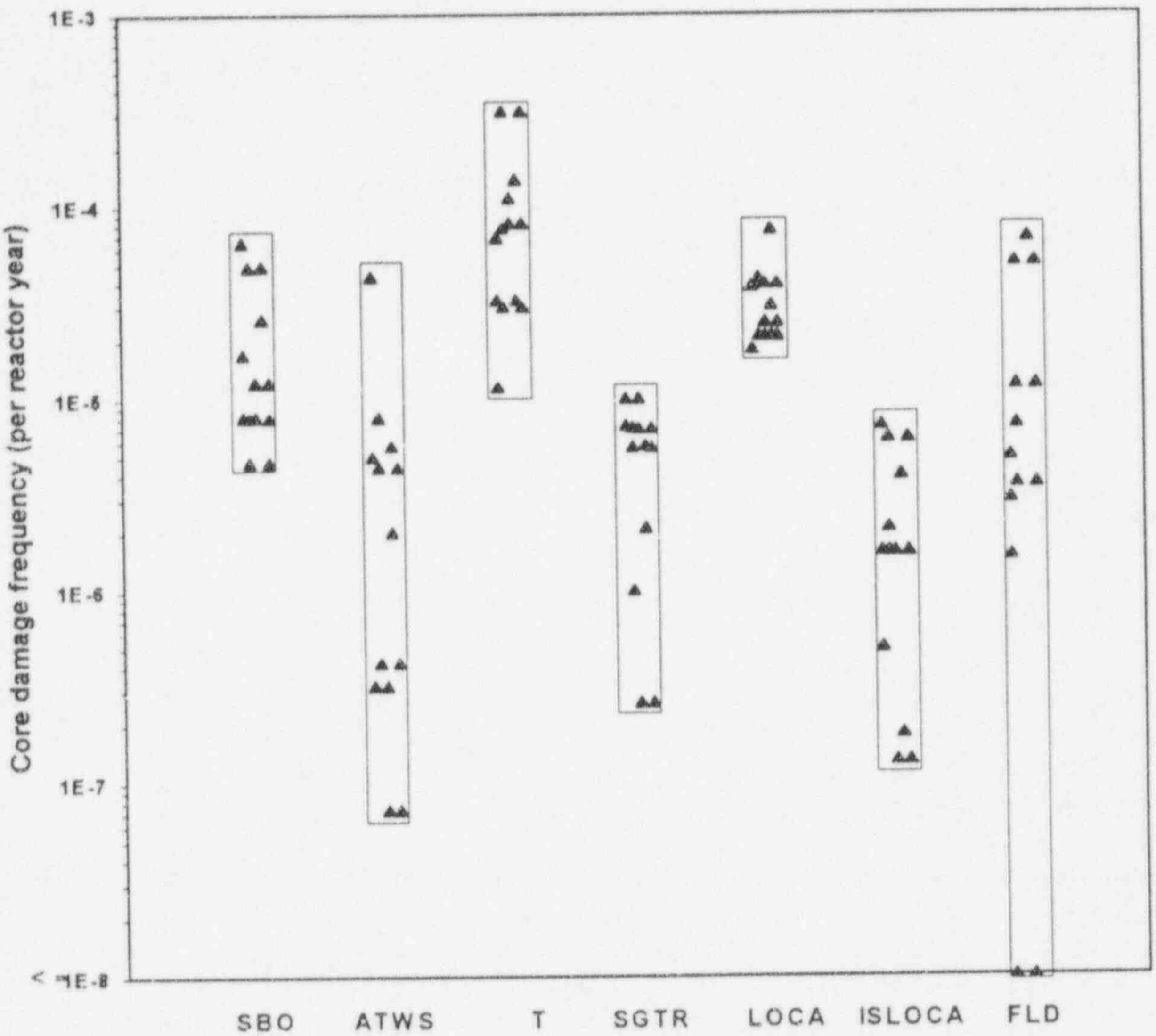


Figure 11.10 Sequence results for Westinghouse 3-loop plants.

Table 11.32 **Key IPE observations for Westinghouse 3-loop plants.**

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
	ATWS accidents	
Not dominant for most Westinghouse 3-loop plants	Plant operation with PORV block valves closed	ATWS CDFs range from 7E-8/ry to 4E-5/ry. Average CDF is 6E-6/ry. Contribution to total plant CDF ranges from negligible to 20%

Table 11.32 Key IPE observations for Westinghouse 3-loop plants.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
	Transient accidents	
Important for all Westinghouse 3-loop plants	Degree of HPI and AFW 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)	Transient CDFs range from 1E-5/ry to 3E-4/ry. Average CDF is 1E-4/ry. Contribution to total plant CDF ranges from 15% to 85%
	SBO accidents	
Important for many Westinghouse 3-loop plants	AC power reliability (e.g., 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	SBO CDFs range from 5E-6/ry to 7E-5/ry. Average CDF is 2E-5/ry. Contribution to total plant CDF ranges from negligible to 30%
	LOCAs	
Important for most Westinghouse 3-loop plants	Most plants have automatic switchover for LPR, but require manual actions for switchover to HPR Ability to depressurize the RCS by aggressively cooling down using steam generator ADVs so LPI can be used when HPI fails Ability to refill the RWST	LOCA CDFs range from 2E-5/ry to 8E-5/ry. Average CDF is 3E-5/ry. Contribution to total plant CDF ranges from 10% to 45%
	Internal flood accidents	
Important for some Westinghouse 3-loop sites	Plant layout: separation of mitigating system components and compartmentalization	Flood CDFs range from negligible to 7E-5/ry. Average CDF is 2E-5/ry. Contribution to total plant CDF ranges from negligible to 40%

Table 11.32 Key IPE observations for Westinghouse 3-loop plants.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
SGTR accidents		
Not important for Westinghouse 3-loop plants	Modeling of operator actions to isolate the rupture and provide long-term heat removal	SGTR CDFs range from 3E-7/ry to 1E-5/ry. Average CDF is 5E-6/ry. Contribution to total plant CDF ranges from negligible to 10%
ISLOCAs		
Not important for any Westinghouse 3-loop plant	Compartmentalization and separation of equipment	ISLOCA CDFs range from 1E-7/ry to 7E-6/ry. Average CDF is 3E-6/ry. Contribution to total plant CDF is less than 5% for all plants

11.3.4.2 SBO Perspectives

SBO occurs when a plant loses off-site AC power, and the on-site backup AC power sources (almost always diesel generators) also fail to function. Because so many safety systems rely on AC power either directly (for motive power) or indirectly (e.g., cooled by systems that rely on AC power), SBO causes most safety systems to be unavailable. Most SBO scenarios do have DC power available, however, through use of station batteries. For PWRs, the loss of all AC power normally leaves turbine-driven AFW as the only available means for core cooling. Turbine-driven AFW can operate until the batteries deplete, leading to a loss of control. If AC power is not recovered soon after loss of control, core damage will follow. Some licensees assume that AFW can continue even after battery depletion, by manually controlling feedwater. Even with such manual control, the SBO sequence might still proceed to core damage if there is not a continuous supply of water available for feedwater injection.

An additional, important, complication for Westinghouse plants involves the potential for leakage from RCP seals. SBO results in loss of cooling for the RCP seals. The prolonged exposure to high temperatures can fail the seals, leading to a small LOCA through the pump seals. If a seal LOCA occurs, injection systems are needed to provide makeup to the RCS. Since these are not normally available during an SBO, pump seal LOCAs will lead to core damage if off-site power is not restored in time. Chapter 17 of this report provides further discussion of SBO sequences.

SBO is an important contributor to plant CDF for many, but not all, of the Westinghouse 3-loop IPEs. Several factors are important for SBOs for the Westinghouse 3-loop plants, with differing combinations of these factors

driving the results for the individual plants in this group. The factors that have the biggest impact on the results primarily represent actual plant characteristics, but the modeling used in the IPEs is also important. The key factors are the modeling of RCP seal LOCAs (probability of leak, size of leak, and timing of subsequent core uncovery), availability of independent emergency power (e.g., emergency response facility) that provides support for AFW, battery life (actual plant design as well as load shedding assumptions), and cross-ties (existence of cross-tie capability and whether it was credited in the IPE).

SBO sequences can be categorized as short-term, long-term, or involving RCP seal LOCA. On average, long-term SBO sequences with RCP seal LOCAs are most frequent. To understand the reasons for differences in SBO CDFs, it is helpful to categorize SBOs as follows:

- short-term SBO - sequences in which turbine-driven AFW pump trains fail to operate and AC power is not recovered before the core is damaged

- long-term SBO - sequences in which turbine-driven AFW pump trains operate initially, but ultimately fail (normally because battery depletion causes loss of DC control power) before recovery of AC power

- SBO with RCP seal LOCA - sequences involving RCP seal LOCAs that are caused by loss of cooling to RCP seals with failure to recover AC power before the core is uncovered.

Short-term SBO sequences typically lead to core damage within about 2 hours, while long-term SBO sequences typically lead to core damage in more than 2 hours. SBO sequences with RCP seal LOCAs can be either short-term or long-term (depending on the model used for the timing and magnitude of a RCP seal LOCA), but they are typically long-term sequences.

The relative fractions of the SBO accident class that fall within these categories is not readily available from the IPE submittals for most plants. However, most submittals qualitatively indicate that short-term SBO sequences are not major contributors to the SBO CDF, because more failures are needed for the short-term case (e.g., turbine-driven AFW must fail). The exceptions are North Anna 1&2 and Surry 1&2. For the plants that are dominated by long-term SBO, all are dominated by RCP seal LOCAs. Therefore, on average, the SBO CDF for the Westinghouse 3-loop plants is dominated by long-term SBOs with RCP seal LOCAs.

The SBO results are driven by a combination of factors involving both plant-specific features and IPE modeling characteristics. Some of the system and component failures that are dominant contributors to the SBO CDF are common across the plants, while others are highly plant-specific. The failures that contribute most to SBO for the Westinghouse 3-loop plants can be grouped into the categories listed below:

- LOSP initiator
- loss of on-site AC power
- failure to recover AC power
- failure of turbine-driven AFW
- transient-induced LOCA

The system and component failures that contribute to each of the categories are discussed in the following paragraphs.

The initiator for SBO is direct LOSP or failure of switchgear ventilation leading to SBO. Higher frequencies for LOSP will lead to higher SBO CDFs. The frequencies used for LOSP for each of the Westinghouse 3-loop plants are listed in Table 11.33. As can be seen, the frequencies vary by a factor of about 3. SBO sequences resulting from failure of switchgear ventilation are significant contributors for Beaver Valley 1 and North Anna 1&2.

Table 11.33 Key parameters affecting SBO CDF for Westinghouse 3-loop plants.

Plant	SBO CDF (1/ry)	No. EDGs unit/other/ x-tie[1]	Battery life (hrs)	LOSP initiator frequency	Pump seals[3]	Other features
Beaver Valley 1	7E-5	2/1/N	3	.0664		• dedicated feedwater pump powered off emergency response facility • has diesel generator backup for AFW • emergency switchgear room needs HVAC
Beaver Valley 2	5E-5	2/1/N	8	.0744		• dedicated feedwater pump powered off emergency response facility • has diesel generator backup for AFW • emergency switchgear room needs HVAC
Farley 1&2	1E-5	1/3/N	2	.047	1	• 1 diesel generator dedicated to each unit, 3 swing • old o-ring material for RCPs
H. B. Robinson	3E-5	2/1/N	8	.061	2	• dedicated diesel generator for powering 1 SW, CCW and charging pump for seal cooling and makeup
North Anna 1&2	8E-6	2/0/Y	2	.114	1,2	• diesel generators are self-contained and self-cooled • emergency switchgear room needs HVAC • turbine-driven AFW pump operates for 9 hr after battery depletion • can cross-tie seal cooling and HPI
Shearon Harris	2E-5	2/0/N	4	.05		
Summer 1&2	5E-5	2/0/N	4	.073	1	• turbine-driven AFW pump continues to operate after battery depletion • diesel-driven firewater serves as backup for cooling charging
Surry 1&2	8E-6	1/1/Y	4	.0769	1	• 1 diesel generator dedicated to each unit and 1 swing • emergency switchgear room needs ventilation
Turkey Point 3&4	5E-6	2/0/Y	Unk.	.17		• diesel generators have self contained cooling • has 5 black start diesel generators

[1] The number of diesel generators dedicated to each unit, other diesel generators such as shared or safe shutdown facility, and the presence of cross connects between units (Y for cross connects, N for no cross connects) are noted.
[2] Unk. = Unknown; information could not be obtained from IPE submittal.
[3] 1 = lower seal LOCA leak rate (primarily plants using Westinghouse seal LOCA model), 2 = backup means for cooling RCP seals

Failure of on-site AC power is also part of the SBO definition, and since diesel generators are used as the backup AC power source for all of the Westinghouse 3-loop plants, diesel generator failures are dominant contributors to SBO CDF for all of the Westinghouse 3-loop plants. The most common reasons for diesel generator failures involve the failure of the diesel generator to start (multiple random failures or common cause failure), failure of the diesel generator to continue to run for a sufficient time interval after successfully starting, diesel generator unavailability because of testing or maintenance, or failure of diesel generator cooling. The redundancy introduced by having additional diesel generators reduces the probability of failing all diesel generators, so reduces the SBO CDF. The number of diesel generators available for each of the Westinghouse 3-loop plants is listed in Table 11.33.

Some multi-unit sites have the ability to cross-tie emergency AC power between units, which reduces the SBO CDF by providing a redundant, somewhat independent power source. Not all licensees that have this capability actually took credit for the action to cross-tie in the IPE, however. Licensees that took credit for the ability to cross-tie are identified in Table 11.33. Note that the three IPEs with the lowest SBO CDFs are the only IPEs for which the capability to cross-tie between units is both present and credited in the analysis.

Although the probability of restoring off-site power is important to the SBO CDF, and would therefore be desirable to compare among the plants, the necessary information is not readily available from most of the IPE submittals.

Early failures of AFW lead to short-term SBO sequences while delayed failures lead to long-term SBO sequences. Early failures normally involve pump failure to start or run. These failures have sufficiently low frequency to cause most plants to have a larger contribution to SBO CDF from long-term SBO sequences rather than these short-term sequences.

Late AFW failures are most commonly caused by the inability to control AFW when DC power is lost as a result of battery depletion. Thus, plants with a longer battery life have more time available to restore off-site power before core cooling is lost, and have lower threat from SBO. Table 11.33 lists the battery life for the Westinghouse 3-loop plants. In some cases, the CST capacity is more limiting than the battery life so that late AFW failures are dominated by the loss of a feedwater source when the tank supply is exhausted. Similarly, some licensees take credit in the IPE for manually controlling AFW after battery depletion, so that the battery life is not modeled as being critical to successful plant shutdown. For these plants without a dependence on battery life, the CST capacity and whether the plant has the ability to refill the condensate tank are important. Information on these factors is provided in Table 11.33.

SBO sequences with stuck-open PORVs are only listed in a few of the submittals, and even then they are minor contributors. Pump seal LOCAs are dominant contributors to the SBO CDF for many of the Westinghouse 3-loop IPEs. The size and timing of the leaks varies considerably among the submittals, but pump seal LOCAs always increases the SBO CDF by reducing the time available for recovering off-site power before core damage would result. Three factors tend to reduce the SBO CDF that involves seal LOCAs:

- using a seal LOCA model that gives lower leak rates and probabilities (e.g., Westinghouse seal LOCA model)
- having a backup source for pump seal cooling (see following paragraph)
- crediting installation of new, temperature-resistant o-rings.

Table 11.33 indicates whether any of these factors apply to a particular plant in the Westinghouse 3-loop group. It is important to note that the IPE submittals do not always contain a thorough description of seal LOCA modeling, and so some of the factors may also apply to other plants besides those indicated in the table. From the table, it can be seen that many of the plants used the Westinghouse seal LOCA model, which tends to give lower seal leakage probabilities than models such as the NUREG-1150 model. None of the licensees appear to have taken credit for the new o-rings, but some performed sensitivity studies that are reported in the submittals and indicate that the new RCP seals would significantly reduce the SBO CDF. Two plants have a backup source for pump seal cooling (alternate diesel generator or cross-ties). The table indicates that these seal LOCA considerations have an important effect on the SBO CDF.

Differences in results for the plants with the lowest and highest SBO CDFs are primarily driven by IPE modeling characteristics. The plants with the lowest and highest SBO CDFs are Turkey Point 3&4 and Beaver Valley 1, respectively. The low Turkey Point 3&4 SBO contribution reflects the availability of 5 black start diesel generators. The IPE submittal indicates that these additional diesel generators reduce the SBO CDF from 2E-5/ry to 5E-6/ry. Beaver Valley has a higher SBO contribution than the other Westinghouse 3-loop plants because of the need for switchgear room cooling at Beaver Valley. Loss of switchgear ventilation leads to a consequential SBO at Beaver Valley, increasing the SBO CDF.

11.3.4.3 Transient Perspectives

Transient sequences involve events that cause the reactor to trip (initiators) followed by failure to bring the reactor to safe shutdown (excluding SBO sequences, which are treated separately). The transient accident class is a broad category, covering both general initiators (e.g., reactor trip or loss of main feedwater) as well as support system initiators (e.g., loss of SW or AC/DC bus). After reactor trip, decay heat must be removed from the RCS. Normally, this would be provided by steam generator heat removal, with a fallback of primary system feed-and-bleed should secondary heat removal fail. If neither of these succeeds, the RCS inventory will boil off, leading to core damage. A second possible path to core damage for transients involves an induced LOCA during the transient (SORV or RCP seal leak) with failure to make up the RCS inventory. For both types of sequences (failure of DHR and transient-induced LOCA with failure of reactor coolant makeup), long-term operability must also be maintained (e.g., switchover from injection to recirculation must succeed), or core damage will result.

Transients are the largest contributor on average to plant CDF for the Westinghouse 3-loop plants (average contribution of 45%), which is not unexpected since the transients encompass the most frequent initiators. The CDF from transients ranges by about a factor of 30. There is wide variability as to the specific failures that lead to core damage for transients, but there is a significant contribution from support system failures (either as initiators or as failures subsequent to some other initiator). The results indicate that plant-specific dependencies are important to the results, but also, different IPE modeling affects the results. The results are mostly heavily driven by differences in SW and CCW dependencies, RCP seal cooling designs, RCP seal leakage modeling in the IPEs, and plant-specific configurations for system dependencies and cross-ties. The remainder of this section provides more details regarding these perspectives for transients.

Transients cover a broad group of sequences, including both general and plant-specific initiators. Some initiators occur from loss of support systems such as SW, CCW, HVAC, instrument air, or AC/DC buses. Because these support systems are needed for a large number of front-line systems, their unavailability can simultaneously fail numerous front-line systems, leaving few options for successful plant shutdown. The dependencies are very plant-specific. The remaining initiators are more general, such as reactor trip, turbine trip or loss of main feedwater. The general initiators typically have higher frequencies than the support system initiators, but more failures are needed (subsequent to the initiator) to result in core damage. LOSP is an exception in that it is usually a lower frequency event, but often has a higher potential to proceed to core damage because failures of one diesel generator often lead to unavailability of full trains of safety systems. Table 11.34 lists the relative contribution of general transients and transients initiated by support system failures, and indicates that both types of transients are important for the Westinghouse 3-loop plants.

Table 11.34 Contributors to Westinghouse 3-loop transients.

Plant	CDF (1/ry)		Key plant characteristics
	General transients associated with balance of plant	Support system initiating events	
Beaver Valley 1	4E-5[2]	4E-5[2]	• charging also serves as HPI
Beaver Valley 2	4E-5[2]	4E-5[2]	• charging also serves as HPI
Farley 1&2	2E-5[2]	6E-5[2]	• old o-rings in RCPs • charging also serves as HPI • charging cooled by CCW, with firewater as backup
H. B. Robinson	7E-5[2]	8E-5[2]	• small CST; alternate AFW source needed long-term • charging can be cooled by firewater
North Anna 1&2	1E-5	2E-5	• charging also serves as HPI • charging is cooled by SW • no heat removal in ECC systems; containment cooling is through spray system cooling • can cross-tie HPI
Shearon Harris	5E-6[2]	4E-6[2]	• charging also serves as HPI • HPI has sufficient head to inject at SRV set point, so can feed-and-bleed without PORVs • large CST; can supply AFW for mission time
Summer	8E-5	2E-5	• charging also serves as HPI • turbine-driven main feedwater • charging can be cooled by firewater
Surry 1&2	1E-5	2E-5	• charging also serves as HPI • can feed-and-bleed with charging pumps • cross-ties for CCW and seal injection • charging cooled by separate system from CCW or SW • HPR and LPR depend on spray heat exchangers for cooling
Turkey Point 3&4	4E-5	3E-4	• AFW and HPI are shared between units • numerous cross-ties, including feedwater, instrument air, CCW, electric power

[1] Total Transient CDF = Generic Transients CDF + Specific Transients CDF. Transient CDF does not include SBO CDF and does not include ATWS CDF.
[2] CDF was estimated for these entries.

For transient sequences with failure of heat removal through the steam generators and failure of primary feed-and-bleed, the dominant contributors vary widely, representing considerable variability in plant design and operation as well as variability in IPE modeling. By examining the results reported in the IPE submittals, it was found that there is wide variability as to the specific failures that lead to core damage for transients with failure to remove decay heat in the Westinghouse 3-loop plants. The results indicate that plant-specific dependencies are important to the results, but also different IPE modeling affects the results. Some of the observed contributors to failures of steam generator heat removal and failure of primary feed-and-bleed are discussed in the following paragraphs.

The more common reasons for failure of steam generator heat removal that are listed in the IPE submittals include depleting the CST with operator failure to provide alternate water supply, feedwater pump failure to start, feedwater pump cooling failures, operator failure to recover main feedwater, and failure of feedwater actuation signal with operator failure to manually actuate feedwater.

The most common failures listed in the IPE submittals for failure to provide feed-and-bleed cooling if steam generator heat removal is lost are failure of injection pumps because of SW failure, operator failure to initiate feed-and-bleed, operator failure to establish recirculation, and failure during switchover from the volume control tank to the RWST. The failures primarily occur in the injection phase, rather than in recirculation.

For transient sequences with RCP seal LOCAs, the dominant contributors also widely vary, representing considerable variability in plant design and operation as well as variability in IPE modeling. The IPE submittals indicate a wide variability as to the specific failures that lead to core damage for transients with induced LOCAs (primarily RCP seal LOCAs) for the Westinghouse 3-loop plants. The results indicate that plant-specific dependencies are important to the results, but also different IPE modeling affects the results. Some of the observed contributors to RCP seal leakage and the subsequent failure to provide makeup are discussed in the following paragraphs.

The RCPs can be cooled by either CCW or by the charging pumps for the Westinghouse 3-loop plants. The charging pumps are normally cooled by either SW or CCW. For plants with charging pumps cooled by CCW, loss of CCW causes both a seal LOCA and loss of the charging system that would normally be used to provide makeup flow from the induced LOCA (effectively, such loss of CCW sequences behave like SBO sequences). The probability of the seal LOCA occurring for such sequences depends on a number of factors, including:

- the seal LOCA model used in the IPE (Westinghouse seal LOCA model gives lower probability of failure than NUREG-1150 model)
- whether backup systems are available to provide RCP cooling
- credit given for operator actions such as cross-tieing systems or cooling down the RCS to reduce the seal LOCA threat

Although these factors introduce variability into the IPE results, CCW failures are commonly seen as dominant contributors to the transient CDFs, which reflects the relatively high dependency of other plant systems on CCW. For example, the dominant transient sequence at Turkey Point 3&4 is loss of CCW leading to a seal LOCA that cannot be mitigated. Similarly, SW cooling is often found to be a dominant contributor because its loss compromises the ability to cool many plant systems (SW is generally also the heat sink for CCW). These sequences are the highest contributors on average to the Westinghouse 3-loop plant group, and the relatively large contribution from them drives the average Westinghouse 3-loop plant CDF above the average CDFs for the other plant groups.

Other contributors to transients with induced-LOCAs for Westinghouse 3-loop plants include failures during recirculation, failure of charging to switch from the volume control tank to the RWST, and PORV failure resulting from battery depletion.

The results for the plants with the highest and lowest CDFs for transients are driven by both differences in actual plant characteristics and IPE modeling. Turkey Point 3&4 has the highest transient CDF for this plant group. Loss of CCW causes loss of cooling to the RCP seals, causing a seal LOCA. The seal LOCA can not be mitigated because ECCS recirculation from the containment sump is also cooled by CCW, and recirculation fails for this sequence.

The low CDF for transients at Shearon Harris is primarily attributable to plant design characteristics. The charging/HPI pumps at Shearon Harris are cooled by SW, which introduces less vulnerability to transients with RCP seal LOCAs than designs with pump cooling provided by CCW. Also, the Shearon Harris charging/HPI pumps provide adequate flow for feed-and-bleed cooling at the SRV setpoint so that the PORVs are not necessary for success of this action. In addition, the plant has a large CST for AFW, eliminating the need for operator action to supply alternate sources of feedwater in the long term.

11.3.4.4 LOCA Perspectives

LOCA core damage sequences encompass any breaks in the RCS. Normally, the licensees consider stuck-open pressurizer relief valves or reactor coolant seal leaks that initiate an accident as a LOCA. If either of these occur following another initiator (e.g., LOSP), they are normally categorized as an SBO or transient, as appropriate. After a LOCA is initiated, inventory makeup is needed to prevent the core from uncovering and proceeding to core damage. In addition, it is necessary to remove the decay heat from the RCS. Larger breaks can remove the decay heat through the break so that only inventory makeup is of concern. Smaller breaks exhaust less energy, so supplemental cooling is needed. This is normally provided by steam generator cooling or by primary system feed-and-bleed. Injection to the RCS is initially from a source outside containment (e.g., RWST), but this source is ultimately depleted so the plant must switch from this injection mode to the recirculation mode, in which water is drawn from the containment sump. Unless the injection mode is successful, however, there will not be sufficient water in the sump to provide RCS makeup during recirculation. For all of the Westinghouse 3-loop plants, recirculation using high-pressure pumps is achieved by aligning the pumps to draw suction from low-pressure pumps, which draw suction from the containment sump.

LOCAs are a significant contributor to plant CDF for the Westinghouse 3-loop plants. The LOCAs are most often small LOCAs, and recirculation failures are more common than injection failures. Both actual plant characteristics and IPE modeling affect the results for this plant group. The factors that have the greatest impact on the variation in LOCA CDFs among the plants are (1) whether there is an automatic, semi-automatic, or manual switchover to recirculation, (2) the size of the RWST (for plants with manual or semi-automatic switchover), (3) whether the plant has the capability for refilling the RWST, (4) whether the plant has the capability for depressurizing the RCS through the steam generator secondaries for small LOCAs, and (5) whether credit was given for such actions in the IPEs when the capability exists at the plants.

The most common type of LOCA is a small LOCA, but one plant is instead dominated by medium/large LOCAs. Failures during injection are dominant for about half of the Westinghouse 3-loop IPEs, while failures during recirculation are dominant for the other half. For this report, the following sizes of LOCAs are considered:

- large - large enough to remove decay heat through the break, and also large enough to depressurize the system on its own, which allows injection from low-pressure systems (typically, greater than a 6-inch diameter)

- medium - the break is large enough to remove decay heat through the break, but not large enough to depressurize the system (typically, a 2- to 6-inch diameter)

- small - the break is not large enough to depressurize the system before core damage would occur and so the RCS must be depressurized by some other means if LPI is needed; the break does not remove sufficient energy to cool the RCS, so some other means is required to provide the energy removal (typically, a 1/2- to 2-inch diameter).

Many IPE submittals report results with a more detailed break size categorization, but these were collapsed into the more general categories listed above to provide a consistent basis for comparison of the IPE results.

The CDF contributions from small, medium, and large LOCAs are listed in Table 11.35 for the Westinghouse 3-loop plants. Small LOCAs are the dominant break size for all but one of the IPEs. Recirculation failures are dominant for about half the IPEs.

Table 11.35 LOCA contributors for Westinghouse 3-loop plants.[1]

Plant	CDF from large LOCAs (1/ry) [3]	CDF from medium LOCAs (1/ry)	CDF from small LOCAs (1/ry)
Beaver Valley 1	small	small	2E-5
Beaver Valley 2	small	small	4E-5
Farley 1&2	4E-6	3E-6	2E-5
H. B. Robinson	2E-5	5E-5	7E-6
North Anna 1&2	4E-6	7E-6	1E-5
Shearon Harris	3E-6	4E-6	2E-5
Summer	3E-6	8E-6	3E-5[4]
Surry 1&2	5E-6	5E-6	1E-5
Turkey Point 3&4	2E-6	5E-6	3E-5

[1] Not including SGTR and ISLOCAs
[2] Total LOCA CDF = Large LOCA CDF + Medium LOCA CDF + Small LOCA CDF
[3] Includes Vessel Rupture
[4] CDF was estimated for these entries.

The most common system and component failures for LOCAs are divided fairly evenly between failure during recirculation and failures during injection. Some of the system and component failures that are dominant contributors to the LOCA CDF are common across the plants, while others are highly plant-specific. The failures that contribute most to LOCAs for the Westinghouse 3-loop plants can be grouped into the categories listed below. The system and component failures that contribute to each are discussed in the following paragraphs.

- LOCA initiator
- failure of injection
- failure of recirculation
- failures of alternate actions that could be used to mitigate the above failures

The initiator frequencies for the small, medium, and large break sizes for the Westinghouse 3-loop plants are listed in Table 11.36. The plants generally use generic data to quantify the LOCA initiator frequencies. For plants that subdivide small LOCAs into two categories, the small LOCA initiator frequency represents the sum of the individual initiator frequencies.

Table 11.36 Important characteristics for LOCAs for Westinghouse 3-loop plants.

Plant	LOCA initiator frequency			Recirculation switchover	HPR draws suction from LPR[1]
	Large	Medium	Small		
Beaver Valley 1	2E-4	5E-4	2E-2	auto	yes
Beaver Valley 2	2E-4	5E-4	2E-2	auto	yes
Farley 1&2	3E-4	8E-4	5E-3	manual	yes
H. B. Robinson	5E-4	3E-3	2E-2	manual	yes
North Anna 1&2	5E-4	1E-3	2E-2	manual	yes
Shearon Harris	5E-4	6E-4	2E-2	semi-auto	yes
Summer	3E-4	8E-4	8E-3	semi-auto	yes
Surry 1&2	5E-4	1E-3	2E-2	auto	yes
Turkey Point 3&4	1E-5	1E-4	2E-3	manual	yes

[1] HPR = high-pressure recirculation, LPR = low-pressure recirculation

LOCAs with failure in recirculation are a slightly larger contributor than LOCAs with injection failure. This is reasonable because the use of recirculation is normally more complicated than injection. For example, the operators may have to realign the systems such that the low-pressure pumps draw suction from the sump and align the high-pressure pumps to take suction from the low-pressure pumps.

The dominant contributors found in the IPE submittals for injection failures include common cause and random actuation system failures, operator failure to open accumulator discharge valves after being in cold shutdown, common cause failure of HPI when miniflow line fails closed, and common cause failure of check valves in cold leg safety injection lines.

Table 11.36 summarizes differences among the Westinghouse 3-loop plants with regards to recirculation during LOCAs. Most of the plants require manual actions to initiate recirculation because HPR draws suction from low-pressure systems during recirculation. Manual switchover is less likely to succeed than automatic switchover. The most common dominant contributors to recirculation failures in LOCAs that were found in the IPE submittals are pump failures (random or common cause), operator error in recirculation switchover, failure of motor operated valves in LPIS, and operator failure to depressurize.

Some licensees credit alternate actions that could be used to mitigate a LOCA when the standard approaches discussed above do not succeed. For example, in the H. B. Robinson and North Anna 1&2 IPEs, actions are included for depressurizing the RCS using the steam generator relief valves when HPI fails during a LOCA. Another action considered in some IPEs is refilling the RWST if recirculation fails.

The plants with the highest and lowest LOCA CDFs have important design differences that account for the difference in results, but IPE modeling differences are also important. The Beaver Valley 1 IPE has the lowest CDF from LOCAs and the H. B. Robinson IPE has the highest. The major design difference between the two plants related to LOCAs is that ECCS switchover from injection to recirculation is automatic at Beaver Valley 1 and it is manual at H. B. Robinson. In addition, the large and medium LOCA initiator frequencies that are used in the H. B. Robinson analysis are higher than the values used for Beaver Valley 1.

11.3.4.5 ATWS Perspectives

An ATWS is an accident that is initiated by an event, such as loss of feedwater or turbine trip, followed by failure of the reactor to scram. The failure to scram can result either from failures of electrical components in the RPS or in the final control elements that de-energize the CRD mechanisms, or from mechanical failures involving failure of de-energized control rods to drop into the core. Partial scram failures are possible, involving insertion of a subset of the control rods; however, in most PRAs it is assumed that the scram failure is total and involves all of the rods. With failure of the rods to insert, power greatly in excess of decay heat loads is still being generated and overpressurization of the RCS is possible in some cases.

ATWS is not a major contributor to the total CDF for most plants in the group. ATWS frequencies for this group range from 7E-8/ry to 4E-5/ry and, except for one plant, contribute less than 10% to the plant CDF.

The plant in the group with the highest CDF from ATWS, Beaver Valley 1, operates with two of three pressurizer PORV block valves closed. This reduces the relief capacity of the primary system in the early phase of an ATWS accident to that provided by the safety valves alone.

11.3.4.6 Internal Flood Perspectives

An internal flood sequence involves a release of water into a plant location such that a plant trip is induced and safety systems are compromised at the same time. For example, if a flood causes water to enter an area containing electrical switchgear, a plant trip can occur along with failure of all plant systems dependent upon that switchgear. Systems that are considered to be independent may all fail, if they in fact all contain equipment within the flooded location. The effects of internal flooding are highly plant-specific, depending on the layout of equipment within the plant and the relative isolation of rooms. Often, the systems most affected are support systems such as electric power and SW, which have plant-specific designs. Because of this diversity of design and layout, we expect that each plant will have different vulnerabilities to flooding and do not expect to draw many generic conclusions.

Internal flooding is an important contributor to the total CDF at two of the Westinghouse 3-loop plants. The importance of internal flooding to the total CDF is highly plant-specific, depending on the layout of water bearing lines and vessels, flood propagation, and the locations of equipment needed to provide for shutdown cooling. Typically, important internal floods are those that affect important support systems, such as SW and electric power. Newer plants sometimes reflect better design characteristics, generally better separation, and tend to be less susceptible to floods.

The contribution from flooding varies from negligible to 40% of the CDF. The CDF from flooding ranges from negligible to 7E-5/ry. Internal flooding is not a major contributor to the CDF for most plants, but for two of the IPEs (3 units) it is an important contributor, comprising as much as about 40% of the total CDF.

The highest CDF from internal flooding for the group is 7E-5/ry at H. B. Robinson. For one of the IPEs, internal flooding is a negligible contributor to CDF but the IPE submittal does not report an actual number, so the lowest CDF from internal flooding for the group is not known. The lowest CDF actually reported from internal flooding for the group is 2E-6/ry for Summer. The average CDF from the reported values is 2E-5/ry.

The plants with the highest CDFs from internal flooding are H. B. Robinson and Surry 1&2. Internal flooding, with a CDF of 7E-5/ry, contributes about 20% to the total CDF for H. B. Robinson. The IPE submittal indicates that the high CDF from internal flooding is attributable to the relatively small size of the auxiliary building and the location of safety related equipment in areas in that building are sensitive to flooding. Internal flooding, with a CDF of 5E-5/ry, contributes approximately 40% to the total CDF for Surry 1&2. This contribution is associated with breaks in SW or CCW lines that cannot be isolated by closing valves. These breaks lead to flooding of the turbine building, switchgear room, and auxiliary building.

11.3.4.7 SGTR Perspectives

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. There are several actions that are normally taken to prevent the tube rupture initiator from developing into a core melt sequence. To mitigate the leak, the affected steam generator is normally isolated, and if this does not succeed, actions are taken to depressurize the primary system so that leakage is minimized. The primary system is depressurized either by aggressively cooling down through the unaffected steam generator(s), cooling down the primary system using the pressurizer sprays, or by depressurizing by initiating feed-and-bleed through the pressurizer PORV(s). If the affected steam generator is isolated, long-term core heat removal can be established without inventory makeup, but if the leak is not isolated, inventory makeup must also be provided.

SGTR sequences are small contributors to plant CDF at the Westinghouse 3-loop plants but can be important contributors to risk because the releases bypass containment. The higher CDFs reflect either plant weaknesses identified in the IPEs or IPE modeling that includes more possibilities for SGTRs combined with other plant threats (e.g., ATWS).

SGTR sequences are low contributors to plant CDF but can be significant to risk because the releases bypass containment. Overall, SGTR sequences are minor contributors to plant CDF, with an average SGTR CDF of 5E-6/ry. The average contribution to plant CDF is less than 5%. The individual plant results vary from a high of 1E-5/ry to a low of 3E-7/ry.

The Westinghouse 3-loop plant with the highest SGTR CDF is Surry, and the plant with the lowest SGTR CDF is Farley 1&2. The largest contributor to the Surry SGTR CDF involves failure of instrument air which fails steam dump valves, pressurizer PORVs, and auxiliary pressurizer sprays. Also important are sequences involving steam generator overfill leading to a stuck-open steam generator relief valve and eventually to core damage when the injection water source is depleted. The difference in SGTR CDFs between the Surry and Farley is primarily attributable to different modeling assumptions used in the IPEs.

11.3.4.8 ISLOCA Perspectives

An ISLOCA occurs when valves that normally isolate the RCS from low-pressure systems (e.g., LPI) fail, resulting in backflow from the RCS through the low-pressure piping. If the low-pressure piping or other components (e.g., seals, relief valves, or flanges) can not withstand the resulting pressurization, then a LOCA results. If the breach occurs in a portion of the piping that is outside containment, then a LOCA that bypasses containment results, with effluent from the LOCA being discharged into the reactor or auxiliary building. Because it is not possible to recirculate the coolant through the RCS for this type of LOCA, coolant injection will eventually be lost (leading to core damage) unless some source of sustained makeup is available. These scenarios are typically low probability events, because they involve multiple valve failures (typically two check valves and one motor-operated valve in series). However, ISLOCAs can be important to risk because the containment bypass leads to larger fission product releases.

ISLOCAs are minor contributors to plant CDF in all of the IPE submittals but in some cases, there is a large enough contribution to be risk significant. The average ISLOCA CDF for the Westinghouse 3-loop plants is 3E-6/ry and, on average, the fractional contribution is a negligible fraction of plant CDF. The individual plant results vary from a low of 1E-7/ry to a high of 7E-6/ry, with a percent contribution to plant CDF ranging from negligible to less than 5%. Although there are important differences in plant characteristics that affect the results, the variability in results appears to be predominantly influenced by differences in modeling among the plants.

11.3.5 CDF Perspectives for Westinghouse 4-loop Reactors

As indicated in Table 11.37, the largest group of plants is the Westinghouse 4-loop group, which consists of twenty IPE submittals. Six of these submittals are for single-unit sites, 11 are for dual-unit sites with the same results reported for each unit, one is for a dual-unit site (Salem 1&2) but with slightly different results reported for each unit, and two cover a dual unit site (Indian Point 2 and Indian Point 3) where each unit is operated by a different utility. Accounting for the dual unit sites, the Westinghouse 4-loop group comprises thirty-two plant units. All of the plants in this group have large dry containments except that Millstone 3 has a subatmospheric containment and the following plants have ice condenser containments: Catawba 1&2, D.C. Cook 1&2, McGuire 1&2, Sequoyah 1&2, and Watts Bar 1. These containment differences have not been found to significantly affect the plant CDFs,[11.2] and so the plant group was not subdivided by containment type.

[11.2]Containment differences have some influence on small LOCA CDFs for ice condenser plants, where sprays are initiated for small containment pressure increases, leading to the need for early switchover to recirculation. However, the overall impact on plant CDF is small.

Table 11.37 Plants in Westinghouse 4-loop plant 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	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	Indian Point 3

11.3.5.1 Summary of Results and Perspectives for Westinghouse 4-loop Reactors

The total CDFs for the plants in the Westinghouse 4-loop group are shown in Figure 11.6. The CDFs for this plant group show a large variability, with about a factor 50 spread between the highest and lowest CDFs. The average CDF is 6E-5/ry, which is about the same as the average of all PWR CDFs. The contribution of individual accident classes to plant CDF also shows considerable variability, as shown in Figure 11.11. The variation in the accident class CDFs is attributable to a combination of plant design differences and modeling assumptions. Overall, the largest contributors are transients, LOCAs, SBOs, and SGTRs. The IPE submittals generally report smaller contributions from ATWS, internal flooding, and ISLOCA sequences.

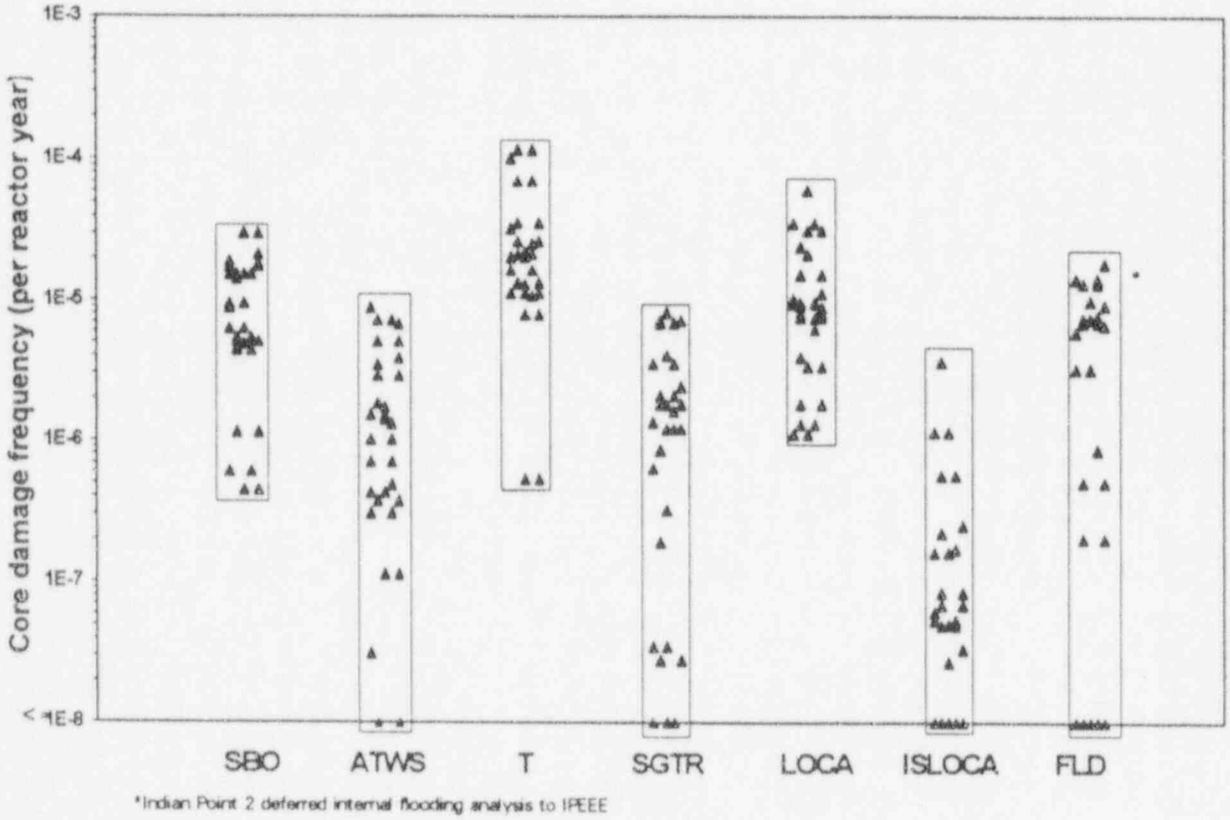


Figure 11.11 Sequence results for Westinghouse 4-loop plants.

Table 11.38 summarizes the perspectives obtained from examination of the Westinghouse 4-loop plants. The details that provide the basis for these perspectives are contained in the remainder of Section 11.3.5. The results for the accident classes are discussed, giving more details on the factors driving the CDFs, particularly for the plants with the highest and lowest CDFs. Design and operational factors along with differences in modeling assumptions are addressed.

Table 11.38 Key IPE observations for Westinghouse 4-loop plants.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
Transient accidents		
Important for nearly all of the Westinghouse 4-loop plants	Degree of HPI and AFW dependence on SW and CCW Dependence of RCP seal cooling on SW or CCW Modeling of RCP seal LOCA probability and size	Transient CDFs range from 5E-7/ry to 1E-4/ry. Average CDF is 3E-5/ry. Contribution to total plant CDF ranges from about 15% to 85%
LOCAs		
Important for many Westinghouse 4-loop plants	Degree of automation of switchover of ECC to recirculation Design of HPR (draw directly from sump vs. drawing suction from LPR) Size of RWST Ability to depressurize the RCS by aggressively cooling down using steam generator ADVs so LPI can be used when HPI fails Ability to refill the RWST	LOCA CDFs range from 1E-6/ry to 6E-5/ry. Average CDF is 1E-5/ry. Contribution to total plant CDF ranges from less than 5% to 55%
SBO accidents		
Important for many Westinghouse 4-loop plants	AC power reliability (e.g., 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	SBO CDFs range from 4E-7/ry to 3E-5/ry. Average CDF is 1E-5/ry. Contribution to total plant CDF ranges from negligible to about 60%

Table 11.38 Key IPE observations for Westinghouse 4-loop plants.

Accident importance	Important design features, operator actions, and model assumptions	Summary of results
Internal flood accidents		
Important for some Westinghouse 4-loop sites	Plant layout: separation of mitigating system components and compartmentalization	Flood CDFs range from 4E-9/ry to 2E-5/ry. Average CDF is 5E-6/ry. Contribution to total plant CDF ranges from negligible to about 30%
ATWS accidents		
Not dominant for most Westinghouse 4-loop plants	Plant operation with PORV block valves closed	ATWS CDFs range from 1E-8/ry to 9E-6/ry. Average CDF is 2E-6/ry. Contribution to total plant CDF ranges from negligible to 20%
ISLOCAs		
Not important for any Westinghouse 4-loop plant	Compartmentalization and separation of equipment	ISLOCA CDFs range from 2E-9/ry to 4E-6/ry. Average CDF is 3E-7/ry. Contribution to total plant is less than 5% for all plants
SGTR accidents		
Not important for most Westinghouse 4-loop plants	Modeling of operator actions to isolate the rupture and provide long-term heat removal	SGTR CDFs range from 9E-9/ry to 8E-6/ry. Average CDF is 2E-6/ry. Contribution to total plant CDF ranges from negligible to 30%

11.3.5.2 SBO Perspectives

SBO occurs when a plant loses off-site AC power, and the on-site backup AC power sources (almost always diesel generators) also fail to function. Because so many safety systems rely on AC power either directly (for motive power) or indirectly (e.g., cooled by systems that rely on AC power), SBO causes most safety systems to be unavailable. Most SBO scenarios do have DC power available, however, through use of station batteries. For PWRs, the loss of all AC power normally leaves turbine-driven AFW as the only available means for core cooling. Turbine-driven AFW can operate until the batteries deplete, leading to a loss of control. If AC power is not recovered soon after loss of control, core damage will follow. In some IPEs, AFW is assumed to continue even after battery depletion, by manually controlling the system. Even with such manual control, the SBO sequence might still proceed to core damage if there is not a continuous supply of water available for AFW injection.

An additional and important complication for Westinghouse plants involves the potential for leakage from RCP seals. In most Westinghouse 4-loop plants, SBO results in loss of cooling for the RCP seals. The prolonged exposure to high temperatures can fail the seals, leading to a small LOCA through the pump seals. Some plants have less of a threat from seal failure because the plants are using new o-rings in the pumps, which are less susceptible to this failure mode. If a seal LOCA occurs, injection systems are needed to provide makeup to the RCS. Since these are not normally available during an SBO, pump seal LOCAs will lead to core damage if off-site power is not restored in time. Chapter 17 of this report provides further discussion of SBO sequences.

SBO is an important contributor to plant CDF for many, but not all, of the Westinghouse 4-loop submittals. Several factors are important for SBOs for the Westinghouse 4-loop plants, with differing combinations of these factors driving the results for the individual plants in this group. The factors that have the biggest impact on the results primarily represent actual plant characteristics, but modeling used in the IPEs is also important. The four key factors are (1) the modeling of RCP seal LOCAs (probability of leak, size of leak, and timing of subsequent core uncovery), (2) availability of independent emergency power (e.g., safe shutdown facility) that provides support for AFW and RCP seal cooling, (3) battery life (actual plant design as well as load shedding assumptions), and (4) cross-ties (existence of cross-tie capability and whether it was credited in the IPE).

SBO sequences can be categorized as short-term, long-term, or involving RCP seal LOCA. On average, long-term SBO sequences with RCP seal LOCAs were most frequent. To understand the reasons for differences in SBO CDFs, it is helpful to categorize SBOs as follows:

- <u>short-term SBO</u> - sequences in which turbine-driven AFW pump trains fail to operate and AC power is not recovered before the core is damaged

- <u>long-term SBO</u> - sequences in which turbine-driven AFW pump trains operate initially, but ultimately fail (normally because battery depletion causes loss of DC control power) before recovery of AC power

- <u>SBO with RCP seal LOCA</u> - sequences involving RCP seal LOCAs that are caused by loss of cooling to RCP seals with failure to recover AC power before the core is uncovered.

Short-term SBO sequences typically lead to core damage within about 2 hours, while long-term SBO sequences typically lead to core damage in more than 2 hours. SBO sequences with RCP seal LOCAs can be either short-term or long-term (depending on the model used for the timing and magnitude of a RCP seal LOCA), but they are typically long-term sequences.

The relative fractions of the SBO accident class that fall within these categories are not readily available from the IPE submittals for most plants. However, most submittals qualitatively indicate that short-term SBO sequences are not major contributors to the SBO CDF, because more failures are needed for the short-term case (e.g., turbine-driven AFW must fail). The exceptions are Callaway, Comanche Peak, Indian Point 2, Indian Point 3, South Texas, and Watts Bar. For the plants that are dominated by long-term SBO, all are dominated by RCP seal LOCAs except Millstone 3, Vogtle, and Wolf Creek. Therefore, on average, the SBO CDF for the Westinghouse 4-loop plants is dominated by long-term SBOs with RCP seal LOCAs.

The SBO results are driven by a combination of factors involving both plant-specific features and IPE modeling characteristics. Some of the system and component failures that are dominant contributors to the SBO CDF are common across the plants, while others are highly plant-specific. The failures that contribute most to SBO for the Westinghouse 4-loop plants can be grouped into the categories listed below:

- LOSP
- loss of on-site AC power
- failure to recover AC power
- failure of turbine-driven AFW
- transient-induced LOCA

The system and component failures that contribute to each of the categories are discussed in the following paragraphs.

The initiator for SBO is, of course, loss of off-site AC power. Higher frequencies for loss of off-site AC power will lead to higher SBO CDFs. The frequencies used for each of the Westinghouse 4-loop plants are listed in Table 11.39. As can be seen, the frequencies vary by a factor of 4.

Table 11.39 Key parameters affecting SBO CDF for Westinghouse 4-loop plants.

Plant	SBO CDF (1/ry)	No. EDGs unit/other/x-tie[1]	Battery life (hrs)	LOSP initiator frequency	Pump seals[3]	Other features
Braidwood 1&2	6E-6	2/0/Y	N/A[2]	.045	3	• diesel-driven AFW • submittal says 1 diesel generator can power both units • can cross-tie CST submittal • credits CST for at least 24 hours
Byron 1&2	4E-6	2/0/Y	N/A[2]	.044	3	• diesel-driven AFW • submittal says 1 diesel generator can power both units • can cross-tie CST • submittal credits CST for at least 24 hours
Callaway	2E-5	2/0/N	8	.046	1.3	• CST supply for 14 hours • turbine-driven AFW manually controlled after battery depletion, but less reliable
Catawba 1&2	6E-7	2/1/N	1	.035	2	• shared diesel generator from safe shutdown facility • low probability for failure to restore off-site power

Table 11.39 Key parameters affecting SBO CDF for Westinghouse 4-loop plants.

Plant	SBO CDF (1/ry)	No. EDGs unit/other/x-tie[1]	Battery life (hrs)	LOSP initiator frequency	Pump seals[3]	Other features
Comanche Peak 1&2	2E-5	2/1/Y	4	.035		• swing diesel generator • submittal states off-site power is very reliable
D.C. Cook 1&2	1E-6	2/0/Y	4	.040	1	• submittal states off-site power is very reliable • did not credit diesel generator cross-tie • no credit for cross-tie for CSTs, each last 6 hours • turbine-driven AFW manually controlled after battery depletion
Diablo Canyon 1&2	5E-6	2/1/N	12	.091	1	• shared diesel generator • portable generator to provide continued AFW control
Haddam Neck	9E-6	2/0/N	9	.090	1	
Indian Point 2	5E-6	3/2/N	3	.068		• shared gas turbine generators • turbine-driven AFW manually controlled after battery depletion
Indian Point 3	5E-6	3/1/N	8	.069	1,2	• appendix R diesel that can cool RCPs • CST supplies AFW for 24 hours at shutdown • can use city water for AFW • credits depressurizing steam generator to reduce RCP seal leakage • turbine-driven AFW manually controlled after battery depletion, credits 8 hours of AFW during SBO
McGuire 1&2	9E-6	2/0/Y	3	.070	2	• standby shutdown facility that can provide seal cooling
Millstone 3	5E-6	2/0/N	6	.112		• did not credit air-cooled diesel generator being added for SBO rule

Table 11.39 Key parameters affecting SBO CDF for Westinghouse 4-loop plants.

Plant	SBO CDF (1/ry)	No. EDGs unit/other/x-tie[1]	Battery life (hrs)	LOSP initiator frequency	Pump seals[3]	Other features
Salem 1	2E-5	3/1/N	8	.060		• standby gas turbine that can provide AC to both units
Salem 2	2E-5	3/1/N	8	.060		
Seabrook	1E-5	2/0/N	6	.049		• need 2 diesel generators because of SW/EDG arrangement
Sequoyah 1&2	5E-6	2/0/N	4	.046		• can cross-tie DC to operate turbine-driven AFW
South Texas 1&2	2E-5	3/1/N	8	.132	2	• tech support center diesel generator can provide RCP seal cooling
Vogtle 1&2	3E-5	2/0/N	4	.051	3	• turbine-driven AFW manually controlled after battery depletion • high diesel generator failure rates because of operating history
Watts Bar 1	2E-5	2/0/N	4	.036		
Wolf Creek	2E-5	2/0/N	8	.051		
Zion 1&2	4E-7	2/1/N		.046		• shared diesel generator

[1] The number of diesel generators dedicated to each unit, other diesel generators such as shared or safety shutdown facility, and the presence of cross connects between units (Y for cross connects, N for no cross connects) are noted.
[2] Not applicable because AFW is diesel driven.
[3] 1 = lower seal LOCA leak rate (primarily plants using Westinghouse seal LOCA model), 2 = backup means for cooling RCP seals, 3 = credited installation of new high-temperature o-rings. Note information was not available from IPE submittal for all plants

Failure of on-site AC power is also part of the SBO definition, and since diesel generators are used as the backup AC power source for all of the Westinghouse 4-loop plants, diesel generator failures are dominant contributors to SBO CDF for all of the Westinghouse 4-loop plants. The most common reasons for diesel generator failures involve the failure of the diesel generator to start (multiple random failures or common cause failure), to continue to run for a sufficient time interval after successfully starting, unavailability because of testing or maintenance, failure of diesel generator cooling, or failure of diesel generator room cooling. The redundancy introduced by having additional diesel generators reduces the probability of failing all diesel generators, and reduces the SBO CDF. The number of diesel generators available for each of the Westinghouse 4-loop plants is listed in Table 11.39.

Some multi-unit sites have the ability to cross-tie emergency AC power between units, which reduces the SBO CDF by providing a redundant, somewhat independent power source. However, not all licensees that have this capability actually took credit in the IPE for the action to cross-tie. Licensees that credit in their IPEs for the ability to cross-tie are identified in Table 11.39. Note that five of the six plants with the highest SBO CDFs either do not have or do not credit cross-tie capability.

Normally, the SBO sequences that proceed to core damage involve the failure to recover off-site power. However, a few submittals list sequences in which off-site power is recovered but injection systems are not fully recovered so that the sequence still proceeds to core damage. Although the probability of restoring off-site power is important to the SBO CDF and would, therefore, be desirable to compare among the plants, the necessary information is not readily available from most of the IPE submittals. Table 11.39 does indicate cases where abnormally high or low values are found in the IPE submittals.

Early failures of AFW lead to short-term SBO sequences while delayed failures lead to long-term SBO sequences. Early failures normally involve either pump failure or steam generator overfill. These failures have sufficiently low frequency to cause most plants to have a larger contribution to SBO CDF from long-term SBO sequences rather than these short-term sequences.

Late AFW failures are most commonly caused by the inability to control AFW when DC power is lost as a result of battery depletion. Thus, plants with a longer battery life have more time available to restore off-site power before core cooling is lost, and have lower threat from SBO. Table 11.39 lists the battery life for the Westinghouse 4-loop plants. In some cases, the CST capacity is more limiting than the battery life so that late AFW failures are dominated by the loss of a feedwater source when the tank supply is exhausted. Similarly, some licensees take credit in their IPEs for manually controlling AFW after battery depletion, so that the battery life is not critical to successful plant shutdown. Also, two sites use diesel-driven AFW, so that battery life is once again not important. For these plants without a dependence on battery life, the CST capacity and whether the plant has the ability to refill the CST are important. The plants that have unusually large or small CST capacity, credit manual control of AFW following battery depletion, have diesel-driven AFW, or have credited refilling the CST are indicated in Table 11.39.

SBO sequences with stuck-open PORVs are only listed in a few of the submittals, and even then, they are minor contributors. Pump seal LOCAs are dominant contributors to the SBO CDF for many of the Westinghouse 4-loop IPEs. The size and timing of the leaks varies considerably among the submittals, but pump seal LOCAs always increase the SBO CDF by reducing the time available for recovering off-site power before core damage would result. Three factors tend to reduce the SBO CDF that involves seal LOCAs:

(1) using a seal LOCA model that gave lower leak rates and probabilities (e.g., Westinghouse seal LOCA model)

(2) having a backup source for pump seal cooling (see following paragraph)

(3) crediting installation of new, temperature-resistant o-rings

Table 11.39 indicates whether any of these factors apply to a particular plant in the Westinghouse 4-loop group. It is important to note that the IPE submittals do not always contain a thorough description of seal LOCA modeling, and so some of the factors may also apply to other plants besides those indicated in the table. From the table, it can be seen that many of the plants use the Westinghouse seal LOCA model, which tends to give lower seal leakage probabilities than models such as the NUREG-1150 model. For seven of the plants, credit is taken for the new o-rings, and seven plants have a backup source for pump seal cooling. The table indicates that these seal LOCA assumptions have an important effect on the SBO CDF. Most of the plants using these assumptions have SBO CDFs in the lower half of the Westinghouse 4-loop group, while those that fall in the upper half have some other particular weakness (i.e., high diesel generator failure rate) or modeling characteristics (i.e., dominant sequence has AC recovered, but failure to restore equipment).

Some plants have alternate power sources that can significantly reduce the SBO CDF. Catawba has a safe shutdown facility that has a source of AC and DC power that is independent from the plant normal or emergency power sources. The safe shutdown facility AC power can be used to operate a small charging pump to provide seal injection (which removes the seal LOCA concern) and primary system makeup flow. The safe shutdown facility DC power can power the turbine-driven AFW pump so that AFW failure caused by battery depletion is not a concern. The South Texas technical support center diesel generator provides similar capability. At South Texas, a pump is available for seal cooling that is self-cooled and can be powered off the technical support center diesel generator. Also, the high head safety injection pumps do not require external cooling. Therefore, SBO does not necessarily lead to a seal LOCA at South Texas, unless additional failures occur.

Differences in results for the plants with the lowest and highest SBO CDFs are primarily driven by IPE modeling characteristics. The plants with the lowest and highest SBO CDFs are Zion and Vogtle, respectively. There are differences in plant characteristics that can partially explain the difference in SBO CDFs between the two plants (e.g., both Vogtle and Zion have two diesel generators per unit, but Zion has an additional backup diesel generator that can be used at either unit), but the differences do not fully account for the two orders of magnitude difference in SBO CDF. The differences in SBO CDFs between these two plants are mostly driven by differences in IPE modeling. The key assumptions driving the difference are discussed in the following paragraphs.

The Vogtle IPE included the ability to manually operate the turbine-driven AFW pump on loss of DC power, which lowers the SBO CDF. In this case, the operators control the pump locally, at the pump, rather than from the control room. However, the benefit from this modeling is offset by Vogtle's use of pessimistic modeling for the CST inventory. In the Vogtle IPE analysis, it is assumed that the CST inventory would only be sufficient to provide AFW suction supply for 8 hours, with no credit for CST makeup. However, each Vogtle Unit has two CSTs, and it appears that the inventory of a single CST would be sufficient to provide DHR and RCS cooldown at a single unit for about 24 hours. Zion does not appear to model long-term failures of AFW from either battery depletion or from emptying the CST.

Neither Zion nor Vogtle have a large contribution to the SBO CDF from seal LOCAs, but the reasons for this result are different. The Vogtle IPE reflects installation of new temperature-resistant RCP o-rings, which are being installed at both units. The Zion seal LOCA CDF is small because the licensee assumed that the leakage would be small, and so that even with an RCP seal LOCA, the core would not uncover for 24 hours.

11.3.5.3 Transient Perspectives

Transient sequences involve events that cause the reactor to trip (initiators) followed by failure to bring the reactor to safe shutdown (excluding SBO sequences, which are treated separately). The transient accident class is a broad category, covering both general initiators (such as reactor trip or loss of main feedwater) as well as support system initiators (e.g., loss of SW or AC/DC bus). After reactor trip, decay heat must be removed from the RCS. Normally, this would be provided by steam generator heat removal, with a fallback of primary system feed-and-bleed should secondary heat removal fail. If neither of these succeeds, the RCS inventory will boil off, leading to core damage. A second possible path to core damage for transients involves an induced LOCA during the transient (SORV or RCP seal leak) with failure to make up the RCS inventory. For both types of sequences (failure of DHR and transient-induced LOCA with failure of reactor coolant makeup), long-term operability must also be maintained (e.g., switchover from injection to recirculation must succeed) or core damage will result.

Transients are the largest contributor on average to plant CDF for the Westinghouse 4-loop plants (average contribution of 40%), which is not unexpected since the transients encompass the most frequent initiators. For most of the Westinghouse 4-loop plants, the CDF from transients falls within a range spanning about an order of magnitude. Zion, however, falls about an order of magnitude lower than the other plants. There is wide variability as to the specific failures that lead to core damage for transients, but there is a significant contribution from support system failures (either as initiators or as failures subsequent to some other initiator). The results indicate that plant-specific dependencies are important to the results, but different IPE modeling also affects the results. The results are mostly heavily driven by differences in SW and CCW dependencies, RCP seal cooling designs, RCP seal leakage modeling in the IPEs, and plant-specific configurations for system dependencies and cross-ties. The remainder of this section gives more details regarding these perspectives for transients.

Transients cover a broad group of sequences, including both general and plant-specific initiators. Some initiators occur from loss of support systems such as SW, CCW, HVAC, instrument air, or AC/DC buses. Because these support systems are needed for a large number of front-line systems, their unavailability can simultaneously fail numerous front-line systems, leaving few options for successful plant shutdown. The dependencies are very plant-specific. The remaining initiators are more general, such as reactor trip, turbine trip, or loss of main feedwater. The general initiators typically have higher frequencies than the support system initiators, but more failures are needed (subsequent to the initiator) to result in core damage. LOSP is an exception in that it is usually a lower frequency event, but often has a higher potential to proceed to core damage because failures of one diesel generator often lead to unavailability of full trains of safety systems. Table 11.40 lists the relative contribution of general transients and transients initiated by support system failures, and indicates that both types of transients are important for the Westinghouse 4-loop plants.

Table 11.40 Contributors to Westinghouse 4-loop transients.

Plant	CDF (1/ry)	
	General transients associated with balance of plant	Support system initiating events
Braidwood 1&2	2E-5	8E-7
Byron 1&2	2E-5	3E-6
Callaway	6E-6	5E-6
Catawba 1&2	6E-7	3E-5
Comanche Peak 1&2	6E-6	5E-6
D.C. Cook 1&2	8E-7	2E-5
Diablo Canyon 1&2	5E-5	2E-5
Haddam Neck	7E-5	3E-5
Indian Point 3	2E-5	5E-6
Indian Point 2	1E-5	1E-6
McGuire 1&2	2E-6	1E-5
Millstone 3	1E-5	1E-5

Table 11.40 Contributors to Westinghouse 4-loop transients.

Plant	CDF (1/ry)	
	General transients associated with balance of plant	Support system initiating events
Salem 1 Salem 2	1E-5 2E-5	2E-6 2E-6
Seabrook	2E-5	1E-5
Sequoyah 1&2	3E-5	8E-5
South Texas 1&2	6E-6[2]	1E-5[2]
Vogtle 1&2	7E-6	3E-7
Watts Bar 1 (revised)	6E-6	1E-5
Wolf Creek	5E-6	5E-6
Zion	5E-7	7E-9

[1] Total Transient CDF = Generic Transients CDF + Specific Transients CDF. Transient CDF does not include SBO CDF and does not include ATWS CDF.
[2] CDF was estimated for these entries.

For transient sequences with failure of heat removal through the steam generators and failure of primary feed-and-bleed, the dominant contributors widely vary, representing considerable variability in plant design and operation as well as variability in IPE modeling. By examining the results reported in the IPE submittals, it was found that there is wide variability as to the specific failures that lead to core damage for transients with failure to remove decay heat in the Westinghouse 4-loop plants. The results indicate that plant-specific dependencies are important to the results, but also different IPE modeling affects the results. Some of the observed contributors to failures of steam generator heat removal and failure of primary feed-and-bleed are discussed in the following paragraphs.

The more common reasons for failure of steam generator heat removal that are listed in the IPE submittals include combinations of failures in which one failure compromises the availability of multiple systems and then a key remaining system fails independently, leading to core damage. For example, LOSP with successful operation of one diesel generator but independent failure of one AFW pump (because of failure to start or maintenance unavailability) is a common contributor. Also seen as contributors are sequences with either common cause failure of feedwater pumps or failure of a support system (e.g., HVAC failures in switchgear room) that result in failure of multiple systems.

The most common failures listed in the IPE submittals for failure to provide feed-and-bleed cooling if steam generator heat removal is lost, are failure of injection pumps resulting from the same types of failures as listed above for steam generator heat removal, or operator failure to successfully perform the operation. The failures primarily occur in the injection phase, rather than in recirculation.

For transient sequences with RCP seal LOCAs, the dominant contributors also widely vary, representing considerable variability in plant design and operation as well as variability in IPE modeling. The IPE submittals indicate a wide variability as to the specific failures that lead to core damage for transients with induced LOCAs (primarily RCP seal LOCAs) for the Westinghouse 4-loop plants. The results indicate that plant-specific dependencies are important to the results, but also different IPE modeling affects the results. Some of the observed contributors to RCP seal leakage and the subsequent failure to provide makeup are discussed in the following paragraphs.

There is considerable variability in pump cooling configurations, but at most of the Westinghouse 4-loop plants the RCPs can be cooled by either CCW or by the charging pumps. The charging pumps are normally cooled by either SW or CCW. For plants with charging pumps cooled by CCW, loss of CCW causes both a seal LOCA and loss of the charging system that would normally be used to provide makeup flow from the induced LOCA (effectively, such loss of CCW sequences behave like SBO sequences). The probability of the seal LOCA occurring for such sequences depends on a number of factors, including:

- the seal LOCA model used in the IPE (Westinghouse seal LOCA model gives lower probability of failure than NUREG-1150 model)
- whether backup systems are available to provide RCP cooling
- credit given for operator actions such as cross-tieing systems or cooling down the RCS to reduce the seal LOCA threat
- whether the plant has credited installation of the new temperature-resistant o-rings for the RCPs

Although these factors introduce variability into the IPE results, CCW failures are commonly seen as dominant contributors to the transient CDFs, which reflects the relatively high dependency of other plant systems on CCW. For example, the dominant transient sequence for D. C. Cook is the loss of CCW leading to a seal LOCA that cannot be mitigated. Similarly, SW cooling is often found to be a dominant contributor because its loss compromises the ability to cool many plant systems (SW is generally also the heat sink for CCW). Other dominant contributors to transients with RCP seal LOCAs include LOSP combined with CCW failure.

The results for the plants with the highest and lowest CDFs for transients are driven by both differences in actual plant characteristics and IPE modeling. Sequoyah has the highest transient CDF for this plant group. The CDF from transients at Sequoyah is dominated by vibration-induced seal LOCAs. Loss of CCW causes loss of cooling to the RCP motors, and failure to promptly trip the running pumps leads to a seal LOCA. The seal LOCA can not be mitigated because ECCS recirculation from the containment sump is also cooled by CCW, and so recirculation fails for this sequence.

Haddam Neck has the highest plant CDF in the Westinghouse 4-loop group. There are two major reasons for the relatively high CDF from transients at Haddam Neck. First, seal LOCAs during transient sequences are an important contributor, and the submittal indicates that loss of seal cooling is primarily caused by either loss of power or mechanical failures in the seal cooling systems. Second, failures associated with one motor control center, MCC-5, are important because the ECCS injection valves for both trains of ECCS are powered off this motor control center.

The low CDF for transients at Zion is primarily attributable to the modeling used in the IPE. The Zion IPE submittal describes its modeling as realistic, but on average, the Zion failure frequencies are much lower than the frequencies used for other plants, and fewer systems are required for successful plant shutdown in the Zion IPE. The most important modeling assumptions appear to be low values for common cause failures and operator failures.

11.3.5.4 LOCA Perspectives

LOCA core damage sequences encompass any breaks in the RCS. Normally, the IPEs considered stuck-open pressurizer relief valves or reactor coolant seal leaks that initiate an accident as a LOCA. If either of these occur following another initiator (e.g., LOSP), they are normally categorized as an SBO or transient, as appropriate. After a LOCA is initiated, inventory makeup is needed to prevent the core from uncovering and proceeding to core damage. In addition, it is necessary to remove the decay heat from the RCS. Larger breaks can remove the decay heat through the break so that only inventory makeup is of concern. Smaller breaks exhaust less energy, and so supplemental cooling is needed. This is normally provided by steam generator cooling or by primary system feed-and-bleed. Injection to the RCS is initially from a source outside containment (e.g., RWST). This source is ultimately depleted, and so the plant must switch from this injection mode to the recirculation mode, in which water is drawn from the containment sump. Unless the injection mode is successful, however, there will not be sufficient water in the sump to provide RCS makeup during recirculation.

LOCAs are a significant contributor to plant CDF for the Westinghouse 4-loop plants. The LOCAs are most often small LOCAs, and recirculation failures are more common than injection failures. Both actual plant characteristics and IPE modeling affect the results for this plant group. The six factors that have the greatest impact on the variation in LOCA CDFs among the plants are (1) whether there is an automatic, semi-automatic, or manual switchover to recirculation, (2) whether HPR is dependent on low-pressure systems, (3) the size of the RWST (for plants with manual or semi-automatic switchover), (4) whether the plant has the capability for refilling the RWST, (5) whether the plant has the capability for depressurizing the RCS through the steam generator secondaries for small LOCAs, and (6) whether credit is given for such actions in the IPEs when the capability exists at the plants.

The most common type of LOCA is a small LOCA with failure of recirculation, but many plants are instead dominated by medium or large LOCAs. For this report, the following sizes of LOCAs are considered:

- <u>large</u> - large enough to remove decay heat through the break, and also large enough to depressurize the system on its own, which allows injection from low-pressure systems (typically, greater than a 6 inch diameter)

- <u>medium</u> - the break is large enough to remove decay heat through the break, but not large enough to depressurize the system (typically, a 2- to 6-inch diameter)

- <u>small</u> - the break is not large enough to depressurize the system before core damage would occur and so the RCS must be depressurized by some other means if LPI is needed; the break does not remove sufficient energy to cool the RCS, so some other means is required to provide the energy removal (typically, a 1/2- to 2-inch diameter).

Many IPE submittals report results with a more detailed break size categorization, but these are collapsed into the more general categories listed above to provide a consistent basis for comparison of the IPE results.

The CDF contributions from small, medium, and large LOCAs are listed in Table 11.41 for the Westinghouse 4-loop plants. Small LOCAs are the dominant break size for about half the IPEs, with medium and large LOCAs each being dominant at about a quarter of the plants. Recirculation failures are dominant for most IPEs.

Table 11.41 LOCA contributors for Westinghouse 4-loop plants.[1]

Plant	CDF from large LOCAs (1/ry) [3]	CDF from medium LOCAs (1/ry)	CDF from small LOCAs (1/ry)
Braidwood 1&2	4E-7	1E-7	6E-7
Byron 1&2	4E-7	2E-7	8E-7
Callaway	3E-6	4E-6	4E-6
Catawba 1&2	1E-6	7E-7	5E-6
Comanche Peak 1&2	3E-6	1E-6	5E-6
DC Cook 1&2	1E-6	4E-6	3E-5
Diablo Canyon 1&2	3E-6	5E-6	9E-7
Haddam Neck	3E-5	2E-5	2E-5
Indian Point 3	3E-6	8E-7	5E-7
Indian Point 2	3E-6	2E-6	6E-6
McGuire 1&2	3E-6	2E-6	1E-5
Millstone 3	8E-6	1E-5	2E-6
Salem 1 Salem 2	2E-6 1E-6	3E-6 4E-6	3E-6 4E-6
Seabrook	2E-6	1E-6	4E-6
Sequoyah 1&2	2E-6[4]	3E-6[4]	3E-5[4]
South Texas Project 1&2	small	1E-6	2E-6
Vogtle 1&2	2E-6	4E-6	3E-6
Watts Bar 1 (revised)	3E-6	2E-6	2E-5
Wolf Creek	2E-6	2E-6	7E-7
Zion	1E-6	4E-7	2E-7

[1] Not including SGTR and ISLOCAs
[2] Total LOCA CDF = Large LOCA CDF + Medium LOCA CDF + Small LOCA CDF
[3] Includes Vessel Rupture
[4] CDF was estimated for these entries.

The most common system and component failures for LOCAs are related to the switchover to recirculation, but failures during injection are also important. Some of the system and component failures that are dominant contributors to the LOCA CDF are common across the plants, while others are highly plant-specific. The failures that contribute most to LOCAs for the Westinghouse 4-loop plants can be grouped into the categories listed below. The system and component failures that contribute to each are discussed in the following paragraphs.

- LOCA initiator
- failure of injection
- failure of recirculation
- failures of alternate actions that could be used to mitigate the above failures

The initiator frequencies for the small, medium, and large break sizes for the Westinghouse 4-loop plants are listed in Table 11.42. The plants generally use generic data to quantify the LOCA initiator frequencies. For plants that subdivide small LOCAs into two categories, the small LOCA initiator frequency represents the sum of the individual initiator frequencies.

Table 11.42 Important characteristics for LOCAs for Westinghouse 4-loop plants.

Plant	LOCA initiator frequency			Recirculation switchover	HPR draws suction from LPR[1]
	Large	Medium	Small		
Braidwood 1&2	3E-4	8E-4	6E-3	Semi-Auto	yes
Byron 1&2	3E-4	8E-4	6E-3	Semi-Auto	yes
Callaway	5E-4	1E-3	1E-2	Semi-Auto	yes
Catawba 1&2	3E-4	3E-4	4E-3	Auto	yes
D.C. Cook 1&2	3E-4	9E-4	7E-3	Manual	yes
Diablo Canyon 1&2	2E-4	5E-4	2E-3	Semi-Auto	yes
Haddam Neck	4E-4	6E-4	1E-2	Manual	yes
Indian Point 2	2E-4	5E-4	2E-2	Manual	yes
Indian Point 3	5E-4	9E-4	7E-3	Manual	yes
McGuire 1&2	3E-4	3E-4	4E-3	Auto	yes
Millstone 3	4E-4	6E-4	9E-3	Manual	no
Salem 1	5E-4	1E-3	2E-2	Manual	yes
Salem 2	5E-4	1E-3	2E-2	Manual	yes
Seabrook	2E-4	5E-4	2E-2	Semi-Auto	yes
Sequoyah 1&2	2E-4	5E-4	2E-2	Semi-Auto	yes
South Texas Project 1&2	2E-4	5E-4	2E-2	Auto	no
Vogtle 1&2	3E-4	8E-4	7E-3	Semi-Auto	yes
Watts Bar 1&2	2E-4	5E-4	3E-2	Semi-Auto	yes
Wolf Creek	5E-4	1E-3	3E-3	Semi-Auto	yes
Zion 1&2	3E-4	1E-3	7E-3	Manual	yes

[1] HPR = high-pressure recirculation, LPR = low-pressure recirculation

LOCAs with failure in recirculation are generally a larger contributor than LOCAs with injection failure. This is reasonable because the use of recirculation is normally more complicated than injection. For example, the operators may have to realign the systems such that the low-pressure pumps draw suction from the sump and align the high-pressure pumps to take suction from the low-pressure pumps. The dominant contributors found in the IPE submittals for injection failures include pump failures, actuation failures, valve failures, and pneumatic supply failures. Table 11.42 summarizes differences among the Westinghouse 4-loop plants with regard to recirculation during LOCAs. Most of the plants require manual actions to initiate recirculation because HPR draws suction from low-pressure systems during recirculation. Manual switchover is less likely to succeed than automatic switchover, and plants that require both high and low-pressure systems to operate for recirculation have a higher probability of failure because of the number of systems that must succeed. The most common dominant contributors to recirculation failures in LOCAs that are found in the IPE submittals are pump failures, pump cooling failures, and operator error in recirculation switchover.

The ice condenser plants generally have higher contributions from small LOCAs with failures during recirculation. Because of the lower design pressure of the ice condenser containments sprays are actuated earlier than for large-dry containments, which results in water being drawn more rapidly from the RWST. This shortens the time availablefor the switchover from the RWST to the containment sump, which decreases the probability of the operators successfully performing the manual portion of the recirculation switchover for HPR.

Some licensees credit alternate actions that could be used to mitigate a LOCA when the standard approaches discussed above do not succeed. Some licensees take credit in the IPE for depressurizing the RCS using the steam generator relief valves when HPI fails during a LOCA. Other actions credited in some IPEs as refilling the RWST if recirculation fails or to delay the need for switchover to recirculation.

The plants with the highest and lowest LOCA CDFs have important design differences that account for the difference in results, but IPE modeling differences are also important. The Braidwood and Byron IPEs have the lowest CDFs from LOCAs and the Haddam Neck IPE has the highest. The following paragraphs discuss the design and modeling differences that drive the results for these plants.

There are two major design differences between the plants related to switchover to recirculation that impact the LOCA CDF. First, Haddam Neck has a manual switchover from injection to recirculation. Braidwood and Byron have semi-automatic switchover. That is, the switchover is automatic for LPR but requires operator action to align high-pressure pumps to draw suction from the low-pressure pump discharge to accomplish HPR. The partial automation of recirculation for Braidwood and Byron tends to reduce the probability of recirculation failure relative to Haddam Neck. Second, the Haddam Neck RWST capacity is small (100,000 gallons) providing operators limited time to accomplish the manual switchover of ECCS from injection to recirculation which further decreases the probability of successful execution. The discussion of the dominant core damage sequences in the IPE for Haddam Neck indicates that about 50% of the overall CDF related to LOCAs is attributable to the failure of operators to successfully perform ECCS switchover for medium and large LOCAs. Braidwood and Byron have larger RWSTs (about 400,000 gallons) giving more time for the operators to accomplish switchover when HPR is needed. Additionally, the IPE submittals indicate that for small LOCAs, recirculation will not be needed for Braidwood and Byron because of the larger RWSTs.

The Braidwood and Byron IPEs also include credit for two strategies not considered for Haddam Neck. Braidwood and Byron have procedures to refill the RWST if recirculation fails, and the IPEs include credit for this action in the IPEs. For small LOCAs with HPI failure, the Braidwood and Byron IPEs also include credit for actions to depressurize the RCS using the steam generators so that LPI can be used. Like most of the IPEs, the Haddam Neck

IPE does not include credit for this strategy. Plant-specific analyses (that consider relief valve capabilities and secondary heat removal capabilities) would be needed to determine whether this depressurization strategy is viable for Haddam Neck or for the Westinghouse plants in general.

11.3.5.5 ATWS Perspectives

An ATWS is an accident that is initiated by an event, such as loss of feedwater or turbine trip, followed by failure of the reactor to scram. The failure to scram can result either from failures of electrical components in the RPS or in the final control elements that de-energize the CRD mechanisms, or from mechanical failures involving failure of de-energized control rods to drop into the core. Partial scram failures are possible, involving insertion of a subset of the control rods; however, in most PRAs it is assumed that the scram failure is total and involves all of the rods. With failure of the rods to insert, power greatly in excess of decay heat loads is still being generated and overpressurization of the RCS is possible in some cases.

ATWS is not a major contributor to the total CDF for most plants in the group. ATWS frequencies for this group range from 1E-8/ry to 9E-6/ry and except for one plant contributes less than about 10% to the total CDF. The moderator temperature coefficient is a key factor in determining the susceptibility of a plant to ATWS events. Plants are generally more susceptible early in core life. For all of the plants with the highest CDFs from ATWS in the group, the following three types of failures contribute to core damage following an ATWS are listed in order of decreasing importance (1) inadequate pressure relief, (2) failure to borate, and (3) failure of AFW. Many of the factors affecting ATWS frequencies are actual plant-specific differences. One area where modeling is important is in the assumptions and modeling approach used to estimate the frequency of scram failures. Data are limited in this area and modeling assumptions can be quite important, particularly for the plants with the lowest ATWS frequencies.

The moderator temperature coefficient is critical in determining the susceptibility of a plant to ATWS events. The actual power level during an ATWS will vary, depending upon the reactivity coefficients (feedback coefficients that usually lead to power decrease as a result of fuel temperature increases and moderator density decreases). The most important coefficient is often the MTC, which can have unfavorable values (not sufficiently negative) early in core life. During this time, the unfavorable moderator temperature coefficient can prevent the power decrease necessary to preclude overpressurization of the RCS.

To successfully mitigate an ATWS, three functions are required (1) pressure control, (2) heat removal, and (3) reactor shutdown. Without pressure control the RCS can overpressurize and fail, causing a LOCA that cannot be mitigated. Heat removal in combination with reactor shutdown is required over the long-term to cool the core. The most demanding ATWS accident is loss of main feedwater followed by failure to scram. Loss of main feedwater results in loss of sufficient heat removal through the steam generators (AFW is insufficient). If the ATWS mitigation system actuation circuitry (AMSAC) fails to trip the turbine, the accident is made worse since this leads to early steam generator dryout; early steam generator dryout results in higher primary system pressure.

Heat removal following an ATWS caused by a loss of main feedwater requires use of AFW. Also, the fission process must be shutdown to lower power to where AFW cooling can match energy addition. Shutdown is accomplished by injection of borated water or, if the ATWS does not involve mechanical-related failures to scram, further operator action to insert or drop the control rods.

The core damage sequences involving ATWS can be categorized as follows:

- sequences involving early overpressurization
- sequences involving loss of AFW
- sequences involving failure to borate.

All of the ATWS sequences lead to early core damage, defining 2 hours or less as early. Sequences of the first type, overpressurization, occur very early.

Closed PORV block valves, AFW failures, and operator failure to borate produce the highest ATWS frequencies for this plant group. ATWS is typically a small contributor to the total CDF for a given plant. The following plants in the group have the highest CDFs from ATWS: Indian Point 3, Seabrook, Sequoyah, and Comanche Peak. Table 11.43 summarizes the important failures for the dominant ATWS sequences for these plants.

The plant in the group with the highest CDF from ATWS, Indian Point 3, operates with both pressurizer PORV block valves closed. This reduces the relief capacity of the primary system in the early phase of an ATWS accident to those provided by the safety valves alone. Operation with the PORV block valves closed is a major reason for Indian Point 3 having the highest CDF from an ATWS of all plants in the group. The model used in the Seabrook IPE for mitigation of an ATWS requires turbine trip (if above 40% power and main feedwater is lost), cooling of the steam generators, and operator action to shutdown the reactor. The dominant ATWS core damage sequences for Seabrook each involves failure of one of these three functions. The CDF from an ATWS at Sequoyah, 7E-6/ry, is close to that for Seabrook. The CDF from an ATWS at Sequoyah is dominated by failure to borate either because of hardware failures or because of failure of operator action to initiate boration.

Table 11.43 Contributors to ATWS for Westinghouse 4-loop plants with relatively high ATWS CDFs.

Plant	CDF from ATWS (1/ry)	Important failures contributing to CDF from ATWS
Indian Point 3	9E-6	Inadequate pressure relief
Seabrook	8E-6	Inadequate pressure relief [1], Failure of AFW, Failure to borate
Sequoyah	7E-6	Failure to borate, Inadequate pressure relief
Comanche Peak	5E-6	Failure of AFW, Inadequate pressure relief

[1] Caused by failure to trip turbine.

The plants with the lowest CDF from ATWS in the group are Zion (1E-8/ry) and Wolf Creek (3E-8/ry). The CDF from ATWS at Zion appears to be low because of lower-than-typical frequencies or probabilities used for initiating events of importance, failure of reactor trip, and failure of turbine trip. The CDF from ATWS at Wolf Creek is lowbecause of credit for two additional operator actions to manually insert control rods following failure to scram. These include (1) operator action to insert control rods from the control room and (2) operator action to local-manually open circuit breakers to remove power from CRD motor generator sets. The latter action lowers the CDF for the two dominant ATWS sequences by about a factor of 6.

11.3.5.6 Internal Flood Perspectives

An internal flood sequence involves a release of water into a plant location such that a plant trip is induced and safety systems are compromised at the same time. For example, if a flood causes water to enter an area containing electrical switchgear, a plant trip can occur along with failure of all plant systems dependent upon that switchgear. Systems that are considered to be independent may all fail, if they all contain equipment within the flooded location. The effects of internal flooding are highly plant-specific, depending on the layout of equipment within the plant and the relative isolation of rooms. Often, the systems most affected are support systems such as electric power and SW, which have plant-specific designs. Because of this diversity of design and layout, we expect that each plant will have different vulnerabilities to flooding and do not expect to draw many generic conclusions.

Internal flooding is an important contributor to the total CDF at some plants. The importance of internal flooding to the total CDF is highly plant-specific, depending on the layout of water bearing lines and vessels, flood propagation, and the locations of equipment needed to provide for shutdown cooling. Typically, important internal floods are those that affect important support systems, such as SW and electric power. Newer plants sometimes reflect better design characteristics, generally better separation, and tend to be less susceptible to floods.

The contribution from flooding varies from negligible to about 30% of the CDF. The CDF from flooding ranges from negligible to 2E-5/ry. Internal flooding is not a major contributor to the CDF for many plants, but for some plants it is an important contributor, comprising as much as about 30% of the total CDF.

The highest CDF from internal flooding for the group is 2E-5/ry at Callaway. For some of the IPEs, internal flooding is a negligible contributor to CDF and so the IPE submittals do not report an actual number, and the lowest CDF from internal flooding for the group is not known. The lowest CDF actually reported from internal flooding for the group is 4E-9/ry for Braidwood. The average CDF from the reported values is 6E-6/ry.

The plants with the highest CDFs from internal flooding are Callaway, Catawba, and Comanche Peak. Internal flooding, with a CDF of 2E-5/ry, contributes about 30% to the total CDF for Callaway. Three types of floods contribute to the CDF from internal flooding (1) rupture of SW piping in the area containing SW valves in the control building, (2) ruptures of SW and firewater piping in the AC switchgear rooms in the control building, and (3) ruptures of SW and firewater piping in the DC power, battery, and inverter rooms in the control building. Fairly high values are used in the IPE for failure of operator action to isolate or recover from flooding. Internal flooding, with a CDF of 1E-5/ry, contributes approximately 25% to the total CDF for Catawba. The IPE submittal attributes the high CDF from internal flooding to the location of all 6.9/14.6 KV transformers in the turbine building basement. Therefore, a flood in the turbine building can cause the loss of much of the plant electrical system. At Comanche Peak the CDF from internal flooding is 1E-5/ry, contributing about 25% to the total CDF. The important flooding sequences are as follows. A pipe from the RWST fails causing flood-induced loss of motor-driven AFW and ECCS pumps. A flood in the SW intake structure causes loss of all SW.

Vogtle represents the lower end of the CDF range. Results from the Vogtle IPE show that flooding-related CDF is negligible compared with the CDF from other accident classes. The Vogtle IPE attributes the negligible contribution of internal flood to the fact that Vogtle is one of the most recently licensed US nuclear plants, and as such has been designed to mitigate and limit effects associated with internal floods.

11.3.5.7 SGTR Perspectives

SGTR sequences involve leakage from the primary system to the secondary system through a ruptured steam generator tube, followed by either failure to mitigate the leak or failure to establish long-term core heat removal. There are several actions that are normally taken to prevent the tube rupture initiator from developing into a core melt sequence. To mitigate the leak, the affected steam generator is normally isolated, and if this does not succeed, actions are taken to depressurize the primary system so that leakage is minimized. The primary system is depressurized either by aggressively cooling down through the unaffected steam generator(s), cooling down the primary system using the pressurizer sprays, or by depressurizing by initiating feed-and-bleed through the pressurizer PORV(s). If the affected steam generator is isolated, long-term core heat removal can be established without inventory makeup, but if the leak is not isolated, inventory makeup must also be provided.

SGTR sequences are a minor contributor to plant CDF at the Westinghouse 4-loop plants, but can be important contributors to risk because the releases bypass containment. The higher CDFs reflect either plant weaknesses identified in the IPEs or IPE modeling that includes more possibilities for SGTRs combined with other plant threats (e.g., ATWS).

SGTR sequences are low contributors to plant CDF, but can be significant to risk because the releases bypass containment. Overall, SGTRs are a minor contributor to plant CDF, with an average SGTR CDF of 2E-6/ry. The average contribution to plant CDF is less than 5%. The individual plant results vary from a high of 8E-6/ry to a low of less than 1E-10/ry.

The Westinghouse 4-loop plant with the highest SGTR CDF is Haddam Neck. The largest contributors to the Haddam Neck SGTR CDF involve steam generator overfill leading to a stuck-open steam generator relief valve, and eventually, to core damage when the injection water source is depleted. For the plants with the lowest SGTR CDFs, Catawba and McGuire, such scenarios are assessed to have negligible frequency.

11.3.5.8 ISLOCA Perspectives

An ISLOCA occurs when valves that normally isolate the RCS from low-pressure systems (e.g., LPI) fail, resulting in backflow from the RCS through the low-pressure piping. If the low-pressure piping or other components (e.g., seals, relief valves, or flanges) can not withstand the resulting pressurization, then a LOCA results. If the breach occurs in a portion of the piping that is outside containment, then a LOCA that bypasses containment results, with effluent from the LOCA being discharged into the reactor building or auxiliary building. Because it is not possible to recirculate the coolant through the RCS for this type of LOCA, coolant injection will eventually be lost (leading to core damage) unless some source of sustained makeup is available. These scenarios are typically low probability events, because they involve multiple valve failures (typically two check valves and one motor-operated valve in a series). However, ISLOCAs can be important to risk because the containment bypass leads to larger fission product releases.

ISLOCAs are minor contributors to plant CDF in all of the IPE submittals, but in some cases, there is a large enough contribution to be risk significant. Although there are important differences in plant characteristics that affect the results, the variability in results appears to be predominantly influenced by differences in modeling among the plants. Further details regarding the variability are provided in the remainder of this section.

ISLOCAs are low contributors to plant CDF but can be a significant risk because releases bypass containment. The CDFs from ISLOCAs are generally low, with an average ISLOCA CDF for the Westinghouse 4-loop plants of 4E-6/ry and an average fractional contribution to plant CDF of less than 1%. The individual plant results vary from negligible ISLOCA contribution (less than 1E-8/ry) to a high of 4E-6/ry, with a percent contribution to plant CDF ranging from negligible to less than 5%.

The IPE submittals show considerable variability for ISLOCA CDFs and dominant ISLOCA contributors. This is because of both differences in modeling among the IPEs, as well as differences in plant characteristics. Modeling differences are particularly large for ISLOCA sequences because several aspects of ISLOCA modeling involve considerable uncertainty, such as the probability of pipe ruptures or the ability of motor-operated valves to close under pressure. Some IPE submittals include detailed descriptions of their ISLOCA modeling but most include very little discussion, making it difficult to determine the key drivers for those plants. Because of this limited discussion of ISLOCAs in the majority of the IPE submittals, this section will focus on simply discussing the factors (both modeling and plant characteristics) that appear to be most important, on the basis of the information that could be obtained from the submittals with detailed descriptions of ISLOCA. Factors affecting the initiator frequency are discussed first, followed by factors that affect the ability of the plant to respond to the ISLOCA.

There is considerable variability in the initiator frequency for ISLOCAs, varying over about four orders of magnitude. However, much of this variability simply reflects the different level of reporting ISLOCA results among the IPE submittals. Some plants report ISLOCA initiator frequencies that represent only failures of the valves that are designed to isolate the low-pressure systems from the RCS, while others also include the probability of rupturing the low-pressure system as part of the initiator frequency. Because of the limited descriptions in most IPE submittals, it is not possible to back out the values used for the valve failures and rupture probabilities for all plants. However, on the basis of the IPE submittals where this information is provided, there appears to be more variability in the rupture probabilities than in the valve failure probabilities.

To determine the potential for a rupture, some IPEs included detailed analyses of the potential for ISLOCAs in the various lines, considering factors such as the strength of the various pipes and the relative length of piping inside and outside of containment. However, some licensees simply used the ISLOCA frequency from WASH-1400. The modeling of the ruptures in the lines also varies among the IPEs. Some licensees distinguish between pipe ruptures and other components (e.g., seals, relief valves, or flanges), giving more credit for recovering from smaller ruptures.

Although the variability in valve failure rates does not appear to be as large as the variability in rupture probabilities, there are significant differences in plant characteristics that can affect the frequency of the multiple valve failures that lead to the ISLOCA initiator. Typical valve arrangements are two-check valves and a motor-operated valve in series, two check valves in series, or one check valve plus a motor-operated valve. There is also variability as to whether the motor-operated valves are normally open or closed. Normally open valves lead to a higher probability of ISLOCAs. In addition, valve testing is performed while at full power at some plants, while at other plants the valves are only tested during shutdown. Although testing the valves at full power would be expected to increase the probability of ISLOCAs, most of the IPEs do not reflect an increased risk because the licensees take credit for system interlocks that are instituted during testing.

Reactor building or auxiliary building design (e.g., compartmentalization) affects the plant's ability to cope with an ISLOCA. In some plants, harsh environments resulting from the ISLOCA will fail some or all injection systems. In other plants, only the system directly involved in the ISLOCA is unavailable.

There is considerable variability among the IPEs regarding the treatment of operator actions to mitigate an ISLOCA. The major actions considered are manually closing motor-operated valves so that the failed line is isolated, depressurizing the RCS so that the leakage is minimized, or replenishing water supplies for injection. In some cases, the licensees give a large amount of credit for one or more of these actions, while in other cases, none of the actions are credited.

12. CONTAINMENT DESIGN PERSPECTIVES

This chapter presents the perspectives obtained on the treatment and results of containment performance reported in the Individual Plant Examination (IPE) submittals. These perspectives are summarized in Chapter 4. Additional details are provided in this chapter on the factors that play a significant role in determining the containment failure probabilities and frequencies reported. The key design and operational features that affect the containment performance and the impact and influence of methods and assumptions on containment performance are provided for different containment types. Perspectives regarding the bypass, early and late containment failure modes, defined in Table 12.1, are obtained for each containment type. Quantitative information involving ranges of probabilities and frequencies of containment failure modes and releases are presented. Perspectives generally applicable to boiling water reactor (BWR) pressure suppression containments and pressurized water reactor (PWR) containments were also obtained. These general perspectives are presented first followed by the more detailed discussions for each containment type.

Table 12.1 Definition of containment failure mode classes.

Failure mode	Containment failure mode definition
Bypass	Involves failure of the pressure boundary between the high-pressurereactor coolant system and a low-pressure auxiliary system. For PWRs it can also occur because of 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	Involves structure failure of the containment before, during or slightly after reactor vessel failure, usually within a few hours of the start of core damage. A variety of mechanisms can cause structure failure such as direct contact of the core debris with the containment, rapid pressure and temperature loads, hydrogen combustion and fuel-coolant interactions. Failure to isolate containment and an early vented containment post core damage is classified as an early containment failure.
Late	Involves structural failure of the containment several hours after reactor vessel failure. A variety of mechanisms can cause late structure failure such as gradual pressure and temperature increases, hydrogen combustion, and basemat melt-through by the core debris. Venting containment late in an accident is classified as a late containment failure.

12.1 General Containment Performance Perspectives

Containment performance is often measured by calculating the conditional containment failure probability (CCFP).[12.1] These probabilities were reported in most of the IPEs or could be calculated from the reported results. The CCFPs are reproduced in Figure 12.1 for various containment failure modes and for BWR and PWR plants. When the accident progression analyses in the IPEs are globally viewed, they are, for the most part consistent with containment performance analyses previously performed for probabilistic risk analyses (PRA). Failure mechanisms, identified in the past as being important, are also shown to be important in the IPEs. 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 have early structural failures than the smaller BWR pressure suppression containments. However, as indicated in Figure 12.1, there is a considerable variability within each containment class in the conditional containment failure probabilities reported. This variability also exists in the reported frequencies, as shown in Figure 12.2.

[12.1] 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.

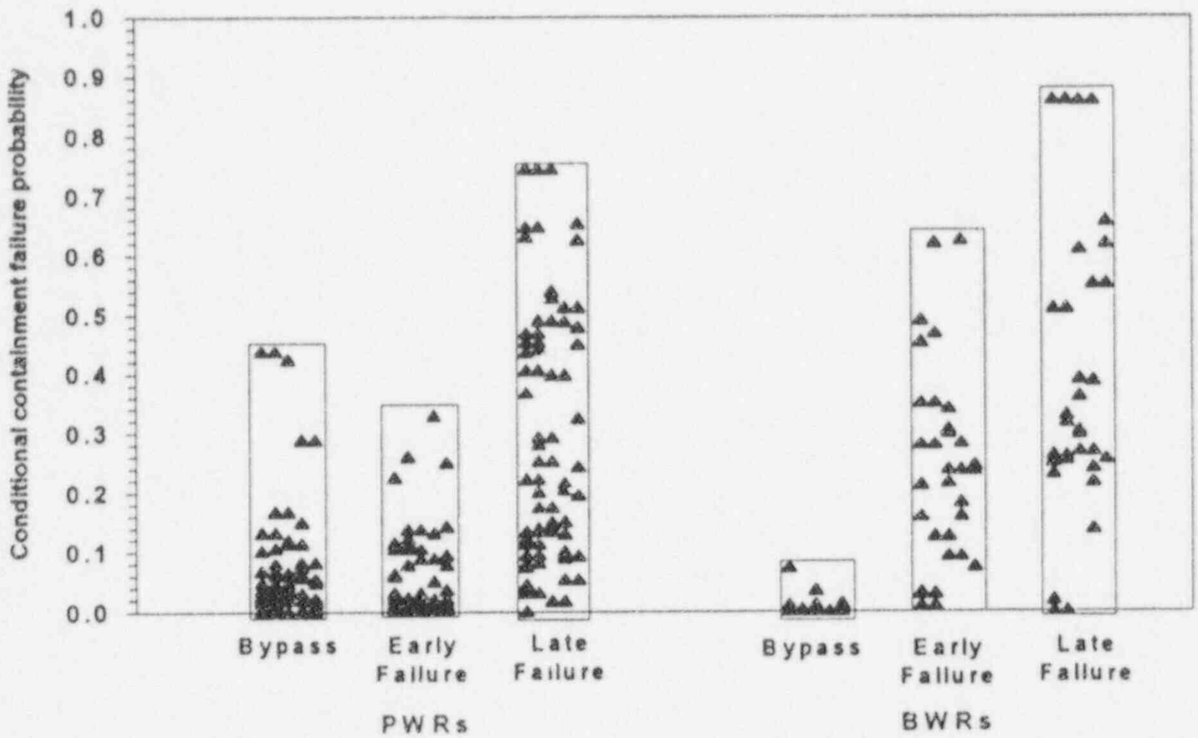


Figure 12.1 Reported IPE CCFPs (given core melt) for all plants.

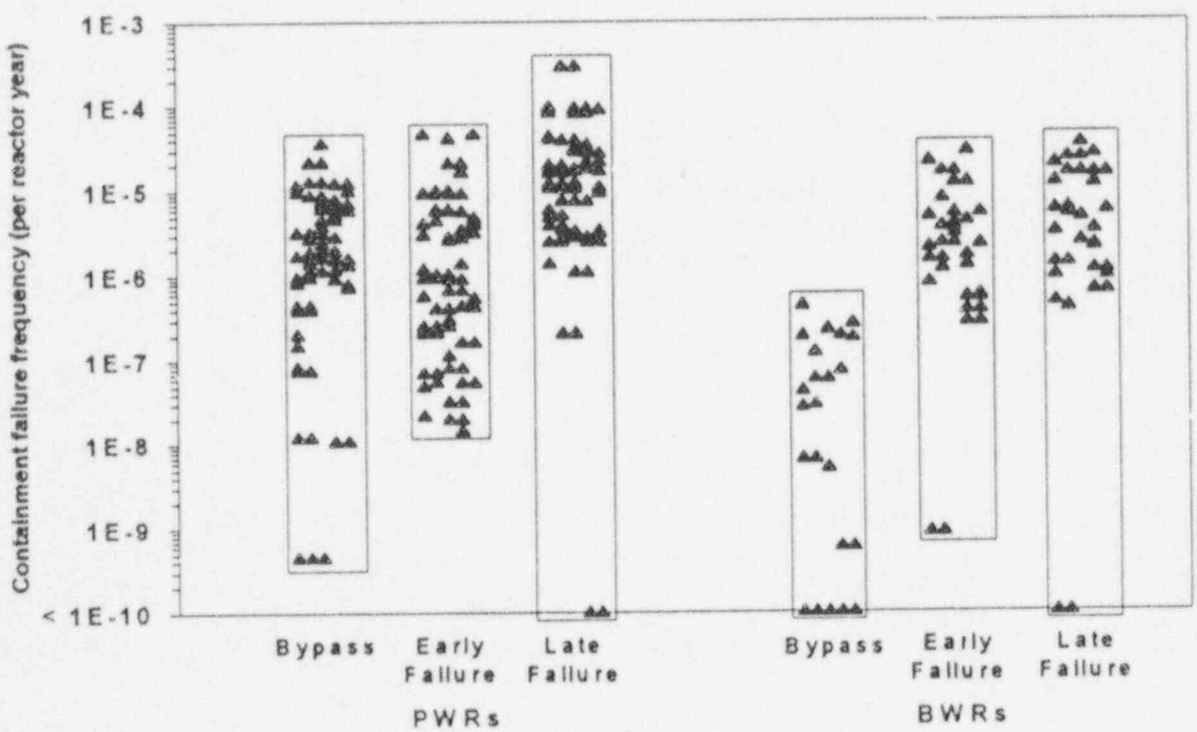


Figure 12.2 Reported IPE containment failure frequencies for all plants.

The importance of early radionuclide releases to all risk measures, (i.e., acute and latent health effects including land contamination), has been established in past PRAs which included consequence calculations. In keeping with the significance of such early releases, the containment performance analysis descriptions in the IPE submittals emphasize the phenomena, mechanisms, and accident scenarios which 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.

On the one hand, the PWR large dry containments analyzed in the IPEs have significantly smaller conditional probabilities of early structural failure than the BWR pressure suppression containments analyzed. On the other hand, containment bypass, as well as isolation failures are, in general, more significant for the PWR containments. As Figure 12.1 shows, these general trends are often not true for individual IPEs, because of the considerable range in the results. For instance, CCFPs for both early and late failure found for a number of PWR large dry containments are higher than those reported for some of the BWR pressure suppression containments.

Differences in containment designs account for much of the differences in failure probabilities indicated in Figure 12.1. This is true for the variations between containment classes but also for differences between individual plants in the same containment class. In a significant number of cases unique, plant-specific containment features are identified in the analyses as leading to important failure mechanisms. However, differing assumptions in the accident progression modeling also play a major role in explaining the significant range in the obtained results.

Since there is still considerable uncertainty regarding the loads imposed on containments by the phenomena postulated in an accident progression analysis, differences in modeling assumptions are not surprising. Key observations on containment performance are summarized in Table 12.2.

12.2 BWR Containment Performance Perspectives

The BWR plants are separated into three groups, according to the type of pressure suppression containment used, for the purpose of identifying containment performance perspectives from the submitted IPEs (1) BWR Mark I Containments, (2) BWR Mark II Containments, and (3) BWR Mark III Containments. The BWR plants in each group are indicated in Table 12.3. One early BWR, Big Rock Point, is housed in a large dry containment and therefore is discussed in Section 12.3.1.

The results indicated in Figure 12.3 for containment failure probabilities and in Figure 12.4 for failure frequencies follow expected trends and indicate that the early Mark I containments are, in general, more likely to fail during a severe accident than the later Mark II and Mark III designs. However, the ranges of predicted failure probabilities are quite large for all containment designs and there is significant overlapping of the results. The variability in the results can be expected partly because of containment design differences involving containment design pressure, ultimate pressure, containment volume, containment construction (e.g., steel vs. concrete), and reactor thermal power. Table 12.4 shows the values of these parameters for all domestic BWR containments. Figure 12.5 shows the range of design pressure and ultimate pressure within the BWR pressure suppression containments and compares these ranges between containment types. Figure 12.6 indicates the range of the ratio of containment volume to thermal power in the BWR plants grouped by containment type. However, the variability in the containment failure results of Figures 12.3 and 12.4 is also attributable to other differences such as the reactor pedestal and drywell floor configuration, ability to flood the drywell, and combustible gas control; as well as modeling assumptions and differences in recovery actions that could be taken during a severe accident.

Table 12.2 **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
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.

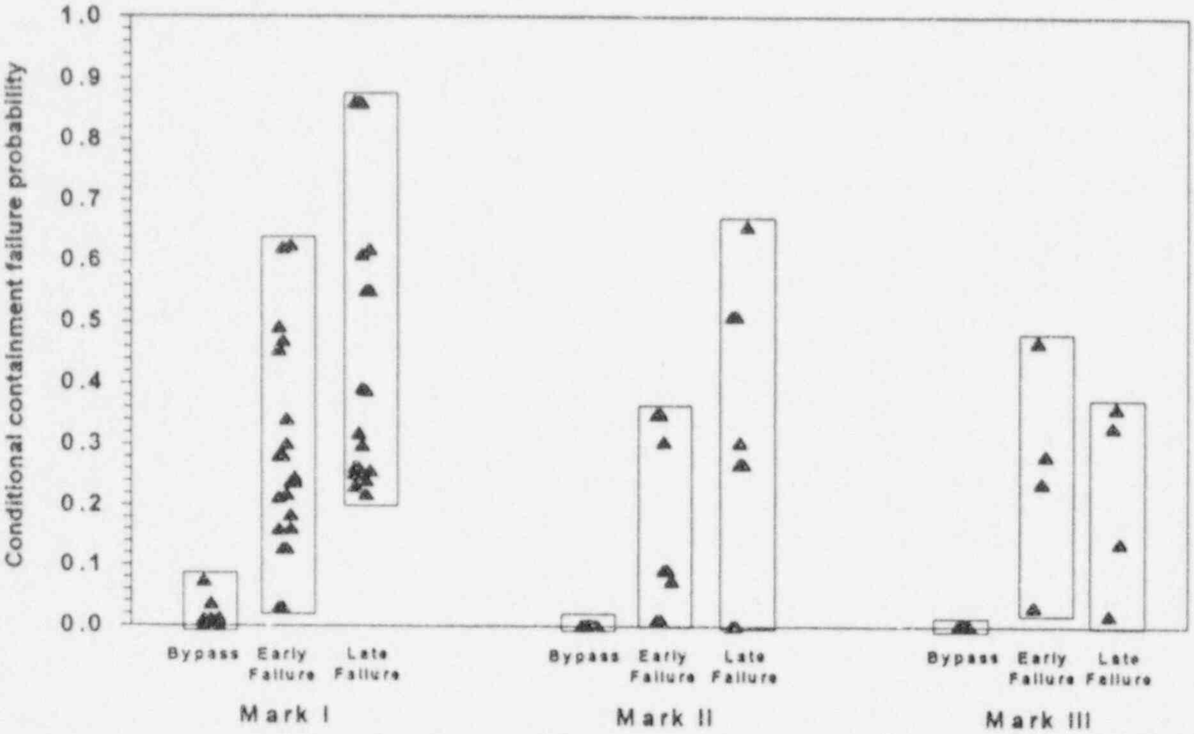


Figure 12.3 Reported IPE CCFPs (given core melt) for BWR plants

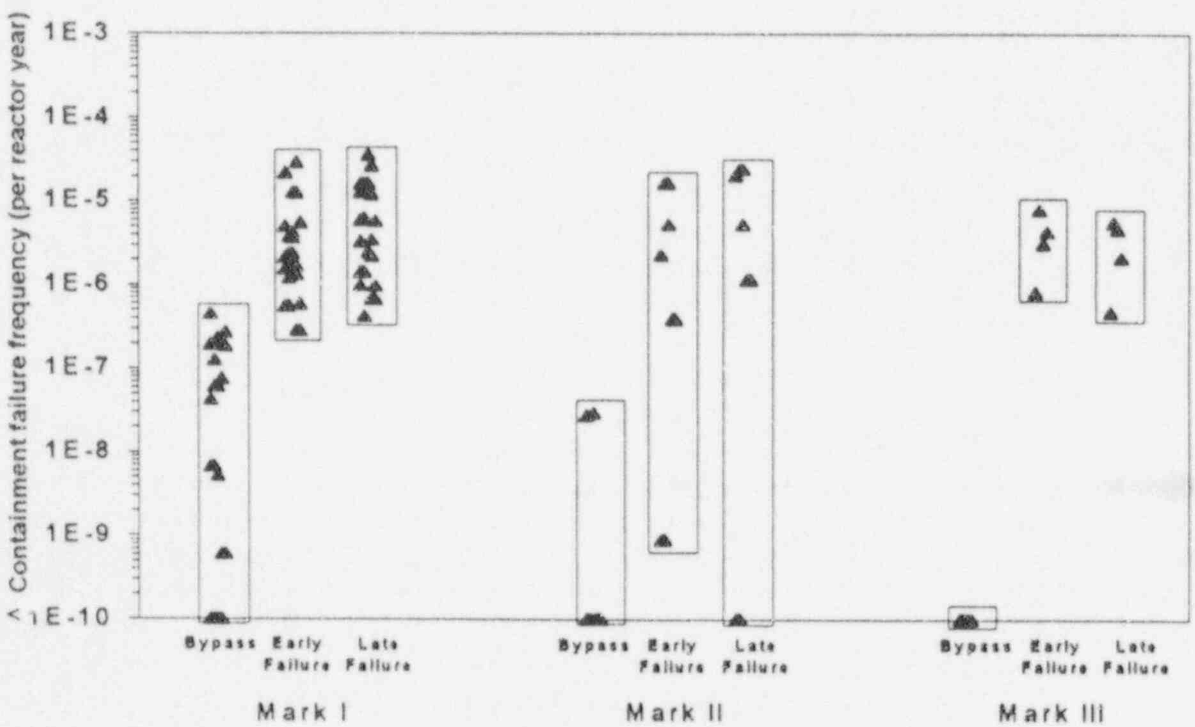


Figure 12.4 Reported IPE containment failure frequencies for BWR plants.

Table 12.3 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.

A large variability exists for each containment group in the contributions of the different failure modes. 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.

Important factors that impact the probabilities and frequencies of the failure modes in Figures 12.3 and 12.4 are discussed for each BWR containment group in Sections 12.2.1 through 12.2.3. In general, the factors that influence the failure modes are not the same for each containment group. This is often because of differences in containment design between the three groups. For example, shell melt-through (caused by contact with the 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 the core debris can reach the containment wall (which is usually steel). The core debris cannot easily reach the containment wall in Mark II and Mark III containments, and therefore, other failure mechanisms are found to be important for these designs. Accidents in which containment heat removal is lost or is inadequate are found to be important contributors to early failure in Mark II plants. Early failure is primarily caused by energetic events such as fuel-coolant interactions and hydrogen combustion events in Mark III plants. Hydrogen combustion is unlikely in Mark I and II plants because their atmospheres are inerted during operation.

Table 12.4 Containment properties for BWRs

Type	Plant name	Containment design pressure (psig)	Ultimate pressure (psig)	Containment volume (ft^3)		Construction*	Thermal power (MWt)	Volume/ thermal power (ft^3/MWt)
				Dry well	Wet well			
L-DRY	BIG ROCK POINT	27	79	1,150,000		S	240	4792
MK I	BROWNS FERRY 2	56	190	159,800	126,200	S	3293	87
MK I	BRUNSWICK 1&2	62	141	164,100	124,000	R	2436	118
MK I	COOPER	56	175	132,250	110,300	S	2381	102
MK I	DRESDEN 2&3	62	125	158,236	117,245	S	2527	109
MK I	DUANE ARNOLD	56	140	109,400	94,270	S	1593	128
MK I	FERMI 2	56	140	163,730	130,900	S	3293	89
MK I	FITZPATRICK	56	140	150,000	114,000	S	2436	108
MK I	HATCH 1	56	98	146,010	112,900	S	2436	106
MK I	HATCH 2	56	98	146,266	109,800	S	2436	105
MK I	HOPE CREEK	56	120	178,000	142,600	S	3293	97
MK I	MILLSTONE 1	62	149	146,900	110,600	S	2011	128
MK I	MONTICELLO	56	120	134,000	106,000	S	1670	144
MK I	NINE MILE POINT 1	62	119	180,000	120,000	S	1850	162
MK I	OYSTER CREEK	35	121	180,000	121,300	S	1930	156
MK I	PEACH BOTTOM 2&3	56	140	175,800	127,700	S	3293	92
MK I	PILGRIM 1	56	98	147,000	110,000	S	1998	129
MK I	QUAD CITIES 1&2	56	NP	158,236	117,000	S	2511	110
MK I	VERMONT YANKEE	56	140	134,200	108,250	S	1593	152
MK II	LA SALLE 1&2	45	191	209,300	164,500	R	3293	114
MK II	LIMERICK 1&2	55	140	243,580	147,670	R	3293	119
MK II	NINE MILE POINT 2	45	141	306,200	190,600	R	3323	150
MK II	SUSQUEHANNA 1&2	53	140	239,600	153,800	R	3293	119
MK II	WNP 2	45	148	200,540	144,184	S	3323	104
MK III	CLINTON	15	94	246,000	1,550,000	R	2894	621
MK III	GRAND GULF 1	15	56	270,000	1,400,000	R	3833	436
MK III	PERRY 1	15	64	276,500	1,160,000	S	3579	401
MK III	RIVER BEND	15	63	251,000	1,200,000	S	2894	501

*S = Steel; R = Reinforced Concrete
NP = Not provided in submittal
L-Dry = Large-dry containment
MK = Mark
MWt = Megawatt thermal

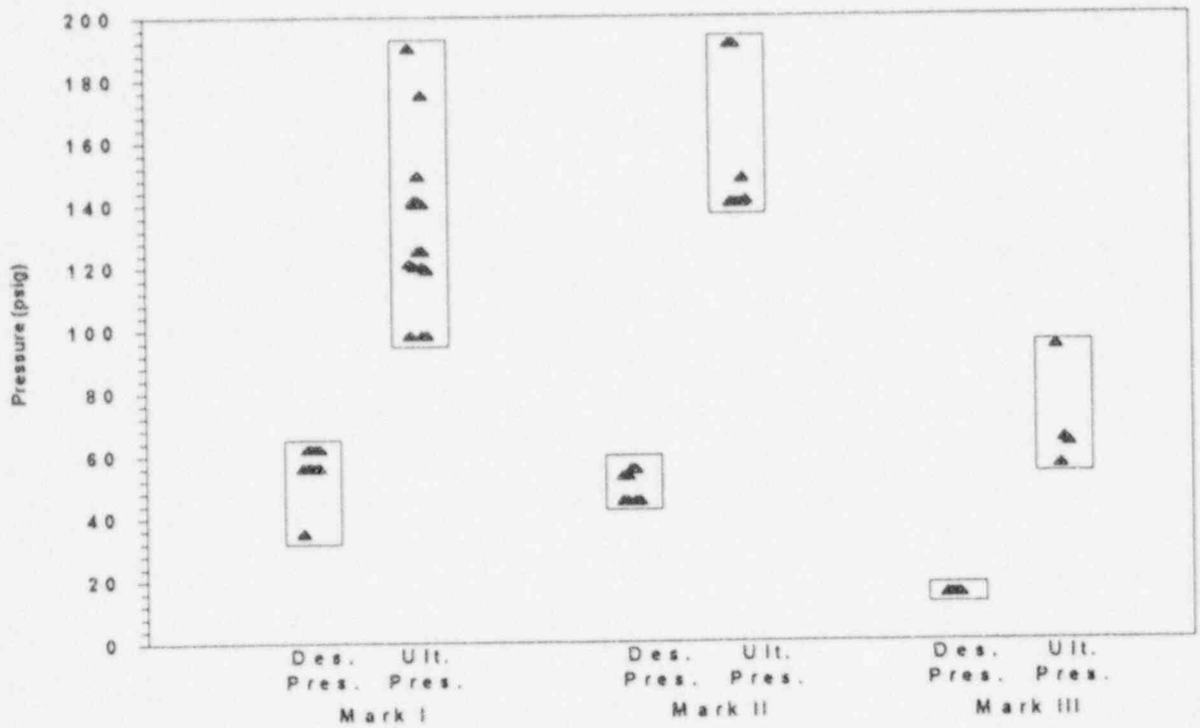


Figure 12.5 Range of containment properties for BWR plants.

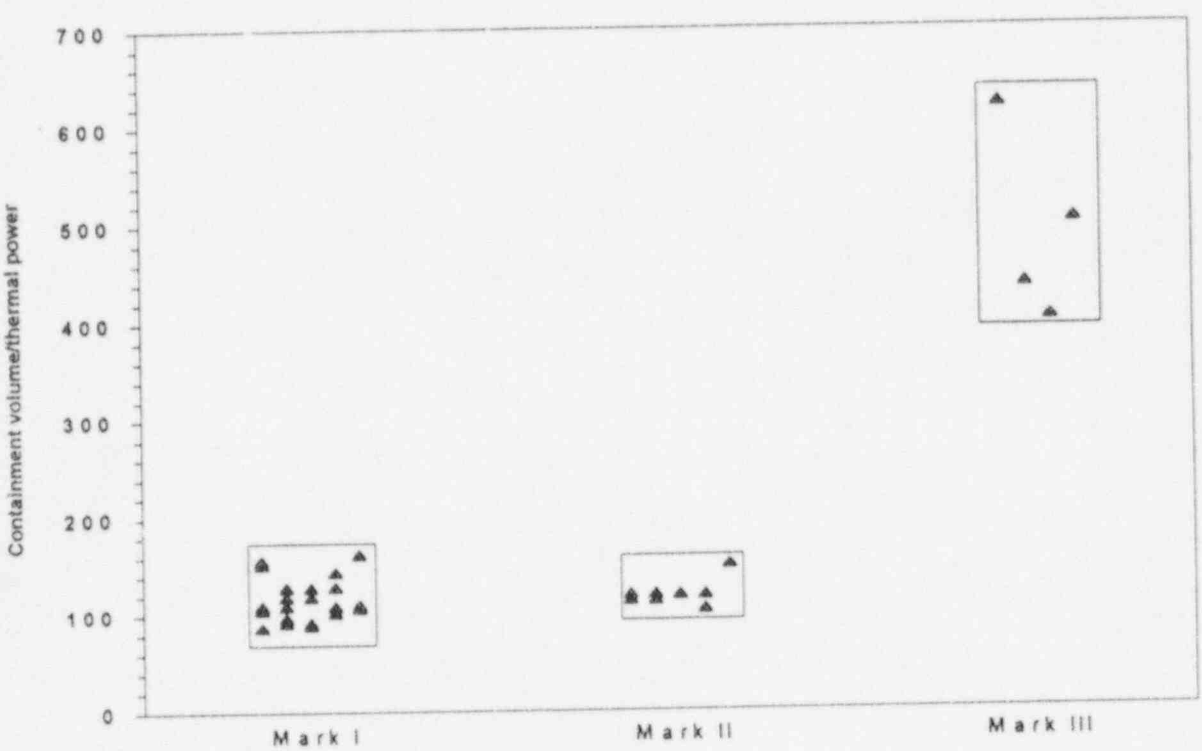


Figure 12.6 Range of containment volume to thermal power ratio for BWR plants.

Late containment failure can be caused by gradual pressure buildup because of non-condensible gas release, basemat melt-through or hydrogen combustion events. Gradual pressure buildup caused by core-concrete interactions 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 Mark III containments. Finally, venting can be important to the probability of loss of containment integrity (early and late in an accident sequence) for some BWR plants. Differences in the modeling of venting contribute to the variability of the results. Perspectives for all operating BWRs using pressure suppression containments are summarized in Table 12.5.

Table 12.5 Summary of performance for BWR containments.

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-pressureloads 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 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. 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.	Alternate 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-pressureand 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 also is found to lead to late failure in some cases. Late combustible gas burns are important in some Mark IIIs. 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

12.2.1 BWR Mark I Perspectives

Twenty-two BWR units (17 IPE submittals) are housed in Mark I containments, as listed in Table 12.6. 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.

Table 12.6 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	

12.2.1.1 Summary of Results and Perspectives for BWR Mark I Containments

The conditional probabilities of the various containment failure modes reported in the IPEs are provided in Figure 12.7 and the reported frequencies of containment failure are shown in Figure 12.8. Results are summarized in Table 12.7. The average conditional containment failure probabilities for the various containment failure modes for all Mark I containments considered are about 0.3 for early failure, 0.01 for bypass, 0.4 for late failure, and 0.3 for no containment failure. The distributions of the various containment failure modes found in the 17 IPE submittals show a significant spread in the results, with the conditional probability varying from negligible to about 0.1 for bypass, from 0.03 to 0.6 for early failure, from 0.2 to 0.9 for late failure and from 0.1 to 0.6 for no containment failure. The probability of early failure for most of the Mark I plants is greater than 0.15, two plants have probabilities below 0.15. As indicated in Figure 12.8, the frequencies of the various containment failure modes vary from negligible to 5E-7/ry for containment bypass, from 3E-7/ry to 3E-5/ry for early failure, and from 4E-7/ry to 4E-5/ry for late failure.

Shell melt-through is a significant contributor to early containment failure in most of the IPE submittals reviewed. The importance of shell melt-through varies among the Mark I IPEs depending on plant specific features, but also on whether it was considered in the base case analyses and what underlying assumptions were made regarding the transport and cooling of core debris. The IPEs reported that even when shell melt-through has a small probability there usually is still a significant probability of early failure from other causes. Early containment venting, which is used during implementation of containment flooding procedures, sometimes involving drywell venting or venting through the reactor pressure vessel (RPV) as the predominant venting mode, is the dominant early release mode in one IPE. Containment bypass is caused mainly by interfacing systems loss-of-coolant accidents (ISLOCA), but one IPE identified the failure of the emergency condenser tubes because of high temperature creep rupture as another bypass mode. Isolation failure is small and negligible for all Mark I IPEs, partly because of the fact that the containment is inerted and isolation failure can be easily detected. Containment venting is most commonly used for containment pressure control, and therefore, usually results in a late release.

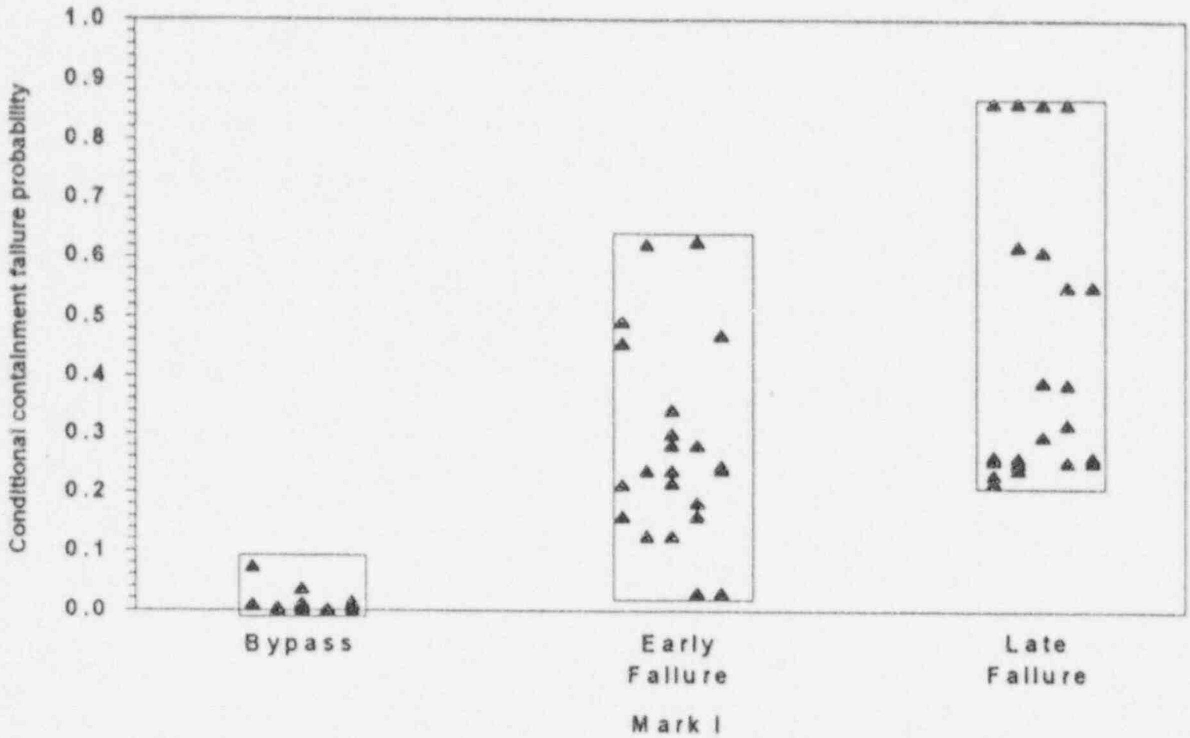


Figure 12.7 Reported IPE conditional probabilities of failure for BWR Mark I containments.

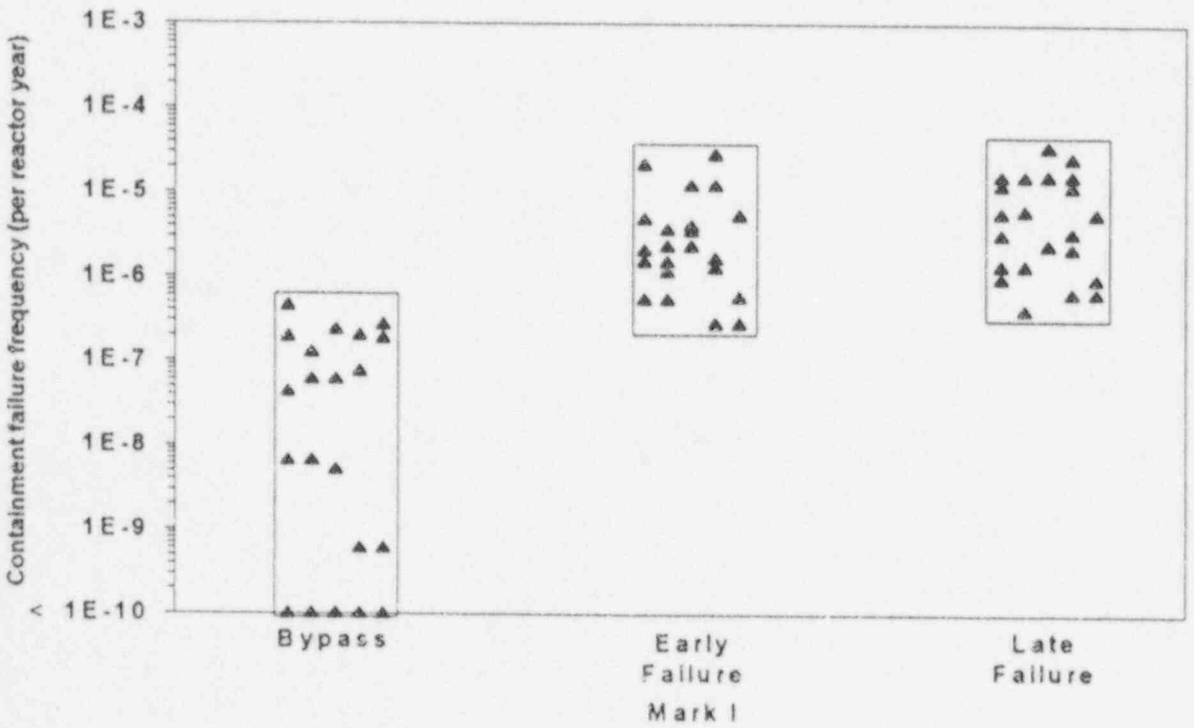


Figure 12.8 Reported IPE containment failure frequencies for BWR Mark I containments.

Table 12.7 Summary of performance for BWR Mark I containments.

Failure mode importance	Important design features, operator actions, and model assumptions	Summary of results
Early failure		
High probability for most plants in this group	Shell melt-through is found to be the most important contributor to early failure for Mark I plants, whose steel containment and reactor pedestal to drywell communication makes them susceptible to direct impingement by core debris. 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-pressureloads at the time the core debris melts through the reactor vessel are also a significant contributor to early failure. ATWS sequences are found to be important contributors in some IPEs. Isolation failures are not important.	Early failure frequencies range from 3E-7/ry to 3E-5/ry. Average frequency is 5E-6/ry. Early failure CCFPs range from 0.03 to 0.6. Average probability is 0.3.
Bypass		
Not important for this plant group.	Bypass via emergency condenser is identified by one IPE.	Bypass frequencies range from negligible to 5E-7/ry. Average frequency is 9E-8/ry. Bypass CCFPs range from negligible to about 0.1. Average probability is 0.01.
Late failure		
Significant probability for most plants in this group.	High-pressureand temperature loads caused by CCI are an important failure mode. Specific design features (e.g., 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.	Late failure frequencies range from 4E-7/ry to 4E-5/ry. Average frequency is 9E-6/ry. Late failure CCFPs range from 0.2 to 0.9. Average probability is 0.3.

The results indicate a significant probability of early and/or late containment failure for most of the Mark I containments. Those 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 depend on plant-specific features and in some cases on modeling assumptions. However, the following mechanisms are 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
- 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 means radionuclides released from the damaged core bypass the suppression pool (significant retention can occur if aerosol radionuclides pass through a suppression pool). The relatively short time to radionuclide release and the magnitude of the release means 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 include 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. In a few IPEs other failure mechanisms are identified as being important. Drywell failure caused by gradual pressure (and temperature) buildup because of gases and steam released during CCI is important in some IPEs.

In other IPEs, venting is found to be an important contributor. However, accidents that bypass containment (such as ISLOCA) or involve containment isolation failure are not important contributors to the core damage frequency (CDF) in any of the IPEs for Mark I plants. These accidents are 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.

12.2.1.2 Early Failure Perspectives

The conditional probabilities of early failure for the Mark I containments vary from 0.03 for Dresden to over 0.6 for Fitzpatrick and Hope Creek (refer to Figure 12.7). Early failure frequencies, as shown in Figure 12.8, vary from about 3E-7/ry for Quad Cities 1,2 to 3E-5/ry for Hope Creek.

The early failure category presented in Figures 12.7 and 12.8 includes containment isolation failure, early containment venting, and containment failure before or shortly after vessel breach (VB), which includes shell melt-through, as well as other early containment failures such as those as a result of overpressure. Shell melt-through is a significant contributor to early containment failure in most of the IPE submittals reviewed. Early containment venting plays an important role in early failure in one IPE, but in general, venting is used for containment pressure control later in the accident and is usually grouped with late failure.

Isolation failure is small and negligible for all Mark I IPEs, partly because of the fact that the containment is inerted and an isolation failure can be easily detected.

In addition to the contributions from shell melt-through and early venting, the values for early failure presented in Figures 12.7 and 12.8 also include early containment failures because of other phenomena that would occur during a severe accident. The potential containment failure modes suggested in NUREG-1335[12.2] for early containment failure include (1) overpressurization because of non-condensible gases and steam or because of direct containment heating (DCH), (2) missiles and pressure loads as a result of steam explosion, and (3) vessel thrust force because of blowdown at high pressure. For Mark I containments, these include the containment failure induced by reactor pedestal failure caused by either a quasi-static overpressurization of the pedestal cavity or a dynamic loading on the pedestal walls from an ex-vessel steam explosion following vessel breach. Although unlikely, the containment may also fail by hydrogen burns if the containment is deinerted, and this possibility is discussed in most of the IPE submittals. Other containment failure modes considered in the IPEs include the probability of containment implosion as a result of drywell spray initiation (the external design pressure for Mark I containments is about 2 psig), recriticality, and debris impingement on the containment boundary as a result of high-pressure ejection from the vessel.

In a number of cases, statements are made in the IPE submittals dismissing a number of failure modes without much discussion or presentation of quantitative analysis. A typical example can be found in the Hatch IPE submittal:

> "The containment challenges and uncertainties presented by steam explosion, direct containment heating, hydrogen detonation and deflagration, and thrust forces at vessel breach are well within the structural capability of the containment."

Another example is the Millstone 1 submittal which finds that high-pressuremelt ejection and the associated potential for early failure from direct containment heating are found not to be threats to the containment. Similarly, neither ex-vessel steam explosions or hydrogen combustion are of major concern.

The level of detail and the type of information presented in the various IPE submittals are not consistent. For example, in some IPE submittals, although the total contributions from shell melt-through and containment venting are provided, their contributions to early failure and late failure categories cannot be obtained. Furthermore, the definition of the time frames used for early failure and late failure are different among the IPEs. Early failure is defined as occurring either (1) before or shortly after VB, (2) within a certain time of core damage (CD), or (3) within a certain time (e.g., 6 hours) of accident initiation. The first two definitions are, in general, consistent because the time between CD and VB is usually within a reasonable time range if the sequence proceeds to VB. The third definition may be consistent with the other two definitions if the sequences are not ones involving the loss of containment heat removal (CHR). In the loss of CHR sequences, containment failure may occur before VB but many hours after accident initiation.

Shell melt-through was 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 because, for most Mark I containments, the reactor pedestal and the drywell floor are at the same level and openings exist between the pedestal region and the floor. This design allows the core debris to flow across the drywell floor and fail the steel drywell shell either by direct melt-through or via creep rupture.

[12.2]USNRC, "Individual Plant Examination: Submittal Guidance," NUREG-1335, August 1989.

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 is found to mitigate shell melt-through in all of the submittals. The benefit of water on the drywell floor before vessel failure as a mitigating mechanism for shell melt-through was found to be significant in NUREG/CR-5423[12.3] 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 prevent it from reaching the containment boundary. In the Oyster Creek drywell, a concrete curb prevents or limits the core debris from reaching the containment shell. Also, the Brunswick containment is unique among Mark I designs because it is of concrete (with a steel liner) rather than steel construction. 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 core debris at high temperature 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 eliminate the uncertainty regarding containment failure caused by this failure mechanism by ensuring a flooded drywell floor.

A number of utilities are being proactive and are identifying minor hardware modifications and changes in procedures to ensure a flooded drywell floor before reactor vessel melt-through. The availability of alternate water sources to the drywell spray header, such as water from a diesel- driven fire pump during a station black-out (SBO), is shown to significantly reduce the likelihood of early failure in the Browns Ferry IPE. Another example is the Monticello plant where connections are available that enable the operators to use residual heat removal (RHR) service water for containment spray. The Nine Mile Point 1 submittal mentions the potential benefit of supplying the drywell sprays from external sources such as the containment spray raw water pumps. 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 Duane Arnold and Monticello, also discuss the possibility of relaxing the restrictions on drywell spray initiation in the current emergency operating procedures (EOPs), providing greater assurance that there would be water on the drywell floor.

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

Shell melt-through usually results in early failure.[12.4] The distribution of conditional probability of shell melt-through is compared to the distribution of total conditional probability of early failure for a number of Mark I IPEs in Figure 12.9. Since the probability value of shell melt-through is grouped into other categories (e.g., the early failure or late failure category) and is not provided separately in some of the IPE submittals, simple calculations (e.g., using the results presented in the containment event tree (CET) figures) and interpretation were sometimes needed to derive the data for this report. Although these derived values may not be precise, they should provide a reasonable picture of the significance of shell melt-through in the IPE submittals for plants with Mark I containments. Figure 12.9 shows significant variation in the shell melt-through data from the various IPE submittals, with values ranging from zero to close to 0.5.

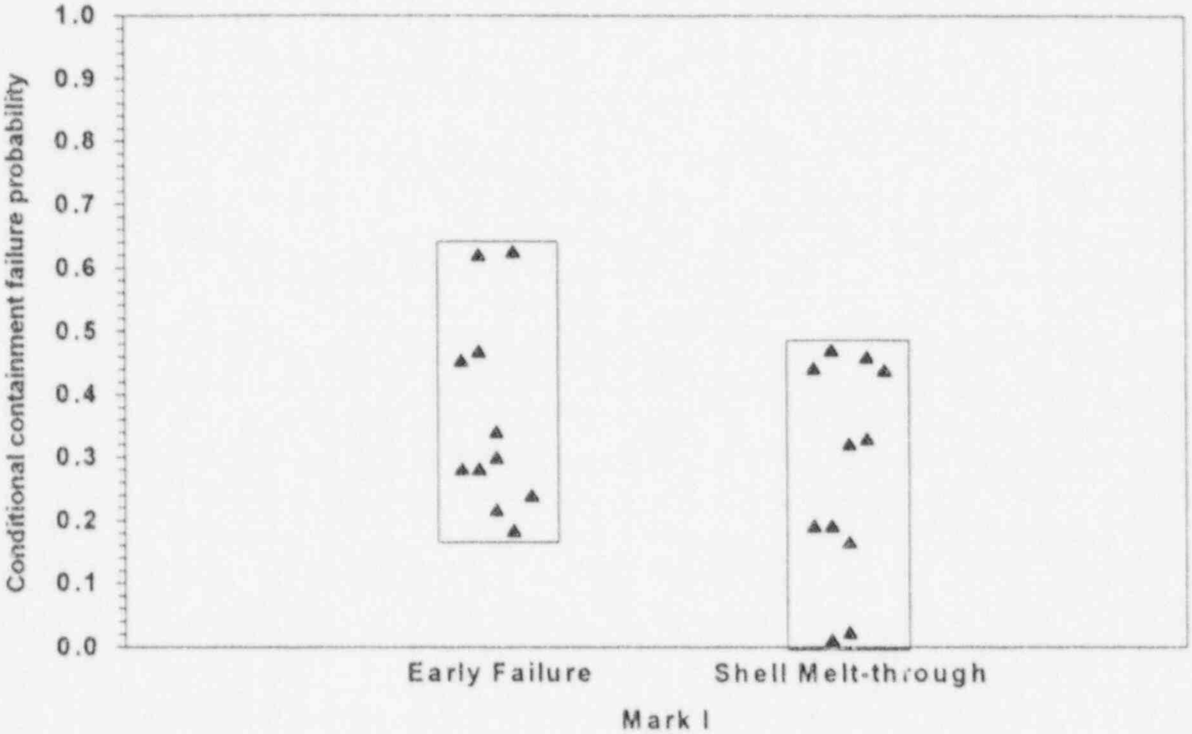


Figure 12.9 Comparison of some Mark I IPE reported probabilities for shell melt-through and total early failure.

While shell melt-through is usually an important contributor to early failure, there are many exceptions because of modeling assumptions. Since the probability of shell melt-through involves significant uncertainty, some IPEs do not consider this failure mode in the base case analysis but consider it in the sensitivity study. Shell melt-through is treated as a sensitivity issue in two of the IPEs, Dresden and Quad Cities, accounting for 4 units. As previously noted, shell melt-through is not considered an important issue in the Brunswick IPE because the drywell for Brunswick is of concrete construction. Shell melt-through is also found to be negligible in the Hatch, as well as in the Monticello, and the Oyster Creek IPEs mentioned earlier.

[12.4] The exception appears to be the Nine Mile Point Unit 1 submittal, where the total probability assigned to "Shell and Drywell Head" failure (grouped with the shell melt-through category in this report) is about 0.4, greater than the total early failure probability of about 0.2. Apparently, some of this shell melt-through failure probability is assigned in the IPE to the late failure category.

According to the Hatch IPE submittal, plant-specific considerations indicate that shell melt-through will not occur for sequences where the core debris can be covered by an overlying pool of water for the duration of the sequence.

It is further assumed in the Hatch IPE that, for dry sequences, a thin debris layer could reach the shell but may not be deep enough or hot enough to ablate the shell and fail the containment. The probability of shell melt-through is, therefore, negligible in the Hatch IPE. In the Monticello IPE, shell melt-through is considered negligible because the containment sumps are large in relation to the size of the core such that the volume of the core debris expected to be released into the pedestal when the vessel fails is not enough to overflow the sumps. Consequently, molten core debris will not contact the shell and melt-through will not occur. In the Oyster Creek IPE, a drywell floor concrete curb, which prevents or limits the debris reaching the containment shell, is the main contributor in reducing the likelihood of shell melt-through.

In addition to the above IPEs, where the contribution of shell melt-through to containment failure is considered negligible, the shell melt-through probabilities are also small, about 0.02, in the IPEs for Duane Arnold and Fermi, partly because of the assumption that shell melt-through will not occur if water is available to the core debris.

The mean value of the conditional probability of shell melt-through failure obtained in the NUREG-1150[12.5] study for Peach Bottom was about 0.5. As can be seen from Figure 12.9, the conditional probabilities for shell melt-through in the IPEs are below that reported for Peach Bottom in NUREG-1150, but many IPE values are comparable (about 0.2 to 0.5). The higher IPE shell melt-through values appear in those IPEs which used NUREG-1150 type of assumptions for this failure mode, while the lower values reflect modeling with less pessimistic assumptions, as found in NUREG/CR-5423. An example of an IPE with a high shell melt-through probability is the Fitzpatrick IPE. The analysis there used Peach Bottom data although the submittal states that the Fitzpatrick core is smaller than at Peach bottom while the sumps are larger than the ones at Peach Bottom.

Because of a high containment pressure capability[12.6] and the energy absorbing capacity of the suppression pool, a typical Mark I containment is unlikely to fail because of overpressure early in the accident sequence. However, accidents in which both containment heat removal and containment venting are not available or inadequate (such as occurs in some sequences in which the reactor vessel fails at high pressure, or in some anticipated transient without scram (ATWS) sequences) can cause early containment failure. For these sequences, containment may fail either before or at VB because of the high containment pressures. For other sequences, early containment failure is more likely caused by shell melt-through. Since the detailed composition of the early failure category is not provided in most of the IPEs, the contributions of the various failure modes to early failure are not always known.

As shown in Figure 12.7, the probabilities of early failure for most of the Mark I plants are greater than 0.15. The exceptions are Dresden (0.03) and Brunswick (0.1). The small probability value for Dresden can be partially attributed to the exclusion of shell melt-through as a failure mode in the analysis (treated as a sensitivity issue).

The IPEs show that even if shell melt-through has a small probability, early failure probabilities may still be significant. If shell melt-through and early containment venting are excluded, the early containment failure

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

[12.6]The design pressure for Mark I containments is either 56 psig or 62 psig and the ultimate pressure is usually greater than 100 psig. For BWR2 plants, the torus has a lower design pressure (35 psig) than the drywell (62 psig).

probability found in the Mark I IPEs varies from 0.01 to 0.3. For instance, the remaining early containment failure probability for Browns Ferry, with venting and shell melt-through excluded, is less than 0.01. However, this reduction in containment failure probabilities by removing shell melt-through may not be as significant as it appears because some other early failure mode would likely be shown to occur by the accident progression analysis.

High-pressureand 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, then failure will occur at the weakest location either in the wetwell or the drywell.

The RCS pressure at vessel melt-through, the containment failure location, and modeling assumptions regarding the rate of RCS depressurization and amount of core debris dispersed determine whether this failure mechanism is a significant contributor to early containment failure for individual Mark I containments. The most important accident characteristics, design features, and modeling assumptions are discussed below.

- RCS pressure at time of vessel melt-through — Containment failure via this mechanism is prevented if the RCS is depressurized before the core debris melts through the reactor vessel. The importance of this failure mechanism to risk 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 (ADS)). Enhancing the depressurization capability of the RCS is explored by a number of utilities but adverse effects are identified which need to be carefully considered.

- containment failure location — The containment failure location can significantly influence the importance of this failure mechanism. If failure occurs in the wetwell, then 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, then 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 the failure mechanism. If rapid depressurization is assumed (caused by a large opening in the reactor vessel) then high-pressurepulses can occur that have a high likelihood of containment failure. In addition, if a large amount of high temperature core debris is assumed to be released and dispersed into the containment atmosphere then it can directly heat it and containment failure is very likely to occur. Containment failure does not occur if lower depressurization rates combined with less core debris dispersal are assumed.

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

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

Containment challenges from ATWS sequences are important in a number of IPEs for plants with Mark I containments. These sequences belong to an accident class in which containment heat removal and containment venting are inadequate. In ATWS events the energy deposited to the containment can overwhelm the normal containment heat removal mechanisms as well as the available vent paths leading to early core damage and containment failure. The inability to remove heat from the containment causes containment failure to occur before core damage. The containment failure can lead to the loss of emergency core cooling systems (i.e., because of a loss of net positive suction head for pumps drawing from the suppression pool) with resulting core damage and vessel failure. Depending on the accident progression, core damage could occur first, but containment failure follows quickly. These accidents have been found risk significant in past PRAs since core damage, vessel failure and containment failure can occur within a short time interval, thus producing conditions for significant release to the environment. However, many IPE submittals indicate 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.

The distribution of containment venting to the early release and late release mode is not always spelled out in the IPE submittals and is somewhat complicated. In some IPE submittals a separate category is used for containment venting and it is not grouped to either early failure or late failures. For these cases, the probability and frequency of venting is divided between the early failure and late failure categories in Figures 12.7 and 12.8 depending on the timing indicated for the venting.

In general, containment venting for containment pressure control (the mode modeled in most IPEs) results in late release, while containment venting for combustible gas control is characterized as an early release, but the latter is usually negligible or zero in the submittals. Other types of venting are drywell or RPV venting used in conjunction with the containment flooding process.

When the containment flood contingency is successfully implemented and completed, either drywell venting or venting through the RPV to the condenser is needed. Since drywell venting does not have the benefit of suppression pool scrubbing, these release can be significantly higher than those from wetwell venting. In RPV venting, the release bypasses the containment and may be even more severe. Drywell or RPV venting is grouped in either the early failure or late failure category depending on the treatment in the IPE.

Early drywell venting is the predominant venting mode in one IPE. In the Duane Arnold IPE submittal, the results show that a large fraction of release involves early drywell venting (about 0.4 of all release or 0.3 of the total CDF). This large fraction of drywell venting for Duane Arnold results from the successful completion of drywell flooding, which is carried out by the operators following the procedures using external water during core melt progression.

Accidents that involve failure to isolate containment are not important for Mark I containments. Isolation failures can be preexisting or occur at the time of the initiating event. If the isolation failure is large and if core melt occurs, then radionuclide release can also be large. In addition, because the containment is open at the time of core damage the off-site consequences can be significant. These events are not important in BWR Mark I plants because of their relatively low frequencies. Pre-existing isolation failures in Mark I plants can be precluded because the containment atmosphere is inerted with nitrogen. Therefore, any loss of containment atmosphere because of pre-existing leaks can be easily 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. Values of isolation failures indicated in the IPEs for Mark I plants are typically less than 1% of the core damage frequency. Some IPE submittals, such as those of Duane Arnold and Oyster Creek do not report isolation failure values separately from early failures.

12.2.1.3 Containment Bypass Perspectives

Accidents that bypass containment are not important for Mark I containments. The bypass category includes ISLOCA and tube rupture of the isolation condenser (for some IPEs).

If the pressure boundary between the high-pressureRCS and a low-pressure auxiliary system fails (called an ISLOCA) then a LOCA outside containment can occur. If water cannot be supplied to the reactor, core damage will occur and a direct path can exists to the environment. Therefore, these accidents can lead to a significant early release of radionuclides. However, ISLOCAs are found to be not important for BWR Mark I containments because of their relatively low frequency compared with the frequency of accidents that dominate the CDF and that can lead to early structural failure.

The IPEs report bypass CDF frequencies that are about an order of magnitude lower in BWR plants than in PWRs. The higher PWR frequencies are in part because the contribution from steam generator tube rupture in the PWR systems. The conditional probability of containment bypass varies from zero to about 0.1 in the Mark I IPEs (Figure 12.7). The highest conditional probability of containment bypass was reported in the Oyster Creek IPE submittal. The CDF for Oyster Creek is approximately 4E-6/ry and sequences totaling about 3E-6/ry are considered in the accident progression analysis. Dominant frequencies for ISLOCA come from the overpressurization of the reactor water cleanup system (approximately 8E-8/ry) which results in a release to both the suppression pool and the reactor building equipment drain tank, and from overpressurization of the core spray system (approximately 3E-8/ry). Containment bypass can also occur, according to the Oyster Creek submittal, because of failure of the scram discharge volume during loss-of-feedwater or loss of off-site power transients (approximately 7E-8/ry).

With the exception of Nine Mile Point, Unit 1 (NMP1), containment bypass results only from the ISLOCA sequences obtained from the core damage frequency analysis. In the NMP1 IPE failure of the emergency condenser tubes as a result of high temperature creep rupture is identified as leading to containment bypass. This failure contributes 1% to total failure probability and accounts for 3% of the significant early release for NMP1. In a degraded core accident failure between the primary and secondary side of the emergency (or isolation) condenser provides a pathway for release similar to a steam generator tube rupture in PWRs. This failure mode is found to have a relatively low frequency (compared with the frequency of early structural failure) at NMP1, and is not important. Although Oyster Creek, as well as Dresden and Millstone 1, also use isolation or emergency condensers, failure of the condenser tubes was not addressed in these submittals. Presumably this bypass accident is also applicable to these plants and it should be determined whether this failure mechanism also has a low frequency compared with the frequency of early structural failure in these plants.

12.2.1.4 Late Failure Perspectives

Generally, late failure mechanism have been found less risk significant than the early failure mechanisms discussed above because of the longer time available for radioactive decay, natural deposition processes, and for accident response. However, even for late failures, if the failure location is in the drywell then, significant radionuclide release can still occur and this failure mechanism has been found important to longer term risk measures (i.e., latent health effects and land contamination) in several past PRAs. The late failure category presented in Figures 12.7 and 12.8 includes contributions from both late containment failure and late containment venting. As shown in Figure 12.7, the probability of late failure for the Mark I IPEs varies from about 0.2 to 0.9. The high late failure probabilities in some cases reflect the small, early failure probabilities reported in the IPEs. For example, Dresden has the lowest early failure probability (0.03) but the highest late failure probability (0.9). Figure 12.8 shows that the frequency of the late failure category varies from 4E-7/ry to 4E-5/ry.

The containment phenomena that can cause late containment failures, as described in NUREG-1335, include (1) overpressurization with high temperatures as a result of non-condensible gases and steam, (2) melt-through because of basemat penetration by core debris, and (3) vessel structural support failure because of core debris erosion. The failure of the reactor pedestal as a result of erosion may subsequently cause the failure of the drywell. Although all of the above failure modes are discussed and included in the IPE accident progression analyses, their contributions to the late failure category are not provided in most of the submittals. In addition to the above failure modes, other modes are considered in some cases. For example, the probability of containment failure because of the discharge of safety relief valves (SRVs) to the suppression pool at high temperature (greater than 260°F) is considered in some IPEs. The Duane Arnold IPE is an example of an analysis where containment failure as a result of high suppression pool temperature can occur. In the Quad Cities, IPE late failure because of failure of the SRV tailpipe in the wetwell, and loss of vapor suppression, is also considered.

High-pressureand temperature loads caused by core-concrete interactions are a significant contributor to late containment failure for Mark I containments. Gradual pressurization at high temperatures caused by non-condensible gases and steam released from the drywell floor during core-concrete interactions can fail Mark I containments several hours after vessel melt-through. 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.

The significance of this failure mechanism to late containment failure is determined by whether or not the drywell is flooded, the design configuration of the drywell, the availability of sprays or venting, and modeling assumptions regarding the quantity and temperature of core debris dispersed across the drywell floor. The most important accident characteristics, design features, and modeling assumptions are discussed below.

- drywell floor flooded - 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 limit gas generation so that steam is the main driving force for containment pressurization. The drywell spray and CHR systems are designed to condense steam and remove heat from containment, and therefore, can control the containment pressure under these circumstances.

- drywell floor not flooded - If the drywell floor is not flooded (and 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 significant gas release. The heat from this core-concrete interaction can raise the temperature of the drywell to a range where the structural capacity of the steel containment shell is significantly reduced. The quantity of gases released from this interaction also depends on the type of concrete used. For example, limestone concrete releases significantly more gases than basalt concrete. The drywell spray and CHR systems cannot control the pressure in containment if the driving force for pressurization is non-condensible gases. Under these circumstance, the only way to control pressure is to relieve gases via venting (preferably from the wetwell in order to benefit from pool scrubbing).

- containment design configuration - The design of the drywell and pedestal region can limit contact between the water and core debris in some Mark I containments. For example, large sumps in the pedestal region produce deep pools of molten core debris, which are difficult to cool with water. Forming a coolable debris bed is particularly difficult if the water is added after the core debris is in the sump. 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 of core debris released to the drywell and its temperature determine the extent of core-concrete interactions. If a large amount of core debris at high temperature is assumed released, from the reactor vessel then extensive concrete erosion is predicted in the IPEs. 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, then 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 build-up 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 limit non-condensible gas generation. If these early flooding strategies are not successful, then most utilities explored other ways of flooding the drywell floor. For instance, the Monticello IPE submittal notes that debris cooling with an alternate injection source, such as fire water, limits the temperature rise in containment and extends the time to containment failure by overpressurization. In all the IPEs, containment sprays are 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 core-concrete interactions. High temperature effects are also addressed in other ways in some IPEs. The Nine Mile Point 1 IPE submittal mentions raising the pre-load 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.

A breach of containment integrity, most often grouped with late failure in the IPEs, is containment venting. Venting differs from the other containment failure modes in that it is an intentional, and presumably controlled, opening of the containment boundary. Vent paths through the suppression pool will result in a "scrubbed" release (i.e., one where a sizeable fraction of the radionuclides originally in the vented fluids are removed before release to the environment).

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 risk, and it is also an important element of the CPI program. Containment venting is sometimes credited for preventing core damage in accidents involving loss of containment heat removal. It is also used to prevent late structural failure for those accidents in which the core melts through the reactor vessel. However, a few utilities state in their IPEs that their analyses indicate that the installation of a hardened vent does not significantly impact risk and is only of marginal benefit. In one case, a utility stated that the installation of a hardened vent led to less than 1% reduction in the CDF.

In response to the recommendations in Generic Letter (GL) 89-16,[12.7] 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.

Venting, after core damage has occurred, as a way of preventing structure failure of the containment is considered to be a last resort by most utilities because it can involve significant radionuclide release. The advantage of venting from the wetwell (benefit of pool scrubbing) is emphasized in most IPE strategies. The pressure at which venting should be started is also examined in detail by several utilities. The impact of high temperatures on the structural capability of the drywell is also noted. For example, the NMP1 IPE reports that at 400°F the containment could fail at pressures below the current venting pressure in the emergency operating procedures (EOPs). Further analysis is recommended that could 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, then 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 licensees express concern about the current BWR Owners Group guidelines for containment flooding (filling the containment solid with water to a level equal with the top of fuel in the RPV) and the venting necessary to carry it out. Since drywell (i.e., unscrubbed) venting is needed to relieve the pressure increase resulting from the compression of the gas space during containment flooding, there is the potential of an early release of significant magnitude associated with the flooding strategy. A number of utilities speculated that other actions, or even no action, is preferable to carrying out the containment flooding strategy.

Accidents with successful reactor scram, but loss of containment heat removal, are found to be relatively unimportant in all the 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 containment heat removal and containment failure is relatively long in these sequences, allowing time for emergency measures on and offsite.

The treatment of containment venting differs considerably among the Mark I IPEs. The information regarding containment venting is not specified uniformly in the IPE submittals. The probability values for containment venting are sometimes grouped into other categories (e.g., the early failure or late failure category) and are not provided separately in some of the IPEs. In some submittals a separate category is used for containment venting and the allocation to either the early or late time frame is not provided. Again, simple calculations and interpretation were used to derive some of the data needed for this report, and while some of these values may not be precise, they should provide a reasonable picture of the significance of containment venting in the Mark I IPEs.

As previously noted, drywell or RPV venting, which is part of the containment flooding process, is grouped in either the early failure or late failure category depending on the treatment in the IPE. Drywell venting plays a significant role in the Monticello IPE. The reason is the same as previously discussed for Duane Arnold, but in the Monticello IPE, drywell venting is grouped with late rather than early failure.

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

The distribution of venting as reported in 16 Mark I IPEs are presented in Figure 12.10[12.8] which shows a significant spread in the data. The conditional probability varies from zero to close to 0.6 for venting. To indicate the contribution of venting to the late containment failure category, Figure 12.10 also shows the conditional probability of late failure reported for these 16 plants.

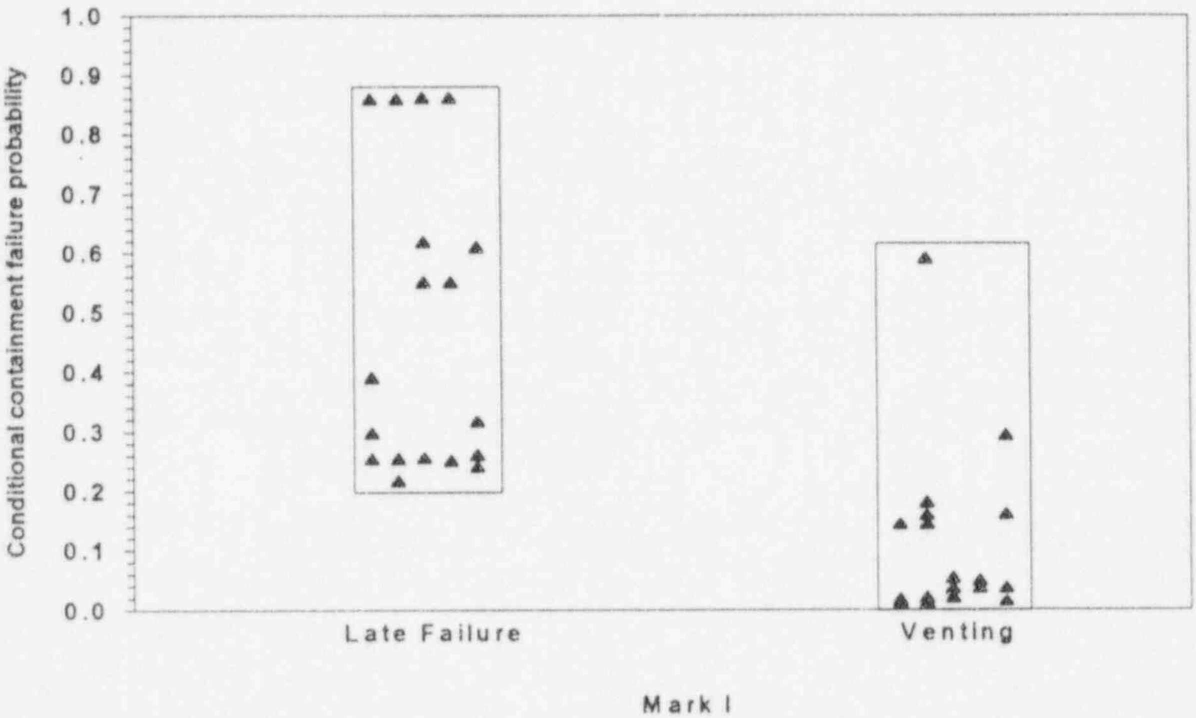

Figure 12.10 Comparison of some Mark I IPE reported probabilities for venting and total late failure.

The small containment venting probability found in some of the IPEs may be partly because the high probability of containment failure even with the containment being vented. Since most venting associated with pressure control would occur late in the accident progression, containment temperatures are likely to be quite high. Containment strength at these elevated temperatures can decrease, increasing the possibility of containment failure at pressures below the venting pressure limit, which is determined on the basis of strength at normal temperature. Another possibility is that venting is followed by shell melt-through. These cases with multiple containment failure modes usually are not grouped with the containment venting category in the submittals, but are grouped with the "uncontrolled" containment failure categories because of the more severe releases associated with the latter.

[12.8]Venting data could not be obtained from all the IPEs. The Vermont Yankee IPE is a typical example of one not included in Figure 12.10. Although shell melt-through and venting are considered in this IPE, results for the various containment failure modes are not provided in the submittal. Containment failure modes are provided only for the large early release category (0.2 of total CDF). In this category, 0.09 (or 0.02 of total CDF) is due to drywell venting, 0.2 (or 0.04 of total CDF) is due to shell melt-through, and 0.1 (or 0.03 of total CDF) is due to energetic events. The probability of containment venting cannot be separated from late failure, and probably has a small value.

The probability of containment venting is negligible in the Browns Ferry 2 IPE because no credit is taken for operator venting in the five key accident classes which comprise over 95% of the total CDF. Venting is assumed not to occur for three of the five classes because of its dependency on electric power and plant air vent valves. For the remaining two classes, it is assumed that, in the absence of a hard vent, no manual venting is possible. In response to NRC questions regarding the IPE submittal the licensee stated that a hardened wetwell vent has been installed in Unit 2, and operator procedures have been revised to reflect its utilization. It must be noted that data from the original Browns Ferry 2 IPE submittal Revision 0 are reported here. This IPE, which was prepared specifically for Unit 2, has been updated by the Tennessee Valley Authority (TVA), the licensee for Browns Ferry, with a PRA that considers shared plant systems among units. According to the submittal for this updated PRA, the study did not pursue an accident progression analysis. An examination of the updated submittal shows that containment venting was considered in the study and new plant damage states were created. There is no information in the updated study which would allow the resolution of the new plant damage states to the various containment failure modes. Obviously, the sharing of systems between units should be considered in the accident progression analysis to ascertain how shared supports affect containment heat removal, venting, and containment sprays.

Containment venting is a dominant contributor to the frequency of late release for a few plants. Containment venting is not normally a dominant contributor to late failure in most of the IPEs; however, there are some exceptions. For example, about 5% of the late failure probability (about 0.6) for Pilgrim is because of containment venting, and most (about 95%) of the containment venting probability involves drywell or RPV venting as a consequence of performing the containment flooding procedures. In some IPEs, the probability of wetwell venting for pressure control is small because containment failure occurs at a containment pressure below the vent pressure of the EOPs, (i.e., the primary containment pressure limit (PCPL)), because of the weakening of the containment pressure capability at high temperature.

12.2.2 BWR Mark II Perspectives

Eight BWR units (represented by five IPE submittals) are in the Mark II containment group and are listed in Table 12.8. 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.

Table 12.8 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	

12.2.2.1 Summary of Results and Perspectives for BWR Mark II Containments

The IPE reported results for this group are shown in Figure 12.11 (conditional containment failure probabilities) and Figure 12.12 (containment failure frequencies).

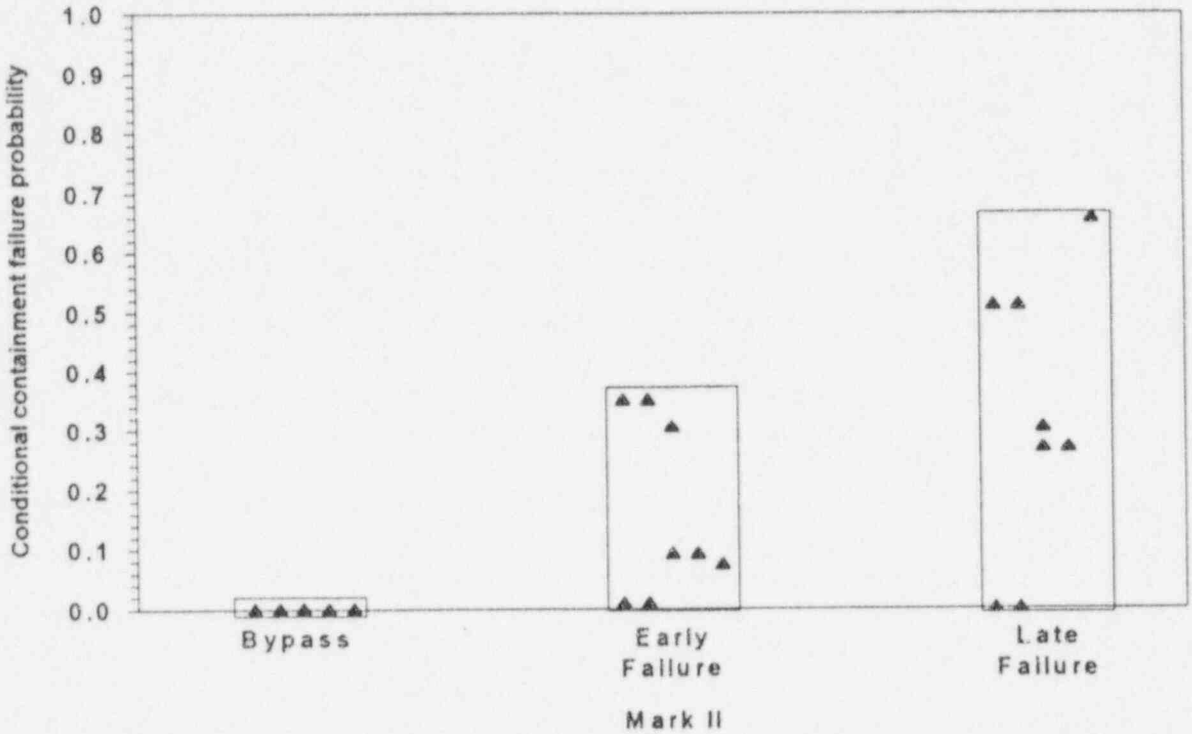

Figure 12.11 Reported IPE CCFPs for BWR Mark II containments.

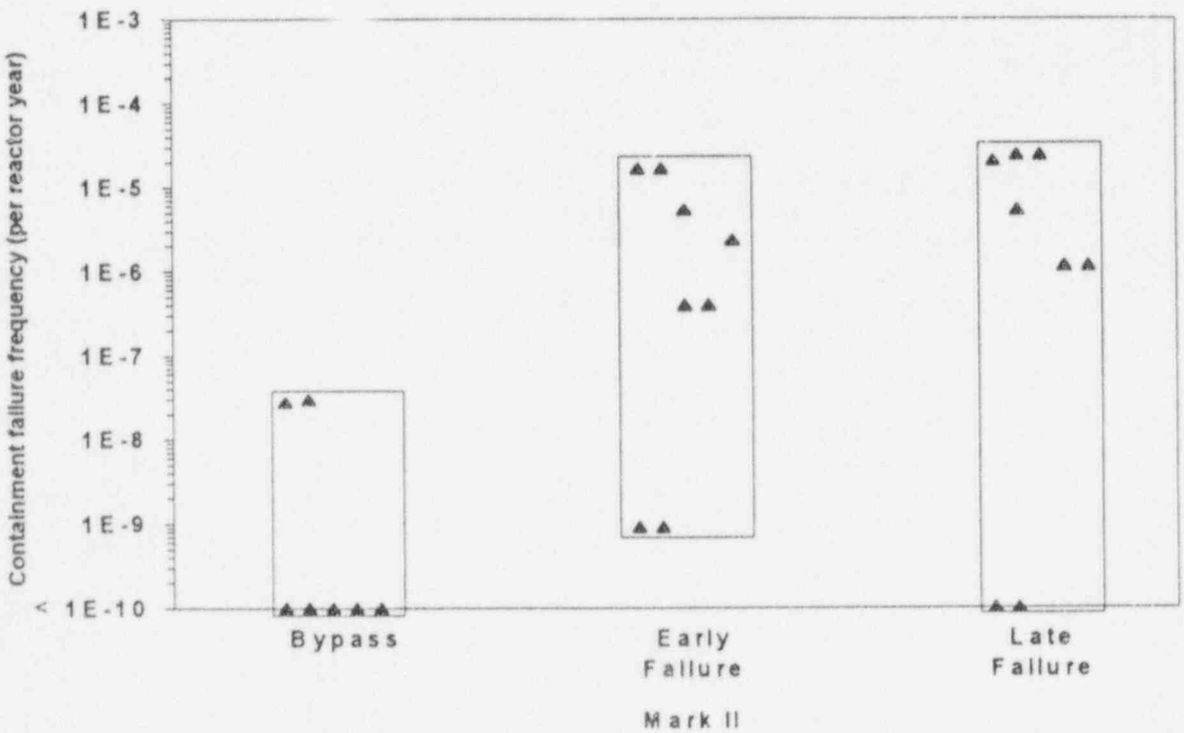

Figure 12.12 Reported IPE containment failure frequencies for BWR Mark II containments.

The average conditional probability is about 0.2 for early failure, 0.3 for late failure, and 0.5 for no containment failure. The probability of containment bypass was reported to be negligible for all Mark II IPE submittals. The distributions of the various containment failure modes for the five IPEs, as presented in Figures 12.11 and 12.12, show significant spread of the data. The conditional probability varies from about 0.01 to 0.4 for early failure, negligible to 0.7 for late failure, and 0.1 to close to unity for no containment failure.

The total CDF of these plants varies from 9E-8/ry for Susquehanna 1&2 to 5E-5/ry for LaSalle 1&2, a several order of magnitude difference. The frequency of early failure varies from 9E-10/ry (Susquehanna 1&2) to 2E-5/ry (LaSalle 1&2), the frequency of late failure varies from negligible (Susquehanna 1&2) to 2E-5/ry (LaSalle 1&2), and the frequency of containment bypass is reported as negligible for all the Mark II IPEs.

Key perspectives are summarized in Table 12.9

Table 12.9 Performance summary for BWR Mark II containments.

Failure mode importance	Important design features, operator actions, and model assumptions	Summary of results
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 and important role in many analyses Assumptions regarding the likelihood and magnitude of some severe accident phenomena vary considerably in the analyses	Early failure frequencies range from 9E-10/ry to 2E-5/ry. Average frequency is 5E-6/ry Early failure CCFPs range from 0.01 to 0.4. Average probability is about 0.2
Bypass		
Not important for this plant group	Low frequency is reported in CDF analysis	Negligible
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	Late failure frequencies range from negligible to 2E-5/ry. Average frequency is 1E-5/ry. Late failure CCFPs range from negligible to 0.7. Average probability is 0.3.

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. 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.

12.2.2.2 Early Failure Perspectives

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. The design and construction of the primary containment for all Mark II plants are in general similar and all of them are inerted with nitrogen. However, there are differences that may affect accident progression and the resultant containment failure modes as discussed below.

The early failure category presented in Figures 12.11 and 12.12 includes containment isolation failure, early containment venting, and early containment structural failure. The time of early containment failure is defined in the IPEs as either the time before or shortly after vessel breach or within 6 hours after accident initiation. Among the Mark II plants, LaSalle 1&2 and WNP2 use the former definition, while Limerick 1&2 and Nine Mile Point 2 use the latter definition. These two definitions are consistent except for the loss of containment heat removal sequences. In such a sequence, containment failure occurs before vessel breach, but may occur many hours after accident initiation.

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 Washington Nuclear 2 IPE, the licensee assumed that if the reactor vessel fails at elevated pressure and a high-pressure melt ejection (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 core debris 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, core debris 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 1&2 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 core debris attack or FCI after vessel breach. The failure of the drain tubes will result either in direct containment failure or in 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 to be released at vessel breach. On the one hand, 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 FCI when the vessel fails.

Small variation in these design features can be important for accident progression. In the LaSalle 1&2 containment, for instance, there are two 4-inch drain pipes that drain water from the upper cavity sumps to the reactor water clean-up system. The isolation valves on these lines are outside the containment. Thus, failure of these lines outside the containment can establish a pathway for radionuclides to escape the containment.

In the Limerick 1&2 analysis the melt-through of the drain plates on the diaphragm floor is not considered a challenge to containment failure if CHR via suppression pool cooling is available. According to the containment event tree, containment integrity is maintained if CHR is available and coolant injection to the reactor vessel or the drywell is successful. A sensitivity study was performed in the IPE for the amount of core debris entering the suppression pool via the failed in-pedestal drywell drain lines or ex-pedestal downcomers. The conclusion of the sensitivity study was that the assumptions used in the base case appear to be bounding. When the wetwell is vented the CHR is only credited as a radionuclide scrubbing feature in the analysis.

For Nine Mile Point 2, after vessel breach, the molten debris will be discharged into the suppression pool via the RPV blowdown through the in-pedestal downcomers (this is not explicitly modeled in the code model used, but this is considered to be "conservative" by the IPE analysts), or via overflow of the debris in the pedestal floor through the downcomers into the wetwell. The suppression pool may be bypassed if molten core debris causes the in-pedestal downcomers to fail at high temperature. According to the IPE submittal, under conditions with suppression pool bypass, the accident management actions that can be taken to minimize the impact of radionuclide releases include the use of suppression pool sprays to scrub radionuclides from the wetwell atmosphere or the use of drywell sprays to scrub radionuclides from containment. Currently, Nine Mile Point 2 EOPs indicate that sprays will not be used unless adequate core cooling is assured. Accident management may be focused in a different direction if RPV breach has occurred and venting is imminent. Another Nine Mile Point 2 specific design feature is the limited communication between the in-pedestal water and the ex-pedestal water in the wetwell. Therefore, suppression pool cooling may have limited effectiveness in maintaining a subcooled pool and preventing a long-term containment pressurization challenge for this containment. The IPE analysis indicated the best return paths to use during suppression pool cooling was to take suction from the suppression pool and return the flow to the RPV. An alternative may be to conduct the return through the drywell sprays. The operation of the low-pressure core injection with in-line heat exchanger is in general already directed by the EOPs.

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. In the WNP2 containment the pedestal region is recessed relative to the drywell floor. There are two 8 ft by 6 ft sumps cast into the pedestal floor which each have 3/8" thick stainless steel covers and normally contain water to a depth of about 17 inches. The sumps have drain lines which are routed beneath the surface of the suppression pool before exiting the containment. The drain lines are closed as part of containment isolation. There are no downcomers in the pedestal region. 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 3% involve vessel breach, 1% involve containment failure, and much less than 1% involve 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.

Several assumptions used in the Susquehanna 1&2 IPE contribute 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 (1) the core debris is quenched, and CCI is prevented if a water pool is present and continuously resupplied when the debris is released from the RPV, (2) that downcomer and drywell shell melt-through are prevented if there is an overlying water pool for the core debris, and, (3) 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. In the LaSalle 1&2 analysis the containment failure probability because of this failure mechanism is about 0.1. The major contributors to this failure mode are transient sequences because of the dominant contribution of transient sequences to total CDF (about 90%). After containment failure, the harsh environmental conditions in the reactor building cause the loss of all injection systems and, consequently, core damage. In the Limerick 1&2 analysis the major contributor to this containment failure mode is ATWS (8% of CDF, or about 90% of early containment failure). For WNP2, sequences during which injection is successful but all viable means of containment heat removal are unavailable contribute about 20% of the total release frequency (or about 15% of CDF). Containment pressure continues to increase in these sequences until it reaches the failure pressure. The energetic release of steam from the containment into the reactor building basement leads to consequential failure of all injection system and core melt. For Nine Mile Point 2, ATWS events are the leading contributor to early containment failure (3% of CDF).

At the time of vessel breach, containment failure may be caused by (1) a missile from an in-vessel steam explosion, (2) fast containment pressurization because of DCH, ex-vessel steam explosions, or reactor blowdown, (3) failure of the reactor pedestal as a result of either a quasi-static pressurization accompanying vessel breach or dynamic loads associated with ex-vessel steam explosions (failure of reactor support and subsequent gross motion of the RPV will place a large amount of stress on the pipe penetrations, resulting in failure), and for some IPE, (4) an ex-vessel steam explosion that fails the cavity drain pipe beyond the containment wall. While the failure mechanisms are not always explicitly discussed in the IPE submittals, it seems that for most of the Mark II IPEs the contribution from in-vessel steam explosions and HPME (e.g., DCH, blowdown loads, debris impingement) is small.

For LaSalle 1&2, the probability of containment failure at vessel breach is about 0.2. It seems that the majority of this is because of failure of the cavity drain pipe (by ex-vessel fuel coolant interaction), which is classified as a leak failure in the IPE. The contribution from energetic events in the LaSalle 1&2 IPE is relatively small. For Limerick 1&2 the conditional probability of containment failure caused by energetic events associated with vessel breach is about an order-of-magnitude lower than the LaSalle 1&2 probability. For WNP2, the reported dominant containment failure modes include one in which a rapidly flowing jet of hot debris impacts the shell, and one resulting from a steam explosion. The steam explosion occurs when core debris is forcefully ejected at high-pressure into the water pool in the pedestal cavity, which is flooded because of operation of the drywell spray. It is postulated in this IPE that such an explosion results in severe overpressure in the cavity, collapse of the pedestal and gross structural failure of the containment. For Nine Mile Point 2, the contribution from early failure from energetic events is less than 1% of total CDF.

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 0.06).

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 1 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 (1) if power is restored, low-pressure injection (LPI) systems become available as sources of core cooling, and (2) 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.

Containment isolation failure and early containment venting are not significant contributors to early failure for Mark II containments. The probability of isolation failure is small for all Mark II IPEs apparently because these containments are inerted and any leak is easily detected and corrected. The reported probability of isolation failure is about 0.01 for Nine Mile Point 2 and for WNP2. Nine Mile Point 2 has a special containment isolation system feature that involves the use of AC powered motor operated valves (MOVs) on the drywell equipment and floor drains. The use of AC powered MOVs as isolation valves means that under SBO conditions these valves would not be able to automatically close. It may be useful during implementation of SBO specific procedures in operator training or in future accident management guidance to emphasize the need to go locally to these MOVs to close the valves and provide containment drywell isolation. Except in the LaSalle 1&2 IPE, early containment venting is also negligible. For LaSalle 1&2, there is a small early containment venting probability, which comes primarily from ATWS sequences.

12.2.2.3 Containment Bypass Perspectives

If the pressure boundary between the high-pressureRCS 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 significant 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.

12.2.2.4 Late Failure Perspectives

The late failure category presented in Figures 12.11 and 12.12 include the contributions from both late containment failure and late containment venting. If the Susquehanna 1&2 analysis is excluded, containment venting that is grouped in the late failure category in these IPEs varies from a conditional probability of 0.04 for Nine Mile Point 2 to more than 0.4 for LaSalle 1&2. Late venting is used frequently in the LaSalle 1&2 IPE for transient events.

High-pressure and temperature loads caused by core-concrete interactions are significant contributors to late containment failure for Mark II containments. The containment phenomena that may cause late containment failure as described in NUREG-1335 include (1) overpressurization with high temperatures because of non-condensible gases and steam, or as a result of combustion processes, (2) containment basemat melt-through because of basemat penetration by core debris, and (3) vessel structural support failure as a result of core debris erosion, which, for Mark II containments, includes the failure of the reactor pedestal caused by CCI. Gradual pressurization at high temperatures caused by non-condensible 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 (exceeding 260°F). This assumption is on the basis 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.

Partly because of the unique approach used in the Susquehanna 1&2 IPE, the probabilities of containment failure for Susquehanna 1&2 are low for late failure as well as for the previously discussed early failure. The total containment failure probability for Susquehanna 1&2 is only 0.01, all of which comes from ATWS events (classified here as early failure). The probability of late failure probability for Susquehanna 1&2 is zero. It is assumed in the Susquehanna 1&2 IPE that the debris is cooled if water is supplied to the ex-vessel core debris. Water supplied to the core debris is also assumed to prevent the downcomers from failing, which would result in a suppression pool bypass.

Excluding Susquehanna 1&2, Figure 12.11 shows that the probability of the late failure category varies from about 0.3 for Limerick 1&2 to over 0.6 for Nine Mile Point 2. Excluding containment venting, the probability of late containment failure varies from about 0.1 for LaSalle 1&2 to over 0.6 for the Nine Mile Point 2.

For LaSalle 1&2, the reactor cavity is large enough to contain all the core debris that is released at vessel breach. Besides containment venting, the probability of late containment failure is primarily from the transient sequences. For Limerick 1&2, late failure is from drywell or RPV venting (classified in the IPE as intermediate failure) or overpressure or late venting because of failure of CHR (late failure). Venting contributes about 10% (it seems most of it is drywell or RPV venting) and structural failure contributes 15% (primarily because of a loss of CHR). For Nine Mile Point 2, the probabilities of intermediate and late containment failure are about 0.4 and 0.3, respectively. The total is about 0.7, of which 0.05 is from venting. For WNP2, late containment failure seems to be primarily from containment overpressure failure. The leading sequences for late containment failure involve the loss of CHR capability.

Figure 12.12 shows that, excluding the Susquehanna 1&2 data, the frequency of late failure varies from about 1E-6/ry for Limerick 1&2 to 2E-5/ry for LaSalle 1&2.

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 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 de-inerted.

Figure 12.13 shows the probability of venting compared to that of total late containment failure as reported in the Mark II IPE submittals. 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, (with about 90% 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 venting probability occurs for sequences without vessel breach.

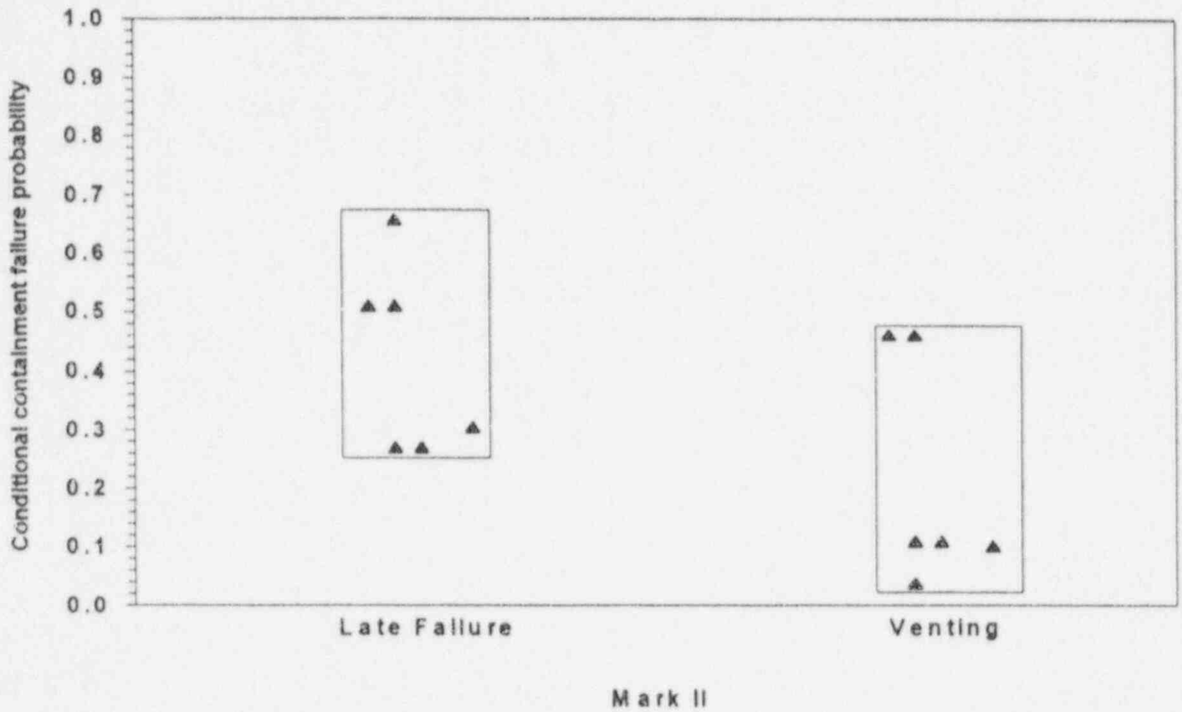

Figure 12.13 Comparison of some Mark II IPE reported probabilities for venting and total late failure.

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 is associated with drywell venting and is 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 39 psig. However, if the differential pressure across the valve disk in the preferred vent path exceeds 49 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 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 licensee 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. Alternative actions were reported in the IPE to produce substantially lower releases and much longer times to failure. The IPE submittal reported that 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.

12.2.3 BWR Mark III Perspectives

Four single-unit BWRs described in four separate IPE submittals make up the Mark III containment group, as indicated in Table 12.10. All four plants use a BWR 6 design.

Table 12.10 Plants (per IPE submittals) in Mark III containment group.

Clinton	Grand Gulf 1	Perry 1	River Bend

12.2.3.1 Summary of Results and Perspectives for BWR Mark III Containments

The containment performance results reported in the IPE submittals for this group are shown in Figure 12.14 (conditional containment failure probabilities) and Figure 12.15 (containment failure frequencies). The containment failure modes are grouped into early failure, containment bypass, and late failure. The containment failure time for the early failure category is before, at, and shortly after vessel breach (VB), and the early failure category includes isolation failure, early containment venting, and early containment structural failure. The bypass category includes ISLOCA. The late containment failure category includes late containment venting, late containment overpressure or high temperature failure, and basemat melt-through.

The average conditional probability for this containment group is approximately 0.3 for early failure, 0.2 for late failure, and 0.5 for no failure. The probability of containment bypass for the group is negligible.

The distributions of the various containment failure modes reported in the four IPE submittals and presented in the figures show significant spread in of the reported results among the IPEs. For example, the conditional probability varies from about 0.03 to 0.5 for early failure, and from about 0.02 to about 0.4 for late failure. The significant spread in the results is partly because of the particularly small containment failure probability predicted in the Clinton IPE. If the Clinton results are excluded, the conditional probabilities for the other Mark III plants vary from about 0.2 to 0.5 for early failure, and from about 0.1 to 0.4 for late failure. Figure 12.15 presents the frequencies of the various containment failure modes for this group and it shows that the frequency of containment bypass is very small for all BWR plants with Mark III containments. Key perspectives are summarized in Table 12.11.

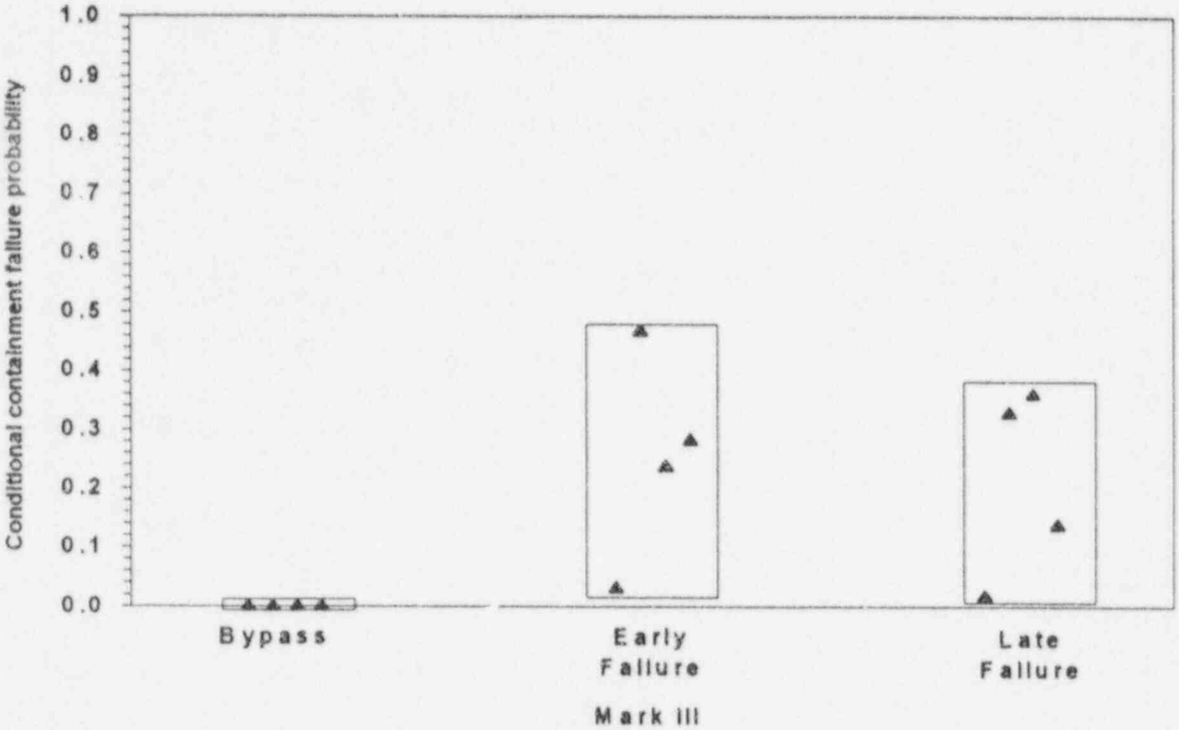

Figure 12.14 Reported IPE CCFPs for BWR Mark III containments.

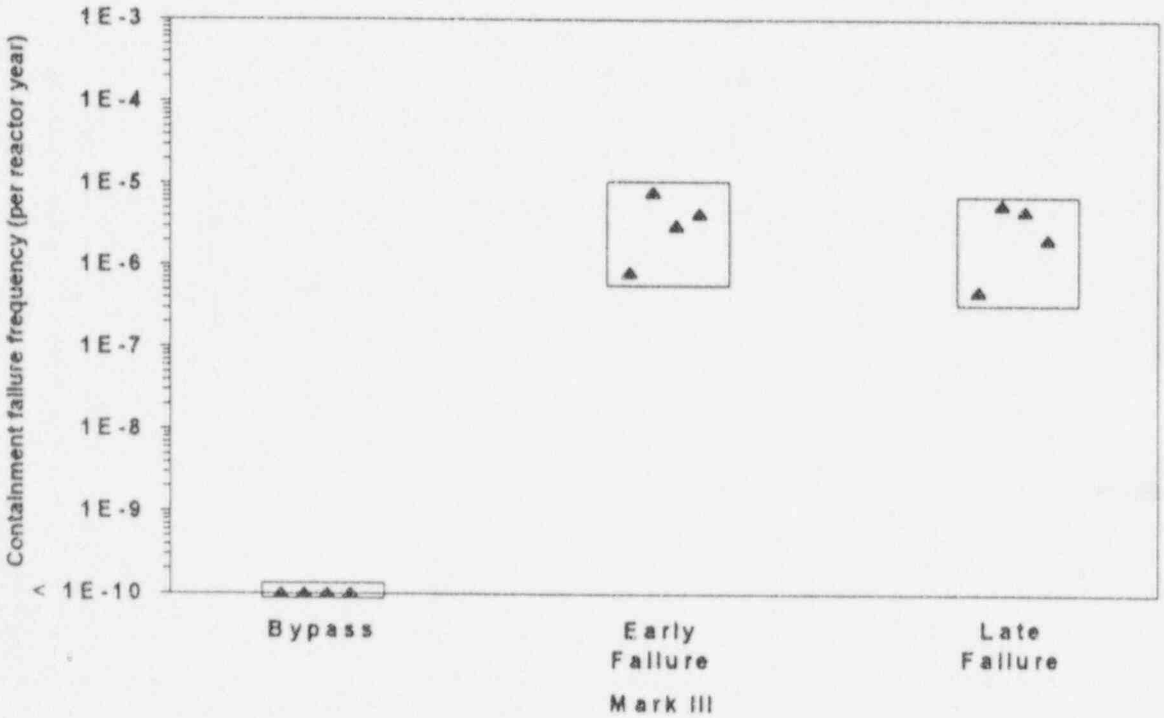

Figure 12.15 Reported IPE containment failure frequencies for BWR Mark III containments.

Table 12.11 Performance summary for BWR Mark III containments.

Failure mode importance	Important design features, operator actions, and model assumptions	Summary of results
	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	Early failure frequencies range from 8E-7/ry to 8E-6/ry. Average frequency is 4E-6/ry. Early failure CCFPs range from 0.03 to 0.5. Average probability is 0.3
	Bypass	
Not important for this plant group	Low frequency is reported in the CDF analysis	Negligible
	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	Late failure frequencies range from 5E-7/ry to 6E-6/ry. Average frequency is 3E-6/ry. Late failure CCFPs range from 0.02 to 0.4. Average probability is 0.2

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 releases 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, i.e., comparable to early structural failure.

12.2.3.2 Early Failure Perspectives

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.

A number of plant features play an important role in the accident progression in Mark III containments. Although the main internal structure are, for the most part, common to all plants, there are differences that may influence accident progression.

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 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.

Vacuum breakers are provided between the drywell and the containment to limit the buildup of negative pressure in the drywell. They are provided for Mark Is and Mark IIs between the drywell and wetwell primarily to protect drywell integrity. For Mark IIIs, a negative drywell pressure will cause the suppression pool water level to drop, clearing of the top rows of the vents (a reverse vent clearing), and gas flow from the containment to the drywell. In this process, some of the suppression pool water will be transported to the drywell and remain there. This reduces the water level in the suppression pool and causes flooding of the drywell cavity. The vacuum breakers for Mark IIIs are used to equalize the pressure between the drywell and the containment to prevent these scenarios. The failure of the vacuum breakers (i.e., failing open) may result in suppression pool bypass and containment failure. This failure mode is considered in the Level 2 analysis in some Mark III IPEs. Vacuum breakers are provided in all Mark IIIs except River Bend. River Bend relies on reverse vent clearing to eliminate drywell negative pressure.

Another unique feature of the River Bend containment is the reactor pedestal configuration. The access door to the pedestal region at River Bend is water tight and the door is kept closed while the plant is operating. The pedestal doors at the other Mark III plants are not water tight or are kept open. Therefore, the probability of water accumulation in the pedestal before vessel failure is much lower at River Bend than at other Mark III plants.

The suppression pool make-up system in Mark III containments provides a means of gravity feeding the suppression pool from the upper containment pool to compensate for any water loss associated with a LOCA and to ensures that there is an adequate water volume in the suppression pool to keep the suppression pool vents covered for all break sizes. The volume in the upper pool is sufficient to account for all conceivable post-accident entrapment volumes and still maintain long-term coverage of the vents. The failure of this system may cause a suppression pool bypass and/or inadequate net positive suction head (NPSH) for the pumps taking suction from the suppression pool.

Again River Bend differs from other Mark IIIs in that its design does not incorporate suppression pool make-up via an upper pool dump. According to the IPE submittal, the River Bend design contains sufficient water in the suppression pool so that an upper pool dump is not required. As discussed above, the River Bend cavity design reduces the volume of water available in the pedestal cavity. This helps to maintain water in the suppression pool and reduces the potential of a flooded cavity and eliminates failure modes associated with ex-vessel steam explosions.

All Mark III containments, with the exception of River Bend, have containment sprays. River Bend does not have containment or drywell sprays. Instead, post accident heat removal from the containment is achieved by operating one of two safety related containment unit coolers, and, with guidance provided in the EOPs, drywell heat removal can be accomplished by the use of non-safety related drywell unit coolers. Relative to the operation of containment sprays, this method of heat removal reduces the radionuclide scrubbing capability but also reduces the likelihood of rapid condensation of containment steam, and the potential of de-inerting the containment.

Hydrogen production in these plants during a severe accident can be significant. In a BWR core there is a large inventory of zirconium and the Mark III plants have the largest (BWR6) cores. 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 can be produced from the oxidation of this metal during the core damage process. If the hydrogen ignition system is not working, hydrogen will accumulate in the containment and the potential of containment failure from hydrogen combustion significantly increases. Containment challenge from hydrogen combustion is particularly severe for accidents in which the suppression pool is subcooled and the steam released from the RPV is condensed in the pool. The lack of steam in the containment atmosphere in combination with the large amount of hydrogen released during core degradation process results in a high hydrogen concentration in the containment. Subsequent ignition of the hydrogen by either random sources or by 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.

Assumptions made in the Mark III IPE analyses play a significant role in the accident progression reported. Examples are assumptions regarding such diverse items as the reliability of the power source of the igniters, the likelihood of in-vessel recovery of cooling for a damaged core, and the mission time used in the analysis.

Supplement No. 3 of GL 88-20[12.9] identified that Mark III containments are expected to evaluate the vulnerability to interruption of power to the hydrogen igniters. In the Clinton IPE analysis, containment failure as a result of a hydrogen burn is assumed to be almost eliminated if an alternate power supply is provided for the hydrogen igniter system. However, according to the IPE submittal, the reduction in containment failure probability on the basis of a 90% availability of the alternate power source does not justify the installation of an alternate power supply.

According to the River Bend analysis, the modification of the electrical supply to the hydrogen igniters to ensure availability during SBO would remove the possibility of high containment loads from hydrogen deflagrations and detonations. The overall impact would be to reduce the probability of penetration failures and containment structural

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[12.9] 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.

failure by about 10%. According to the IPE, the risk of hydrogen burns can also be reduced by modifying the SBO procedure to instruct the operators to turn off the igniters if AC power is unavailable. An alternate power supply for the igniters is not believed to be a preferable option according to the IPE because the least cost options should be explored prior to attempting more expensive options.

For Grand Gulf, a sensitivity analysis simulating the presence of a backup power supply is conducted in the IPE. The effect is not significant because early failure is dominated by MSIV venting. Besides MSIV venting, which results in an early medium/high release, the probability of an early high release is reduced by the installation of the backup power supply by about 60% and the probability of early low release is reduced by about 30%. However, a cost-benefit analysis performed in the IPE shows that the addition of the power supplies would not be justified at this time.

For Perry, the modification of the electrical supply to the hydrogen igniters to ensure availability during SBO is assumed to remove the possibility of containment failure from hydrogen burns. The overall impact is to reduce the RPV failure and early containment failure with pool bypass frequency by 2% and containment structural failure by 7%.

In-vessel 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 in-vessel recovery is not reported, in-vessel 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 in-vessel 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 0.15 for early failure, 0.1 for late failure, and 0.35 for no containment failure. Even with containment failure, the release associated with in-vessel 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.

Hydrogen combustion events or energetic events at vessel failure are the leading contributors to early containment failure for Mark III containments. With the exception of the Grand Gulf IPE results (discussed further below) where MSIV venting dominates the early releases, the early failure category shown in Figures 12.14 and 12.15 is largely the result of early containment structural failures. All of the IPEs for plants with Mark III containments considered early containment failure modes caused by overpressurization from non-condensible gases, hydrogen combustion processes, DCH, and missile and pressure loads resulting from steam explosions. While the causes of early containment failures are not discussed in detail in most of the 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 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 0.2 to 0.5.

ATWS loads are identified in the Clinton IPE as the only mechanism capable of causing an early containment failure. The Clinton IPE submittal discussed and dismissed containment failure mechanisms associated with vessel blowdown forces, in-vessel 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. According to the Clinton IPE submittal, containment venting and suppression pool cooling, although called for by the EOPs, have not been demonstrated as being effective in preventing containment failure in some ATWS scenarios. The dismissal of other failure mechanisms may be attributable in part to design differences between Clinton and other Mark III containments.

As noted above, both early and late containment failure probability in the Clinton IPE analysis are 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.

The Clinton IPE results indicate that 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. 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.

A venting scheme considered in the Grand Gulf IPE analysis 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 submittal, 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.

A sensitivity study in the Grand Gulf IPE shows that besides the MSIV venting, the early release of radionuclides is also governed by hydrogen deflagration. As noted above, the sensitivity study shows that although the use of a backup DC power supply or an independent backup AC power supply for hydrogen igniters does not have a significant effect on the probability of total early failure (which includes MSIV venting) it does significantly reduce the probability of early containment structural failure.

The Perry IPE analysis used the same event tree analysis code applied in the Grand Gulf NUREG-1150 study. The result of the Perry containment performance differ from those in the Grand Gulf NUREG-1150 study because of differences in containment failure modes, phenomenological assumptions, and plant damage state group frequencies. While the NUREG-1150 study for Grand Gulf was dominated by SBO sequences and hydrogen combustion was the dominant cause for containment failure, the results in the Perry IPE are dominated by shutdown ATWS sequences (ATWS with successful standby liquid control) and other transients. The contribution of hydrogen combustion to total containment failure is not provided in the Perry IPE submittal. However, since steam pressurization is likely to be the dominant cause of containment failure for ATWS events, the importance of hydrogen combustion is correspondingly diminished. A sensitivity study in the Perry IPE submittal addressing the hydrogen igniter system backup power supply indicates that containment failure is not significantly reduced by the installation of such a system.

The probability of early containment failure reported in the submittal for River Bend is 0.25. The contribution of hydrogen burns to early containment failure is not provided. A sensitivity study in the River Bend IPE shows that containment failure probability is moderately reduced (by about 10%) if an alternate power supply is used for the hydrogen igniter system. It should be noted that the conditional probability of SBO sequences found in the River Bend IPE analysis (approximately 0.9) is comparable to that obtained in the Grand Gulf NUREG-1150 study.

Except for the results reported for Clinton, the probability of containment isolation failure is much lower than that from other early containment failure modes. For Clinton, the conditional probability of early failure is about 0.03, of which 90% is from isolation failure and the remainder from other early containment failure modes. The 0.03 probability of containment isolation failure is comparable to that reported in the River Bend IPE but higher than those reported in the other Mark III IPE submittals. For Clinton, the SBO sequences can impair the containment isolation function because there are containment isolation valves that would fail open under loss of power conditions, and they would have to be manually isolated (i.e., local manual actions by area operators) to ensure that a radioactive release from the containment would not occur. For River Bend, the relatively high isolation failure probability is because of the containment isolation failure probability for SBO sequences (about 0.03) and the high contribution to the plant CDF from SBO sequences (about 90% contribution). While this probability of isolation failure is small compared to early containment failure for River Bend, it is of the same order as the small probability of early structural failure calculated in the Clinton IPE.

For Grand Gulf, the probability of isolation failure is briefly discussed and dismissed because it is believed that it is below the screening criteria (using a conditional probability of 1E-3 for screening). In the Perry analysis, the mostly likely mechanism identified for loss of isolation is if the normally open fuel pool cooling and cleanup vent path fails to isolate following an SBO. This path is utilized in the Plant Emergency Instruction for containment venting. The conditional probability for isolation failure for Perry is estimated in the IPE to be negligible.

12.2.3.3 Containment Bypass Perspectives

The probability of containment bypass was found to be negligible in all the Mark III IPEs. However, the MSIV venting considered in the Grand Gulf IPE, and discussed above, leads to a release that bypasses containment and is the most important failure mode in that analysis with respect to radionuclide release.

12.2.3.4 Late Failure Perspectives

The late failure category shown in Figures 12.14 and 12.15 includes the contributions from both late containment failure and late containment venting. The containment phenomena that may cause late containment failures as described in NUREG-1335 include (1) overpressurization with high temperatures because of non-condensible gases and steam, or as a result of combustion processes, (2) containment basemat melt-through because of basemat penetration by core debris, and (3) vessel structural support failure as a result of core debris erosion. If containment venting (discussed further below) is excluded, the probability of late containment failure reported is about 0.1 for Perry, 0.15 for River Bend, and 0.25 for Grand Gulf. As with early containment failure, the probability of late containment failure reported for Clinton (less than 0.01) is much smaller than that found in the other Mark III IPEs. The Clinton IPE results indicate that 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).

Principal contributors to late failures in Mark III containments are late combustible gas burns and phenomena associated with core-concrete interaction. Most of the IPE submittals did not provide a detailed discussion on the causes of late containment failure; therefore, contributions to late containment failure as a result of specific containment phenomena are not known. Late failures can be inferred to result primarily from late combustible gas burns, 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.

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 MSIV venting scheme used in the Grand Gulf IPE analysis has been discussed under the early failure perspectives. It is not reported in the other IPE submittals. Most of the venting described in the IPEs for Mark III plants is late venting scrubbed by the suppression pool; therefore, the releases associated with this venting are small. Figure 12.16 compares the conditional probability of venting reported in the IPE submittals with the reported conditional probability of total late failure. The highest value in the venting column is that for Grand Gulf which includes the early venting discussed above.

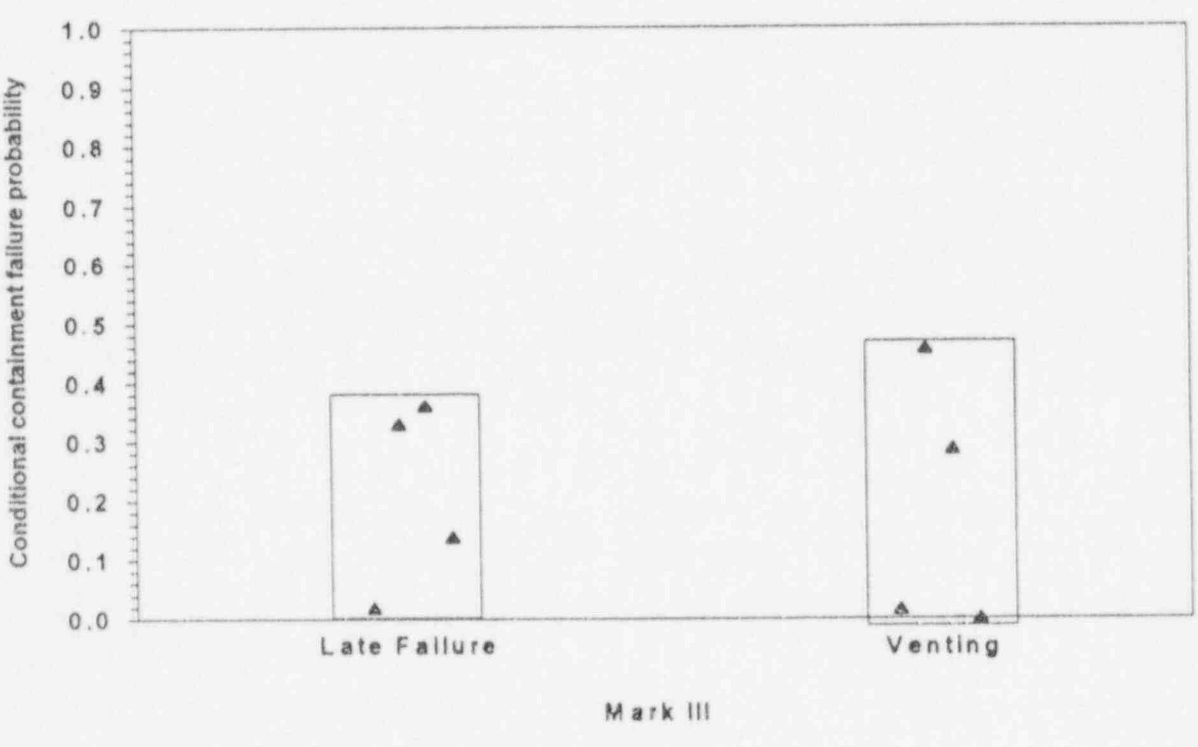

Figure 12.16 Comparison of some Mark III IPE reported probabilities for venting and total late failure.

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. In the River Bend analysis the effect of increased containment venting capability via installation of a hardened 10-inch diameter vent is investigated. Although the installation of such a vent would reduce containment failure, it would increase the probability of radiological release through the vent. According to the licensee, this issue cannot be resolved without performing a Level 3 PRA, and venting is not believed to be a preferred option by the IPE analysts

Although the probability of late venting was less than 0.02 in the Clinton IPE submittal, the total venting probability indicated in the Clinton CETs was more than 0.1. However, most of this venting 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.

12.3 PWR Containment Performance Perspectives

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 (1) large dry containments (including those operating with a subatmospheric internal pressure) and (2) ice condenser containments. In addition to the PWRs, one early BWR (Big Rock Point) is housed in a large dry containment. The PWR plants in each group are indicated in Table 12.12.

Containment performance results for all PWRs in the two groups are shown in Figures 12.17 (conditional containment failure probabilities) and Figure 12.18 (containment failure frequencies). The results indicate that most of the containments in both PWR groups have relatively low conditional probabilities of early failure.

Significant variability exists in the contributions of the different failure modes for both containment groups. This variability results in part from such different containment design parameters as containment design pressure, ultimate pressure, containment volume, containment construction (i.e., steel versus concrete), and reactor thermal power.

Table 12.13 shows the values of these parameters for all domestic PWR containments. Figures 12.19 shows the range of design pressure and ultimate pressure within the PWR containments and compares these ranges between containment types. Figure 12.20 indicates the range of the ratio of containment volume to thermal power in the PWR plant groups. However, the variability in the containment failure results of Figures 12.17 and 12.18 is also attributable to other plant specific design differences as well as modeling assumptions made as part of the analysis. 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 in-vessel 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.

Important factors that impact the probabilities of the failure modes in Figures 12.17 and 12.18 are discussed for the two groups in Sections 12.3.1 and 12.3.2. 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.

Table 12.12 Summary of PWR containment classes and associated nuclear power plants.

Class	IPE submittals			
Large Dry and Sub-atmospheric	• 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	• Seabrook	• San Onofre 2&3
	• 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	• Salem 1&2
	The large dry and subatmospheric containment group consists of 65 units of which 7 have containments which are kept at subatmospheric pressures. These containments rely on structural strength and large internal volume to maintain integrity during an accident. Most of these containments utilize a reinforced concrete or post-tensioned concrete design with a steel liner. A few units are of steel construction.			
Ice Condensers	• Catawba 1&2	• DC Cook 1&2	• McGuire 1&2	• Sequoyah 1&2
	• Watts Bar 1			
	The ice condenser containment is a pressure suppression containment which 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.			
*Although Big Rock Point has a BWR, it is housed in a large dry containment, and therefore, for containment classification purposes, it is considered as a PWR containment.				

In contrast to the BWR IPE results, containment bypass, especially associated with steam generator tube rupture (SGTR), is important for most of the PWR containments and is the major contributor to large, early releases in many of the PWR IPEs. Containment bypass for the PWR containments can result from ISLOCA events or SGTR events.

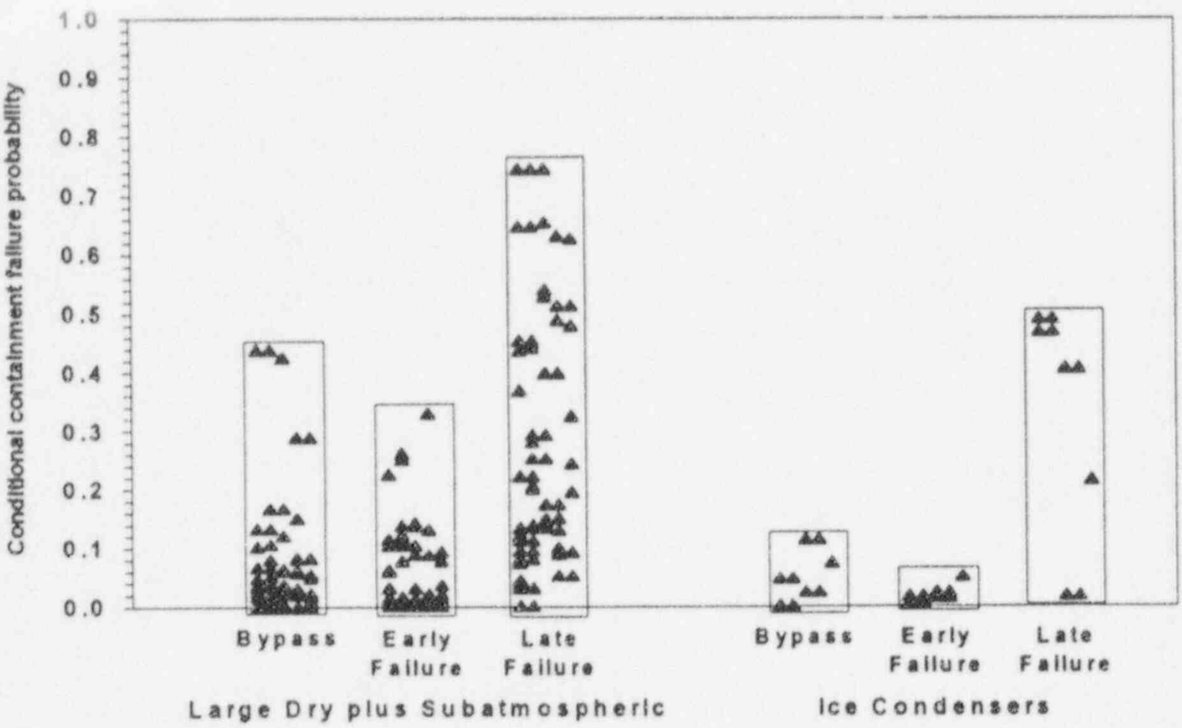

Figure 12.17 Reported IPE CCFPs (given core melt) for PWR plants.

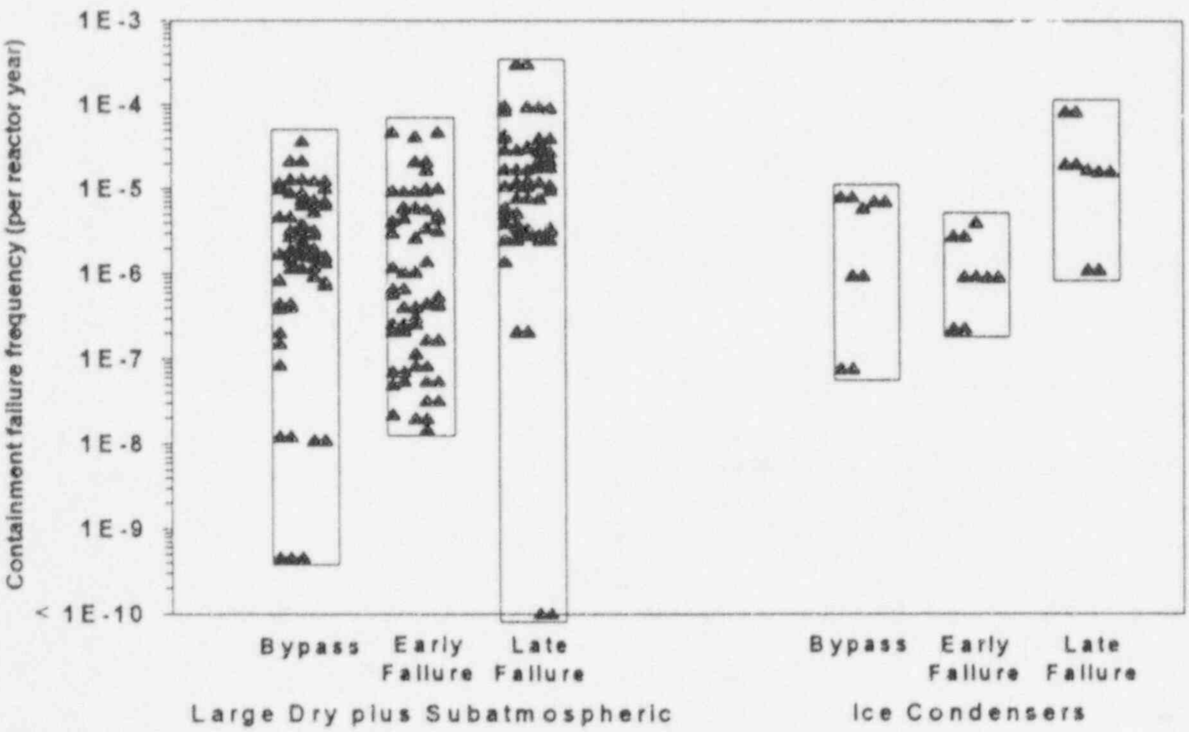

Figure 12.18 Reported IPE containment failure frequencies for PWR plants.

Table 12.13 Containment properties for PWRs.

Plant	Containment design pressure (psig)	Ultimate pressure (psig)	Containment volume (1E6 ft^3)	Construction*	Thermal power (MWt)	Ratio volume/ thermal power (ft^3/MWt)
Large dry and subatmospheric containments						
ARKANSAS NUCLEAR ONE 1	59	154	1.81	P	2568	705
ARKANSAS NUCLEAR ONE 2	54	141	1.78	P	2815	632
BEAVER VALLEY 1	54	126	1.76	R	2652	664
BEAVER VALLEY 2	54	126	1.72	R	2652	649
BRAIDWOOD 1&2	61	108	2.80	P	3411	821
BYRON 1&2	61	108	2.80	P	3411	821
CALLAWAY	60	135	2.50	P	3565	701
CALVERT CLIFFS 1&2	50	121	2.00	P	2700	741
COMANCHE PEAK 1&2	50	114	2.99	R	3425	873
CRYSTAL RIVER 3	55	122	2.00	P	2544	786
DAVIS-BESSE	40	95	2.87	S	2772	1035
DIABLO CANYON 1&2	47	NP	2.63	R	3400	774
FARLEY 1&2	54	116	2.03	P	2652	765
FORT CALHOUN 1	60	190	1.05	P	1500	700
GINNA	60	129	0.97	P	1520	638
H.B. ROBINSON 2	42	130	2.10	P	2300	913
HADDAM NECK	40	90	2.23	R	1825	1222
INDIAN POINT 2	47	126	2.61	R	3071	850
INDIAN POINT 3	47	134	2.61	R	3025	863
KEWAUNEE	46	151	1.32	S	1650	800
MAINE YANKEE	55	122	1.86	R	2630	707
MILLSTONE 2	54	150	1.85	P	2700	685
MILLSTONE 3	45	117	2.30	R	3411	674
NORTH ANNA 1&2	45	128	1.73	R	2893	598
OCONEE 1,2&3	59	160	1.91	P	2568	744
PALISADES	55		1.60	P	2530	632
PALO VERDE 1,2,&3	60	169	2.60	P	3800	684
POINT BEACH 1&2	60	162	1.07	P	1518	705

Table 12.13 Containment properties for PWRs.

Plant	Containment design pressure (psig)	Ultimate pressure (psig)	Containment volume (1E6 ft^3)	Construction*	Thermal power (MWt)	Ratio volume/ thermal power (ft^3/MWt)
PRAIRIE ISLAND 1&2	41	165	1.32	S	1650	800
SALEM 1	47	123	2.62	R	3411	768
SALEM 2	47	123	2.62	R	3411	768
SAN ONOFRE 2&3	60	157	2.34	P	3390	690
SEABROOK	52	187	2.70	R	3411	792
SHEARON HARRIS 1	45	150	2.10	R	2775	757
SOUTH TEXAS PROJECT 1&2	56	NP	3.30	P	3800	868
ST. LUCIE 1	44	95	2.50	S	2560	977
ST. LUCIE 2	44	95	2.50	S	2560	977
SUMMER	55	142	1.84	P	2775	663
SURRY 1&2	45	126	1.80	R	2441	737
TMI 1	55	140	2.00	P	2535	789
TURKEY POINT 3&4	59	150	1.55	P	2200	705
VOGTLE 1&2	52	145	2.76	P	3411	809
WATERFORD 3	44	135	2.68	S	3390	791
WOLF CREEK	60	128	2.50	P	3411	733
ZION 1&2	47		2.86	P	3250	880
Ice condensers containments						
CATAWBA 1&2	30	85	1.22	S	3411	358
D.C. COOK 1&2	12	36	1.20	R	3411	352
MCGUIRE 1&2	28	77	1.29	S	3411	378
SEQUOYAH 1&2	11	60	1.22	S	3411	358
WATTS BAR 1	15	95	1.19	S	3411	349

*S = Steel; P = Pre-stressed; R = Reinforced
NP = not provided in submittal
MWt = megawatt thermal

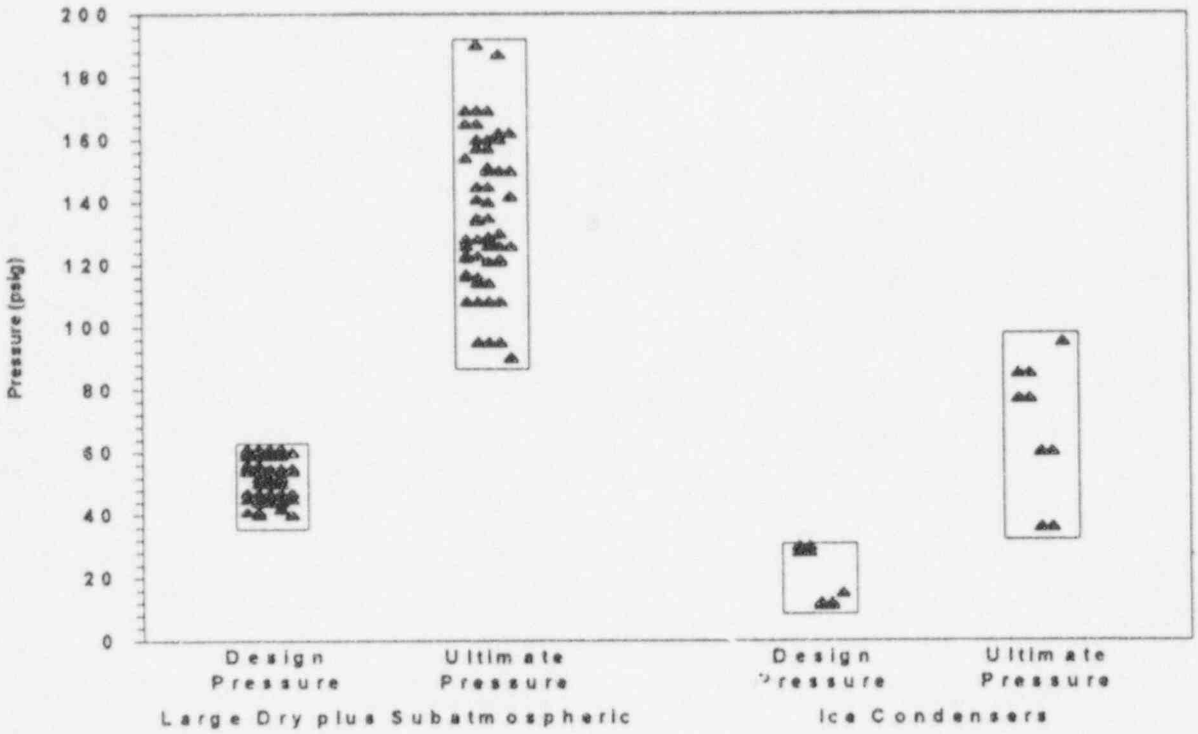

Figure 12.19 Range of containment properties for PWR plants.

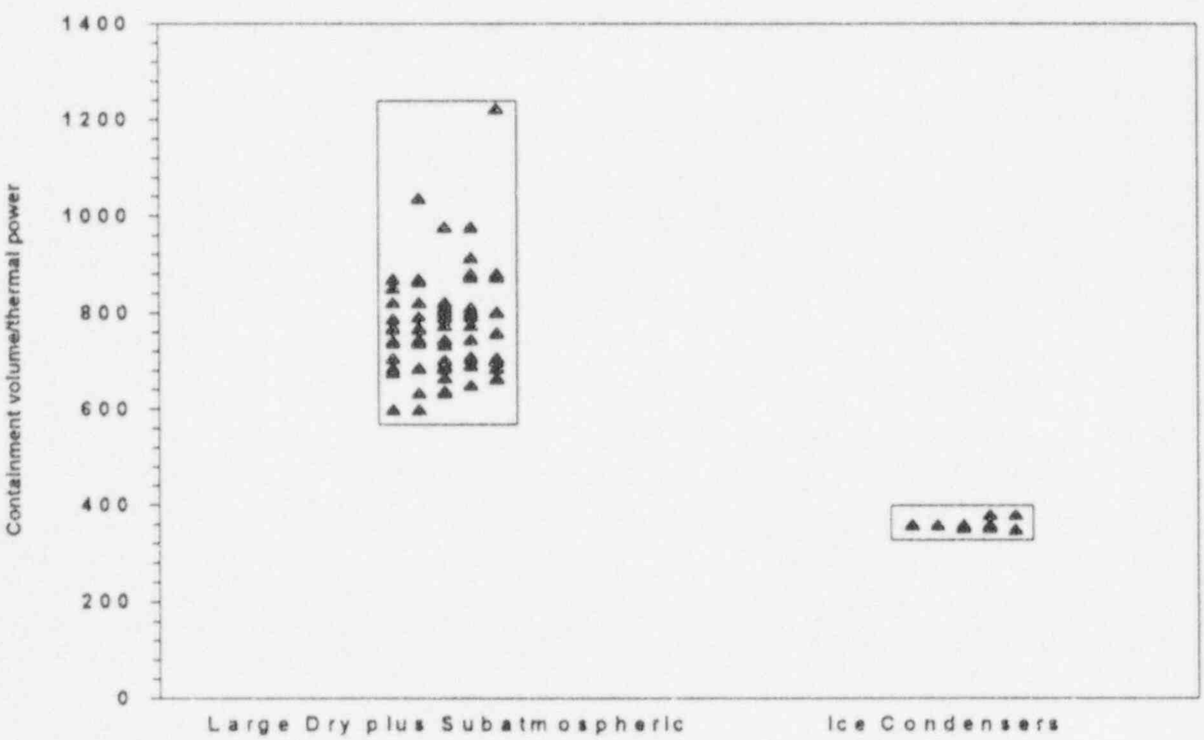

Figure 12.20 Range of containment volume to thermal power ratio for PWR plants.

Figure 12.21 shows the distribution of initiating event frequencies as well as core damage frequencies associated with ISLOCA for each PWR containment group as reported in the PWR IPE submittals. Figure 12.22 presents the same data for SGTR events. For the majority of cases the CDF shown in these figures translates directly to a containment bypass frequency. It is interesting to note that the reported ISLOCA initiating event frequencies as well as the CDF are quite low except for a few cases. Almost all values are below 1E-5/ry and most are less than 1E-6/ry. However, SGTR initiating event frequencies are in the 1E-2/ry range and their CDF contributions, although still relatively small compared to total PWR CDFs (See Chapter 11), are significant for many plants since they represent containment bypass sequences which usually involve sizeable releases. Many of the plants with the higher values of SGTR, and therefore bypass frequency, are also the ones which are relevant for the safety goal discussion of Chapter 16.

Induced SGTR (i.e., an SGTR which is not an accident initiator but which results from the conditions of a severe accident) 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, as analyzed in the IPEs.

The key perspectives coming out of the reported IPE results for the entire family of PWR containments are summarized in Table 12.14.

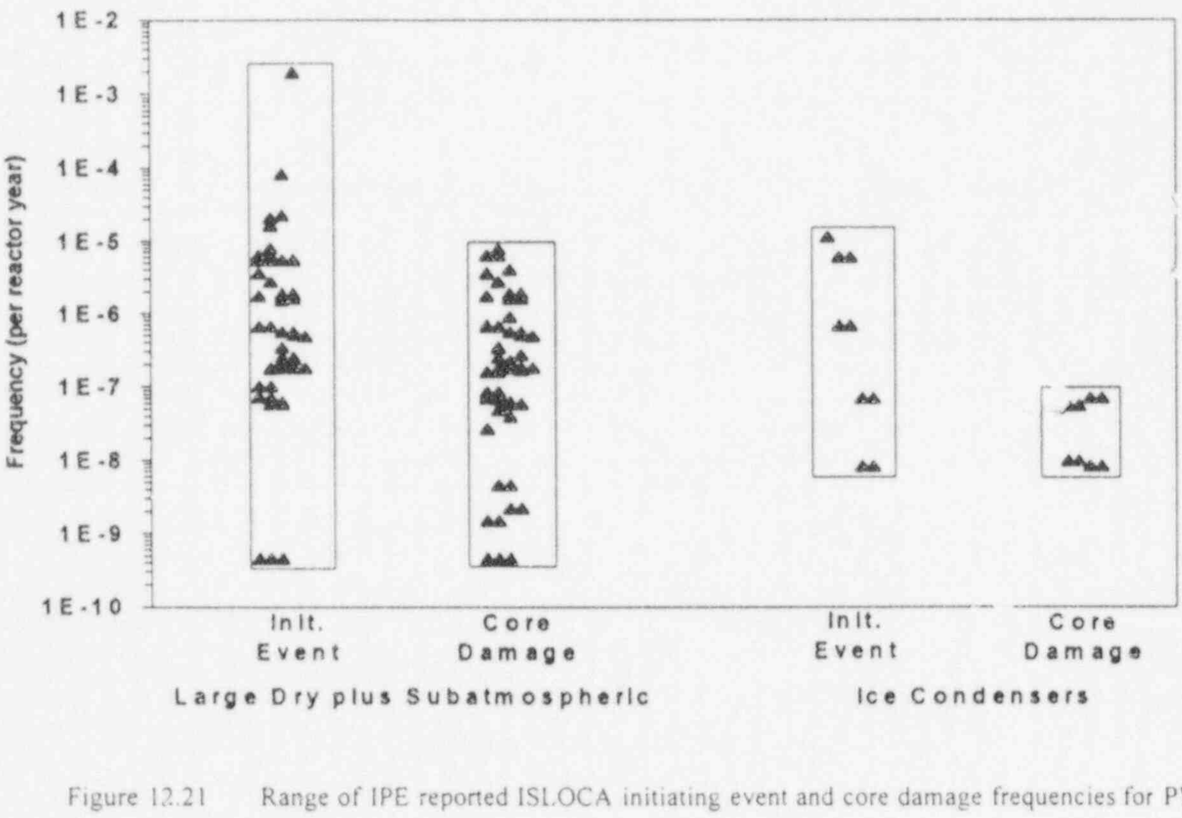

Figure 12.21 Range of IPE reported ISLOCA initiating event and core damage frequencies for PWR plants.

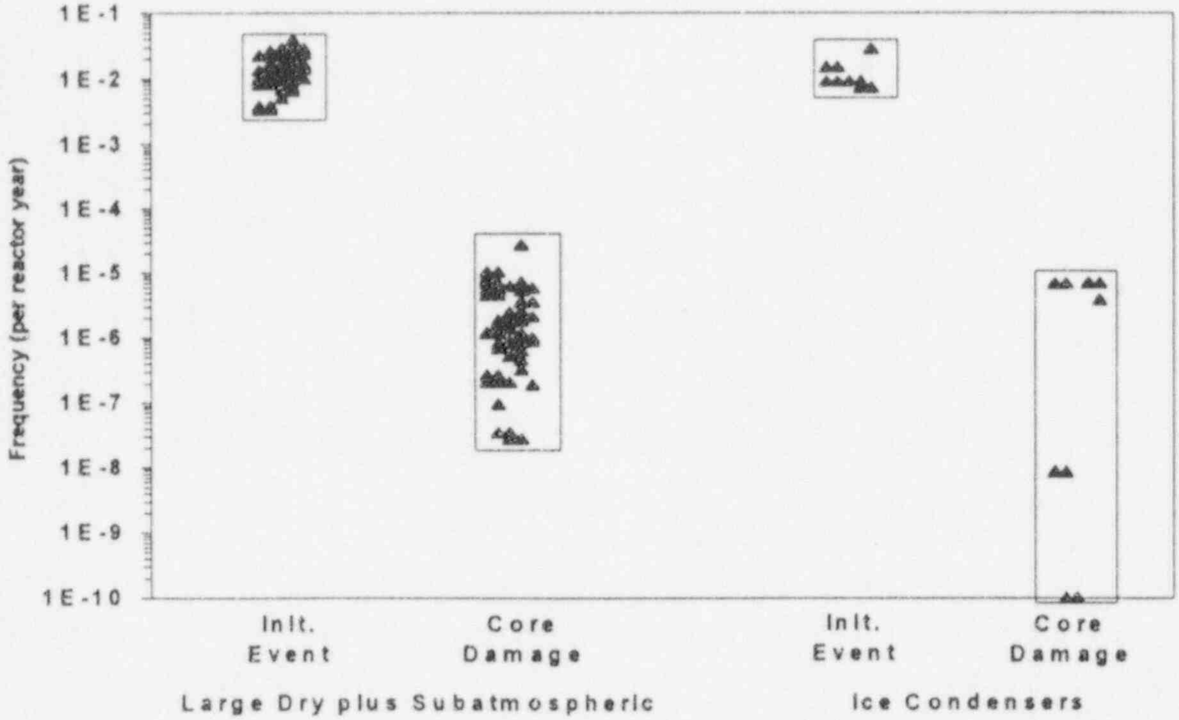

Figure 12.22 Range of IPE reported SGTR initiating event and core damage frequencies for PWR plants.

Table 12.14 Summary of performance for PWR containments.

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

Table 12.14 Summary of performance for PWR containments.

Failure mode importance	Important design features, operator actions, and model assumptions	Important plant improvements
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

12.3.1 PWR Large Dry and Subatmospheric Perspectives

Sixty-four PWR reactor units and 1 BWR unit (described in forty-four submittals) are housed in large dry containments as indicated in Table 12.15.

Table 12.15 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.

12.3.1.1 Summary of Results and Perspectives for PWR Large Dry and Subatmospheric Containments

The IPE reported values of conditional containment failure probability for this group are shown in Figure 12.23 while containment failure frequencies are shown in Figure 12.24.

The containment failure modes in Figure 12.23 and 12.24 are grouped into early failure, containment bypass, and late failure. The early failure category includes both isolation failure and containment failure before and shortly after vessel breach. The bypass category includes ISLOCA, SGTR, and temperature-induced SGTR. The late containment failure category includes containment overpressure or overtemperature failure and basemat melt-through.

The average conditional probability for the group is 0.05 for early failure, 0.07 for containment bypass, and 0.3 for late failure. On average there is a 0.6 conditional probability for no containment failure. Distributions of the various containment failure modes for the 38 IPEs show significant spread in the data from the IPEs. For example, the conditional probability varies from negligible to over 0.3 for early failure, from negligible to over 0.4 for containment bypass, and from negligible to over 0.7 for late containment failure.

Figure 12.24 indicates that the early failure frequencies reported range from 2E-8/ry to 5E-5/ry, frequency of containment bypass, which is primarily established via the CDF analysis, varies from negligible to almost 4E-5/ry, and late failure frequencies range from negligible to 3E-4/ry.

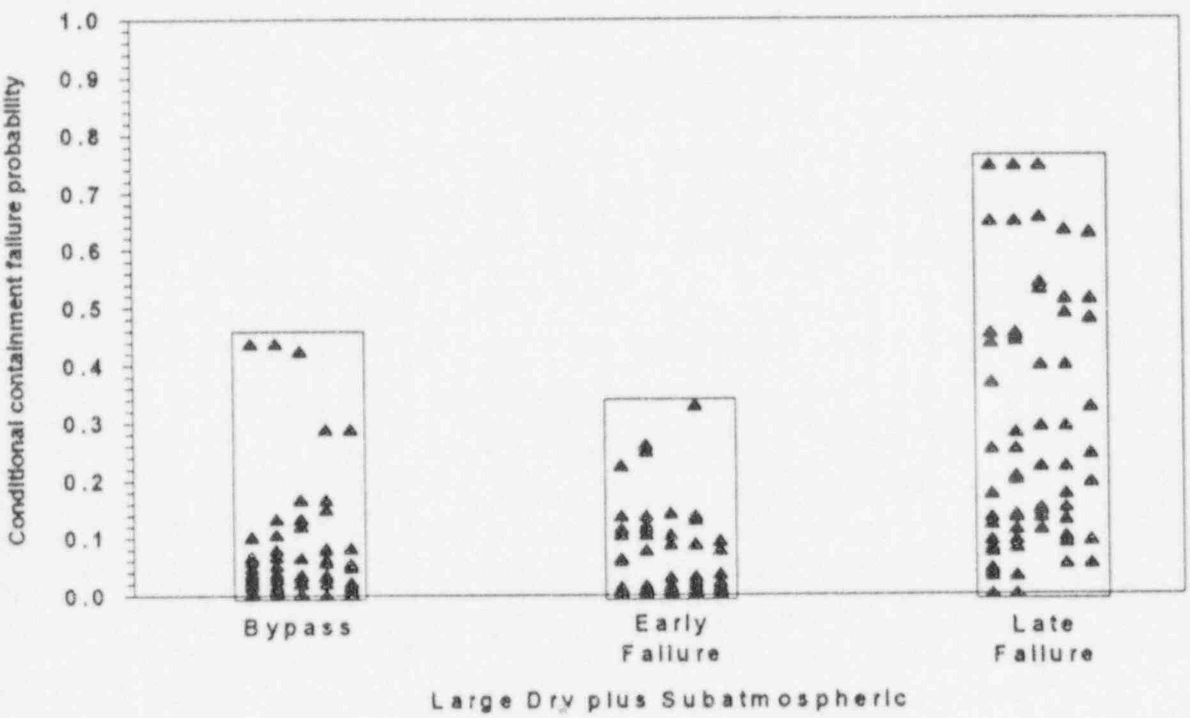

Figure 12.23 Reported IPE CCFPs for PWRs in large dry and subatmospheric containments.

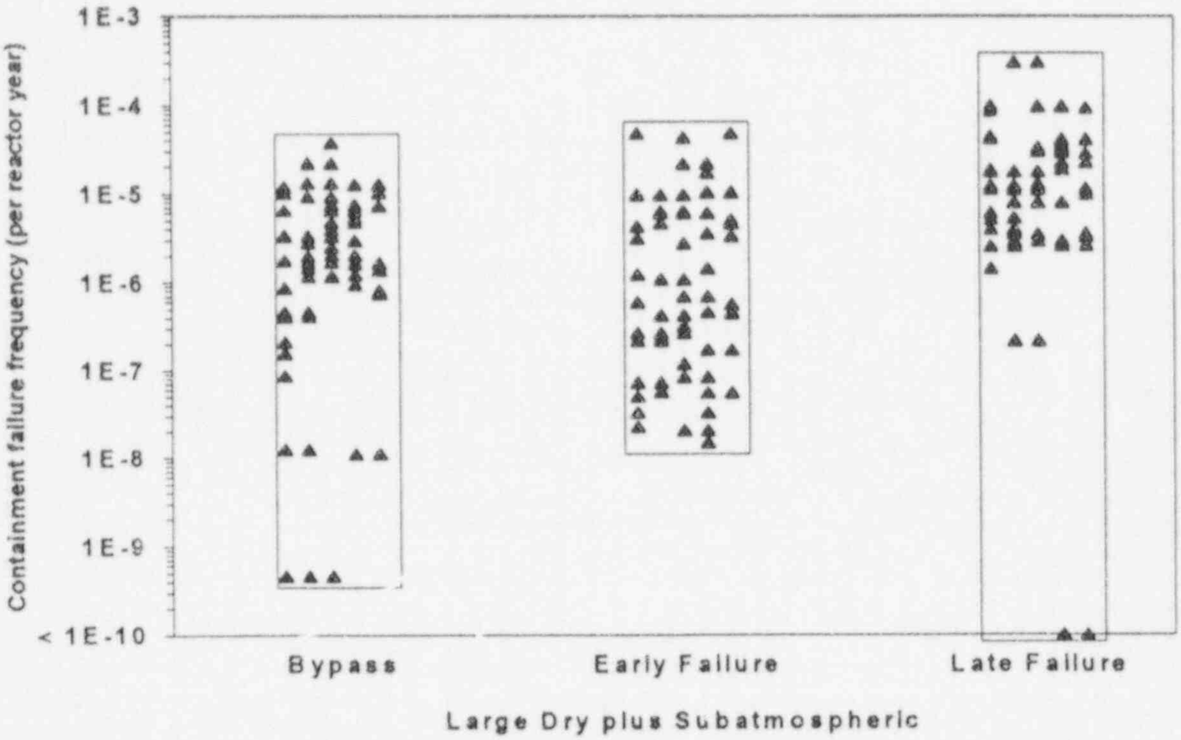

Figure 12.24 Reported IPE containment failure frequencies for PWRs in large dry and subatmospheric containments.

Both the probabilities and frequencies of containment bypass and early containment failure are roughly comparable in many of these plants. 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 12.3.2) 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 calculated containment strength, 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. Results and key perspectives are summarized in Table 12.16.

Table 12.16 Performance summary for PWR large dry and subatmospheric containments.

Failure mode importance	Important design features, operator actions, and model assumptions	Summary of results
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	Early failure frequencies range from 2E-8/ry to 5E-5/ry. Average frequency is 5E-6/ry Early failure CCFPs range from negligible to over 0.3. Average probability is 0.05
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 or larger than early structural failure in many PWR IPEs	Bypass frequencies range from negligible to 4E-5/ry. Average frequency is 5E-6/ry. Bypass failure CCFPs range from negligible to over 0.4. Average probability is 0.07
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	Late failure frequencies range from negligible to 3E-4/ry. Average frequency is 3E-5/ry. Late failure CCFPs range from negligible to over 0.7. Average probability is 0.3

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.

12.3.1.2 Early Failure Perspectives

The potential containment failure modes of large dry containments involving early containment failure include (1) overpressurization because of non-condensible gases and steam, as a result of hydrogen combustion processes and because of DCH, (2) missiles and pressure loads as a result of steam explosion, and (3) missile thrust forces as a result of blowdown at high pressure. Containment shell melt-through because of direct contact between core debris and the containment wall is also a possibility in some cases.

Because of their high containment pressure capabilities and large containment volume to thermal power ratios, large dry containments are not likely to fail before vessel breach by slow pressurization from non-condensible gases and steam or early hydrogen combustion. 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 (1) blowdown of reactor coolant system steam and hydrogen inventory into the containment, (2) combustion of hydrogen released before and during HPME, (3) interaction between molten core debris and water on the containment floor, and (4) DCH. This combined load is referred to as the DCH load in some IPEs. These containment failure modes are evaluated in all the IPEs submittals under discussion, although treated with varying degrees of thoroughness.

In analyzing containment performance against these loading conditions, plant-specific design features as well as assumptions made in the IPE analyses play a significant role for the obtained results. Some of the more important issues are discussed below:

Reactor Vessel and Reactor Cavity Design — The parameters for reactor vessel and reactor cavity design that are important for containment failure include (1) lower head penetrations in the reactor vessel, (2) the communication paths between reactor cavity and the containment atmosphere, (3) the floor area of the reactor cavity, (4) the flow of water from the containment floor to the reactor cavity. The IPEs reflect current thinking that a reactor vessel with lower head penetrations is more likely to develop a leak type of vessel failure while a reactor vessel without lower head penetration may take a longer time to melt-through with a higher potential of a rupture type failure. This affects core debris dispersion and consequently the challenge to containment failure because of high-pressuremelt ejection. The potential of HPME is also affected by the communication paths between the reactor cavity and containment atmosphere. The flow area between the reactor cavity and the containment atmosphere is usually large for plants that have an instrument tunnel leading from the reactor cavity to the containment volume. Such a configuration was usually taken in the IPEs as promoting core debris dispersion to the containment volume and increasing the challenge of DCH but also increasing the probability of forming a coolable debris in the reactor cavity. For plants with a steel containment, direct attack by the core debris of the containment steel shell is a concern, although such an attack may also pose a challenge for concrete containments. For example, in one IPE the analysis showed that in this plant the reactor building floor and the cavity region are at the same elevation. The cavity area connects with the instrumentation tunnel, and is bounded by the outer reactor building wall. Therefore, during a severe accident direct contact of the molten core material with the reactor building liner is possible. Also, because a path from the reactor cavity to the outer reactor building wall exists via the incore instrument tunnel, there is a probability of containment failure by ex-vessel steam explosion at vessel breach.

The floor area of the reactor cavity and the availability of water to the reactor cavity affect the probability of ex-vessel debris cooling. Water in the reactor cavity before vessel failure also affects the potential challenge at vessel breach from an ex-vessel steam explosion and HPME. On the one hand the reactor cavity for some plants always is assumed to remain dry because of some special feature. For example, in the Millstone 2 analysis, as a result of the presence of the neutron shield ring, only a negligible amount of containment spray water is assumed to actually flow into the cavity via the annulus around the reactor vessel and the reactor cavity type is assumed to remain dry. On the other hand, the reactor cavity is assumed to be usually flooded in other IPEs. For example, in the St Lucie IPE submittal, it is stated that a key feature of the St Lucie containment design is that for almost all accident sequences, the reactor cavity is flooded with water.

The core debris dispersed outside the reactor cavity is assumed to be coolable in most of the IPEs. However, in at least one IPE, the containment spray is required for successful cooling of the debris relocated to the upper containment floors, and for cases where a significant amount of core debris is dispersed to the containment floors, long-term containment failure is assumed if containment spray is not available.

In-Vessel Recovery — In-vessel recovery is considered in some IPEs for PWRs with large dry containment while it is excluded in others. Recovery actions that may result in an in-vessel recovery include recovery of AC power for SBO sequences and RCS depressurization by operator action. RCS depressurization allows the low-pressure injection system, unable to inject because of high RCS pressure, to inject coolant to the RCS. With the recovery of RCS coolant injection, successful in-vessel recovery (i.e., core melt terminated) also requires that the in-vessel core debris is in a coolable configuration. In-vessel recovery via cooling of debris through the vessel wall by the water in the reactor cavity (e.g., ex-vessel cooling) is discussed in some IPEs, but credit for in-vessel recovery by this mechanism is not taken by most because of lack of sufficient supporting data. In-vessel recovery will eliminate the challenges associated with vessel failure such as HPME and ex-vessel steam explosion. It also eliminates the challenge to containment integrity and the source term associated with core-concrete interaction. It should be noted that while RCS depressurization eliminates the challenge to containment integrity associated with HPME, the probability of alpha-mode failure (from an in-vessel steam explosion) increases. Although the NUREG-1150 conditional probabilities are used for alpha-mode failure for some IPE analyses (i.e., 8E-3 for low-pressure sequences and 8E-4 for high-pressure sequences) significantly lower values are used in some other IPEs.

Ex-Vessel Debris Coolability and Mission Time — The probability of late containment failure often depends on the assumption made in the IPE analysis regarding ex-vessel debris coolability and the length of the mission time for which the analysis is carried out. For some IPEs that use a mission time of less than or equal to 48 hours, late containment failure is assumed not to occur even if the debris is not coolable because accident sequence calculations (i.e., by the Modular Accident Analysis Program (MAAP)[12.10] code) show that the containment failure pressure is not reached within the mission time. For example, the lack of water to the core debris in Vogtle may contribute to the lack of containment overpressure failure within the 48-hour mission time used in the IPE. In the Vogtle IPE, the containment failure pressure is not reached for any of the sequences analyzed using the MAAP code within the 48-hour mission time.

For some other IPEs, containment failure (e.g., by non-condensible gas generation or basemat melt-through) is assured if the ex-vessel debris is not coolable. While debris cooling is assured in some IPEs if the debris is covered with water, the probability of successful debris cooling is assigned a low probability in some other IPEs even if the debris is covered with water.

Containment Pressure Capability and Containment Over-Pressure Failure — In the NUREG-1150 analysis, because of the inherent uncertainties, both the containment pressure capability and the containment pressure loads are treated as distributions and the containment failure probability is determined by a sampling method. While this is done in some IPEs, deterministically predicted containment pressure loads are used in most IPEs to determine the containment failure probability (by comparing the pressure load with the containment fragility curve). In some other IPEs, a bounding containment failure pressure, instead of a distribution, is used for containment pressure capability, and predicted containment pressure loads are compared with this bounding pressure to determine containment failure. In some analyses certain containment failure modes (e.g., DCH) are dismissed as unlikely to cause containment failure (and not included in the back-end quantification process) if the predicted pressure load for the containment failure mode is less than the bounding pressure. For a group of IPEs, all early containment failure modes are dismissed as unlikely and consequently the probability of early containment failure in these IPEs is zero. Since a bounding failure pressure is usually taken as the 95th percentile value of the fragility curve, there is a potential for containment failure probability of 0.05 even if there is no uncertainty in the predicted containment pressure loads.

[12.10]Fauske and Associates, Inc., "MAAP Modular Accident Analysis Program User's Manual," Volumes 1-2, IDCOR Technical Report 16.2-3, February 1987.

According to the IPE analyses of most PWR plants the most important challenges to containment integrity before or at vessel breach are those associated with HPME. As noted above, there are significant uncertainties related to the containment pressure loads that can be produced from the energetic events associated with HPME. The pressure of the 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. It was reported in these IPEs 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 5th 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 reported as not credible in these IPEs.

The predicted containment pressure loads are higher in IPEs that reported relatively higher early containment failure probabilities (i.e., from about 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 already to be 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.

As noted before, because of their high containment pressure capabilities and large containment volume-to-thermal power ratios, large dry containments as a group are not likely to fail before vessel breach by slow pressurization from non-condensible 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 non-condensibles or because of a combination of containment pressurization and an early hydrogen burn. For example, according to MAAP calculations cited in the IPE submittal, the containment failure pressure of 169 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 136 to 183 psig for cases without sprays, and from 122 to 136 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 116 psig.

As previously noted, an obvious parameter that affects the probability of early containment failure from the 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 (i.e., about 70% when the RCS is at the pressurizer power operated relief valve (PORV) setpoint pressure and about 3% 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 significant 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 containment performance issues. 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. (Subsequent to a request for additional information from the NRC the licensee for Waterford calculated a much reduced early failure probability using a more conventional fragility curve.)

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, do not result from the high-pressureloads 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. The IPE results indicate that side wall failure could occur if 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. (It should be noted that 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.)

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 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 in-vessel 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 12.23, 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 12.23 also shows that large dry containments as a group are quite robust in response to severe accident challenges. In many analyses most core damage accident sequences did not lead to containment failure. 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 in-vessel recovery actions. (i.e., gross damage and vessel failure are prevented if sufficient coolant injection becomes available after core damage has occurred and the core is cooled in-vessel.) A few in-vessel 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. In-vessel 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 in-vessel cooling and prevention of vessel failure is likely. Individual IPEs include the above in-vessel 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 in-vessel 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).

Isolation failure is assumed to be negligible in some PWR 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 significant early release given a core damage event is a large containment isolation failure. This includes (1) failure to isolate the large supplemental purge penetrations in the unlikely event that a purge is in progress during the accident, and (2) 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 0.15. 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, on the basis of the early results from the IPE, was the changeover from motor-operated to air-operated containment isolation valves in some containment penetrations.

Figure 12.25 shows how the conditional probability of containment isolation failure reported in the IPEs for PWRs with large dry containments compares with the conditional probability of total early containment failure reported. The figure indicates that while isolation failure was found to be quite small in many of these plants it makes up a significant part of the early failure probability in a number of others.

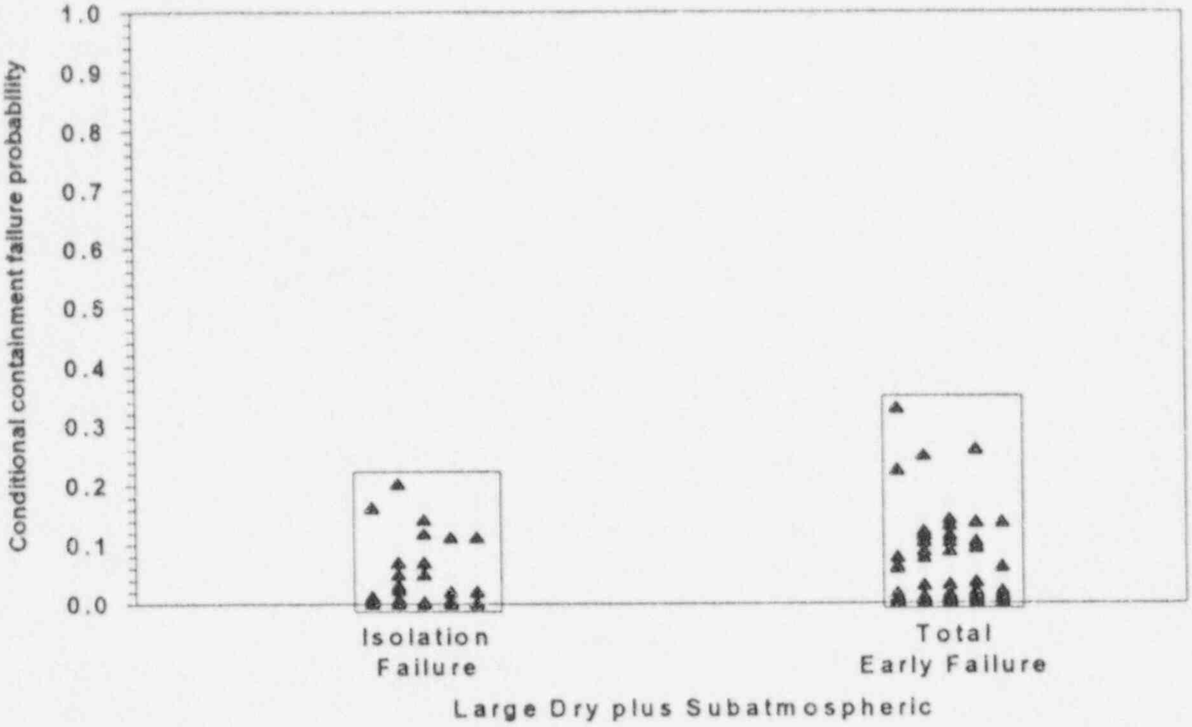

Figure 12.25 Comparison of IPE reported conditional probabilities of isolation failure and total early failure for PWRs with large dry and subatmospheric containments.

12.3.1.3 Containment Bypass Perspectives

The results of both Figures 12.23 and 12.24 indicate that both the probability as well as the frequency of containment bypass is approximately equal to the probability of early failure for many of the PWR plants in large dry containments. Often in the IPE analyses the bypass sequences were found to result in the most severe releases.

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 conditional probability value (given that the RCS is at system pressure) used in the IPEs is about 0.01; therefore, temperature-induced SGTR is generally not found to be 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 the 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).

These submittals report that 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 may further increase the potential for failure. However, 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, and in some that do, a low probability of temperature-induced SGTR is used on the basis of the expected limited duration of RCP operation. This variability in the treatment of temperature-induced SGTR in the IPEs indicates the large uncertainty associated with this issue. Figure 12.26 shows for each plant in this group the fractional contribution to CDF from ISLOCA and SGTR initiators as well as their combined contribution. The figure also indicates the bypass fraction used in the Level 2 analysis. The difference between the Level 1 total contribution and the Level 2 values is principally because of the induced SGTR found in the individual analyses. As the figure shows, this difference is non-existent or small in most cases but there are some significant exceptions.

After the Prairie Island IPE, the highest bypass probability is predicted in the Ginna IPE (approximately 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, derived from the CDF analysis. The IPEs for Braidwood and Byron report 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.

12.3.1.4 Late Failure Perspectives

The containment phenomena that may cause late containment failures in large dry containments include (1) overpressurization with high temperatures due to non-condensible gases and steam or due to combustion processes, (2) containment basemat melt-through due to basemat penetration by core debris, and (3) vessel structural support failure due to core debris erosion.

The IPE results show that the dominant late containment failure mode is containment overpressurization, which occurs when CHR capability is lost. For some of the IPEs, fan coolers, which are not designed as engineered safeguards features, are credited for CHR. Late containment failure probabilities for the large dry containments considered in the IPEs range from negligible to about 0.7 with an average value of about 0.3.

Basemat melt-through occurs when the concrete basemat is penetrated because of CCI. This may happen if CCI is not terminated either because there is no water in the reactor cavity or because the core debris is not coolable even if water is available. Since the basemats of some PWR containments have considerable thickness, eventual penetration of the basemat by the core debris is not certain even if a large fraction of the core is involved in CCI and water is not available. Depending on the amount of core involved in the CCI, and the cavity condition, as well as the type of concrete, basemat melt-through probabilities vary from 0.05 to 0.4 for Surry 1&2 and from 0.05 to 0.8 for Zion 1&2 in NUREG-1150. These values are typical of those used in the IPEs.

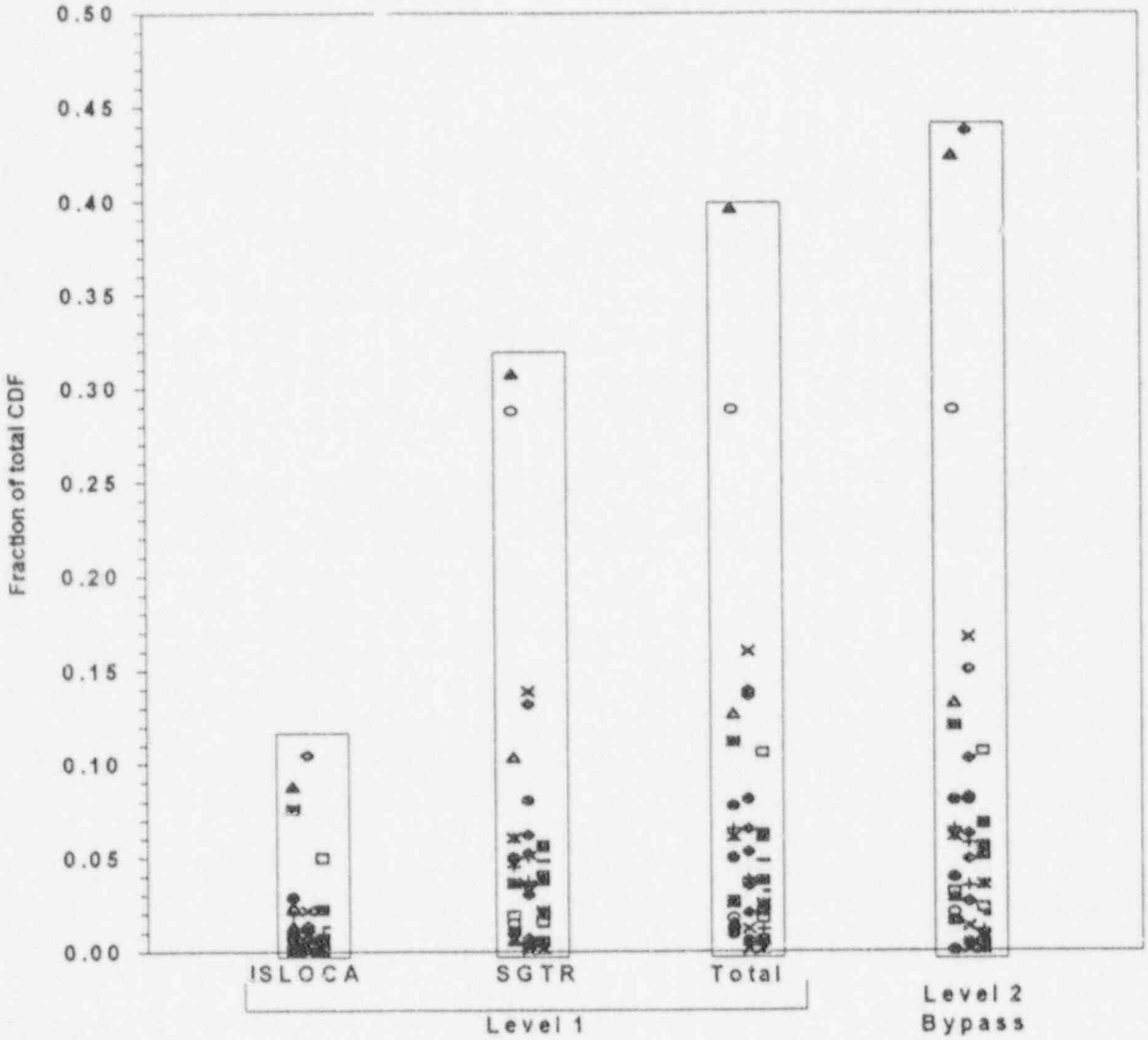

Figure 12.26 Comparison of IPE reported fractional contribution to CDF from ISLOCA and SGTR with bypass fraction for PWR large dry and subatmospheric containments.

Containment failure as a result of reactor vessel support structure failure is not likely for large dry containments. This is because the reactor vessel is usually away from the containment walls and there are structures located between the reactor vessel and the containment boundaries. Therefore, vessel structural support failure, even if it occurs, will not cause a containment failure in large dry containments.

While late failure results from overpressurization in most PWR IPEs, there are exceptions. 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.) In 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.

One important reason for the low probabilities for late failure found in some IPEs is the use of a 48-hour mission time. This is because containment pressurization is largely due to the generation of non-condensible 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 the 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.

Overall the IPE results confirm that the large dry containments are quite robust to severe accident challenges. They are not very susceptible to containment overpressure challenges because of their large volumes and high structural strengths.

For these reasons, there is a high probability that a large dry containment will remain intact during a severe accident. According to the IPE results the average probability of no containment failure for large dry containments is 0.6 and for individual plants the probability varies from 0.2 to over 0.9. The higher probabilities of no containment failure are often because of the use of a 48-hour mission time and the lower values most likely result when there is a large group of sequences involving complete failure of CHR.

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 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 reported in the IPE for Big Rock Point are small, only about 0.01 for containment bypass and 0.04 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 over 0.9. No late failures were reported in the IPE. The probability of early containment failure includes a small contribution 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.

12.3.2 PWR Ice Condenser Perspectives

Nine PWR units (described in five IPE submittals) use ice condenser containments. The plants are indicated in Table 12.17. All of these plants use a Westinghouse four-loop reactor system design.

Table 12.17 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	

12.3.2.1 Summary of Results and Perspectives for PWR Ice Condenser Containments

The conditional probabilities of the various containment failure modes reported in the IPEs for PWRs with ice condenser containments are provided in Figure 12.27 and the reported frequencies of containment failure are shown in Figure 12.28; and the results are summarized in Table 12.18. For the ice condenser containments in the IPEs the average conditional containment failure probability is 0.02 for early containment failure, 0.05 for containment bypass, 0.3 for late containment failure, and 0.6 for no containment failure. In the five ice condenser IPEs the conditional probabilities for the different containment failure modes vary from less than 0.01 to 0.1 for bypass, from less than 0.01 to 0.05 for early failure, and from 0.02 to 0.5 for late failure as indicated in Figure 12.27., and from 0.4 to 0.9 for no containment failure. The frequencies of the various containment failure modes, shown in Figure 12.28, vary from 2E-7/ry to 4E-6/ry for early failure, from 8E-8/ry to 8E-6/ry for bypass, from 1E-6/ry to 8E-5/ry for late failure. Total core damage frequency varies from 4E-5/ry to 2E-4/ry. With one exception, the probability of isolation failure in the ice condenser IPEs is small.

The average conditional early containment failure probability (0.02) of the ice condenser IPEs is smaller than the average values obtained from the large dry and subatmospheric IPEs. In two ice condenser IPEs the leading cause of early failure is direct impingement of core debris on the containment in the seal table room. In two other IPEs principal contributors to early failure are rapid steam generation, DCH, and hydrogen combustion. In the remaining IPE early failure due to DCH, steam explosions and vessel thrust forces is discounted, leaving overpressurization when containment heat removal is not available as the leading early failure mechanism. Although the majority of the ice condenser IPEs used data from the NUREG-1150 Sequoyah 1&2 analysis in their models, additional plant specific models resulted in lower failure probabilities than found in NUREG-1150. Containment bypass is dominated by interfacing systems LOCA and SGTR initiators. But one IPE found induced SGTR to be dominant because of the restart (per procedures) of the RCPs when inadequate core cooling conditions exist. The primary cause of late containment failure was found to be overpressure failure. 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.

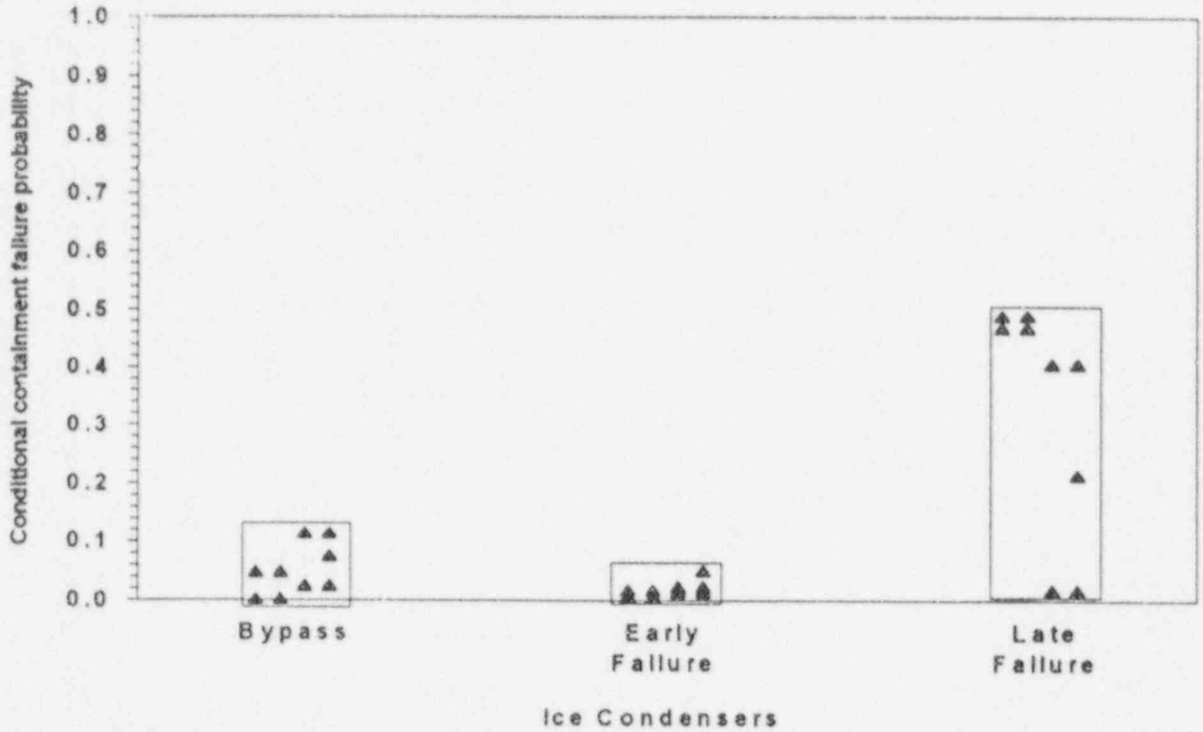

Figure 12.27 Reported IPE CCFPs for PWR ice condenser containments.

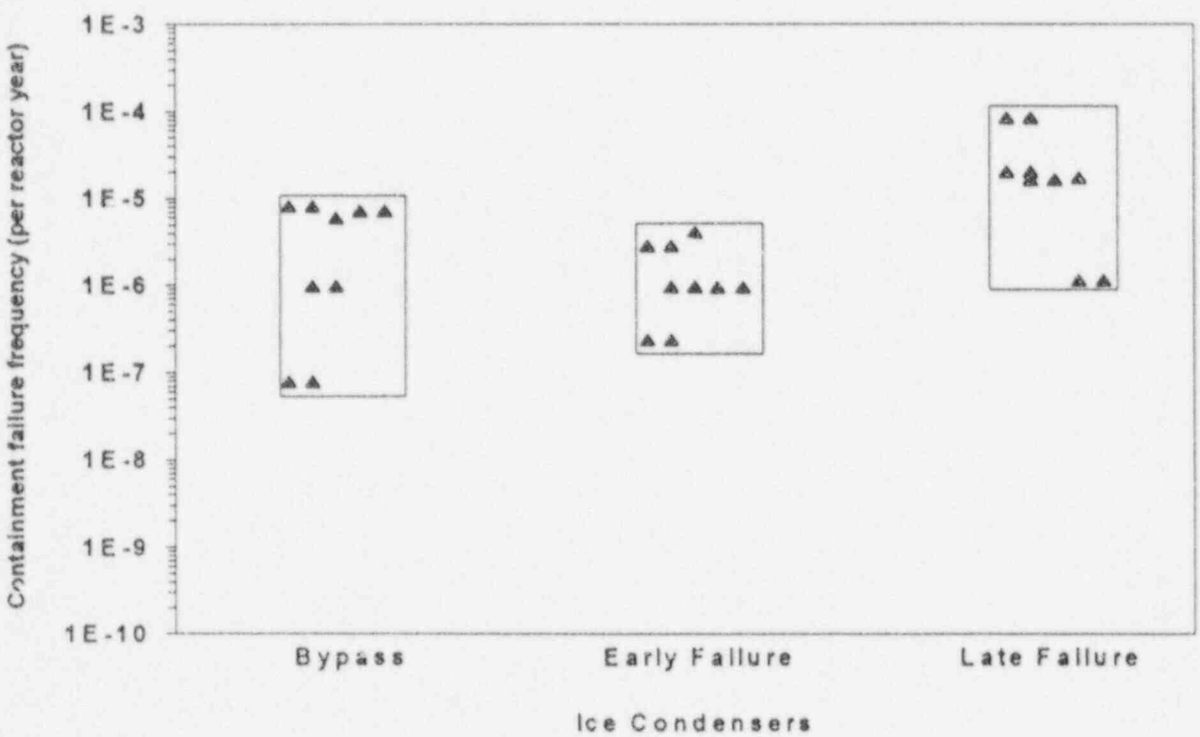

Figure 12.28 Reported IPE containment failure frequencies for PWR ice condenser containments.

Table 12.18 Performance summary for PWR ice condenser containments.

Failure mode importance	Important design features, operator actions, and model assumptions	Summary of results
	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	Early failure frequencies range from 2E-7/ry to 4E-6/ry. Average frequency is 2E-6/ry Early failure CCFPs range from less than 0.01 to 0.05. Average probability is 0.02
	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	Bypass frequencies range from 8E-8/ry to 8E-6/ry. Average frequency is 4E-6/ry Bypass CCFPs range from less than 0.01 to 0.1. Average probability is 0.05
	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	Late failure frequencies range from 1E-6/ry to 8E-5/ry. Average frequency is 3E-5/ry Late failure CCFPs range from 0.02 to 0.5. Average probability is 0.3

12.3.2.2 Early Failure Perspectives

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 through 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 prevent energetic hydrogen events.

Seven of the nine ice condenser units have a cylindrical steel containment surrounded by a concrete secondary containment. The remaining two units (the D.C. Cook plants) feature reinforced concrete containments with steel liners and lack secondary containments.

The early failure category in Figures 12.27 and 12.28 includes isolation failure as well as containment structural failure before and shortly after vessel breach. Figure 12.29 separately shows the conditional probabilities for isolation failure and total early failure. The probabilities of early containment failure (excluding isolation failure) for the five ice condenser containments range from less than 0.01 to slightly over 0.02. The average value is about 0.015.

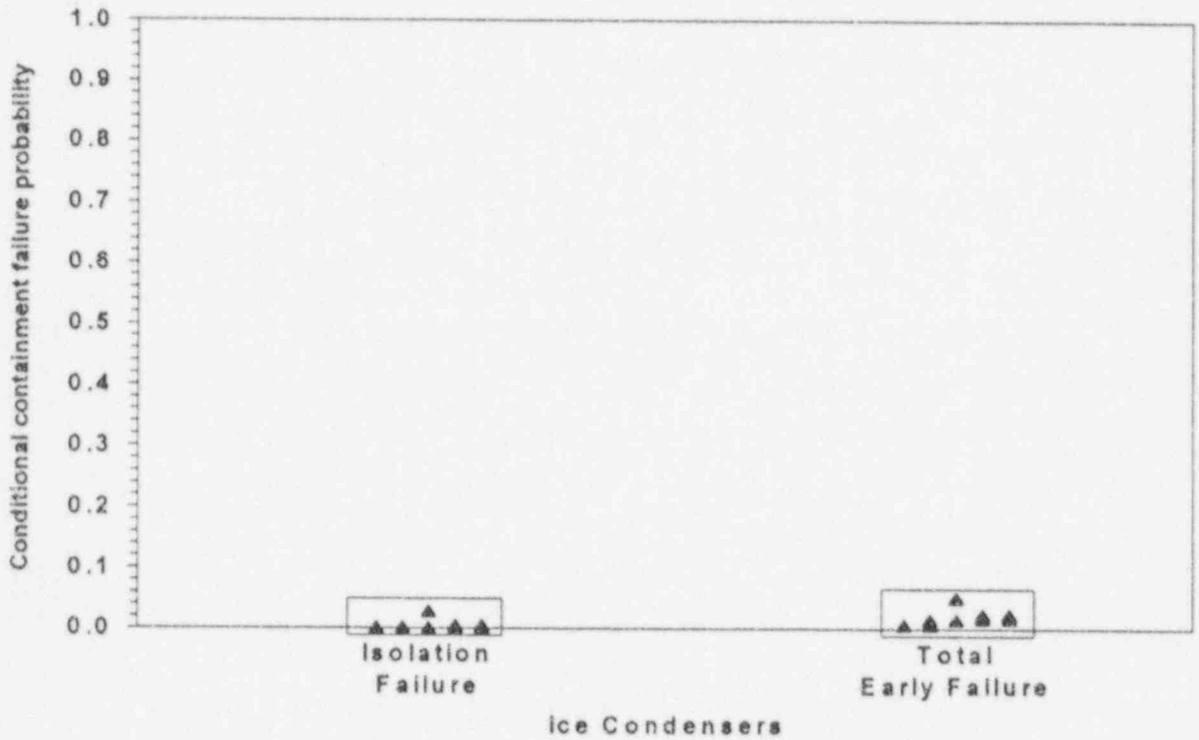

Figure 12.29 Comparison of IPE reported conditional probabilities of isolation failure and total early failure for PWR ice condenser containments.

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 1&2 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 in-vessel steam explosion and HPME/hydrogen burns at vessel breach.

- For Catawba 1&2 and McGuire 1&2, 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 isolation and early failure conditional probabilities for the ice condenser IPEs are, on average, smaller than the values obtained for the large dry and subatmospheric IPEs. 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 PORVs. Successful RCS depressurization may allow the LPI system to inject to the RCS and 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 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 1&2 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 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 on the basis of 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 (about 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 fraction 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 on the basis of 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 SBO probability for McGuire 1&2. Although the total CDFs are similar for the two plants (about 4E-5/ry), the contribution to the total CDF from loss of offsite power is more than 25% for McGuire 1&2 and less than 3% 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 core debris 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 core debris flow path. It is assumed in these IPEs, that the cavity geometry is "likely" to limit the amount of core debris 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 on the basis of 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, in-vessel 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 non-condensible gases.

Isolation failures are small for most of the ice condenser IPEs. As indicated in Figure 12.29, with the exception of one analysis (Watts Bar) the probabilities of isolation failure obtained in the IPEs for PWRs with ice condenser containments 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.

12.3.2.3 Containment Bypass Perspectives

Because of the relatively small early failure probabilities reported in the IPE submittals of the ice condenser plants, Figures 12.27 and 12.28 indicate that containment bypass scenarios are likely to dominate the early releases from these plants according to the analyses conducted during their IPE.

ISLOCA and SGTR (as an initiator) are the major bypass contributors and 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 because of 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 because of temperature-induced SGTR. This high probability for McGuire 1&2 is because of 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 as a result of 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.

Figure 12.30 shows for each plant in the ice condenser group the fractional contribution to CDF from ISLOCA and SGTR initiators as well as their combined contribution. The figure also indicates the bypass fraction used in the level 2 analysis. The difference between the level 1 total contribution and the level 2 values is principally because of the induced SGTR found in the individual analyses. The figure shows that this difference is significant for some of the plants.

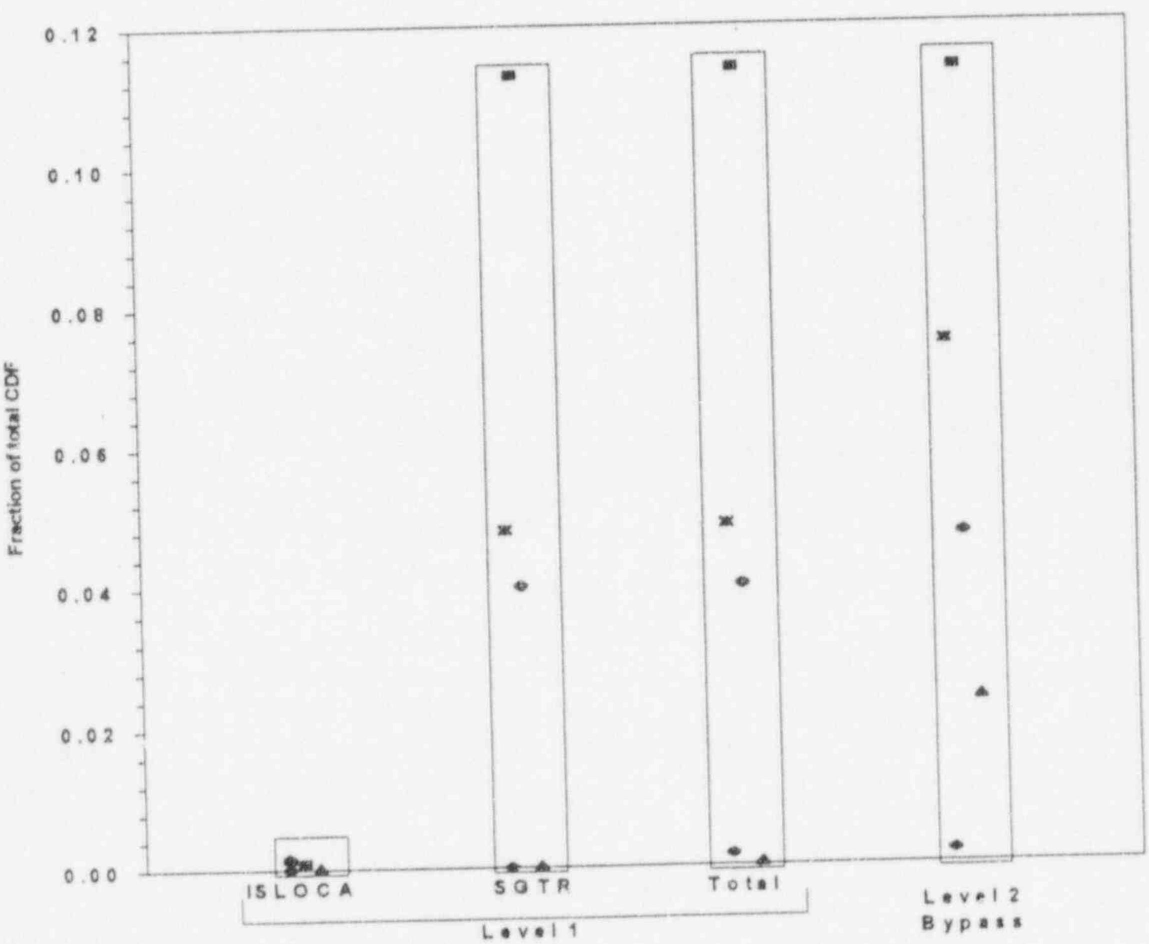

Figure 12.30 Comparison of IPE reported fractional contribution to CDF from ISLOCA and SGTR with bypass fraction for PWR ice condenser containments.

12.3.2.4 Late Failure Perspectives

The probabilities of late containment failure for the five ice condenser containments considered in the IPEs vary from less than 0.02 to almost 0.5. The average value is about 0.3. The containment phenomena that may cause late containment failures for these containments include (1) overpressurization with high temperatures as a result of non-condensible gases and steam or combustion processes and (2) containment basemat melt-through because of basemat penetration by core debris.

The dominant late containment failure mode found in the ice condenser IPEs is overpressurization when CHR 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 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 significantly vary. The extremely low failure probability for Cook is because of 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.

The average conditional probability of no containment failure for all the results of the ice condenser IPEs is about 0.6. This no failure category varies from a low probability of just over 0.4 (for Sequoyah 1&2) to a high of just under 0.9 (for Cook 1&2).

12.4 Radionuclide Release Perspectives

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[12.11] 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. 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 significant early release in the discussion below. Figure 12.31 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.

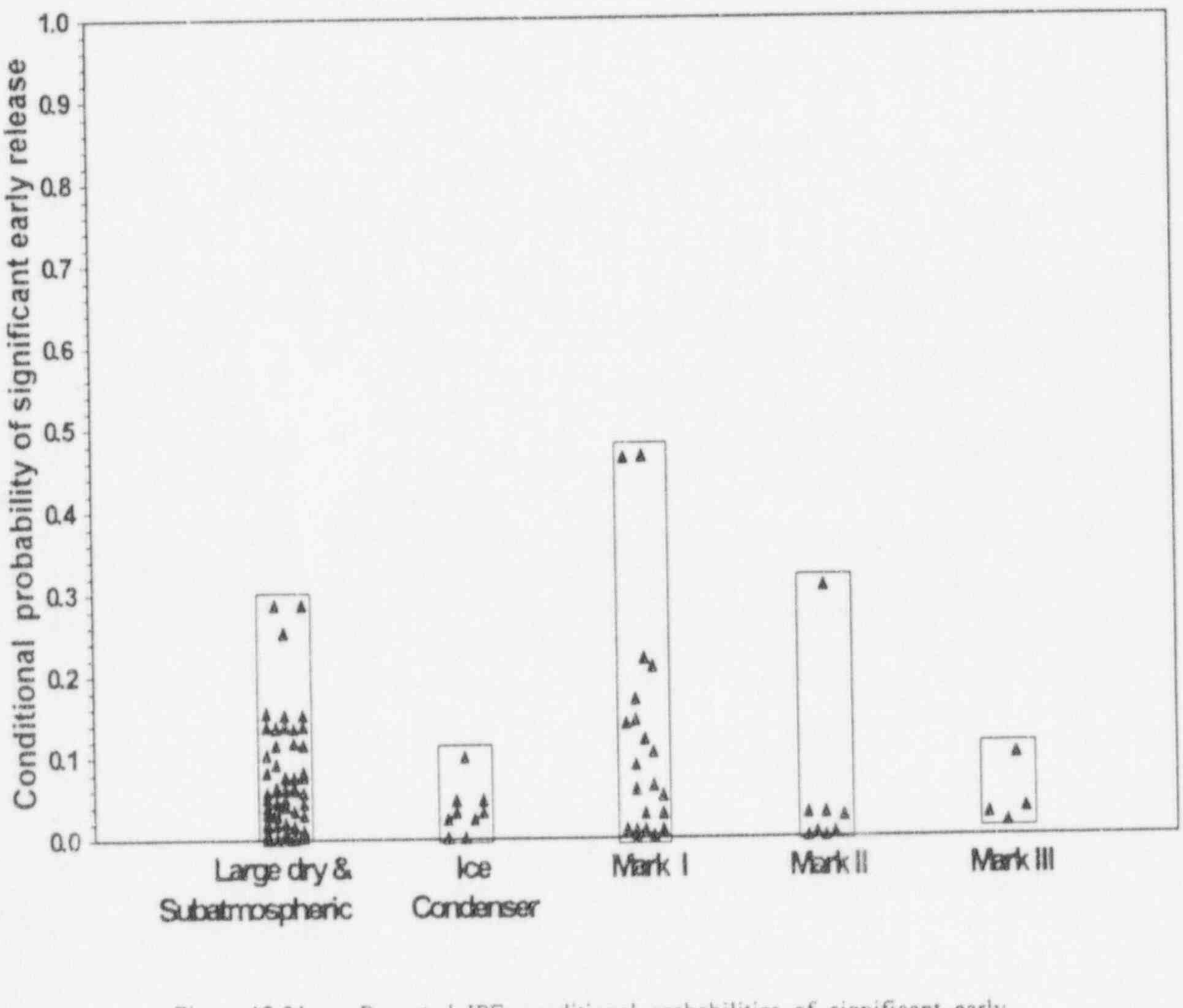

Figure 12.31 Reported IPE conditional probabilities of significant early release by containment type.

[12.11]This definition of a significant release should not be confused with a release fraction threshold used in Chapter 16 to screen for potential off-site early fatalities.

The results presented in Figure 12.31 depend on the containment performance results and therefore also reflect significant variability between the various containment types and within each containment type. Differences in containment designs account for much of the differences indicated in Figure 12.31. In a number of unique cases, plant-specific containment features are identified in the analyses as leading to significant radionuclide releases. However, differing assumptions in the accident modeling also play a major role in explaining the significant range in reported results.

The higher probabilities of significant releases in Figure 12.31 for the PWR plants are driven by bypass accidents with a major contribution from SGTR accidents. The higher probabilities of significant release for the BWR plants are caused by structural failure of the containments close to or shortly after reactor vessel melt-through. For BWR Mark I plants, shell melt-through is the leading cause of early structural failure, while overpressurization events are most important for BWR Mark II plants. More details are given for each containment type in the following sections.

12.4.1 BWR Radionuclide Release Perspective

The conditional probabilities of early containment failure and bypass (i.e., total early release) are compared with the conditional probabilities of significant early release for each BWR containment type in Figure 12.32. Similar information is provided in Figure 12.33 but in terms of frequency per reactor year.

Among the BWR plants, 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 12.32, 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 iodine and cesium less than 0.1. For example, in the Brunswick IPE, the release fractions for iodine and cesium 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 12.32, 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 2 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 (refer to Figure 12.33). The highest frequency for significant early release is from the Browns Ferry 2 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, but no accident progression analysis was carried out for the update, so the results shown are from the original submittal. Except for Browns Ferry 2, the frequencies of significant early release for all other Mark I IPEs are less than 1E-5/ry.

For Mark II containments Figure 12.32 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

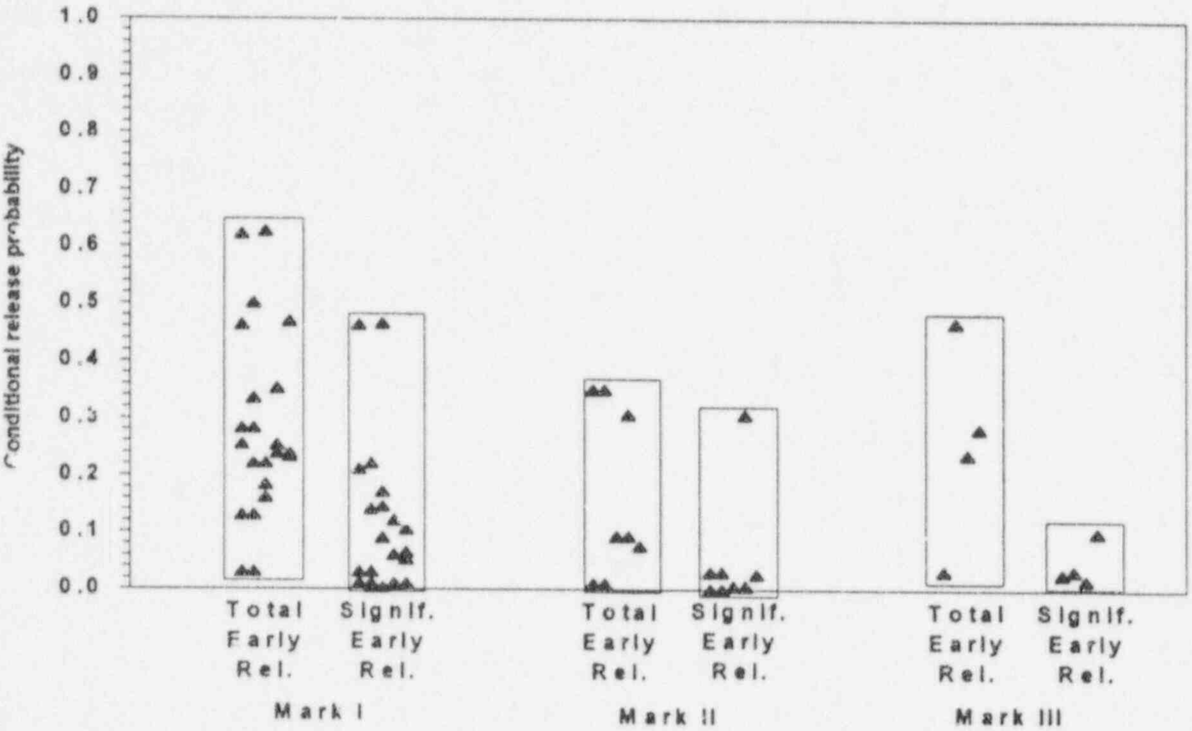

Figure 12.32 Reported IPE conditional probabilities of total and significant early release for BWR plants.

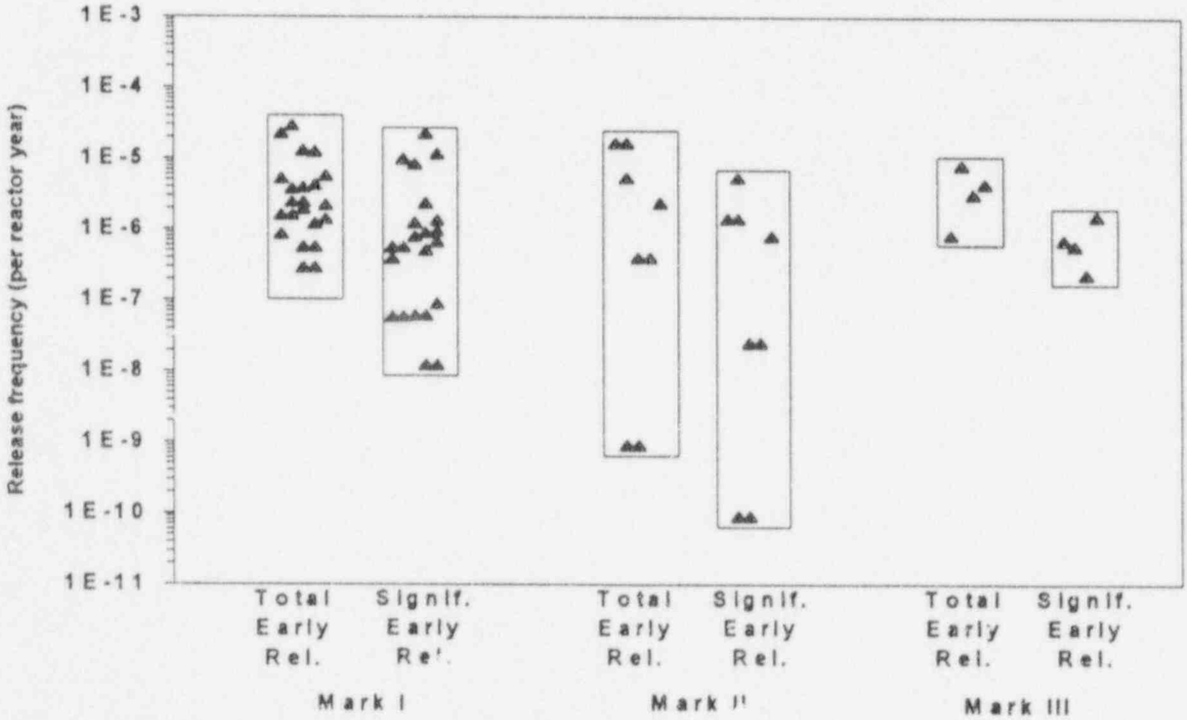

Figure 12.33 Reported IPE frequencies of total and significant early release for BWR plants.

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 12.32. The frequencies of significant early release reported for Mark II containments are negligible for Susquehanna 1&2 and vary from less than 3E-8/ry for Limerick 1&2 to about 5E-6/ry for WNP2.

Figure 12.32 also presents the distribution of the conditional probabilities of total early release and significant early release reported in the Mark III IPE submittals. Except for the River Bend results, the probability of significant release indicated includes those early release cases that involve a release fraction of volatile radionuclides equal or greater than 10% (i.e., the release fraction of either iodine or cesium is equal to greater than 10%). For River Bend, the data presented for significant early releases includes the early failure cases with a large failure size (i.e., containment dome or anchor failure). A different definition is used for River Bend because the release fractions for both iodine and cesium are predicted to be less than 10% for all release categories in this IPE submittal.

Figure 12.32 shows that while the conditional probabilities vary from 0.03 to almost 0.5 for total early release, they vary from less than 0.02 to about 0.1 for significant early release. The high value, 0.1, is obtained for River Bend, for which, as noted above, significant early failure rather than iodine or cesium release fractions greater than 10% is used, so the 0.1 value may be overestimated. Besides River Bend, the conditional probabilities of significant early release are about 2 to 3%. The frequencies of significant early release are shown in Figure 12.33 and vary from about 2E-7/ry reported for Perry to about 2E-6/ry for River Bend. Except for River Bend, the frequency for significant early release reported is less than 1.0E-6/ry for all the Mark III IPEs.

12.4.2 PWR Radionuclide Release Perspective

The conditional probabilities of early containment failure and bypass are compared with the conditional probabilities of significant early release for the PWR containment types in Figure 12.34. The frequencies of early containment failure and bypass are compared with significant early release frequencies in Figure 12.35.

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 12.34 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 12.34.

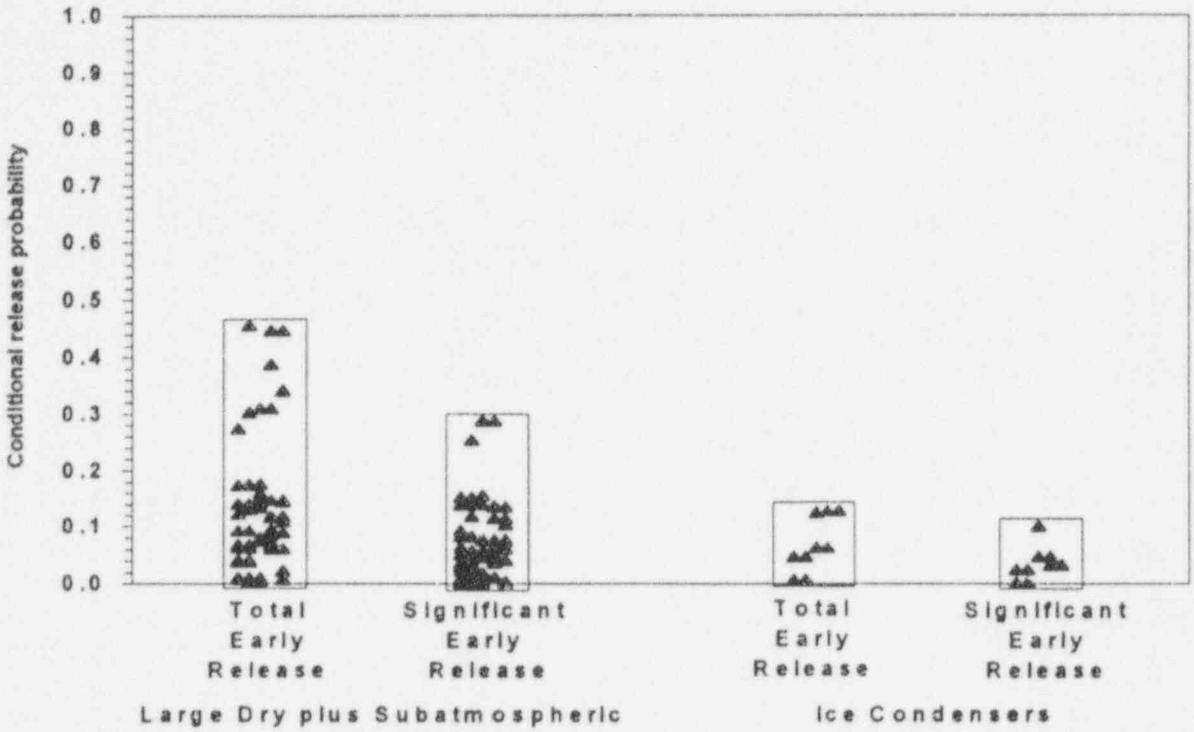

Figure 12.34 Reported IPE conditional probabilities of total and significant early release for PWR plants.

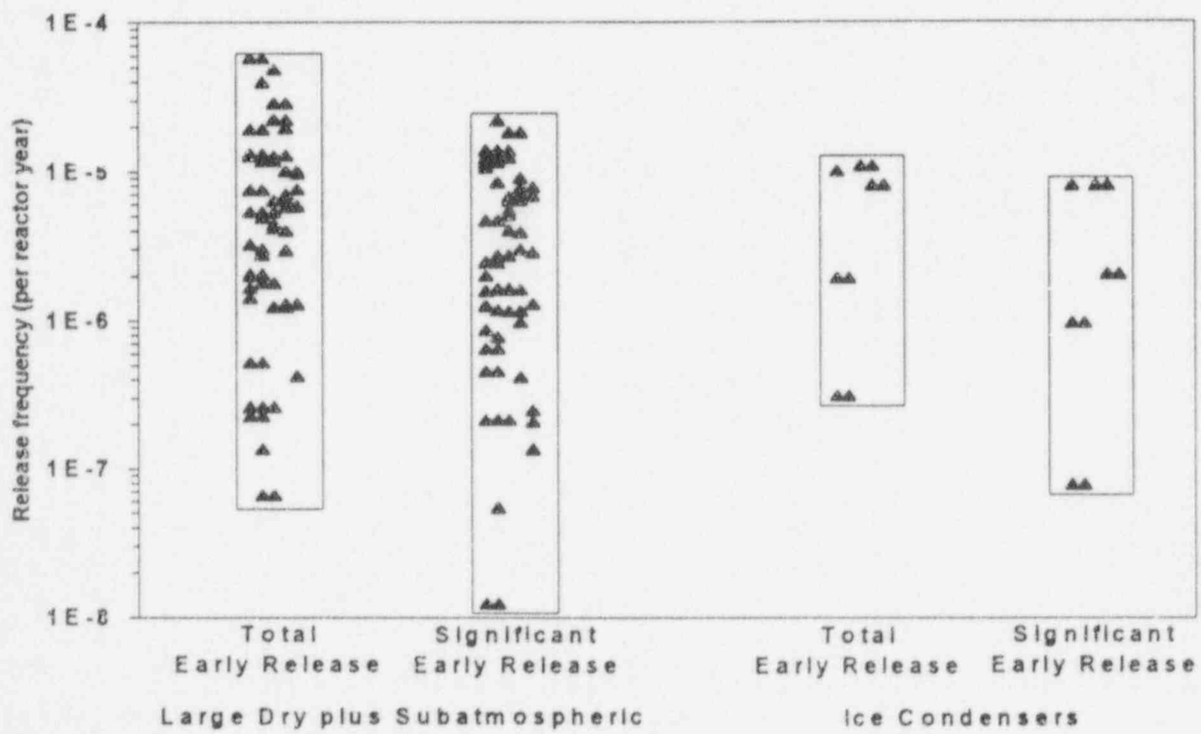

Figure 12.35 Reported IPE frequencies of total and sig. ificant early release for PWR plants.

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 (refer to Figure 12.35).

Figure 12.34 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 12.34 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 12.34 as the probabilities of significant early release for Sequoyah 1&2 and Watts Bar.

The frequency of all early release is 1E-5/ry or less for all the ice condenser IPEs. The frequencies of total early release and significant early releases for five ice condenser containments are presented in figure 12.35. The frequencies for total early releases vary from about 3E-7/ry to about 1E-5/ry. The frequencies for significant early releases vary from less than 1E-7/ry to 8E-6/ry for the five IPEs. As the figure shows, there is not a large difference between total early release and significant early release for most of the ice condenser submittals.

13. OPERATIONAL PERSPECTIVES

Chapter 5 of Part 1 summarizes the key perspectives regarding the importance of the operator's role in core damage frequency estimation and containment performance analysis. Chapter 13 provides a more in-depth discussion of these perspectives addressing the human actions consistently identified as important in the Individual Plant Examinations (IPEs), the human actions found important by only a few licensees because of plant-specific characteristics, and the influence of modeling assumptions and methodologies on the results.

Because of the role of humans in maintaining and operating nuclear power plants and in recognizing and responding to events that could lead to core damage and risk to the environment, an important aspect of the IPE program, as described in Generic Letter 88-20, 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 done for the IPEs. Incorporating human actions into the event and fault tree models and quantifying their failure probabilities can have an important impact on the resulting estimates of core damage frequency (CDF). Not surprisingly, results from the IPEs have indicated not only that human error can be a significant contributor to CDF, but that correct human action can significantly reduce overall CDF. The purpose of this chapter is to provide an overview of the HRAs done for the IPEs, to discuss factors that influence the results of HRA, to address the variability in the results of the HRAs across the different IPEs, and to identify the human actions that are significant contributors to CDF. Important general observations contained in this chapter include the following:

- In the case of both boiling water reactors (BWRs) and pressurized water reactors (PWRs), only a few human actions are consistently important. That is, while many different actions are indicated as being important, relatively few are important in most of the BWR IPEs or in most PWR IPEs.

- The only human actions important in more than 50% of the BWR IPEs are manual depressurization, containment venting, initiation of standby liquid control (SLC), and system alignment for decay heat removal. In PWRs, only switchover to recirculation, feed-and-bleed, and the actions associated with depressurization and cooldown are important in more than 50% of the IPEs.

- Many factors influence which human actions are important. They include plant characteristics, modeling details, sequence specific attributes, dependencies, HRA method and the associated performance shaping factors (PSFs) modeled, assumptions about PSFs, and the biases of both the analysts performing the HRA and the plant personnel from whom selected information and judgments are obtained.

- Examinations of the relationship between plant class differences (e.g., BWR 2s and 3s vs. BWR 5s and 6s) and which actions are important indicate that apparent trends are more likely due to modeling preferences or plant-specific differences than to vendor or design vintage.

- Because of the potential impact unrealistic human error probabilities (HEPs) can have on which accident sequences and human actions are important, the variability in the HEPs for several of the more important human actions was examined across IPEs. The results of this examination indicated that while clear reasons for all of the variability in HEPs for specific events cannot always be determined, most of the variability is due to appropriate factors such as the time available for the operator to respond, initiator and sequence modeling related influences, dependencies, and plant-specific system characteristics. However, it should be noted that 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 for that event may be similar to those obtained for other plants. 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 high quality HRAs, it is possible that licensees obtained HEP values that are not appropriate for their plant.

13.1 Approach

The present analysis examines and compares the most important human actions identified in the submittals. All 75 IPEs submitted to the U.S. Nuclear Regulatory Commission were reviewed and the identified important human actions are discussed in Sections 13.3 and 13.4 for BWRs and PWRs, respectively. The focus of the discussion is on human actions that are frequently important for either of the two different plant types (BWRs vs. PWRs). An examination was also conducted to determine the extent to which the human actions found important are related to plant class differences (e.g., BWR3 vs. BWR6) or to unique, plant-specific characteristics.

In addition, the human error probabilities (HEPs) for some of the important human actions were compared across plants in order to assess the degree of variability in the quantification of similar human actions. The degree of variability is important because of the potential impact HEP values can have on which human actions and accident sequences are important. Assigning unrealistically low failure probabilities to human actions may lead to important accident sequences being screened out, whereas the assignment of unrealistically high HEPs can lead to relatively unimportant sequences having large contributions to CDF. Thus, invalid or unexplained variability in the HEP values for similar events can be problematic. Potential causes for variability in HEPs are discussed in detail in Section 13.2 below. The extent to which the variability in HEPs appears to be reasonable is discussed in Sections 13.3 and 13.4.

Neither the methods used to identify important human actions nor the documentation is consistent across the IPEs. For example, some submittals use Fussel-Vesely or similar measures to identify important actions (and report the resulting indices), while others use a sensitivity analysis approach in which all HEPs less than 0.1 are set to 0.1 and the sequences are requantified. Selected human actions are then systematically returned to their original values and reductions in CDF are examined to determine which actions are having the greatest impact. Other submittals determine which human actions are reducing CDF by an order of magnitude and report those as the important human actions. In some cases the percent contribution to core damage is reported, while in others risk achievement worth or risk reduction values are presented. In some instances a list of important human actions is provided, but the basis for the list is not discussed. Nevertheless, most submittals attempted to provide some indication of which actions are important and the discussion below is based on what is reported in various sections of the IPE submittals.

13.2 HRA Influences and Issues

This section discusses factors which have the potential to influence the results of the HRAs performed for the IPEs and describes some of the differences in how the HRAs are performed for the different IPEs. The extent to which HEP calculations for important human actions may have been influenced by extraneous or inappropriate factors is examined separately for BWRs and PWRs in Sections 13.3 and 13.4, respectively.

Numerous factors can influence the quantification of HEPs and introduce significant variability in the resulting HEPs, even for essentially identical actions. As noted above, 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, assumptions about PSFs, 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, it can be seen that several have the potential for causing inappropriate variability. A discussion of both appropriate and inappropriate influences is presented below.

13.2.1 Plant Classes, Event Sequences, and Initiators

Due to the many similarities in the way nuclear power plants accomplish key functions, it might be expected that there would be a certain degree of commonality in the operator actions found to be important contributors to risk. However, there are a number of factors which can determine whether a human action is important. Different types of plants (e.g., BWRs vs. PWRs) can require different types of human actions because their systems are different. Plants of the same general type can generate different important human actions due to plant class differences (e.g., BWR 2 versus BWR 6) and due to unique, plant-specific characteristics. Even the importance of similar human actions in similar types of plants can vary quite dramatically as a function of the specific initiators involved and the pattern of failures and successes which occur in particular accident sequences. Thus, even though a certain degree of commonality is to be expected, it is critical to realize that valid differences can arise across plants in the human actions found important.

13.2.2 Categorization of Human Actions

The traditional approach in HRA is to separate human actions into two basic categories: pre-initiator actions and post-initiator actions. The post-initiator actions are then usually subcategorized (using various labels) as either "response actions" or "recovery actions." In the context of the PRA, pre-initiator human actions are those which, if performed incorrectly or at inopportune times, can render instrumentation or systems unavailable when they are needed to respond to an accident. These actions typically include failures in calibrating instrumentation or failures in correctly restoring systems after maintenance. Post-initiator human actions are those required in response to initiating events or related system failures. Post-initiator response-type actions are generally distinguished from recovery-type actions in that the response actions are usually explicitly directed by emergency operating procedures (EOPs). Alternatively, recovery actions may entail restoring failed or unavailable systems in time to prevent undesired consequences, using systems in relatively unusual ways, or in some cases going beyond written procedures.

In reviewing the submittals, it is found that while all of the various HRAs address pre-initiator human actions in some way, their treatment varies somewhat across plants. For example, several licensees simply dismiss the pre-initiator human action events by arguing that their failure probabilities are insignificant or that the human failure probabilities associated with such events are contained within the system unavailability data. Some licensees explicitly consider events concerned with the failure to restore systems after maintenance, but dismiss miscalibration events (or at least fail to provide any evidence that they considered them). At least one licensee dismisses all potential restoration faults on the basis of staff-developed screening criteria, but quantifies miscalibration errors. Other plants used a screening approach in which all the pre-initiator events are assigned relatively conservative failure probabilities and are only quantified explicitly if they prove to be important after initial quantification of the accident sequences. At least one licensee calculates HEP values for several general classes of pre-initiator events and applies those values to the relevant actions throughout the fault trees.

Variability is also found in the treatment of the post-initiator response and recovery type actions. While all of the examined IPEs quantify post-initiator actions in some way (e.g., screening values vs. detailed quantification), they do not always make an explicit distinction between response and recovery type actions. Failing to make this distinction makes it difficult to determine the relative importance of recovery actions in eliminating accident sequences and to determine whether recovery actions are treated appropriately from an HRA perspective.

While most of the HRAs done for the IPEs appropriately model relevant human actions and use viable systematic approaches to quantify human failure probabilities, it should be apparent that differences in categorization and the treatment of the various categories (in terms of quantifying the relevant actions) could have had important impacts on the results.

13.2.3 Methods, PSFs, and Assumptions Used in the IPEs

Another important factor that has the potential to lead to differences in results is the HRA method used to derive the HEPs. In quantifying HEPs, several of the licensees use a single HRA methodology, while others use a combination of HRA methods to address different aspects of the analysis. In general, it appears that the different methods that are used to accomplish the HRA can be grouped into six basic categories or groups of methods:

1. ***A <u>modified</u> version of the Success Likelihood Index Methodology (SLIM) (Ref. 13.1) that relies on subjective estimates of the impact of various PSFs on the operator's likelihood of failure.*** This method (often referred to as the failure likelihood index methodology by licensees using this method) explicitly addresses approximately 23 different PSFs which range from fairly common ones such as operator training and experience, procedural direction available, and relevant plant indications to relatively uncommon factors such as the impact of preceding and concurrent unrelated actions on potential operator confusion. In addition to being the only method that consistently relies directly on subjective estimates by experts to derive the HEPs for the post-initiator human actions, this method is also distinguished by the fact that the impact of time on the performance of a task is usually determined on the basis of subjective estimates as opposed to the time reliability correlations (TRCs) used by most other HRA methods. That is, judgments regarding the adequacy of the time available are solicited from experts (e.g., operators) and factored into the HEP derivation process. In contrast, in the methods using TRCs, failure probability as a function of the time available is either derived directly from an equation or simply looked up in a table or figure. In most cases, several different TRC curves can be used depending on various factors which might influence performance (e.g., degree of training for a particular scenario, quality of procedures, operator burden, etc.) A characteristic of SLIM based methodologies that can bias the calculated probabilities of failure is that the SLIM 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 result.

2. ***The Electric Power Research Institute (EPRI) cause-based decision tree method described in EPRI-TR-100259*** [13.1]. The decision tree method considers a number of potential failure mechanisms and associated performance shaping factors in determining the failure probability for the detection, diagnosis, and decision making phase of an operator action. Values from the Technique for Human Error Rate Prediction (THERP) (Ref. 13.2) are used to quantify the execution portion of the human actions. This method generally takes time into account by limiting the consideration of recovery factors (such as checking by additional crew members or the availability of the emergency response facility) to those actions where the time available exceeds some criterion. In some of the applications of this method in the

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[13.1] G. W. Parry, "An Approach to the Analysis of Operator Actions in PRA," Draft, EPRI TR-100259, Electric Power Research Institute, Palo Alto, CA.

submittals, time-critical actions are quantified using the TRCs from the Accident Sequence Evaluation Program (ASEP) (Ref. 13.3) or THERP and only non-time-critical actions are quantified using the decision tree method.

3. ***The Human Cognitive Reliability (HCR) (Ref. 13.4) method or the Operator Reliability Experiments (ORE)-based modification of the HCR method (EPRI NP-6560-L)*** [13.2]. These methods are essentially TRC methods that may also use THERP to quantify the execution portion 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 determine failure probability. The ORE method apparently no longer considers this distinction, but does incorporate data collected from 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 method described in the book by Dougherty and Fragola (Ref. 13.5) and referred to as the operator reliability characterization and assessment method.*** In one submittal that uses this method, it is stated that the method is functionally a combination of SHARP (Ref. 13.6) and HCR, and therefore may in some ways be similar to the third method. 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 part of the task will be the dominant failure mode, while mistakes are where the cognitive part of the task is less reliable. Thus, for most actions only the execution or the cognitive portion is explicitly quantified, not both. Slips are quantified using a simplified THERP method, and mistakes are quantified with a set of TRCs that vary as 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 (which is a method derived from THERP).*** These methods explicitly consider (among other PSFs) 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), a TRC is used to quantify the diagnosis portion of an operator action. Upper and lower bounds of the TRC are used to account for various performing shaping factors 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 is apparently only documented in the IPEP IPE submittals and is distinguished by its lack of emphasis on modeling the diagnosis portion of a task, while creating a PSF referred to as "slack time," which allows credit to be given 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.

[13.2] 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.

These descriptions show that while different HRA methods naturally have many commonalities, they may also consider different factors in deriving HEPs. The different methods may incorporate the effects of time in different ways, they may vary in terms of which and how many PSFs are considered, and they may vary in the degree to which recovery from error is analyzed and credited. 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 "bench-marked" to some degree in order to assess the values they produce relative to other methods, it is clear that the potential for method-based differences does exist.

The fact that different methods have the potential for producing 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 those variables are systematically and appropriately addressed, realistic and consistent HEPs can be expected across different methods. While it might seem reasonable to compare HEPs for similar actions across the different methods used in the submittals in order to determine the impact of method per se, the results of such a comparison can be very misleading. For example, as noted above, analyst bias can influence resulting HEPs, as can plant characteristics, sequence related factors, and modeling details. In addition, given the sparseness of the documentation of many of the submittals regarding the information that will need to be obtained, a valid comparison regarding the specific influence of HRA method will be difficult to achieve. Suffice it to say that at least some "unexplained variability" in resulting HEPs can be created as function of the differences in HRA methods and the ways in which they are applied. The potential impact of methodological effects on HRA results will be addressed in an upcoming NUREG report.

In addition to the basic HRA methodology and the associated PSFs used to quantify the post-initiator HEPs, there are a number of other factors related to how the analysis is conducted that can have an impact on the results. Many of these factors may or may not have a direct impact on the derivation of HEPs, but may reflect on the nature and extensiveness of the analysis performed for the HRA or on how the HRA is incorporated into the PRA. Thus, their influence of these factors can be quantitative, qualitative, or both. Several of these factors and their treatment in the submittals 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 emergency operating procedures 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, to realistically quantify the human potential for failure or success, context effects and dependencies across a given accident sequence should be considered. While most of the licensees clearly consider context and dependencies in analyzing post-initiator actions, some do not. Some plants analyze operator actions only to the extent needed to determine the conditions that will 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 "conservative" value is assigned in all cases where the event occurred. Other licensees address context only in cases where extreme differences in HEPs are expected, and several either fail to consider context or dependency at all, or at least fail to document that they have done so. Obviously, these types of decisions can lead to variability in quantifying HEPs and in the knowledge gained about the importance of operator actions in the IPE.

Other factors having a potential impact on the results of the HRA include whether the analysts conducted simulator exercises to assess the performance of the control room crews in responding to important accident sequences and whether the analysts performed walk-throughs 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 HRAs evaluated simulator exercises and conducted walk-throughs of specific actions and others did not, differences in results might be expected.

13.2.4 Summary

To summarize this section, many factors can influence the results of HRAs. The factors include modeling assumptions, the variables considered in deriving HEPs, and the steps taken by the analysts to verify assumptions and judgements. Although most of these factors introduce appropriate variability in result, it can be seen that several have the potential for causing inappropriate or artifactual variability.

13.3 Important Human Actions For BWRs

This section identifies and summarizes the human actions important in the BWR submittals. Actions important in a relatively high percentage of the submittals are discussed in the first subsection, followed by an examination of the impact specific plant classes, and unique plant designs and characteristics have on what is important. As in other sections of this report, BWR 1s, 2s and 3s with isolation condensers form one class, with the remaining BWR 3s and BWR 4s constituting a second, and BWR 5s and 6s making up a third. Of the 27 BWR IPEs reviewed (covering 35 units), five are in the BWR 1/2/3 class (covering six units), 15 were in the BWR 3/4 class (covering 21 units), and seven were in the BWR 5/6 class (covering eight units). In the final subsection, the HEPs for selected human actions are examined to determine the extent to which variability in HEPs across the BWR IPEs appears reasonable.

13.3.1 Human Actions Generally Important for BWRs

A list of the most important human actions identified in a review of all 27 BWR IPEs submittals is presented in Table 13.1. The table lists the human action event, the accident sequences in which the event is important, the percentage of all BWR units finding the event important, and the percentage of units finding the event important as a function of BWR class.

Only a few human actions are important in a high percentage of the BWRs. The most frequently identified action is manual depressurization of the vessel, which is critical when high pressure injection systems are lost or unavailable. Several anticipated transient without scram (ATWS) related human actions are also important, which is not surprising given that ATWS sequences may require several human actions for success. Decay heat removal (DHR)-related actions, recovery of ultimate heat sink sources, and use of alternate injection sources are also important to many BWRs. Plant characteristics, modeling assumptions, and boundary conditions impact the importance of the sequences where these human actions appeared, and thus affect the importance of human errors to overall CDF. Modeling assumptions and boundary conditions regarding critical phenomena and time limits for successfully performing human actions can also directly impact human error rates. A discussion of the important human actions is provided after Table 13.1.

Table 13.1 Important human actions, accident sequences, and percentage of BWRs finding the action important.

Important human actions	Accident sequences	Percentage of BWRs finding the action important
Manual depressurization	Transients, small and medium LOCAs, SBO (particularly for use of firewater pump for injection), any sequences where high pressure injection has failed.	All BWRs ≈ 80% BWR 2/3 ≈ 80% BWR 3/4 ≈ 80% BWR 5/6 ≈ 60%
Containment venting	DHR sequences in transients and LOCAs	All BWRs ≈ 55% BWR 2/3 ≈ 35% BWR 3/4 ≈ 60% BWR 5/6 ≈ 60%
Align containment cooling/ suppression pool cooling	DHR sequences in transients and LOCAs	All BWRs ≈ 55% BWR 2/3 ≈ 70% BWR 3/4 ≈ 50% BWR 5/6 ≈ 50%
Initiate SLC	ATWS	All BWRs ≈ 50% BWR 2/3 ≈ 70% BWR 3/4 ≈ 50% BWR 5/6 ≈ 40%
Level control in ATWS	ATWS	All BWRs ≈ 25% BWR 2/3 ≈ 50% BWR 3/4 ≈ 30% BWR 5/6 ≈ 0%
Recover ultimate heat sink events (e.g., service water (SW))	All	All BWRs ≈ 20% BWR 2/3 ≈ 20% BWR 3/4 ≈ 20% BWR 5/6 ≈ 25%
Align/initiate alternate injection source (e.g., firewater, safe shutdown makeup pump, standby feedwater system, control rod drive (CRD), feed booster pumps, suppression pool cleanup)	Transients, LOCAs, SBO	All BWRs ≈ 25% BWR 2/3 ≈ 30% BWR 3/4 ≈ 30% BWR 5/6 ≈ 15%
Inhibit automatic depressurization system (ADS) - While only a small percentage of plants listed this event as important, some felt they would go to core damage if this action failed.	ATWS	All BWRs ≈ 20% BWR 2/3 ≈ 20% BWR 3/4 ≈ 20% BWR 5/6 ≈ 25%
Miscalibration of pressure switches (pre-initiator event, in one case for automatic isolation condenser (IC) initiation)	Transients, LOCAs	All BWRs ≈ 15% BWR 2/3 ≈ 20% BWR 3/4 ≈ 15% BWR 5/6 ≈ 10%
Initiation of isolation or emergency condenser makeup	Transients, SBO	All BWRs -- N/A BWR 2/3 ≈ 85% BWR 3/4 -- N/A BWR 5/6 -- N/A

Table 13.1 Important human actions, accident sequences, and percentage of BWRs finding the action important.

Important human actions	Accident sequences	Percentage of BWRs finding the action important
Control feedwater events (e.g., after loss of instrument air (IA)	Transients, small LOCAs	All BWRs ≈ 15% BWR 2/3 ≈ 15% BWR 3/4 ≈ 20% BWR 5/6 ≈ 15%
Manual initiation of core spray or other low pressure system	Transients (e.g., loss of high-pressure injection with ADS success, followed by low pressure failure), LOCAs	All BWRs ≈ 15% BWR 2/3 ≈ 20% BWR 3/4 ≈ 20% BWR 5/6 ≈ 0%
Miscalibration (or failure to restore after test) of low pressure core spray (LPCS) permissive (pre-initiator event)	Transients, LOCAs	All BWRs ≈ 10% BWR 2/3 ≈ 20% BWR 3/4 ≈ 15% BWR 5/6 ≈ 0%
Loss of heating, ventilation, and air conditioning (HVAC) related events (e.g., provide alternate room cooling)	Transients (Loss of HVAC)	All BWRs ≈ 10% BWR 2/3 ≈ 0% BWR 3/4 ≈ 5% BWR 5/6 ≈ 25%
Recovery of injection systems	Transients, LOCAs	All BWRs ≈ 10% BWR 2/3 ≈ 0% BWR 3/4 ≈ 15% BWR 5/6 ≈ 15%
DC load shedding after SBO. Some may have assumed load shedding successful and didn't model. Others may have just assumed a standard battery time.	SBO	All BWRs ≈ 5% BWR 2/3 ≈ 20% BWR 3/4 ≈ 0% BWR 5/6 ≈ 15%
Prevent isolation or loss of isolation (or emergency) condenser (e.g., on high level or before safety relief valves (SRVs) open) following SBO.	Transients, SBO	All BWRs -- N/A BWR 2/3 ≈ 35% BWR 3/4 -- N/A BWR 5/6 -- N/A

Only a few human actions are important in a high percentage of the BWR submittals. That is, while many different events are indicated as being important, relatively few are important to most of the licensees. Thus, an attempt is made to group some of the operator actions according to the function to be accomplished. For example, several licensees find events related to aligning an alternate injection source during transients, loss-of-coolant accidents (LOCAs), and station blackouts (SBOs) to be important. Even though the alternate systems to be used range 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 important for BWRs, these actions with similar functions are grouped and are presented in Table 13.1 along with the other important individual operator actions. It should be noted that not every action identified as important in the submittals is represented in Table 13.1. Some actions which are important occur in single plants and cannot easily be grouped according to function. Some of these actions are addressed in the section below which discusses the impact of unique plant characteristics on HRA.

Manual depressurization of the vessel[13.3] ***is important in most of the BWR IPEs.*** The operator action found important in the greatest number of BWR submittals is the action to manually depressurize the vessel so that low pressure injection systems can be used after a loss or unavailability of high-pressure injection systems. These types of sequences are commonly referred to as "TQUX" sequences in PRA terminology and are important in most BWR IPEs. In addition, 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 due to the fact that most plant operators are directed by the plant EOPs to inhibit automatic actuation of ADS. Thus, operators must manually depressurize the vessel when injection from low pressure systems is required to cool the core. Given the importance of the action after ADS is inhibited, one licensee proposes that inhibition of ADS should constitute a simple resetting of the 2 minute delay that exists in the ADS actuation logic rather than an override of the ADS function. In fact, this was the general means for inhibiting ADS prior to instituting the guidance in the Emergency Planning Guide Revision 4 for inhibiting ADS. The percentage of total CDF accounted for by cutsets including this event ranges from approximately one percent to 45%

While ATWS-related human actions are frequently found in the licensees' list of top ten important events, the contribution of ATWS events to overall CDF is usually relatively small. The human action to inhibit ADS[13.4] is important in the ATWS sequences of several plants. In fact, some plants assume that because of the instabilities created under low-pressure conditions during an ATWS, core damage will occur if the operators fail to inhibit ADS. Given this position, it is somewhat surprising to find that only about 20% of the BWR submittals identify inhibition of ADS as being important. The low percentage is partly due to how inhibition of ADS is modeled. Many plants assume that failure to perform this action has a very low probability. Other plants model the failure to inhibit ADS as only resulting in core damage 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 have the effect of reducing the importance of this type of accident sequence, and thus the importance of the related human errors. The remaining plants model the failure to inhibit ADS during an ATWS as resulting directly in core damage. This human error is noted as being important for approximately 50% of the plants that model inhibition of ADS in any fashion. A comparison of the HEPs for this event between plants that find inhibition of ADS important and those that do not fails to reveal any trends. Most of the plants that fail to find the event important had HEPs for the action similar to those finding it important. However, even for the plants finding this human action important, its contribution to CDF is usually relatively small due to the relatively small contribution from ATWS overall for BWRs.

Two other ATWS-related events are found important in several of the submittals reviewed. The operator action to initiate boron injection during an ATWS[13.5] is important in ~50% of the BWRs and ~25% identify level control as being important. As with ADS inhibit, the modeling of these events partly impacts their importance to core damage. For example, early SLC initiation is modeled in some submittals 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 temperature and are somewhat variable, ranging from 1 minute up to 45 minutes. Some licensees take credit for alternate means of injecting boron and others take credit for level control as a means of reducing core power to acceptable levels following SLC failure. All these variables can contribute to the importance of the failure to manually initiate SLC. At least some licensees think that the need to bring vessel level to the top of active fuel

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[13.3] The variability in HEPs for this event across the BWR IPEs is discussed in Section 13.3.4.1.

[13.4] The variability in HEPs for this event across the BWR IPEs is discussed in Section 13.3.4.2.

[13.5] The variability in HEPs for this event across the BWR IPEs is discussed in Section 13.3.4.3.

or to prevent overfilling of the vessel will be critical. Modeling of level control is highly variable, ranging from not modeling it to modeling level reduction to achieve power reduction in scenarios with or without SLC operation. Due to a large turbine bypass capacity, some plants do not require the additional power reduction associated with level reduction if SLC is operating and if the main condenser is available. Thus level reduction is only modeled in these plants if the main condenser is unavailable. Some plants also model failure to increase the vessel level to promote boron mixing and/or raising the level too high (thereby flushing injected boron from the core).

The modeling of SLC and level control is also highly related in some submittals. For example, one licensee assumes that if an operator fails to initiate SLC, he will also fail to control level. Whether or not these actions are important for particular plants is also to some extent a function of the contribution of the ATWS sequences to overall CDF. The contribution to CDF for these events is usually in the one to three percent range. As with inhibiting ADS, an examination of the HEPs for initiating SLC and controlling level in an ATWS fails to indicate that the HEP value itself plays a dominating role in determining whether or not the event turns out to be important. In fact, the plants having the three lowest HEPs for initiating SLC still find the event to be an important contributor to ATWS sequences. On the other hand, as is discussed later in this report, BWR 5/6s show a slight trend toward having lower HEP values for the initiation of SLC than do the other plant types, and the 5/6s are less likely to find the initiation of SLC to be important.

DHR related human actions are identified in many IPEs as being important. In fact, two of the most frequently identified important actions in BWRs are actions related to DHR sequences in transients and LOCAs. With a loss of the power conversion system and 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 cannot be provided, are found important in over 50% of the submittals. Plant characteristics and modeling differences are important factors in determining the impact of these human actions. For example, plants require actuation of DHR before some adverse conditions are reached. These conditions can range from a high suppression pool temperature that results in loss of emergency core cooling system (ECCS) pumps to 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). For some plants, failure of DHR is modeled as not failing the ability to inject water into the vessel from ECCS or alternate injection systems. For other plants, the steam released following containment failure is identified as having a negative impact on the operability of injection systems. Since some plants 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. Contribution to CDF for these events generally ranges from one to five percent, with one plant indicating a contribution of approximately ten percent.

Alignment/initiation of an alternate coolant injection source is found important in ~25% of the plants reviewed. As noted above, the need for an alternate injection system is relevant to loss of injection sequences (primarily loss of all normal low- and high-pressure systems) and loss of DHR sequences. Different plants have different sources for alternate injection, including service water, firewater, CRD, a unique standby feedwater system, a safe shutdown makeup pump, feed booster pumps, and suppression pool cleanup. Some licensees credit alternate high-pressure injection systems such as CRD but most only credit low-pressure systems such as service water. In addition, alternate injection systems are generally credited only in long-term scenarios when sufficient time is available to perform required alignments and/or when the required coolant injection flow is within the injection system capacity. However, some plants credit alternate systems in the short term when the capacities are within make-up requirements and when the alignment is proceduralized and can be accomplished from the control room. Contribution to CDF for these events tend to range from approximately one to four percent.

Recovery of service water or some other ultimate heat sink is found important. Another action emerging as important in multiple BWRs (~20%) is the action to recover service water or some other ultimate heat sink (e.g., recover lake water screen house intake, start standby cooling tower fans, or start emergency cooling water after loss of general service water). The importance of cooling water systems is variable due to 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 important, these events are usually relevant to multiple classes of accident sequences. This is due to the need for heat removal from components and equipment needed to mitigate all types of accident scenarios. These events are particularly important for loss of DHR scenarios since cooling water systems are required for transporting heat to the ultimate heat sink as well as for cooling equipment such as coolant injection pumps. In general, the data provided on these events is insufficient to assess their contribution to CDF.

Other interesting human actions are found important even at lower frequency. While the remaining human actions listed in Table 13.1 are not found important to a large number of BWRs, several have particular interest. Actions related to the initiation of mak for isolation or emergency condensers are obviously only relevant to BWRs that have these systems. Four of the five BWRs in this group find this event to be important. One plant lists the contribution to CDF as being approximately 14%, while the only other plant providing a value indicates an approximate one percent contribution. While Nine Mile Point 1 is the only plant in this group for which makeup to the emergency condenser (EC) is apparently not critical, the need to prevent isolation of the EC or recover from isolation of the EC is important for Nine Mile Point 1, thus supporting the overall importance of the EC in these types of plants.

One action that is interesting in the sense that only a couple of plants find it to be important, is the action to perform DC load shedding after an SBO. Station blackout scenarios are the dominant accident sequence type for BWRs and are highly influenced by the availability of DC power to operate AC-independent systems such as RCIC and to maintain SRVs open to allow low pressure vessel injection from a diesel-driven firewater pump. Failure to shed unnecessary DC loads reduces the time essential DC power will be available for critical component operation and thus reduces the time to core damage (which determines the time to recover offsite power so that other mitigating systems can be utilized). Most plants simply do not model this event, apparently using either the SBO coping time of one to four hours or a battery depletion time calculated assuming load shedding is successful.

Several actions appearing as important events in some plants are pre-initiator human actions. All of the actions discussed above are post-initiator human actions. However, several actions appearing as important events in some plants were pre-initiator human actions. Over all the BWR submittals reviewed, miscalibrations of various instruments are found important in ~20% of the units. Moreover, 15% of the licensees find that failures to restore certain systems after test or maintenance are important. The most important events related to instrumentation miscalibration (or failure to restore instruments after test) are those events involved with the calibration of various pressure switches. Examples include miscalibration of the vessel low pressure permissive required to open the LPCS and low pressure coolant injection (LPCI) valves (~15% of the submittals) and miscalibration of pressure switches that allow auto-initiation of an IC. Contribution to CDF for these types of events is as high as six percent in several cases. Failures to restore critical systems include events such as failure to correctly align SLC valves after test, failure to restore service water after test, and failure to restore the high pressure core spray after test or maintenance. The fact that these events are not important for more plants may in part be attributable to the apparent failure of some plants 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 will also impact the importance of some of these events.

13.3.2 Relationship Between BWR Class and Important Human Actions

The relationship between BWR class and important human actions is examined to see if particular events tend to be found important only for particular classes of BWRs. While some differences in important human actions appear to correlate with BWR type and are discussed below, it must be kept in mind that larger populations of the different types of BWRs will be necessary before the differences can be taken seriously. Moreover, it is believed that most of the differences are related to modeling or plant-specific differences and not to differences in general BWR vintage designs.

The third column of Table 13.1 presents the percentage of the different classes of BWRs which find a particular human action to be important. The reason for extracting this information is to assess whether there are any interesting relationships between the classes of BWRs and the events found to be important. That is, are certain events found important for some classes of plants but not others? A review of Table 13.1 suggests several instances where 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 related to modeling or plant-specific differences and not to differences in general BWR vintage designs. It should also be kept in mind that the population sizes of the different classes of BWRs are quite small and that apparent differences may not necessarily represent any real differences.

There is some suggestion that ATWS-related events are more often important for the older plants (BWR 1/2/3s and BWR 3/4s as opposed to BWR 5/6s). The initiation of SLC is found important in ~70% of the BWR 1/2/3s, but only shows up as important in ~40% of the BWR 5/6s. Similarly, level control during an ATWS is important to ~50% and ~30% of BWR 1/2/3s and 3/4s respectively, but fails to show up as important in any of the seven BWR 5/6s reviewed. While there are no apparent generic differences in the BWR vintages that account for these differences, there are some plant-specific design differences that may partially account for the variability. For example, one BWR 5 and two BWR 4s have auto-actuated SLC systems. Failure to initiate SLC does not appear as an important human error for these plants, partly explaining why fewer BWR 3/4 and 5/6 plants than BWR 1/2/3s list this human error as important. In addition, as is discussed in more detail in Section 13.3.3, the HEP values for initiating SLC tend to be slightly lower for BWR 5/6s than for the other types of plants and may impact importance.

The turbine bypass capacity is an example of a plant-specific feature that impacts the percentage of plants reporting level control as an important human error. At least one BWR 2 with a turbine bypass capacity of 40%, and one BWR 3 with a turbine bypass capacity of 105% model level control only when the main condenser path is not available. Considering only this smaller set of ATWS scenarios for level control effectively reduces the importance of failing to control level. Other BWR 1/2/3s with 40% turbine bypass capacities model level control regardless of the availability of the turbine bypass path to the condenser. One BWR 6 has only a 10% turbine bypass capacity. Level control is assumed not to be required at this plant if SLC is successful and is not credited if SLC fails, since level control does not reduce core power to within the combined capacities of the turbine bypass and the residual heat removal (RHR) system.

While there is some slight suggestion that pre-initiator miscalibration events are more important to BWR 1/2/3s and that events related to providing alternate room cooling are more important to BWR 5/6s, additional data is necessary before taking the differences seriously. The miscalibration differences might be related to the fact that BWR 1/2/3 miscalibrations include IC actuation sensor miscalibrations, which are only applicable for that vintage.

The final trend evident from Table 13.1 is that the BWR 1/2/3s find events related to the successful operation of the isolation or emergency condensers important. Obviously, these events will not be relevant to BWR 3/4s and 5/6s that do not have these systems.

13.3.3 Human Actions Important at Selected BWRs

Table 13.2 presents a list of important human actions that are iden.ified as being unique because they are in general important for one plant only and/or because they are apparently related to unique plant designs or characteristics. A discussion of several of these human errors and why they are unique to a plant is presented below.

Table 13.2 Human actions found important at individual BWRs.

Plants	Unique important human action	Accident sequences
Big Rock Point, BWR1	Post-Incident recirculation (switchover to recirculation)	DHR
Big Rock Point, BWR1	Trip condensate pumps on low hotwell level. Prevents loss of pumps so they can be used after refill of the hotwell.	LOCAs with core spray failure
Big Rock Point, BWR1	Align firewater system for makeup to the hotwell. Refill of hotwell permits continued use of condensate and feedwater as injection source	LOCAs with core spray failure
Nine Mile Point 1, BWR2, MKI	Recovery of screen house intake (loss of lake intake)	DHR
Nine Mile Point 1, BWR2, MKI	DC load shedding (also see Clinton)	SBO
Nine Mile Point 1, BWR2, MKI	Load shedding emergency diesel in LOCA conditions and SBO	SBO
Millstone, BWR3, MKI	Initiate IC before SRVs open following SBO. Avoids problems with stuck- open relief valves (SORVs).	SBO
Monticello, BWR3, MKI	Alignment of bottled nitrogen to SRVs or restoration of MCC41 on loss of offsite power	Loss of offsite power (LOOP)
Pilgrim, BWR3, MKI	With both DC buses lost, if operator follows the loss of DC power procedure, then both feed pumps will be lost.	Loss of DC power
Pilgrim, BWR3, MKI	4160 volt bus breaker maintenance errors which will prevent breakers from changing state, resulting in loss of feedwater, low pressure injection, DHR, etc.	Loss of containment heat removal, LOCAs
Quad Cities 1&2, BWR3, MKI	Initiate safe shutdown makeup pump as a backup to RCIC for fire scenarios and as an alternate injection source with suction from the condenser storage tank (CST) or firewater system. Has redundant electrical supplies from each unit.	Transients, Small LOCAs
Fermi 2, BWR4, MKI	Manually initiate unique standby feedwater system	Transients, LOCAs, ATWS - any sequences needing level control

Table 13.2 Human actions found important at individual BWRs

Plants	Unique important human action	Accident sequences
Perry, BWR6, MKIII	Align feed booster pump or suppression pool cleanup for alternate injection	Transients (LOOP, loss of IA) for late injection (post-core damage)
Perry, BWR6, MKIII	Reopen motor feed pump control valves and manually depressurize (to get injection during ATWS, possibly using condensate from the condenser hotwell -- do not take credit for HPCS)	ATWS
Clinton, BWR6, MKIII	DC load shedding (also see Nine Mile Point 1)	SBO
River Bend, BWR6, MKI	Operator fails to start standby cooling tower fans	Transients, DHR

Millstone 1 is the only IC plant to identify a human error in failure to manually actuate the IC before SRVs open. Successful IC actuation before SRV opening eliminates the potential for stuck-open SRVs that defeat the use of ICs and lead to the need for another source of coolant injection. Auto-actuation of the IC is also modeled and occurs at a setpoint below the first SRV setpoint. It appears that the other IC plants do not model manual actuation. There is no apparent reason why this human action is modeled only by Millstone 1 and would not be applicable to other IC plants. By taking credit for it, Millstone 1 effectively reduces the occurrence of transients with stuck-open relief valves for which the ICs cannot 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 is only applicable to that plant. Big Rock Point is like a PWR in that it is housed in a dry containment and, during a LOCA, requires a switchover from a coolant injection mode to a coolant recirculation mode. DHR 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. Failure to provide the heat removal is not a concern since the heat can be removed from the containment by passive conduction through the containment structure.

The licensee for Big Rock Point also takes credit for condensate injection for some LOCAs, as did other plants. However, Big Rock Point is the only plant to identify a human error in failure to prevent cycling of the condensate pump breaker from cycling condenser hotwell level signals 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 believed not to be unique to the Big Rock Point plant and can be modeled in other plant IPEs. A human error in failing to replenish the condenser hotwell with firewater is also uniquely identified in the Big Rock Point submittal. It is not known if other plants have the capability to provide condenser hotwell makeup using a firewater or similar system.

As can be seen in Table 13.2, many of the remaining unique human actions are related to actuation of alternate injection systems unique to the plant. The actions required to initiate these systems, the time available to perform the actions, and the circumstances where the system can be used are all plant-specific and therefore result in variable importance for these human errors across the different IPEs.

13.3.4 Examples of Variability in BWR HEPs

In order to examine the variability in HRA results from the submittals and to assess the extent to which variability in results is due to real versus artifactual differences, the HEPs from several of the more important human actions appearing in the submitta's are examined across plants. It should be noted that many of the plants may have had multiple values for a given human action because they consider context and dependency effects, while other plants may have had only a single value or, for various reasons, no value at all. For example, not all BWRs model the failure to inhibit ADS.

13.3.4.1 Failure to Depressurize During Transients

The HEPs for failure to depressurize the vessel during transients are presented in Figure 13.1 for the various submittals.

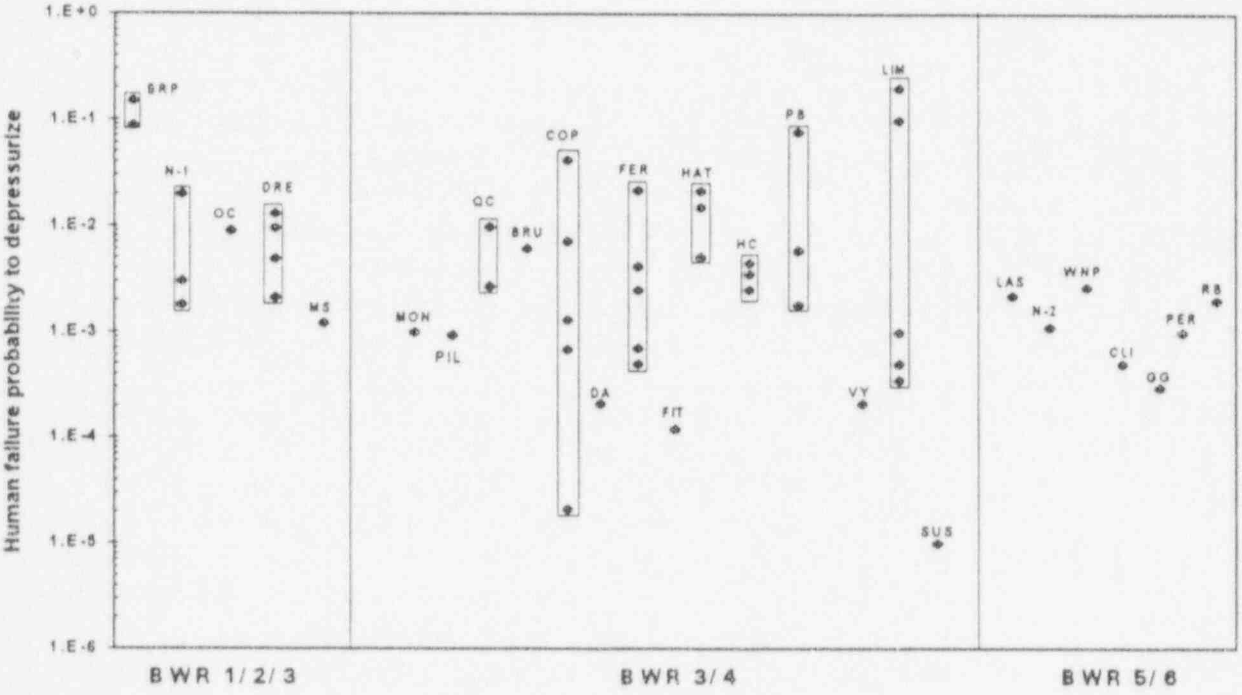

Figure 13.1 HEPs for failure to depressurize by BWR class.

As can be seen from the figure, a relatively large degree of variability exists across the submittals for this event and a significant portion of the variability is difficult to explain. Generally there appear to be reasonable explanations for the HEPs on the high end of the continuum (e.g., the high failure probabilities for Big Rock Point (BRP), Nine Mile Point 1 (N-1), Cooper (COP), Peach Bottom 2&3 (PB), and Limerick 1&2 (LIM)). The high value for Nine Mile Point 1 (N-1) involves depressurization using MSIVs and the condenser, which apparently is not typically modeled. The high value for Peach Bottom 2&3 (PB) and the next to the highest value for Limerick 1&2 (LIM) are for the case when a controlled depressurization is needed to allow use of the condensate system. The highest value for Limerick 1&2 (LIM) is for a recovery of a failed automatic depressurization. While the justification for the high values for Big Rock Point (BRP) is not apparent, the plant is unique relative to the other BWRs in that it has some PWR characteristics. The reason for the high value for Cooper (COP) is not obvious either, but the large range of values for Cooper (COP) is apparently related to the number of SRVs to be used for depressurization. The reasons for the approximately 1.5 to 2 orders of magnitude difference between the HEP values in the middle range are less obvious. As near as could be determined, the variability in the middle range of values appears to be related 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 derive multiple values in order to take specific conditions into account. The specific conditions included LOOP, SBO, loss of DC power, use of turbine bypass valves for depressurization, and loss of feedwater and standby feedwater. Most submittals also have unique values for depressurization during small and intermediate LOCAs and for ATWS scenarios, but those values are not examined since the failure to depressurize is mainly important to transients.

The reasons for the relatively low HEP values (i.e., Cooper (COP), Duane Arnold (DA), Fitzpatrick (FIT), Vermont Yankee (VY), and Susquehanna 1&2 (SUS)) are also not clear. It can be argued that at least the top three or four values from these submittals fall within an acceptable range and there may be plant-specific characteristics that support the HEPs on the lower end of the continuum.

13.3.4.2 Initiation of SLC During ATWS in BWRs

The next specific action examined is the operator action to initiate SLC or add boron during an ATWS in BWRs. As can be seen in Figure 13.2, a large range of values is found for the initiation of SLC during an ATWS. For the 27 BWR submittals reviewed, the lowest and highest values differ by more than three orders of magnitude. At least some of the variation in the HEPs can be attributed to the fact that one of the plants has an automatic initiation of SLC and the operator action is a recovery of this failure by manual initiation. The recovery HEP is relatively higher than most of the other values derived for the initiation of SLC. An important contributor to the differences is that some analyses gave credit for initiation of SLC both early and late. In all cases, early initiation of SLC is determined to have a lower failure probability than late initiation, usually with at least an order of magnitude difference. The assumption appears to be that if the operators fail early, they will also tend to fail late.

Another factor having an impact on the HEP values for initiation of SLC is whether the condenser is assumed to be available. With the condenser available, more time is allowed for initiation of SLC and therefore lower failure probabilities are obtained. Nevertheless, even when such factors are taken into account, there is still more than an order of magnitude difference between the lowest and highest values across all submittals.

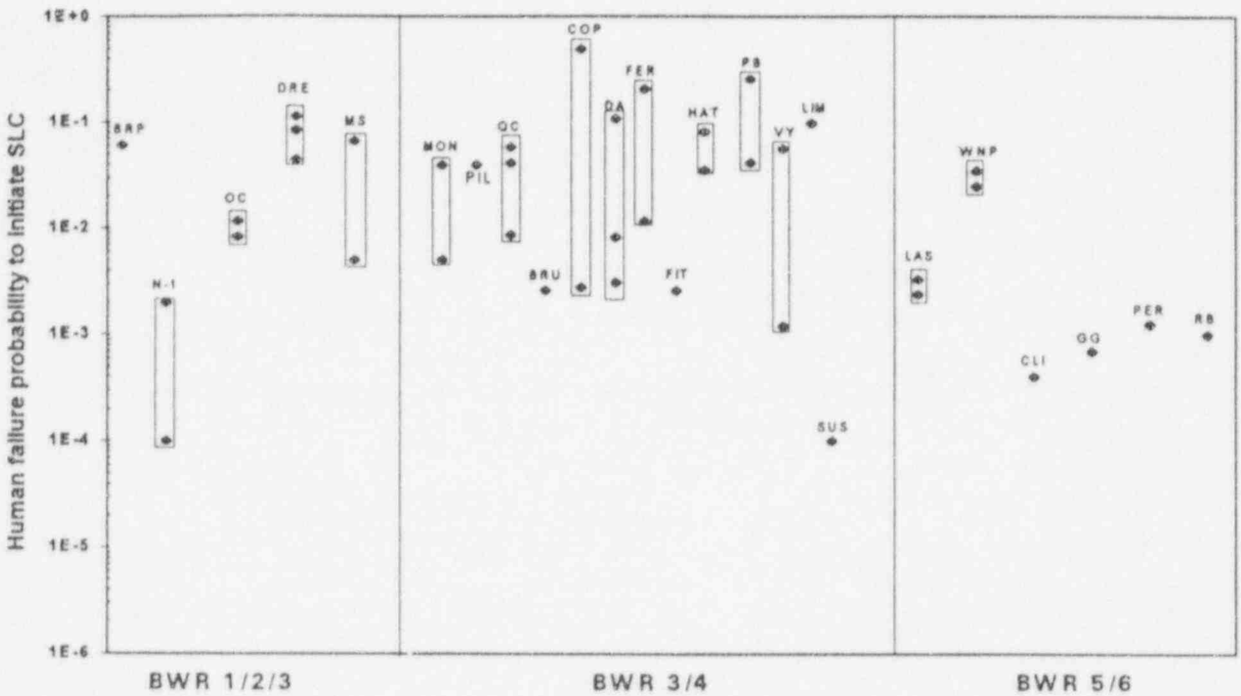

Figure 13.2 HEPs for initiation of SLC by BWR class.

Figure 13.2 displays the HEPs for the initiation of SLC as a function of the different types of BWRs. When the differences are examined in this way, it appears that some of the variation may be related to BWR class. In particular, with the exception of the two outliers from Nine Mile Point 1 (N-1) and Susquehanna (SUS), the HEPs for BWR 5/6s tend to be lower than those for the other plant classes. Some of the more extreme high failure probabilities obtained for the BWR 3/4s are related to the relatively high failure probabilities derived for initiating SLC late or for initiating SLC when the condenser is unavailable. For example, the high HEPs for Cooper (COP) and Fermi 2 (FER) are values for initiating SLC late, and the highest values for Dresden 2&3 (DRE), Millstone 1 (MS), Monticello (MON), Duane Arnold (DA), Hatch 1&2 (HAT), Peach Bottom 2&3 (PB), Vermont Yankee (VY), and WNP 2 (WNP) are for scenarios when the condenser is unavailable. The high value for Limerick 1&2 (LIM) is the recovery value for failure of the automatic initiation of SLC and the second highest value for WNP 2 (WNP) is for a unique case where degraded instrumentation is modeled. In any case, when the more extreme values are ignored, there seems to be a trend for the HEPs to decrease linearly across BWR class. A reason for a downward trend is not obvious.

Regardless, when the HEPs for initiating SLC are examined within BWR class and factors such as condenser availability and early vs. late initiation of SLC are considered, reasonably grouped HEP values (approximately within an order of magnitude of each other) are found. Given the numerous HRA factors noted above which can influence the quantification of HEP values, "unexplained" differences in the range of an order of magnitude do not seem excessive. However, there are no obvious reasons for the relatively low SLC values found for Nine Mile Point 1 (N-1) and Susquehanna 1&2 (SUS).

13.3.4.3 Inhibition of ADS in BWRs

Another specific human action important to the ATWS scenarios in BWRs is the action designated in many of the plant emergency procedures to inhibit ADS. Inhibition of ADS is indicated by the emergency procedures to help avoid activation of low-pressure injection during an ATWS, which could increase reactivity. One reason this action is interesting is that apparently some licensees assume that they will go to core damage during an ATWS if ADS is not inhibited. Others assume this is not the case -- that an ATWS can still be mitigated, and that inhibition of ADS can lead to problems in other scenarios if the operators fail to depressurize. As can be seen in Figure 13.3, there are some fairly wide variations in the HEPs for failing to inhibit ADS.

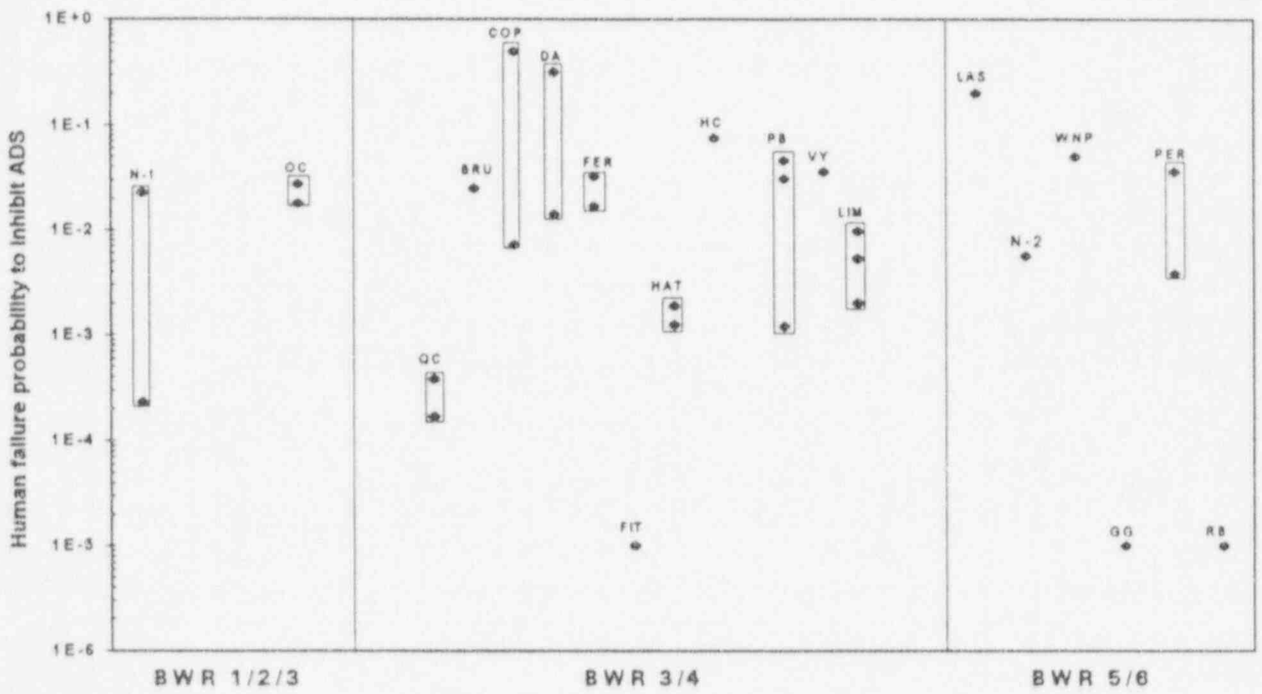

Figure 13.3 HEPS for inhibition of ADS by BWR class.

Much of the difference seems to be caused by apparent outliers on both ends of the distribution. Two of the extreme values (Cooper (COP) and Duane Arnold (DA)), on the high failure probability end of the distribution are related to ATWS events in which no high-pressure makeup is available. Of the other two values on the high end (Hope Creek 1 (HC) and LaSalle 1&2 (LAS)), one is a screening value and the other is from a plant with automatic initiation of SLC, which may be assumed to have an impact on the likelihood of inhibiting ADS. Of the six lowest values (Nine Mile Point 1 (N-1), Quad Cities 1&2 (QC) (2 values), Fitzpatrick (FIT), Grand Gulf (GG), and River Bend (RB)), the three lowest are derived by associated analysis teams. The consistently low values for these three IPEs may reflect a commonality across the three plants in factors that support a high likelihood of inhibiting ADS, but can also reflect a strong belief on the part of the analysts that operators will successfully inhibit ADS. It is difficult to determine which is the most likely case on the basis of the information in the submittals. The analysts for Nine Mile Point 1 (N-1) and Quad Cities 1&2 (QC) may share the same belief or may also have identified factors that support a high likelihood of inhibiting ADS. The higher of the two values for Nine Mile Point 1 (N-1) is for an ATWS with a loss of feedwater, and only one minute is assumed available before the ADS set point will be reached. Since the obtained HEP is relatively low for a one minute response time, it can be argued that even the higher value for inhibiting ADS in Nine Mile Point 1 (N-1) reflects optimism that the operators will be successful. In any case, given the fact that some plants failed to model the event and thereby essentially assume success, the low values are not necessarily outliers. The reasons for the differences in the remaining values are not clear and may simply reflect plant differences in training and attitude toward ATWS scenarios.

The main conclusion of examining the HEPs for specific actions across plants is that although at least some of the variability in HEP values can be an artifact of the HRA methods used and the way in which they are applied, it also appears that in most cases there are justifiable reasons for much of the variability in HEPs and in the results of the HRAs across the different IPEs. However, as noted earlier, such a conclusion 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 extend 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 for that event may be similar to those obtained for other plants. 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 high quality HRAs, it is possible that licensees obtained HEP values that are not appropriate to their plant. Examples of variability in the results from PWRs are presented in the next section.

13.4 Important Human Actions for PWRs

This section identifies and summarizes the human actions important in the PWR submittals. Actions important in a relatively high percentage of the submittals are discussed in the first subsection, followed by examinations of the impact specific plant classes and unique plant designs and characteristics have on what is important. Consistent with the rest of this report, the PWRs are separated into 5 classes: B&Ws, CEs, 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 seven units), ten are CEs (covering 15 units), four are W-2s (covering six units), 9 are W-3s (covering 13 units), and 20 are W-4s (covering 32 units). In the final subsection, the HEPs for selected human actions are examined to determine the extent to which the variability in HEPs across the PWR submittals appears reasonable.

13.4.1 Human Actions Generally Important for PWRs

A list of the most important human actions identified in a review of all 48 PWR IPE submittals is presented in Table 13.3. The table lists the human action event, the accident sequences in which the event is important, the percentage of all PWR submittals finding the event important, and the percentage of submittals finding the event to be important as a function of PWR class.

Table 13.3 Important human actions, accident sequences, and percentage of PWRs finding the action important.

Important human actions	Accident sequences	Percentage of PWRs finding the action important
Switchover to recirculation (plants with manual or semiautomatic switchover)	LOCAs	All PWRs ≈ 80% B&W ≈ 85% CE -- N/A W-2 ≈ 100% W-3 ≈ 55% W-4 ≈ 90%
Feed-and-bleed	Transients with all feedwater failed and steam generator tube ruptures (SGTRs).	All PWRs ≈ 60% B&W ≈ 45% CE ≈ 60% W-2 ≈ 70% W-3 ≈ 45% W-4 ≈ 70%
Depressurization/cooldown	All except ATWS	All PWRs ≈ 50% B&W ≈ 60% CE ≈ 30% W-2 ≈ 100% W-3 ≈ 70% W-4 ≈ 50%
Use of backup cooling water systems (recovery, alignment, restoration)	Transients	All PWRs ≈ 40% B&W ≈ 45% CE ≈ 30% W-2 ≈ 35% W-3 ≈ 60% W-4 ≈ 30%
Provide makeup to tanks for water supply, e.g., borated water storage tank (BWST), refueling water storage tank (RWST), CST. (Excludes auxiliary feedwater (AFW)/emergency feedwater (EFW) water source alignment)	SGTRs, small LOCAs	All PWRs ≈ 35% B&W ≈ 30% CE ≈ 20% W-2 ≈ 35% W-3 ≈ 40% W-4 ≈ 40%
Restore/provide backup for loss of room cooling (HVAC)	Transients	All PWRs ≈ 30% B&W ≈ 15% CE ≈ 50% W-2 ≈ 35% W-3 ≈ 30% W-4 ≈ 30%
Restore/reestablish main feedwater (MFW) or condensate to steam generators	Transients with AFW failure	All PWRs ≈ 30% B&W ≈ 30% CE ≈ 35% W-2 ≈ 35% W-3 ≈ 50% W-4 ≈ 30%
Proper control of AFW/EFW once operating (sometimes under adverse conditions such as loss of IA or no DC power)	Transients	All PWRs ≈ 25% B&W ≈ 30% CE ≈ 40% W-2 ≈ 35% W-3 ≈ 0% W-4 ≈ 30%

Table 13.3 Important human actions, accident sequences, and percentage of PWRs finding the action important.

Important human actions	Accident sequences	Percentage of PWRs finding the action important
Trip reactor coolant pumps (RCPs) upon loss of seal cooling/injection	Transients	All PWRs ≈ 25% B&W ≈ 45% CE ≈ 35% W-2 ≈ 35% W-3 ≈ 15% W-4 ≈ 20%
Pre-initiators	Variety	All PWRs ≈ 25% B&W ≈ 0% CE ≈ 50% W-2 ≈ 0% W-3 ≈ 25% W-4 ≈ 20%
ATWS related reactivity control response (e.g., manually insert rods, emergency boration)	ATWS	All PWRs ≈ 20% B&W ≈ 0% CE ≈ 20% W-2 ≈ 0% W-3 ≈ 10% W-4 ≈ 35%
Lineup of alternate or replenishment of existing water supply for AFW/EFW	Transients	All PWRs ≈ 15% B&W ≈ 0% CE ≈ 40% W-2 ≈ 35% W-3 ≈ 10% W-4 ≈ 5%
Initiation of AFW/EFW (may be recovery of auto-start failure in some cases)	Transients	All PWRs ≈ 15% B&W ≈ 0% CE ≈ 50% W-2 ≈ 0% W-3 ≈ 10% W-4 ≈ 10%
Attempt to maintain/reestablish RCP seal cooling/injection	Transient	All PWRs ≈ 15% B&W ≈ 15% CE ≈ 0% W-2 ≈ 15% W-3 ≈ 15% W-4 ≈ 20%
Isolation of steam generator (SG) in SGTR	SGTR	All PWRs ≈ 15% B&W ≈ 0% CE ≈ 20% W-2 ≈ 15% W-3 ≈ 30% W-4 ≈ 10%
Recovery of specific failures in AFW/EFW (e.g., open/control turbine-driven AFW flow control valves)	Transients	All PWRs ≈ 10% B&W ≈ 0% CE ≈ 30% W-2 ≈ 0% W-3 ≈ 10% W-4 ≈ 5%

Table 13.3 Important human actions, accident sequences, and percentage of PWRs finding the action important.

Important human actions	Accident sequences	Percentage of PWRs finding the action important
Proper control of MFW flow (e.g., prevent SG overfill)	Transients, SBOs	All PWRs ≈ 10% B&W ≈ 15% CE ≈ 15% W-2 ≈ 0% W-3 ≈ 0% W-4 ≈ 15%
Recovery of diesel generators	Transients, SBOs	All PWRs ≈ 15% B&W ≈ 30% CE ≈ 0% W-2 ≈ 0% W-3 ≈ 15% W-4 ≈ 15%
Bus cross-tieing	Transients	All PWRs ≈ 10% B&W ≈ 0% CE ≈ 35% W-2 ≈ 50% W-3 ≈ 0% W-4 ≈ 0%
Restore/provide backup to IA	Transients	All PWRs ≈ 12% B&W ≈ 15% CE ≈ 15% W-2 ≈ 15% W-3 ≈ 30% W-4 ≈ 5%
Use of special/alternate diesel generator or gas turbine	Transients, SBOs	All PWRs ≈ 10% B&W ≈ 15% CE ≈ 0% W-2 ≈ 50% W-3 ≈ 15% W-4 ≈ 5%
Proper breaker positioning/switching during loss of power events (excluding general offsite power recovery	Transients, SBOs	All PWRs ≈ 10% B&W ≈ 0% CE ≈ 9% W-2 ≈ 0% W-3 ≈ 0% W-4 ≈ 0%
Recover or align alternate pump for high-pressure safety injection (HPSI)	Transients, LOCAs, SGTRs	All PWRs ≈ 5% B&W ≈ 30% CE ≈ 30% W-2 ≈ 0% W-3 ≈ 0% W-4 ≈ 0%
Restore DC (battery) power	Transients	All PWRs ≈ 8% B&W ≈ 15% CE ≈ 30% W-2 ≈ 0% W-3 ≈ 0% W-4 ≈ 0%

Table 13.3 Important human actions, accident sequences, and percentage of PWRs finding the action important.

Important human actions	Accident sequences	Percentage of PWRs finding the action important
Recover specific recirculation switchover failures (e.g., manually recover failed switchover, manual or automatic)	LOCAs	All PWRs ≈ 10% B&W ≈ 15% CE ≈ 15% W-2 ≈ 0% W-3 ≈ 10% W-4 ≈ 5%
Establish RHR cooling or provide cooling to RHR heat exchangers	Transients, SGTR	All PWRs ≈ 10% B&W ≈ 15% CE ≈ 0% W-2 ≈ 15% W-3 ≈ 15% W-4 ≈ 0%
Isolation of nonessential SW or component cooling water loads	Transients	All PWRs ≈ 10% B&W ≈ 0% CE ≈ 0% W-2 ≈ 30% W-3 ≈ 0% W-4 ≈ 13%

Only a few human actions are regularly important across the PWR submittals. As with the BWRs, only a few human actions are important in a relatively high percentage of the PWR submittals. The human action most consistently found important is performance of the switchover to recirculation during LOCAs. Other frequently important human actions 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 cited as important by the licensees. Identified relationships between PWR plant classes and important human actions are discussed in Section 13.4.2. Section 13.4.3 addresses events found important in a single or only a few submittals.

Switchover to recirculation[13.6] on low ECCS level is important for LOCA sequences in most submittals for plants with semiautomatic or manual switchover. All of the 10 CE plants (15 units) have an automatic switchover, as do four of the other plants. For the 35 plants (58 units) which require operator actions to complete the switchover (either completely manual or semiautomatic), ~80% of the submittals find this action to be important. One possible reason some licensees fail to find this action important may be the fact that the sizes of refueling water storage tanks (RWSTs) vary from plant to plant. Those licensees with plants with larger RWST capacities may model the small LOCA and long-term transient sequences as not requiring the switch over 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 a preferred action 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 human actions. For the licensees finding the operator action to switch over to recirculation important

[13.6]The variability in HEPs for this event across the PWR IPEs is discussed in Section 13.4.4.1.

(and reporting contribution to total CDF), the contribution to CDF ranges from less than one percent in several cases to as much as approximately 17%, with an average contribution of approximately ten percent.

Initiating the feed-and-bleed operation[13.7] ***is identified by many licensees as being important.*** This event is important in transient and steam generated tube rupture (SGTR) sequences when all feedwater has failed. In addition, a few licensees find the establishment of an reactor coolant system (RCS) bleed path with one pilot operated relief valve (PORV) important in small LOCAs. Approximately 60% of the submittals indicate that feed-and-bleed is one of the most important events. Why some licensees fail to find feed-and-bleed important can be due to many interrelated and not easily discernible reasons. For instance, the relative reliability of each plant's auxiliary feedwater/emergency feedwater (AFW/EFW) system is a factor since it is only in sequences where AFW/EFW has failed that feed-and-bleed becomes another important action in the in-depth defense to providing core cooling. Thus, accident sequences involving AFW/EFW failure (and thus the need to use feed-and-bleed) can vary considerably in frequency, thereby affecting the overall importance of the feed-and-bleed function. Specific support system dependencies can also affect the overall reliability of feed-and-bleed and hence the importance of this human action. For plants with a higher susceptibility of failing feed-and-bleed due to support system failures, this mode of cooling is less reliable, and so the human action of operating feed-and-bleed can be less important. Additionally, many licensees spent considerable effort to also 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 and the related human action. Other factors related to the success criteria needs for feed-and-bleed as well as the HEPs themselves can also contribute to the relative importance of this mode of cooling and the human action. The contribution to CDF for this event ranges from less than one percent to approximately ten percent, with most submittals showing relatively small contributions from this event, resulting in an average contribution to total CDF of approximately five 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 important by more than half of the licensees. This action usually (but not always) involves depressurizing the steam generators to cooldown the RCS and is found important in all types of sequences except ATWS. It is most frequently important in SGTR sequences. Fifty percent of the licensees find the human action important. Reasons for failing to find depressurization and cool down important can be numerous and interrelated and include the reasons given for the feed-and-bleed event. Additionally, not all of the plants even model this mode of cooling — in some cases because of the relatively low capacity to depressurize in some scenarios depending on PORV, atmospheric dump valve, or other equipment sizes. The contribution to CDF 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 contribution to total CDF of approximately three percent.

None of the remaining human actions are found important in more than 40% of the submittals and none of them make consistently large contributions to CDF. As is seen in Table 13.4, none of the remaining human actions are important in a large percentage of the submittals. Recovery and use of 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 important in several submittals and tripping the RCPs upon loss of seal cooling is important in about 25% of the submittals. As with the BWRs, pre-initiator events, including both miscalibration and restoration errors, are found important in some submittals. Interestingly, many of the licensees, finding one pre-initiator important, found

[13.7] The variability in HEPs for this event across the PWR IPEs is discussed in Section 13.4.4.2.

several of them important. Exactly why this is the case is not clear, but as with the BWRs, it appears to be related to modeling differences and the HEP values used. Some licensees use relatively low screening values or do not model pre-initiators, which eliminates potential contributions from pre-initiator events. 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/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.

13.4.2 Relationship Between PWR Class and Important Human Actions

The relationship between PWR class and important human actions is examined to see if particular events tend to be found important only for particular classes of PWRs. It must be kept in mind that larger populations of some of the plant types (especially the B&W and W-2s) are necessary before the differences can be considered statistically significant. Moreover, in most cases, it is believed that the differences are related to modeling preferences or plant-specific differences and not to differences in general PWR vendor or vintage designs.

The third column of Table 13.4 presents the approximate percentage of the different classes of PWRs for which the submittals report 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.

Factors that may affect the relative importances of those actions associated with switchover to recirculation cooling, feed-and-bleed, and depressurization have already been addressed. These factors are related to plant-specific design, reliability, and modeling preferences with no clear cut differences specifically related PWR class related.

The percentage of plants in each PWR class where the submittal reports backup to cooling water systems, to tank water supplies, and to room cooling as important, varies with no obvious strong trends among the PWR classes. These types of plant features tend to be very plant-specific; hence the availability and potential operability of backup trains or other equipment of this type are most likely dependent on detailed design differences among all the plants, rather than on general differences among, as opposed to the various classes of PWRs.

It is somewhat curious that actions related to AFW/EFW (lineup of alternate water supply, initiation of AFW/EFW, control of AFW/EFW, and recovery of specific AFW/EFW failures) are generally important in a higher percentage of CE plants than of the Westinghouse plants, and often are not found important in any of the B&W plants at all. It is not clear that any PWR class-specific reasons exist for this trend as there are many variables that can affect the overall importance of AFW/EFW and any related human actions associated with this system. These include numbers of trains, sizes of tank supplies, design details of support systems, etc., which tend to be plant-specific. The noted trend may simply result from a greater effort on the part of the CE plant analysts to model the potential success (including recovery actions) of the AFW systems at these plants. While it may be more important to do this modeling for plants that cannot feed-and-bleed (such as the CE System 80 plants), specific reasons for the trend cannot be determined based on the information in the submittals.

Tripping the RCPs is generally not identified as important in as great a percentage of the Westinghouse plant submittals as of the other plant class submittals. This might be due to the generally higher susceptibility to seal failure in the Westinghouse plants, even if the RCPs are tripped (per the modeling provided in the submittals). This action is therefore somewhat less relevant to Westinghouse plants. In addition, some plants have changed to the more temperature-resistant seals. Another factor may be that vulnerability to loss of seal cooling can be very plant-specific, depending on the redundancy and support system needs of seal cooling systems at each plant.

Also interestingly, actions related to electric power reliability and recovery vary in importance with no clear trends. The PWR classes vary considerably in the percentage of plants that defined bus cross-tieing, use of alternate AC supplies, proper breaker positioning, and restoring DC power as important. This is probably due to the wide and plant-specific variability in electric power designs, with no specific design trends by PWR class.

One final observation is that approximately half of the CE submittals identified important pre-initiator actions, considerably more than for the other plant classes. This is most likely to be a result of modeling scope and level of detail differences rather than any true differences in the design or in test and maintenance practices at the plants.

In conclusion, many factors can affect why certain actions are relatively more important for some plants than others. These factors are believed to be mostly related to individual plant differences, modeling preferences, and differences in emphasis. There appears to be little to no correlation between the importance of human actions and classes of PWR plants.

13.4.3 Human Actions Important at Selected PWRs

Table 13.4 presents a list of important human actions that are identified as being unique because they were in general important for one plant only (or both units at a single site) and/or because they are apparently related to a unique plant design or characteristic. The licensees used a variety of methods to identify these actions as important. Typically, these actions contribute from one to ten percent to the total core damage frequency presented in each submittal.

Table 13.4 Human actions found important at individual PWRs.

Plants	Human action	Accident sequences
Arkansas Nuclear 1, B&W-2, L-dry	Locally open SW jacket cooling valves to diesel generator upon failure of these normally closed valves to open.	LOSP and SBO
Beaver Valley 1&2, W-3, sub-atmospheric	Prematurely secure HPSI.	Particularly small LOCAs
Beaver Valley 2, W-3, sub-atmospheric	Prevent or otherwise recover from auto- switchover of high-pressure suction from volume control tank to raw water storage tank if raw water pathway fails.	Loss of vital bus initiators that cause switchover
Beaver Valley 2, W-3, sub-atmospheric	Manually align/start safety injection upon failure of both trains of the solid state protection system used for auto safety injection.	Particularly transients (any sequence requiring safety injection)
Calvert Cliffs 1&2, CE-2, L-dry	Block spuriously opened PORV with blocking motor-operated valve.	Particularly transients whenever a PORV is stuck open
Crystal River 3, B&W-2, L-dry	Locally isolate high pressure makeup pump individual recirculation line when common recirculation valves have failed.	Particularly SGTR
Crystal River 3, B&W-2, L-dry	Manually close open valve between the BWST and the containment sump to prevent drainage of BWST used for high pressure injection suction, or use recirculation cooling early.	Particularly SGTR

Table 13.4 Human actions found important at individual PWRs.

Plants	Human action	Accident sequences
Haddam Neck, W-4, L-dry	Isolate core deluge line break to ensure sufficient ECCS recirculation.	Small LOCA (core deluge line)
Haddam Neck, W-4, L-dry	Locally control charging flow on loss of air.	Transients or others involving loss of IA
Kewaunee, W-2, L-dry	Manually isolate makeup valve in path between CST and condenser to ensure initial suction for AFW.	Transients or others involving loss of IA
Maine Yankee, CE-3, L-dry	Close 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).	Particularly SGTR
North Anna 1&2, W-3, sub-atmospheric	Identify and correct safety injection flow diversion (backflow) through inoperative charging pump discharge check valve.	Particularly medium and small LOCAs
Seabrook, W-4, L-dry	Recover from loss of emergency safety features actuation system (ESFAS) by manually starting equipment from control room.	Transients (particularly involving loss of vital bus, thus failing part of ESFAS)
Sequoyah 1&2, W-4, Ice	Isolate component cooling train from spent fuel heat exchanger to ensure adequate cooling during recirculation phase of the accident.	Particularly small LOCA (with one train of component cooling out for T&M)
Summer, W-3, L-dry	Align and start standby charging pump and chiller upon failure of one train of chilled water.	Particularly LOSP when one diesel has failed
Summer, W-3, L-dry	Restore one train of chilled water within 12 hours to support charging.	Particularly Transients with loss of chilled water
TMI-1, B&W-2, L-dry	Throttle high pressure injection flow to avoid PORV demands and possible stuck-open PORV.	Particularly events causing low reactor coolant system pressure (e.g., small LOCAs)
TMI-1, B&W-2, L-dry	Open decay heat removal dropline valves to prevent boron precipitation.	Large and Medium LOCAs

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 be dependent on the specific design details at each plant. In many cases, the specific actions and corresponding responses or recoveries are somewhat dependent on plant-specific design configurations. 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 1 is clearly dependent on the normal state of the diesel generator jacket cooling valves during normal operation of the plant. Other plants may operate with a similar configuration. Vital bus loads and the design specifics for auto-switchover of high pressure suction can make the Beaver Valley 2 auto-switchover action similarly applicable at other plants. The Calvert Cliffs

1 and 2 action concerning use of the PORV block valves to mitigate the effects of a stuck-open PORV will seem applicable to many plants, and yet only the Calvert Cliffs submittal identifies this specific event as among the "important" human actions.

Some actions, while unique, may also be applicable at other plants as well. 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. 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 auto-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, as is described in the submittal, status indication regarding the continued need for injection may be confusing or otherwise uncertain. Whether such "uncertainty" is due to Beaver Valley 2 specifics or response to a conservative modeling approach and whether such an action is applicable to other plants cannot be determined based on just the submittal contents.

The Maine Yankee action concerning closure of reactor coolant loop stop valves to mitigate SGTRs or a reactor coolant pump thermal barrier rupture is uniquely dependent on the Maine Yankee design since other plants do not have reactor coolant loop stop valves.

In all the cases above, specific design configuration details and the time and status information available for performing these actions are plant-specific. This results in variability in the importance of these actions and likely accounts for the unique importances identified at the plants listed above. However, these important human actions may be applicable at other plants as well, although the relative importances will likely vary.

13.4.4 Examples of Variability in PWR HEPs

To assess the variability in HRA results from the submittals and the extent to which variability in results is due to real vs. artifactual differences, the HEPs from two of the more important human actions appearing in the PWR submittals are examined across plants.

13.4.4.1 Failure to Switchover to Recirculation During LOCAs

The first action examined for the PWRs is the operator action to switch from injection with ECCS to recirculation. This action is selected because importance measures indicate that it is a dominant contributor for many PWRs. Figures 13.4 to 13.6 display, by PWR vendor, the HEPs for the switchover to recirculation for each of the 40 PWR submittals which model the event. As can be seen in the figure, a large difference exists in the HEPs for accomplishing this action. The difference between the lowest and the highest value is several orders of magnitude. One reason for the variability in HEPs within a given plant is that success of the switchover is in general (but not always) estimated to be more likely at high pressure (e.g., in small LOCAs) than at low pressure (e.g., in large LOCAs). One advantage for the high pressure case is that in many instances more time is assumed available for the operators to diagnose and accomplish the desired actions, and the operator stress levels should be less than in the large LOCA scenarios. However, in some instances the advantage for the high pressure case is confounded because of "piggybacking." That is, some plants require the high pressure pumps to intake from the low pressure system, which is aligned to the sump. Thus, the operator actions for high pressure can be somewhat more demanding than alignment in the low pressure case. Moreover, the difficulty and complexity of the general process to accomplish the switchover apparently varies significantly across plants.

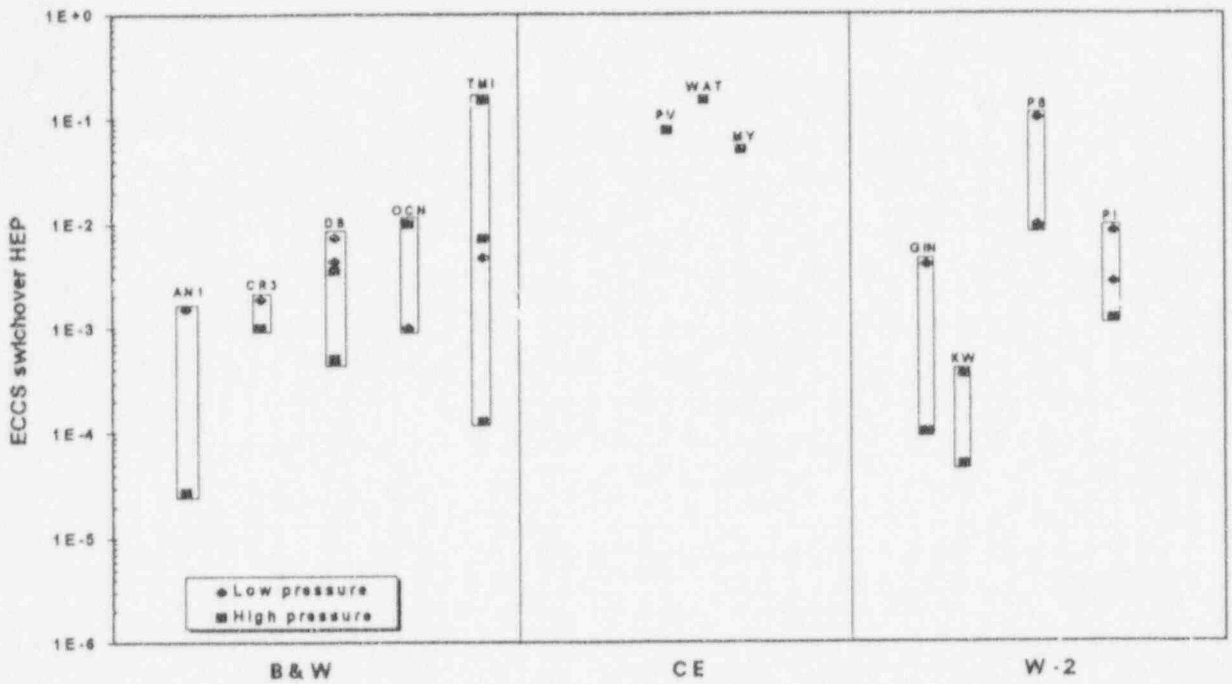

Figure 13.4 HEPs for switchover to recirculation for B&W, CE and W-2 plants.

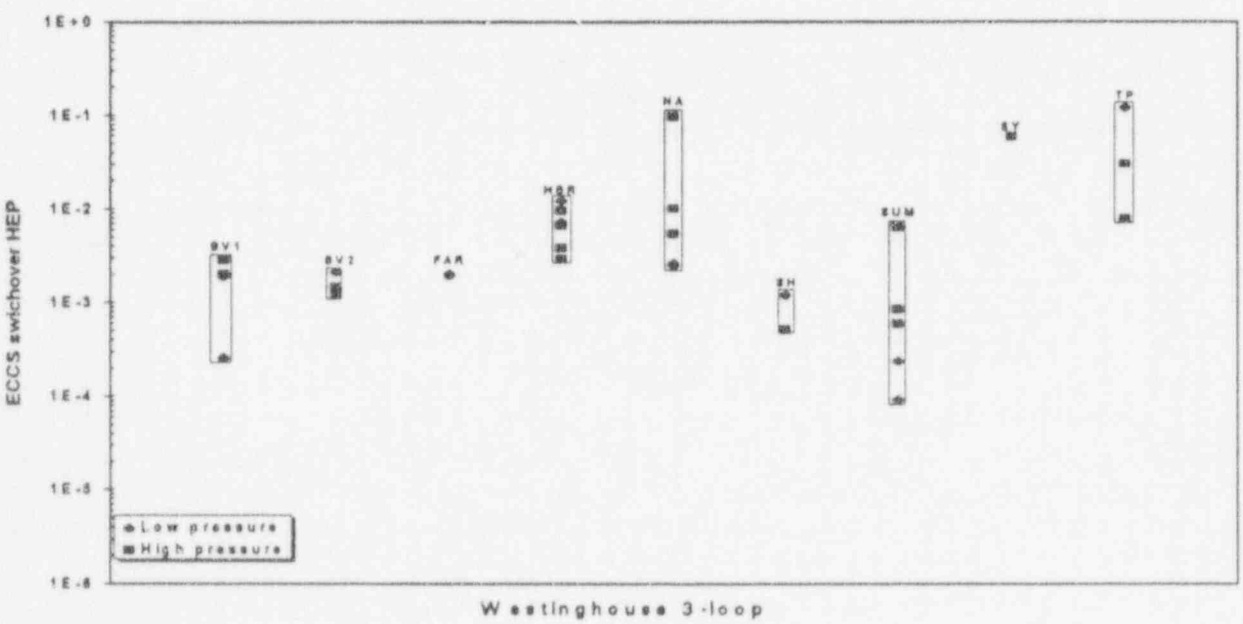

Figure 13.5 HEPs for switchover to recirculation for W-3 plants.

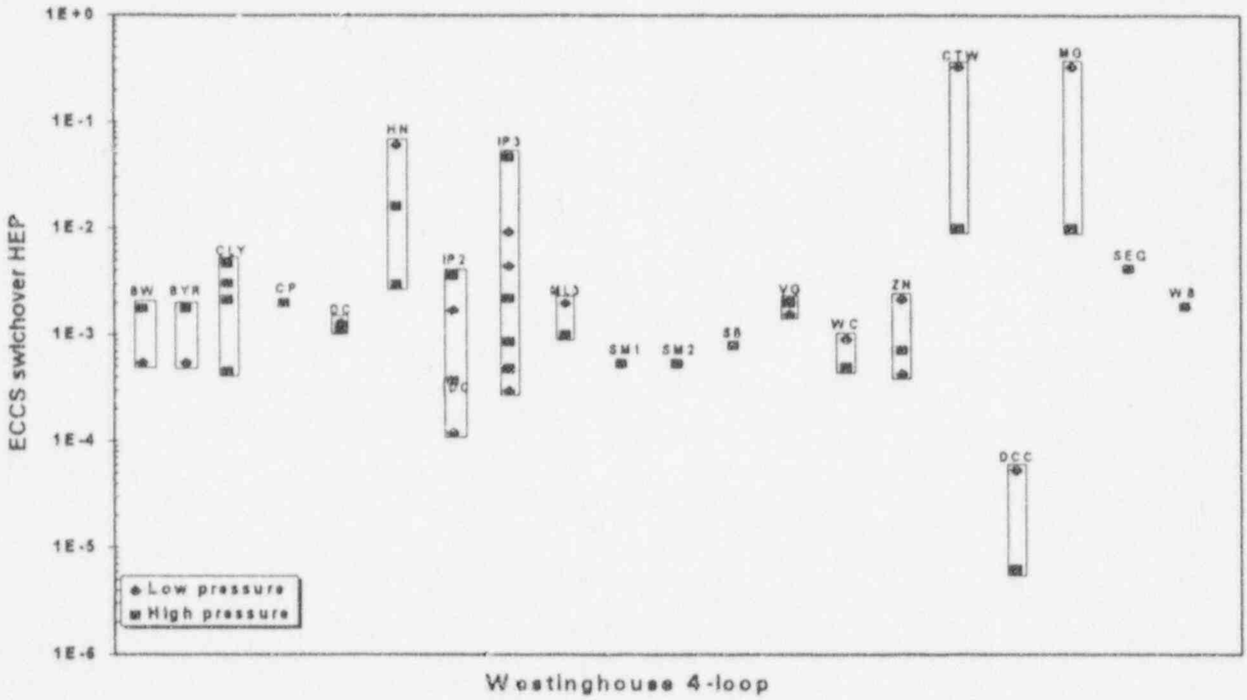

Figure 13.6 HEPs for switchover to recirculation for W-4 plants.

Another reason for the large differences in the HEPs across submittals is that in some cases the switchover is automatic, while in others it is either a semiautomatic or a completely manual operation. For plants with an automatic switchover, some licensees model an operator action to recover a failed automatic actuation, while for the other plants the operators will be conducting a normal (albeit not necessarily simple) activity for the accident scenario. Thus, a difference in the HEPs for these situations will not be surprising. Of the 48 PWR submittals reviewed, apparently 20 require manual alignment and initiation of the switchover, with 13 plants having semiautomatic initiation and 15 being completely automatic. An average of each plants average HEP for the switchover action indicates that the mean HEP for plants requiring manual alignment tends to be lower than that for the semiautomatic and automatic plants (8.4E-3, 3.1E-2, and 5.0E-2, respectively).

The factors discussed above and certain special cases can account for most of the variability in the HEPs for the switchover to recirculation. The high value for Three Mile Island 3 (TMI) is a special case that involves reestablishing recirculation after EFSAS closure of two makeup pump related valves. The high HEP values for

Waterford 3 (WAT), Maine Yankee (MY), and Surry (SY) are recoveries of failed automatic initiations, and the high values for Catawba (CTW) and McGuire (MG) reflect the potential for recovery of the failed automatic part of a semi-automatic process. Palo Verde's (PV's) high value is for a special case in which level transmitters have failed, and Point Beach (PB) and Haddam Neck (HN) argue that their high value reflects a complex and time-critical task during a large LOCA. Similarly, the Turkey Point 3&4 (TP) licensee indicates that all of their HEPs for the switchover are relatively high because the ex-control room actions required to complete the task are not frequently practiced. North Anna 1&2 (NA) has a high value for small break LOCA and transient scenarios after depressurization has occurred, and the high value for Indian Point 3 (IP3) reflects a scenario in which operators fail to depressurize as needed. Thus, all of the higher HEPs appear to have reasonable bases.

Much of the variability in the mid-range clearly reflects the difference between the high and low pressure cases. In addition, some licensees consider whether the RHR pumps have to be stopped (e.g., Summer (SUM)) before initiation of recirculation. Some of the rest of the variability in the mid-range apparently reflects the time available for the switchover as determined by the size of and level in the RWST (e.g., H.B. Robinson (HBR)), the difficulty of the alignment process at the individual plants, or whether the action must occur during a flooding scenario (e.g., Callaway (CLY)). However, at least some of the variability in the values between plants in the mid-range is not clearly explainable. Similarly, reasons for most of the values on the low end of the distribution are not straightforwardly explained. The low values for Arkansas Nuclear (AN1), Three Mile Island (TMI), Ginna (GIN), Kewaunee (KW), Beaver Valley 1 (BV1), Summer (SUM), Indian Point 2 (IP2), and D.C. Cook (DCC) may reflect plant-specific advantages (e.g., simple switchover process), careful analyses, or optimism on the part of the analysts. Regardless, as with the HEP values examined for the BWRs, most of the variability in the HEPs for switchover to recirculation appears reasonable and justified.

13.4.4.2 Operator Action to Perform Feed-and-bleed

The last specific operator action examined is the PWR event for initiating feed-and-bleed. Forty-two of the licensees provided HEP values for this operator action. Some of the B&Ws essentially have an automatic initiation of feed-and-bleed, and therefore no operator action is modeled. In addition, some of the CE plants are unable to feed-and-bleed. For those submittals modeling the event, the difference between the lowest and highest HEPs (see Figures 13.7, 13.8 and 13.9) is several orders of magnitude. However, with the exception of the values from a few of the submittals, the HEPs for failing to feed-and-bleed are reasonably well grouped and in general show greater consistency than those from other actions examined. There are several general factors that lead to the variability in the values for feed-and-bleed. Since the action can be important in a variety of scenarios when AFW is lost, the timing for the action can vary and influence the HEP. In addition, the number of PORVs needed varies by plant and in some cases alternate valves must be used to accomplish the action. Finally, a potential "reluctance factor" exists regarding the initiation of feed-and-bleed because of the impact on the plant. It is unknown how the different licensees address this factor.

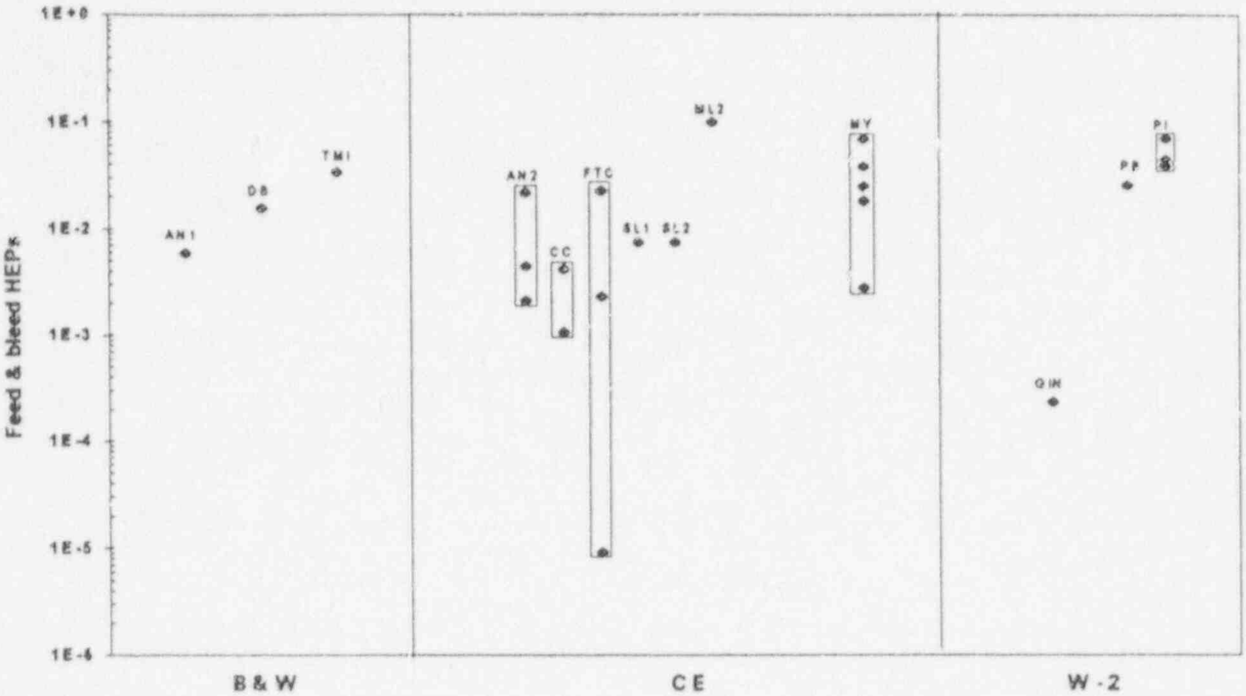

Figure 13.7 HEPs for feed-and-bleed for B&W, CE and W-2 plants.

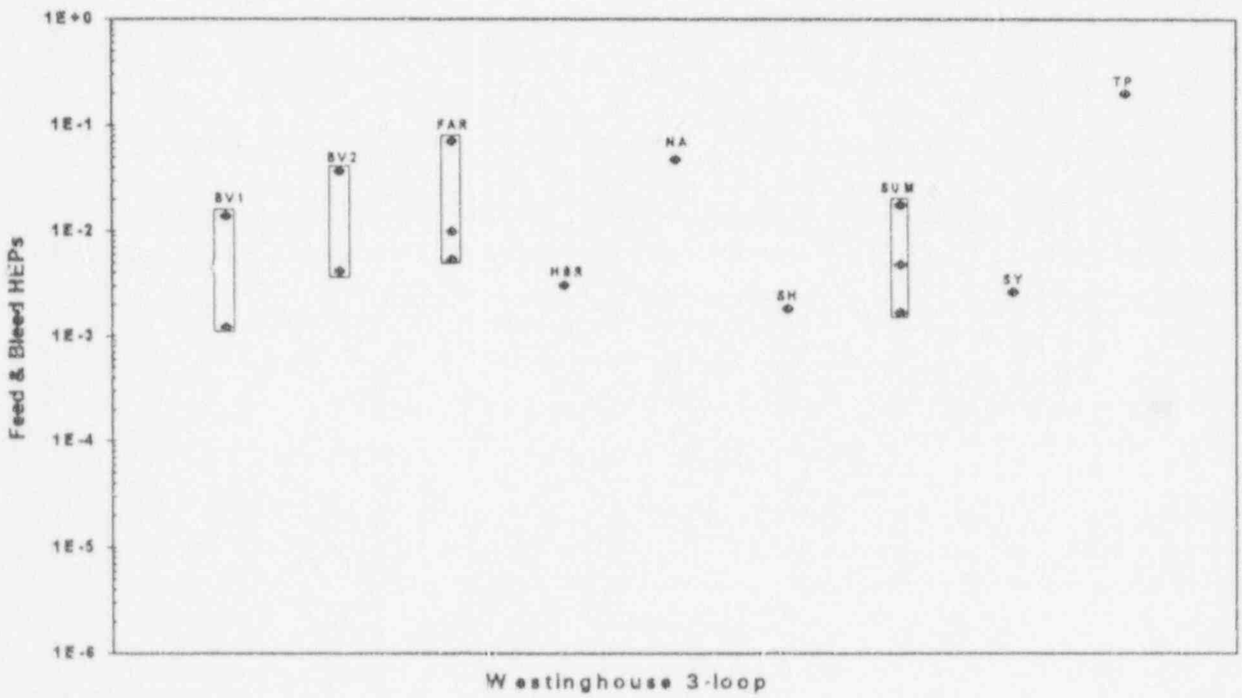

Figure 13.8 HEPs for feed-and-bleed for W-3 plants

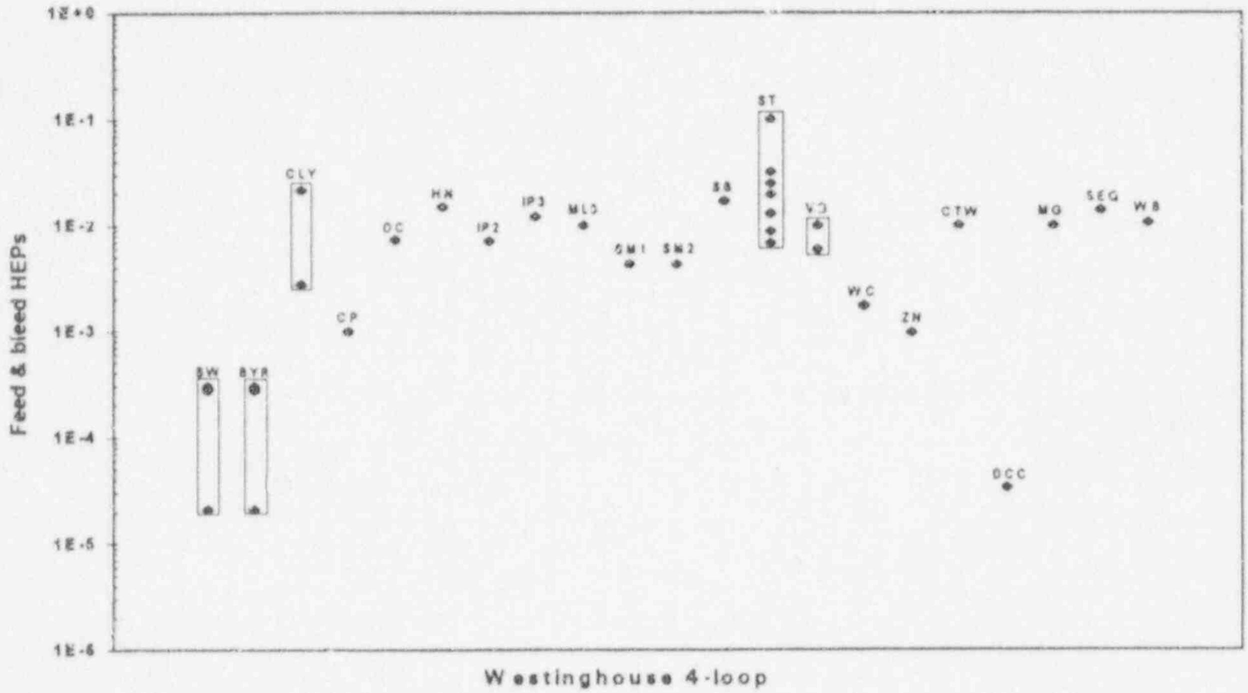

Figure 13.9 HEPs for feed-and-bleed for W-4 plants.

Turning to some of the specific HEPs, the reason for the relatively high value for Millstone 2 (ML2) cannot be clearly determined from the submittal. However, the licensee assumes a relatively short time period for the operator diagnosis, and because Millstone 2 is a CE plant, the alignment may have been relatively difficult. The four values for Maine Yankee (MY) all tend to be somewhat high. The variability between these values is related to PSFs such as timing, stress, and scenario-specific training received by the operators. Similarly, the variability in HEPs for Prairie Island (PI) is related to plant-specific factors, and the relatively high HEPs appear to be related to the time assumed available. The high value for Turkey Point 3&4 (TP) reflects a special case for "dual unit" feed-and-bleed, and the value apparently also includes hardware failure probabilities. A scenario in which all PORVs fail contributes to the high value for South Texas 1&2 (ST).

Regarding the low HEP values for feed-and-bleed, the outlier from Ft. Calhoun 1 (FTC) cannot be explained. However, given the difference between this value and the values for the remaining 60 events modeled in the Ft. Calhoun IPE (all of the other HEPs modeled are greater than an order of magnitude higher), the value reported could be an error. The remaining unusually low values (Ginna (GIN), Byron (BYR), Braidwood (BW), and D.C.Cook 1&2 (DCC)) appear to be related to the HRA methods used and their application. Without additional information, it is impossible to determine whether or not the values are realistic. In any case, disregarding what appears to be outliers, most of the differences in the HEPs appear reasonable in that they are driven by scenario and plant-specific factors as discussed above.

13.5 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, obvious commonalities across the plant types prompt an examination of potential similarities or differences in the operational and HRA-related observations. Several observations follow.

- As noted in the introduction to this chapter, neither BWR nor PWR submittals show a broad range of consistency in terms of which human actions are found important. Given the numerous factors which can influence the results of the IPEs, and the fact that redundancy of function 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 important for one type of plant as opposed to the other.

- Of the events frequently found important in BWRs and PWRs, the only similar actions are those related to depressurization and cooldown.

- Events related to aligning or recovering backup cooling water systems (e.g., SW) are found important in approximately one-third of both the 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 participating in the individual plant examination partnership (IPEP), there is no indication that particular HRA methods are more frequently applied to one type of plant than to another. 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 are related to HRA method per se. The IPEP methods are applied primarily to PWRs.

In summary, it seems that most of the differences in the HRA results of the BWR and PWR submittals are related (not surprisingly) to the differences in the systems of the two types of plants. In terms of methodology, general patterns of results, and overall importance of humans in operating the plants, the BWRs and PWRs are reasonably similar.

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PART 4

IPEs WITH RESPECT TO RISK-INFORMED REGULATION

14. ATTRIBUTES OF A QUALITY PROBABILISTIC RISK ANALYSIS

Chapter 6 of Part 1 summarizes the key perspective regarding the Individual Plant Examinations (IPEs) with respect to risk-informed regulation. Section 6.2 of that chapter summarizes the characteristics that comprise a quality probabilistic risk assessment (PRA). Chapter 14 provides a more in-depth discussion of these characteristics addressing in detail the attributes of each element of a quality PRA.

14.1 Introduction

In Generic Letter (GL) 88-20 (Ref. 14.1) the U.S. Nuclear Regulatory Commission (NRC) requested that licensees perform an IPE to identify any plant-specific vulnerabilities. In that request, the NRC indicated that a 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. 14.2), 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 generic letter, licensees elected to perform PRAs for their IPEs. In addition, most licensees have indicated their intent to maintain and update these PRAs and to use them in internal utility decisions and regulatory activities. In this light, these PRAs provide the foundation for the increased future use of PRAs in risk-informed regulation. Therefore, the potential role of these PRAs needs to be examined and this requires consideration of several issues:

- What is a quality PRA?
- How do the IPEs/PRAs compare to this quality PRA?
- What can be said about the quality of the IPE analyses, given the limited scope of the staff's IPE reviews?

The first issue is discussed in this chapter, but only in light of 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 chapter provides attributes of a quality 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 chapter also provides the attributes for a Level 3 PRA because a few licensees examined off-site consequences (Level 3 PRA). The last two issues are discussed in Chapter 15.

This chapter identifies the attributes of each level of a "quality" or "state-of-the-art" PRA. Before describing the attributes of such a PRA, it is necessary to clarify that a "quality" PRA entails "the best current PRA practices widely used in the industry." Thus, a quality PRA does not have to encompass areas that are currently being developed, such as the treatment of human errors of commission. A quality PRA also represents the current as-built and as-operated plant and realistically models, to the best extent possible, how the plant responds under accident conditions.

This chapter discusses the level of detail, boundary conditions, analytical requirements and documentation that are needed for a quality internal events, (including internal flood), full-power PRA. The attributes associated with an internal events analysis at full power are developed in significant detail. For other elements of a full-scope PRA (e.g., external events during full-power operation and all events during low-power or shutdown conditions), the attributes are not defined or discussed in this chapter.

This chapter is not intended to prescribe guidelines for how to perform a quality PRA. Such guidelines are available in numerous documents including NUREG/CR-2728, NUREG/CR-2815, NUREG/CR-4550, Volume 1, NUREG/CR-4840, NUREG/CR-2300, and NUREG/CR-5259 (Ref. 14.3). This chapter is only intended to provide the attributes of a quality PRA such that a reader can judge if a specific PRA (e.g., IPEs) is a quality PRA.

14.2 General Elements of a Full-Scope PRA

A PRA of a nuclear power plant is an analytical process that quantifies the potential risk associated with the design, operation, and maintenance of the plant, with regard to the health and safety of the public. A full-scope PRA is used to quantify the risk from all internal and external events that can occur at the plant under full-power, low-power, or shutdown conditions. 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. Since the IPE scope only covered full-power operation and internal events (including internal flood, but not internal fire), the attributes are only defined for this scope. Therefore, for simplicity, the term PRA, as used in this chapter, refers to a full-power, internal events PRA.

A PRA involves three sequential parts or "Levels" as shown in Figure 14.1:

- Level 1 - involves the identification and quantification of the sequences of events leading to core damage;
- Level 2 - involves the evaluation and quantification of the mechanisms, amounts, and probabilities of subsequent radioactive material releases from the containment; and
- Level 3 - involves the evaluation and quantification of the resulting consequences to both the public and the environment.

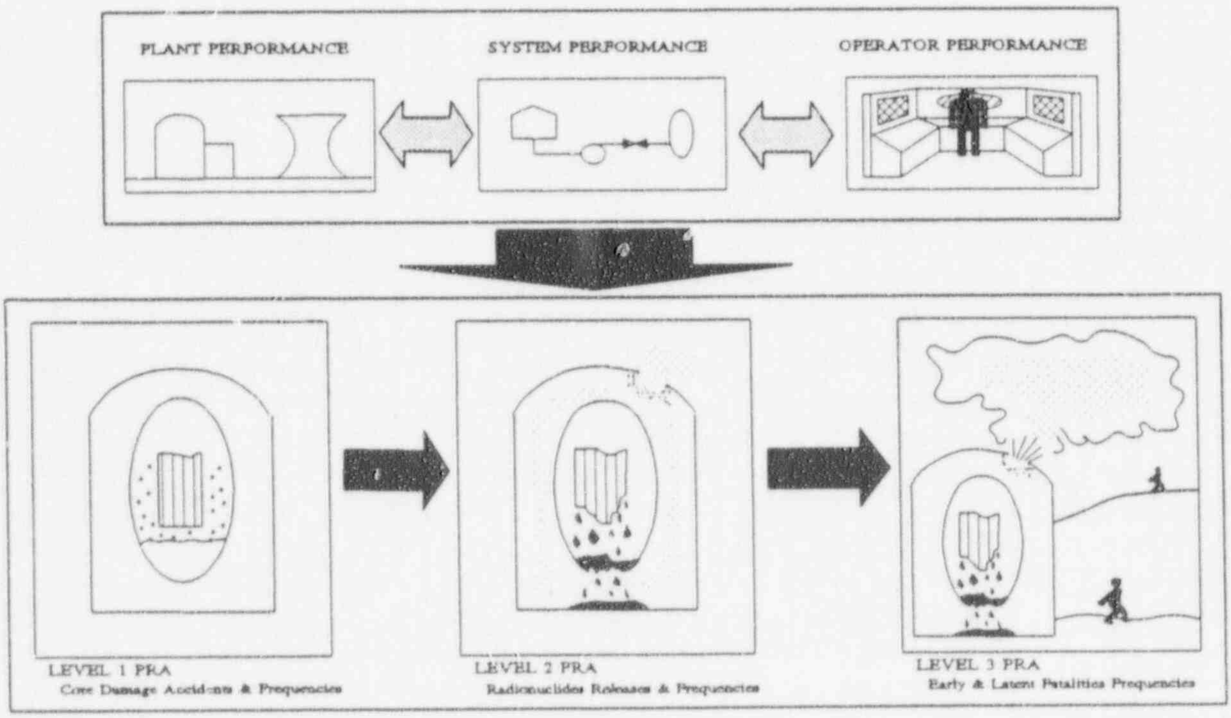

Figure 14.1 The three levels of a PRA.

Certain general assumptions and limitations are imposed on the scope and boundary conditions of a PRA. The following assumptions are usually found in a quality PRA:

- The "average" core damage frequency is calculated based on an average plant configuration (as defined below) rather than a core damage probability (refer to Section 14.3).
- The plant is operating within its Technical Specifications and other regulatory requirements.
- The design and construction of the plant are adequate.
- Plant aging effects are not modeled; that is, constant equipment failure rates are assumed.
- A "freeze" date of the PRA is selected to represent the design, operation and maintenance of the plant. In a quality PRA, to ensure that the PRA model is as current as possible at the end of the analysis, and therefore, represents (as practicable as possible) the as-built and as-operated plant, the design, operation and maintenance of the plant as reflected at the beginning of the PRA analysis is selected as the freeze date. However, plant modifications (that would impact the PRA model) are credited in the PRA if the licensee has committed to implementing (i.e., actually installed) them by the next refueling outage.
- A minimum mission time of 24 hours is used in analyzing the accident sequences in a quality PRA; however, the mission time is extended in a quality PRA when the core melt progression and potential releases have not been terminated and reactor pressure vessel and containment integrity are still challenged.
- The PRA is calculated for an "average" plant configuration. The plant can be in many different configurations (especially during shutdown) for short periods of time and it is not practical to calculate the risk from all of the potential configurations. Instead, the average plant risk is calculated using test and maintenance outage events in the PRA models to represent average unavailabilities of systems (or portions of systems). The average system unavailabilities reflect the availability of the systems during all the different configurations actually experienced in the past operation of the plant (see Section 14.3.1.4 for further discussion on the requirements for calculating test and maintenance unavailabilities). The actual test and maintenance unavailabilities for the plant systems thus must be calculated using plant-specific operational data.

14.3 Attributes of a Level 1 PRA

A quality Level 1 PRA comprises the following three major elements:

- delineation of those sequences of events that, if not prevented, could result in a core damage state and the potential release of radionuclides
- development of models that represent the core damage sequences
- quantification of the models used in estimating the core damage frequency

Figure 14.2 illustrates the relationships between the "analytical" tasks associated with each of the above elements. The first element of a Level 1 PRA delineates those sequences of events that, if not prevented, could result in a core damage state and the potential release of radionuclides. This process typically involves identification of the initiating events and development of the potential core damage accident sequences associated with the initiating events.

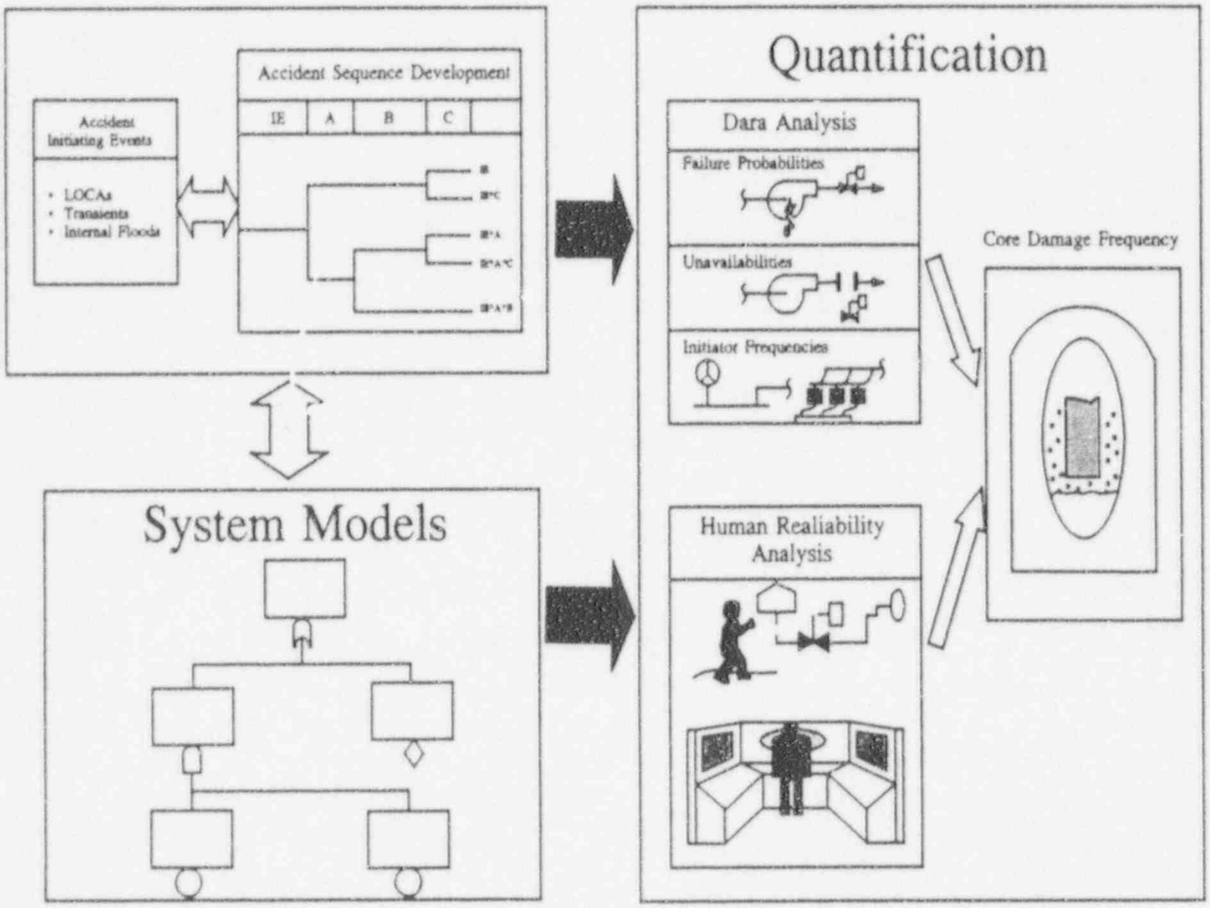

Figure 14.2 Level 1 analytical tasks associated with a quality PRA.

The identification of initiating events focuses on events that challenge normal plant operation and require successful mitigation in order to prevent core damage. Since there can be tens or hundreds of such events, this task also includes grouping the individual events into initiating event classes within which all events have similar characteristics and require the same overall plant response.

Accident sequence analysis involves delineating the different possible sequences of events that can evolve as a result of each initiating event class. 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 development task.

The second element of a Level 1 PRA involves the development of 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 necessary to prevent core damage (as defined by the core damage accident sequences). This modeling process, which usually involve the use of fault trees, defines the combinations of equipment failures, equipment outages (such as for test or maintenance), and human errors that cause failure of the systems to perform the desired functions.

The third element of a Level 1 PRA involves the quantification of the plant's core damage frequency and the associated statistical uncertainty. This process typically involves three tasks: data analysis, human reliability analysis, and quantification and uncertainty analysis.

The data analysis task involves determining initiating event frequencies, equipment failure probabilities, and equipment maintenance unavailabilities. Plant maintenance and other operating records are evaluated to derive plant-specific equipment failure rates and the frequencies of the initiating events.

The human reliability analysis task involves evaluating the human actions that are important to prevent core damage. This evaluation involves identifying the operator actions and quantifying the error probabilities of these actions. Human reliability analysis is a special area of analysis 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 task involves integrating the initiating event frequencies, event probabilities, and human error probabilities into the accident sequence models. Integration of the event tree and fault tree models using the Boolean reduction algorithm results in a set of accident sequence minimal cut sets (CS_i). An accident sequence minimal cut set identifies a combination of events [i.e., initiator (IE_j), hardware failure (BE_{i1}), human error (BE_{i2}), and recovery failure (BE_{i3}) etc.] that result in core damage. Each accident sequence cut set is quantified through the product of the probability of all its elements multiplied by the accident initiator frequency:

$$\text{frequency}(CS_i) = \text{freq}(IE_j) * BE_{i1} * BE_{i2} * \ldots$$

The expected annual core damage frequency (Total CDF) is obtained from the frequencies of the minimal cut sets. Various algorithms are used for aggregating the minimal cut sets frequency; e.g., rare event approximation, minimal cut set upper bound routine, etc. As an example, the rare event approximation results in summation of the frequencies of all contributing minimal cut sets:

$$\text{Total CDF} = \sum_{i=1}^{n} [CS_i]$$

In addition to core damage frequency, core damage probability is also estimated using the PRA quantification algorithm. Two types of core damage probabilities are typically estimated:

(1) Representing the probability of core damage for a period of time. Here, the core damage frequency is treated as the Poisson rate, and the core damage probability for a period of time, T, is estimated by:

$$\text{Probability (CD)} = 1 - Exp(-CDF * T)$$

(2) Conditional on a given plant configuration. The conditional core damage probability is extensively used for evaluating the risk impact of an operational event. As an example, the conditional core damage probability when two components A and B are unavailable for a period of time, w, is estimated by:

$$\text{Probability (CD)} = \text{Total CDF(with components A} \wedge \text{B failed)} * w.$$

14.3.1 Accident Sequence Initiating Event Analysis

The objective of the initiating events task is to identify and determine the frequency of events (occurrences) that upset the existing stable condition of the plant (normal operating state) such that a reactor trip or unplanned controlled shutdown is required with the need for core heat removal. This upset is such that an equipment and/or operator response is required to return the plant to a stable condition. These occurrences have the potential to lead to accident conditions and are called initiating events.

For this task, the following sections provide the attributes of identifying the upset initiating events, excluding events from the PRA, grouping the individual initiating events, and documenting the work.

Identifying Initiating Events

In a quality PRA, all internal events that upset normal plant operation and require a reactor trip or unplanned controlled shutdown with the need for core heat removal are considered as initiating events. These events fall into one of two categories as follows:

- Loss-of-coolant accidents (LOCAs) — All events that disrupt the plant by causing a breach in the primary coolant system with a resulting loss of core coolant inventory are modeled. These events include such occurrences as primary system pipe breaks, steam generator tube ruptures, feedwater pipe breaks, interfacing system loss of coolant accidents, reactor pressure vessel rupture, and steam pipe breaks.

- Transients — All events that disrupt the plant but leave both the core coolant and other water systems' inventory intact are modeled. These occurrences include automatic reactor shutdowns (scrams or trips), unplanned controlled reactor shutdowns (including those caused by degraded equipment configurations), manual reactor trips or scrams, manual operator actions taken in anticipation of degrading plant conditions, and transient-induced loss of coolant accidents. In identifying the transient events, frequently occurring events (such as turbine trips) and more rare events (such as a loss of a support system) are considered.

In ensuring "completeness" in identifying all potential initiating events for a quality PRA, the analyst would perform a comprehensive engineering evaluation that includes the following events:

- Events resulting in a loss of primary core coolant. This includes leaks and ruptures of various sizes and at different locations in the primary system (e.g., primary system pipe breaks, penetration failures, steam generator tube ruptures, and vessel rupture). In addition, a systematic search of the reactor coolant pressure boundary is performed to identify any active component in systems interfacing with the primary system that could fail or be operated in such a manner to result in an uncontrolled loss of primary coolant (commonly referred to as interfacing system LOCAs).

- All actual initiating events that have occurred at the plant. Actual plant scrams and unplanned shutdowns as documented in Licensee Event Reports (LERs) and scram reports are included. These initiators typically involve faults in the nuclear steam supply system (NSSS) and in the turbine-generator and related systems (referred to hereafter as the balance-of-plant).

- All initiating events considered in published PRAs (and related studies) of similar plants. NUREG/CR-4550 Volume 1 contains a list of transient initiating events that have actually led to reactor trips.

- All initiating events that have occurred at conditions other than full-power operation (i.e., during low-power or shutdown conditions) are included unless it is determined that they are not applicable to full-power operation.

- All systems supporting the operation of other plant systems are reviewed to determine if their loss results in an automatic or manual scram, or a controlled shutdown. Failure Modes and Effects Analysis (FMEA) are generally used to determine if an initiating event results from complete or partial failure of the system to operate, or from inadvertent operation of a system. In this method, the analyst determines for ***each*** component in the system: (1) its function, (2) the possible failure modes, (3) the failure mechanisms, and (4) the effects of the failure on the system and the plant.

  A system is evaluated if its loss would disrupt the normal operation of the plant. At a minimum, support systems that are examined include AC and DC buses; cooling water or service water systems; instrument and service air; heating, ventilation, and air conditioning (HVAC) systems throughout the plant (including the control room); and instrumentation/control systems.

  In determining whether the loss of a plant system or component must be treated as a support system initiating event, the expected level of degradation to other plant systems (specifically, accident mitigating systems) is also determined and evaluated. This may require calculations to determine the resulting environment to which the mitigating equipment is exposed and comparison to equipment qualification information.

- Initiating events consisting of multiple equipment failures are included if the equipment failures result from a common cause. For example, the failure of two DC electrical buses is included as an initiating event if the failure results from a common cause.

- For multiple unit sites where systems are shared or can be cross-tied, initiating events that can impact both units are identified in addition to those that will only impact a single unit.

- An interfacing system LOCA (ISLOCA) can be an important accident sequence particularly because of its potentially significant contribution to the releases of radioactivity from the plant due to all possible accident scenarios. Therefore, the modeling of ISLOCAs and particularly the credit given for isolation of the ISLOCA, the resulting size of the ISLOCA (for instance, does it fail overpressurized piping), and the effects of the ISLOCA on other equipment can significantly affect the importance of this type of event.

Excluding Initiating Events

In a quality PRA, not every initiating event that causes a disruption of the plant has to be modeled. That is, accident sequences do not have to be developed for every initiating event. In some cases, it is allowable to exclude initiating events.

Any of the following criteria are used in a quality PRA to exclude initiating events:

- The frequency of the initiating event is less than 1E-7/per reactor-year (ry) when the initiator does not involve either an ISLOCA, containment bypass, or vessel rupture.

- The core damage frequency resulting from the initiating event is estimated to be less than 1% of the total core damage frequency resulting from all accident initiators ***and*** the potential consequences of the associated radionuclide releases due to the excluded initiating events and subsequent system failures must also be less than 1% of the total from either the early or late releases. To estimate the core damage frequency from an excluded initiator, the conditional probability of failure of the mitigating systems for all possible accident sequences must be estimated and the basis for the estimate well documented. The core damage frequency from retained initiating events must also reflect 95% of the total core damage frequency resulting from all initiators (i.e., from retained and excluded events). Also, as with the core damage frequency, initiators can not be excluded if their exclusion results in retention of less than 95% of either the early or late release frequency.

- The resulting reactor trip is not an "immediate" occurrence. That is, the event does not require the plant to go to shutdown conditions until sufficient time has expired during which the initiating event conditions can, with a high degree of certainty (based on supporting calculations), be detected and corrected before normal plant operation would be curtailed (either administratively or automatically).

For example, a steam generator tube rupture event may have a relatively low contribution to the total core damage frequency, but may constitute a significant fraction of total large early releases. ***Initiating events such as these are not excluded in a quality PRA***. The need to understand the potential consequences of an initiating event in order to exclude it from detailed analysis makes the process of excluding initiating events necessarily iterative.

As another illustration, the loss of switchgear room HVAC may not require the operator to initiate a manual shutdown for 8 hours based on a room heatup calculation. During this time, the operator can almost certainly detect and recover the fault using portable cooling equipment (as directed by procedures) and prevent the need for a forced shutdown. In this case, loss of switchgear room cooling could justifiably be eliminated as an initiating event (based on procedural guidance and calculational support).

The basis for excluding initiating events from detailed evaluation is established in a quality PRA and evident to a peer review. The fact that an event has never occurred, by itself, is not a sufficient basis for eliminating an initiating event from evaluation.

Grouping Initiating Events

Numerous events and occurrences can disrupt a plant, and the plant's response to many of the events can be virtually identical. In such cases, it is acceptable to group the initiating events using the following criteria:

- Initiating events resulting in the same accident progression (i.e., requiring the same systems and operator actions for mitigation) can be grouped together. The success criteria for each system required for mitigation (e.g., the required number of pump trains) is the same for all initiators grouped together. In addition, all grouped initiators should have the same impact on the operability and performance of each mitigating system and the operator. Consideration can also be given to those accident progression attributes that could influence the subsequent Level 2 analysis (refers to Section 14.4.1.1).

- In conformance with the above criteria, LOCAs can be grouped according to the size and location of the primary system breach. However, primary breaches that bypass the containment are treated separately.

- Initiating events can be grouped with other initiating events with slightly different accident progression and success criteria if it can be shown that such treatment bounds the real core damage frequency and consequences that would result from the initiator. To avoid a pessimistic assessment of risk and to obtain valid insights, grouping of initiators with significantly different success criteria is avoided. The grouping of initiators requires that the success criteria for the grouped initiators encompass the most stringent success criteria for all of the individual events in the group. Note that in a quality PRA, low-frequency initiators are grouped with other relatively high-frequency initiators, rather than excluding them from further analysis.

Documentation of the Initiating Event Analysis

In general, the documentation of the initiating event task in a quality PRA is sufficient to enable a peer reviewer to reproduce the results. At a minimum, the documentation for a quality PRA includes the following information pertinent to initiating events:

- a list or general description of the information sources that were used in the task
- specific information/records of events (plant-specific, industry experience, "generic" data) used to identify the applicable initiating events
- the initiating events considered including both the events retained for further examination and those that were eliminated, along with the supporting rationale
- any quantitative or qualitative evaluations or assumptions that were made in identifying, screening, or grouping the initiating events as well as the bases for any assumptions and their impact on the final results
- documentation of the FMEA performed to identify support system initiators and the expected effects on the plant (especially on mitigating systems)
- specific records of the grouping process, including the success criteria for the final accident initiator groups

(Note: initiating event frequencies are addressed in the discussion of the data analysis task.)

14.3.2 Accident Sequence Analysis

The objective of the accident sequence analysis task is to determine the possible plant responses (sequences) that could occur as a result of initiating events. These plant responses are defined in terms of the different possible combinations of successful and unsuccessful functions or systems and operator responses required to mitigate an accident initiator. For the Level 1 portion of an analysis, this section provides attributes of selecting the accident sequence model, establishing the success criteria, modeling the accident progression, and documenting the work. These attributes apply specifically to those plant responses or sequences that end either with the plant in a stable state, or when the plant enters into a "severe accident" state in which the onset of core damage is imminent.

Selecting the Accident Sequence Model

Accident sequences are determined by implementing a logical methodology for identifying the different possible plant responses to the various initiating events. The plant safety functions and corresponding plant systems and operator responses that need to occur to mitigate each initiator are used to represent the different possible plant responses (or accident progression sequences).

Different models can be used to represent the accident sequences. Among these, the two principle methods used in a quality PRA are event trees and event sequence diagrams. There are also different types of event trees (e.g., functional versus systemic) and different ways of documenting the response to each accident initiator (e.g., separate event trees for each initiating event group or a general tree with the initiating event impacts included in system fault trees, or inclusion of support systems and shared equipment in the event tree rather than at a fault tree level). All of these different event tree approaches are acceptable in a quality PRA provided that the attributes discussed below are met and verified in a peer review. (The following discussion presents the attributes as applied to the event tree approach since it is the most prevalent technique.)

Establishing Success Criteria

Accident sequence analysis establishes the success criteria for the event tree sequence end states; therefore, core damage is defined. Several different definitions for core damage have been used in past PRAs. For example, peak cladding temperature limits or designated levels in the vessel have been specified as the onset of core damage. In a quality PRA, the onset of core damage is generally defined as occurring when no imminent recovery of coolant injection is anticipated, and therefore, a substantial amount (equivalent to or greater than the design basis) of the radioactive material contained in the gap between the cladding and the fuel is subsequently released. The definition chosen for the onset of core damage is supported by calculations. Note that considerable fuel melting may also be expected in most accident sequences with core damage outcomes.

At a minimum, a quality accident sequence model includes as the event tree headings the necessary safety functions, systems, and operator responses to prevent the onset of core damage. Accident sequence models can also delineate the functions required to protect the containment and influence the amount of radioactive material released.[14.1] The safety functions modeled in a quality Level 1 PRA include reactivity control, reactor coolant system (RCS) overpressure protection, reactor coolant inventory control and heat removal, and containment over-pressure protection (both early and late). The containment over-pressure protection functions are listed in the Level 1 requirements because the containment condition can adversely impact the core heat removal and inventory control functions.

The success criteria for each of these functions required to prevent core damage are established (e.g., the RCS inventory control function is expressed in terms of required flow rate). Once established, the system and operator responses modeled in a quality PRA include those front-line and required support systems and operator actions that can successfully meet the modeled safety function success criteria. The minimum hardware for each identified system (e.g., the number of pump trains) and operator responses required to meet the function success criteria determine the success criteria for responding to each initiating event group.

In a quality PRA, "realistic" success criteria are used for both the safety functions and the individual systems that perform those functions. That is, given that an accident condition actually occurs, the goal of a quality PRA is to model, as closely as possible, how the plant will respond. Therefore, the evaluation does not stop with safety-related systems when non-safety related equipment may be available to perform the needed function, thereby, preventing the onset of core damage. Consequently, in determining the success criteria, those systems, whether safety related or not, that are available and that may realistically be used are modeled in a quality PRA.

For grouped initiators, the accident sequence modeling reflects the most stringent initiator. For example, the coolant injection requirements for LOCA initiators (which usually involve a spectrum of break sizes) are based upon the upper end of the break spectrum. For other functions, the requirements may have to be based upon a different initiator included in the group.

[14.1] The attributes provided in this section do not address event trees where the end state goes past the onset of core damage. Functions required for establishing the containment performance and release of radioactive material are identified in the Level 2 discussion. Further event tree modeling to establish plant damage states is not addressed in this section.

The use of realistic success criteria provides additional assurance that the relative importance of the quantified accident sequences is as accurate as possible. To further ensure a "realistic" analysis, the use of excessively limiting success criteria (such as those used in design-basis assessments) is avoided in a quality PRA. Therefore, the minimum equipment needed to perform the function is modeled in a quality PRA. Safety Analysis Report (SAR) or licensing-basis system performance may be excessively limiting. For example, the licensing basis may require two out of four emergency core cooling pumps when "best estimate" calculations show that only one out of four pumps will prevent the onset of core damage.

The success criteria also reflects the timing of when the system can be used. That is, the success criteria for preventing core damage are dependent on the accident progression. The systems and equipment (e.g., number of pumps) needed to perform the required safety function at the beginning of the accident may be different from those needed later in the accident. For example, for a boiling water reactor (BWR), the control rod drive (CRD) system may not provide sufficient coolant injection into the vessel at the beginning of a small LOCA; however, at four hours into the accident (given that coolant injection has been occurring), the coolant inventory requirements are reduced and CRD flow is adequate. Therefore, in a quality PRA, the success criteria are established for various time periods, as may be necessary to reflect the accident sequence modeling.

In determining "realistic" success criteria, particularly when those criteria differ considerably from the SAR design basis or are not even addressed in the SAR, supporting analyses (e.g., thermal hydraulic calculations) are used as the bases for the success criteria that are credited in the PRA. Representative examples of criteria often used in PRAs that differ considerably or are not addressed by the design-basis criteria are (a) feed-and-bleed mode for pressurized water reactor (PWR) core cooling, (b) primary/secondary system depressurization and use of low-pressure safety injection and/or condensate to the steam generators whenever high-pressure safety injection and/or main and auxiliary feedwater are unavailable in PWRs, and (c) use of alternative injection systems (such as CRD flow or firewater) under conditions when all other injection systems are unavailable in BWRs, among others. These represent conditions that go well beyond the single failure considerations applied in the design basis and hence did not have to be treated in the original licensing basis for the plant. Hence, additional calculational support is required to address these "beyond design basis" considerations in the PRA.

While plant-specific calculations are preferred, non-plant-specific calculations (e.g., use of "similar" plant analyses perhaps with modification) are acceptable provided that appropriate justification is established. The computer codes used to calculate success criteria (either plant-specific or for a similar plant) contain the modeling detail present in codes such as RELAP and TRAC and are verified for the conditions that exist in the success criteria application (e.g., a code not verified for anticipated transient without scram (ATWS) analysis is not used to establish power reduction success criteria).

Modeling Accident Progressions

The modeling of the accident sequence progressions requires that the responses of the plant systems and the operators accurately reflect the system capabilities and interactions, procedural guidance, and timing of the accident sequences. Therefore, the development of the accident sequence models correctly incorporates the planned response to an initiator as specified in the plant emergency and abnormal operating procedures and as practiced in simulator exercises. In fact, the procedural guidance along with timing information obtained through thermal-hydraulic calculations serves as the guide in the actual development of the accident sequence models.

Operator actions required to mitigate an accident sequence (e.g., manual initiation of systems or special actions such as controlling vessel level during an ATWS in a BWR) are modeled (see Section 14.3.5). Therefore, event tree headings are generally chronologically placed in the order that the system or operator action is expected to be challenged. Deviations from the chronological representation of the procedural guidance are well documented.

In developing the accident sequences, the accident progression (as represented by the logic structure of the model) also accounts for dependencies and interfaces between and among the plant safety functions, systems, and operator actions required for accident mitigation. The dependencies and interfaces that are considered in a quality PRA include functional, phenomenological, and operational dependencies and interfaces.

Functional dependencies exist where the success of one function is dependent upon or otherwise affected by the success or failure of another function. There are two dependencies that are addressed in a quality PRA. These dependencies include (1) interaction of the initiating group on mitigating systems and operator actions, and (2) interaction among the mitigating systems and operator actions.

The interactions between the initiating event group and available mitigating systems and actions are accounted for either in the accident sequence model or at the system model level. Both immediate effects (e.g., loss of systems such as the power conversion system (PCS) following loss of off-site power) and delayed effects (e.g., loss of a system as a result of a loss of HVAC) are included. Delayed impacts can be subtle and require consideration of both harsh environment impacts (discussed in more detail below) and protective trip logic. An example of a protective trip logic concern is the occurrence of a steam leak detection trip signal resulting from a high room temperature that could result from a loss of room cooling. The loss of room cooling may occur as a result of various initiators including loss of off-site power, loss of a cooling water systems, or loss of the HVAC system itself.

The interactions among mitigating systems and operator actions is also accounted for either in the accident sequence model or at the system model level. One type of interaction is the successful operation of a system precluding the need for a redundant system to perform the same function. The second type of interaction is the failure of one system precluding the operation of another system. An example of these types of functional dependencies in both a BWR and PWR is the requirement for the success of primary system depressurization before low-pressure coolant injection can be utilized. Alternatively, vessel depressurization may cause the loss of a system as a result of pump runout inducing a subsequent pump trip. Another common functional dependency is that battery depletion during a station blackout precludes continued operation of steam-driven systems.

Phenomenological dependencies manifest themselves where the environmental conditions generated during an accident sequence influence the operability of systems and equipment. Phenomenological impacts can include generation of harsh environments that result in protective trips of systems (e.g., caused by high pressures or temperatures), loss of emergency core cooling system pump net positive suction head (NPSH) when containment heat removal is lost, clogging of pump strainers from debris generated during a LOCA, failure of components outside the containment following containment failure induced by the resulting harsh environment, closure of safety/relief valves (SRVs) in BWRs on high containment pressure, and coolant pipe breaks following containment failure.

Phenomenological impacts can also be indirect. For example, failure of containment heat removal in a BWR should cause the operator to depressurize the vessel per procedures to maintain suppression pool heat capacity limits. Such an action can result in loss of driving steam for systems such as high pressure core injection (HPCI) and reactor core isolation cooling (RCIC). Circumvention of some of these failure modes such as bypassing protective trips, switching suction sources for pumps, and arranging alternative room cooling can be credited either in the accident sequence modeling or in system models if the action can realistically be accomplished considering available staffing, available time to perform the action, and any harsh environment where the actions must be performed. Most of these phenomenological dependencies are identified on an individual system basis as part of the systems analysis (see Section 14.3.3).

Operational dependencies are hardwired or configuration dependencies present for some systems or components. For example, the suppression pool cooling mode of a loop of residual heat removal (RHR) is not available when the system is in the low-pressure coolant injection (LPCI) mode.

In a quality PRA, consideration is given to sequences in which the nature of the accident changes. For example, an initial transient may become a LOCA event as a result of a reactor coolant pump seal failure or a demanded and stuck open primary relief valve. Proper modeling of this progression change accounts for any dependencies among events previously discussed. Transfers to other sequence models to reflect the change in the sequence are made with due consideration to any differences between the modeled initiators. Screening of such transfers can be performed, but follows the truncation discussion provided in Section 14.3.6.

Documentation for Accident Sequence Analysis

The following information concerning the accident sequence modeling is reported in a quality PRA:

- a list or general description of the information sources that were used in the task
- the success criteria established for each initiating group, including the bases for the criteria (i.e., the system capacities required to mitigate the accident and the necessary components required to achieve these capacities)
- event trees or other types of models used (including all sequences) for each initiating event group
- a description of the accident progression for each sequence or group of similar sequences (i.e., descriptions of the sequence timing, applicable procedural guidance, expected environmental or phenomenological impacts, dependencies between systems, and other pertinent information required to fully establish the sequence of events)
- any assumptions that were made in developing the accident sequences, as well as the bases for the assumptions and their impact on the final results
- existing analyses or plant-specific calculations performed to arrive at success criteria, and expected sequence phenomena including necessary timing considerations
- sufficient system operation information (see the following section) to support the modeled dependencies
- input, calculations, etc. (particularly to justify equipment operability beyond its "normal" design parameters and for which credit has been taken)

14.3.3 Systems Analysis

The objective of the systems analysis task is to identify the credible failure modes and unavailabilities for the systems modeled in the accident sequence analysis task. Systems analysis is a systematic process for identifying system failures and integrates equipment failure data, unavailabilities due to test and maintenance, system and component dependencies (including support system dependencies), and human error probabilities.

For this task, attributes are provided for selecting the systems analysis model, establishing systems analysis boundaries, modeling system dependencies and interfaces, screening and excluding components and failure modes, and documenting the work.

Selecting the Systems Analysis Model

Different analytical techniques can be used to perform or support systems analysis. Examples include failure modes and effects analysis, reliability block diagrams, and fault trees. Fault trees are the preferred method for a quality PRA, however, since they are deductive in nature and, if properly performed, can identify all potential failure modes of a system and can thus be used to calculate system unavailability. Therefore, the following attributes are presented in terms of using fault trees in the systems analysis. However, the identified attributes also apply to any other method chosen to perform the systems analysis.

Current PRAs use detailed fault trees (modeling all relevant components and all credible failure modes), simplified fault trees (modeling only the components and failure modes perceived to be critical) and a "black box" in which a system failure is represented by a single event for which the failure probability is determined from an established database. In a quality PRA, detailed fault tree models are generally developed in analyzing the system. Acceptable deviations are discussed below. The basic concepts for constructing fault trees, in a quality PRA, generally follow the guidance in "The Fault Tree Handbook" (Ref. 14.4). Specific attributes to be applied with this methodology are provided below.

In a few cases, it is not necessary to develop detailed fault trees. Typically, simplified system models can be used if there is a high level of component redundancy and complexity and/or there is a lack of available data to properly quantify a complex model. In such cases, the system unavailability can be characterized either by a simplified model covering the perceived dominant contributors or simply by a data value if sufficient experience exists for the system (generic data used in this fashion must represent a common system design). Care must be taken to model aspects of the system that form dependencies with other systems so that dependent or common-cause events are properly handled.

The automatic depressurization system (ADS) system in a BWR is an example of where a simplified fault tree could be used. Past studies have shown that common-cause valve failure and an operator error to manually initiate the system are dominant failure modes for the ADS. Thus, a simplified fault representing these failure modes could be constructed. Since this system is dependent on several support systems (DC power and instrument air) used by other systems, these support system interfaces would have to be modeled.

The reactor protection system (i.e., the failure to scram the reactor) is an example of where a data value is permissible. In this case, the reactor protection system (RPS) system failure modes are independent from other system failures.

Establishing Systems Analysis Boundaries

An accurate representation of the design, operation, and maintenance of each modeled system is essential. In a quality PRA, the design, operation, and maintenance requirements and practices are reviewed to ensure that the system model reflects the as-built and as-operated system. At a minimum, system walkdowns are performed in a quality PRA to confirm the design of the system[14.2]. In addition, a quality PRA requires operator interviews, system procedure (abnormal, operating, maintenance, and testing) reviews, and involvement of plant system engineers.

The failure criteria defining the top event of the fault tree for each system matches the accident sequence success criteria. Note that in some cases, multiple models for the same system may be needed to address different sequences.

As noted above in a quality PRA, the system model considers all equipment and components necessary for the system to perform its function (as defined by the accident sequence success criteria) during the required accident mission time. The system model also defines the boundaries of the necessary equipment and components, and these definitions match the level of detail where statistical data exists in determining their failure probabilities. In addition, the defined boundaries reflect the dependencies and interfaces between equipment and systems.

All relevant and possible failure modes for each component are considered. In a quality PRA, these failure modes generally include the following:

- hardware faults
  - failure to change state
  - failure to operate
- out-of-service unavailability
- common-cause faults
- operator faults
- conditional operability faults, including equipment capability and phenomenological faults

Hardware faults are those physical breakdowns of the equipment such that the system or component cannot function as designed (e.g., pump shaft breaks). There are numerous types of hardware faults. For each component modeled in a quality PRA, the failures that could potentially preclude the component from successfully performing its function both when initially demanded and during the course of the accident are considered. (Criteria for excluding components and failure modes are provided later.) For example, for active components such as pumps and compressors, a change of state and continued operation is required; therefore, two types of failures (failure to start and failure to run for a required mission time) must be modeled. For other active components, such as operating valves and dampers that are required to change state, a change of state and remaining in that state are required; therefore, two types of failures (failure to open/close and failure to remain open/close) are modeled.

In modeling the out-of-service unavailability, a quality PRA considers both planned and unplanned testing and maintenance contributions. The type of testing and maintenance modeled is consistent with the actual practices of the plant for removing equipment for maintenance. Equipment configurations that are not permitted by the plant technical specifications are also considered. These considerations include technical specification equipment configuration control violations, as well as previously identified implementation and program deficiencies of the equipment configuration control process.

[14.2]Confirming the design refers to confirming the existence of the components comprising the system.

Common-cause equipment failures involve multiple failures that result from a single event or failure. The NRC's Office of Analysis and Evaluation of Operational Data (AEOD) report, "Common Cause Failure Data Collection and Analysis System" (Ref. 14.5) presented in six volumes, provides a common cause failure modeling for a quality PRA. Volumes 5 and 6 of that report are particularly useful as they directly apply to the modeling (Volume 5) and the database (Volume 6) applicable to PRA. Given the current state-of-the-art of common-cause failure analysis and the available data, only intrasystem common-cause failures are generally modeled in a quality PRA except for the HPCI-RCIC cases cited in the AEOD report.

How common-cause events are included in the model may vary (e.g., included in the system fault trees, added after initial cut set review of independent failure combinations), but the approach demonstrates that quantitatively important common-cause combinations are not missed. Truncation considerations are consistent with the expectations provided under the quantification discussion in Section 14.3.6 (i.e., truncation of any common-cause events shall be based on low cut set frequency arguments). For cases where the PRA requires the evaluation of common cause among a component type not covered by the AEOD report, the component type closest in design and similarity in the AEOD report is used to perform the evaluation. Human error probabilities also consider common causes and incorporate performance shaping factors to account for dependencies.

Certain types of human events are also considered in the systems analysis. In a quality PRA, these events include, at a minimum, those human actions that cause the system or component to be inoperable when demanded. These events (also referred to as pre-initiator human events) are analyzed as part of the human reliability analysis, and the associated attributes are discussed in Section 14.3.5. The systems analysis models can address other human events, including those actions needed for the operation of the system or component. These events (also referred to as post-initiator human events) are analyzed as part of the human reliability analysis, and the associated attributes are also discussed in Section 14.3.5.

System models also treat conditional faults. These failures are discussed below under system dependencies and interfaces.

Supercomponents or modules can be used in a quality PRA. However, the modularization process is performed in a manner that avoids grouping events (i.e., component failures, testing and maintenance unavailabilities, and human errors) with different recovery potential (e.g., hardware failures that can not be recovered versus actuation signals that can), human error events, events that are mutually exclusive of other events not in the module, and events that occur in other fault trees (especially common-cause events).

Modeling System Dependencies and Interfaces

A quality PRA models the dependencies and interfaces between and among the systems and components. At a minimum, the following dependencies and interfaces are modeled in a quality PRA:

- System Initiation, Actuation and Operation — those systems that are required for initiation, actuation and continued operation of the system (i.e., for both the front-line mitigating systems or other support systems) are identified. This includes those systems required to provide the needed power to the systems and components for initiation, actuation, operation and control. Typical systems providing this type of support include electric power (AC and DC) and instrument air. In modeling the initiation and actuation of a system, conditions needed for initiation and actuation (e.g., low reactor pressure vessel water level) are also addressed. For example, a condition required to initiate a system automatically may not exist in all accident sequences. Thus, failure of that portion of the automatic actuation system has a probability of 1.0 for that sequence.

- System Isolation, Trip or Failure — those conditions that can cause the system to isolate or trip and those conditions that once exceeded can cause the system to fail. At a minimum, conditions to be considered in a quality PRA include environmental conditions, fluid temperature and pressure being processed, external water level status, water and air temperature and pressure, humidity, and radiation levels. These conditions may arise when other system fail to function. Examples of required systems include HVAC, service/component cooling water, heat tracing on piping and tanks to prevent boron solution precipitation, instrumentation (pressure, temperature, level, etc.), and water transfer systems to maintain tank levels.

  Examples of conditions that can isolate, trip or fail a system or component include:

  — For BWRs, high pressure in the reactor pressure vessel (RPV) will cause the low pressure injection system to isolate; that is, the injection motor operated value (MOV) to the vessel will close and the recirculation MOV to the suppression pool will open.

  — A diesel generator can trip when the high jacket water temperature setpoint is reached. This conditions can occur when the supporting cooling water supply to the diesel generator is lost.

  — Inadequate pump NPSH due to low suction source level or high temperatures, clogging of strainers, steam binding of auxiliary feedwater pumps, and steam environment effects are a few example of conditions that can fail pumps.

  Because of the attempted realistic nature of PRAs, there are many examples of where a quality PRA allows for the operability of equipment beyond its design basis. This credit is allowed to account for the design margins built-in to most equipment used in a nuclear power plant and hence to recognize that equipment may function in conditions that are beyond those accounted for in the design basis. Examples include operability of pumps under saturated water suction conditions, steam relief valve operability even when the valve is operating under two-phase flow conditions, battery operability given all charging to the batteries has been lost, human performance under undesirable environment or radiation conditions, etc.

  While crediting the potential for this operability supports the intent to provide a realistic analysis, such judgments of operability can often "drive" the results of the analysis and significantly impact the dominant sequences and contributing equipment that most affect the core damage frequency estimated in the PRA; therefore, such judgments must be supported. Test data, actual plant experience, vendor information regarding experience of similar equipment in other applications, expert elicitation and technical analyses are acceptable evidence that are used in a quality PRA. Otherwise, it is assumed that once the expected conditions in the scenario exceed the design basis limits for the equipment, the equipment then fails with a probability of 1.0.

- System Capability — those conditions that can cause the system, though operable, to not meet the required function. Examples of this nature include flow diversion, insufficient inventories of air, water or power to support continued operation of the system for the required mission time. In a quality PRA, such "failures" are explicitly treated in the modeling process using realistic operability considerations and are supported with analysis; otherwise, it is assumed once these conditions exist that the equipment/system fails with a probability of 1.0.

- Shared equipment — those components and equipment that are shared among systems. Passive components not typically modeled are included when their failure impacts more than one system (e.g., a discharge pipe from a tank feeding two separate systems).

Screening and Excluding Components and Failure Modes

A quality PRA considers all equipment and components necessary to the function of the system important in the accident sequence analyses, as well as all credible failure modes associated with the systems and components. Nonetheless, it is not always necessary to model every component or failure mode.

In screening or excluding components or failure modes, the following criteria are used in a quality PRA:

- Screen/Exclude Component — The ***total*** failure probability of the component (sum of all failure modes) is at least two orders of magnitude lower than the next highest failure probability of another component in the same system train ***and*** the component to be screened/excluded does not have any dependencies or interfaces with other components or systems. Typically, passive components are excluded based on the fact that failure rates for these components are substantially less than those for active components.

- Screen/Exclude Failure Mode — The failure probability of the failure mode is at least two orders of magnitude lower than the next highest failure probability of another failure mode of that component.

Documentation of Systems Analysis

The following systems analysis information is documented in a quality PRA:

- a list or general description of the information that was used in the development of the system models, including a brief discussion of the following:
  - system function and operation under normal and emergency operations
  - actual operational history, indicating any past problems with the system operation
  - system success criteria and relationship to accident sequence models
  - human actions necessary for system operation
  - a list of all testing and maintenance procedures
  - a system schematic illustrating all equipment and components necessary for system operation
  - records/notes of walkdowns and significant discussions with plant staff
  - system dependencies and shared component interfaces documented using a dependency matrix or dependency diagram indicating all dependencies for all component among all system (front-line and support)
  - a table listing failure modes modeled for each component and event quantification
  - general spatial information and layout drawings to support external event analyses
  - assumptions or simplifications made in developing specific system models

- the nomenclature for the basic events modeled

- the freeze date used to represent the design and operation of the plant

- any general assumptions that were made in developing the systems models as well as the bases for the assumptions and their impact on the final results

- a list of all components and failure modes included in the model, along with justification for any exclusion of components and failure modes

- information and calculations to support equipment operability considerations and assumptions

- references to specific controlled input documents used for modeling (e.g., piping and instrumentation diagrams).

- documentation of the modularization process (if used)

- records of resolution of logic loops developed during fault tree linking (if used)

14.3.4 Data Analysis

The objective of the data analysis task is to quantify the input parameters that are necessary to quantify the core damage accident sequences, containment failure probabilities, and resulting quantitative estimates of radionuclide release frequencies and resulting consequences. The input parameters for the Level 1 portion of the PRA include initiating event frequencies, equipment failure probabilities (including common-cause failure probabilities), and equipment outage unavailabilities. For this task, the following sections provide attributes of identifying data sources, selecting data input, quantifying data parameters, and documenting the work.

Identifying Data Sources

Plant-specific information and actual and current plant operating experience are used in a quality PRA to quantify the initiating event frequencies, equipment failure probabilities, and equipment unavailabilities (such as during testing and maintenance). For initiating event frequencies, the number of plant scrams and unplanned shutdowns and the hours the generator is online are identified. The number of scrams and unplanned shutdowns are identified by examining actual scram reports, LERs, operator/control room logs, and other plant information. The number of hours the generator is online is obtained from the NRC Gray Book (Ref. 14.6).

For component failure rates, the number of failures and the number of demands or operating time are required. Sources of information for determining the number of component failures include maintenance records, operator/control room logs, clearance reports, and other pertinent component records. The number of demands and component run times is determined from plant records such as operator/control room logs, technical specifications, clearance orders, maintenance work orders, test surveillance records, and other plant records. Generic data[14.3] can be used in a quality PRA to the extent described below.

Common-Cause Failure Probabilities

The methods and database from the AEOD report entitled "Common Cause Failure Data Collection and Analysis System" (INEL-94/0064) provide a basis for a quality PRA. For cases where the database needs to be expanded to include numbers of components beyond that addressed in the AEOD report (generally six components), it is assumed

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[14.3] Acceptable sources of generic data for initiating events and component failure rates are contained in AEOD-sponsored reports. Other sources of generic data can be used, provided that such data is representative of the nuclear power industry and more specifically, represents the same type of components with the same component boundaries as modeled in the PRA.

that the conditional probability of each subsequent component is the same value as indicated for the sixth component in the AEOD database. Using lower generic common-cause values than those shown in the AEOD report or eliminating a common cause treated in the AEOD report are discouraged and are generally deemed inappropriate.

Selecting Data Input

As indicated above, many plant-specific sources are used in gathering the raw data necessary to quantify the basic event probabilities. In a quality PRA, data from throughout the operating history of the plant are used in determining the initiating event frequencies, component failure rates, and testing and maintenance unavailabilities. Data from the first year of operation can generally be excluded since operation during this period is not representative of operations in later years.

In reviewing the incidents for potential initiators, a quality PRA considers all incidents that require a reactor trip while at power with the need for core heat removal. Incidents occurring during other modes are considered in a quality PRA when they are not specific to the operating mode; that is, they could also occur at full power operation. Scrams events can be excluded from initiating event frequency calculations if a substantial change has occurred in the plant design or operation such that the identified failures can no longer occur. This exclusion is supported by analysis.

In reviewing component performance and quantifying component reliability, a quality PRA considers all incidents that affect the performance of equipment, as well as incidents occurring during all modes of operation and considered relevant to full-power conditions. In quantifying component unavailability (from testing and maintenance), a quality PRA only considers incidents occurring during full-power operation (for a PRA of this scope). In evaluating the applicability of actual plant equipment failures and events, a cause and effect relationship must be established and the cause is known to have been corrected before eliminating the failures from the data-base. For example, a change in component design or testing and maintenance practices can influence the applicability of component failure data. In general, the data used in the component failure rate calculations are representative of the current component design and operation. However, failure data prior to a plant modification made in response to such failures are not excluded unless it can be shown through a trending analysis that the plant modification eliminated the types of failures previously identified. Finally, testing and maintenance unavailabilities reflect current testing and maintenance practices.

Collection of component failure data is also based upon similar component attributes. For example, data on valve failures are grouped by valve type, size, operating pressure, and other pertinent attributes that discriminate one valve from another. In addition, the data collected are consistent with the component boundaries in the model. For example, if the definition of a pump used in the fault tree analysis includes the motor operator, control circuit, and power breaker, failure data for the pump include failures of these subcomponents. Caution must be exercised when using generic data since it may not be consistent with these component boundaries, or it may not be known if the data is consistent with the boundaries. Finally, exact calculations of the number of demands or component operating times (the denominator in the failure rate calculations) is preferred in a quality PRA. However, these values can be estimated provided that a reasonable basis is established for the estimate (e.g., the reported demands reflect the general history of the equipment rather than an exact count of every demand).

Quantifying Data Parameters

In a quality PRA, Bayesian estimation methods are used to combine the plant-specific component failure data with the generic data from the AEOD reports (discussed above) serving as the prior distribution. If no plant-specific data is available because the plant, system, or component is new; then generic data can be used. To the extent possible, calculations of initiating event frequencies and outage unavailabilities reflect, actual plant experience. That is, Bayesian estimates are not performed for initiating event and test and maintenance outages.

Equipment failures, outages, and accident sequence initiating events are modeled as random occurrences; however, the exact value of the failure probability or initiating event frequency is never precisely known. To account for this lack of knowledge, the uncertainty in the values of the failure probabilities and initiating event frequencies are reflected in statistical distributions. Note that the statistical distributions do not address other uncertainties in the model such as modeling assumptions and completeness of the model. The statistical distributions generated for the input parameters reflect the range of values that are believed to be possible. For plant-specific data, the Poisson distribution is used for time-dependent failure rates and the Binomial distribution is used for demand-related events. Either a lognormal or beta distribution is used as a prior distribution in the Bayesian estimation of demand failure rates. Similarly, either a gamma or lognormal distribution is used as a prior distribution in the Bayesian estimation of time-dependent failure rates. These uncertainties in the knowledge of the input parameter values are then propagated through the PRA models when the model is quantified. The results of the quantification are thus expressed in distributions for the top events in the model (e.g., the core damage frequency).

When no plant-specific or generic data exists, data estimates are typically generated using logic models such as fault trees, or expert judgement can be utilized. Fault trees and other logic models are beneficial in that they can be used to decompose a problem into more basic failures for which data may be available. For example, fault trees can be used to generate frequencies for support system initiating events that have never occurred at a plant. Expert judgement can also be used in the analysis of unusual or rare events. To the extent possible, expert judgement is avoided in a quality PRA. However, when no other choice exists, NUREG/CR-4550, Volume 2, provides a methodology for expert judgement elicitation for a quality PRA.

Documentation of Data Analysis

The following data analysis information is reported in a quality PRA:

- initiating event frequencies
- the failure rate (per demand or per hour), failure probability and/or unavailability (as applicable) for each event along with its associated error factor
- the specific time over which the given component or system is required to operate (may be sequence dependent)
- sources used in estimating the event values
- the operating time period during which plant-specific data was gathered
- any assumptions that were made in the data analysis, as well as the bases for the assumptions and their impact on the final results
- raw data records and related interpretations of those records used to derive the data values
- calculations used (including inputs) to produce any data values developed using Bayesian methods

14.3.5 Human Reliability Analysis (HRA)

The objective of human reliability analysis is to identify and evaluate those human actions relevant to the accident scenarios being analyzed. Given the high degree of hardware reliability and redundancy, human interfaces become a critical aspect in causing, preventing and mitigating an accident.

For this task, the following sections provide attributes of selecting an HRA model/method, selecting human events to model, screening/excluding human events, evaluating and quantifying human events, integrating HRA into sequence quantification, and documenting the work.

Selecting HRA Model/Method

Several methods (including databases) are available to perform HRA. A quality PRA considers the strengths and weaknesses of each method and the most appropriate model/method matching the human events and situations being analyzed is selected. Therefore, the model/method selected in a quality PRA has certain inherent characteristics (as described below).

Identifying and Selecting Human Events

Currently, a quality PRA identifies and quantifies relevant errors of omission (errors involving failure to initiate a specific action); however, it does not currently need to address errors of commission (errors involving unintended actions). The relevant errors of omission that are included in a quality PRA are those human actions that can cause a system or component to be unavailable when demanded (referred to as pre-initiators), and those "key" human actions needed to prevent or mitigate core damage given that the initiator has occurred (referred to as post-initiators).

A quality PRA considers all pre-initiator human events that could result in the unavailability of the system or component. At a minimum, these events include restoration errors in returning the system and components to their normal state after completion of testing and maintenance, and miscalibration errors of critical instrumentation (both independent errors and common-cause miscalibration where appropriate). The following criteria are generally used in a quality PRA to identify and select the pre-initiator human events:

- Test, maintenance or calibration is performed on the system or component that is needed to prevent or mitigate core damage.

- The system or component is not needed to physically operate (correctly) when either bringing or operating the reactor ar full power.

- Improper maintenance, restoration or calibration will preclude the system or component from performing its function as modeled in the PRA.

A quality PRA considers both response and recovery post-initiator human events. Response actions include those human actions performed in direct response to the accident (i.e., actions delineated by the emergency operating procedures). Human response actions that are included in a quality PRA are those actions required to manually initiate, operate, control or terminate those system and components needed to prevent or mitigate core damage. The modeled response actions include only those action needed to ensure that the systems or components meet the requirements of the success criteria defined for those systems or components in the systems analysis.

Recovery actions include those human actions performed in ***recovering*** a failed or unavailable system or component. Recovery actions may also include using systems in relatively unusual ways. However, credit for recovery actions are not credited unless at least some procedural guidance is provided or operators receive frequent training that would lead them to perform the required actions. Recovery actions can also include restoration and repair of failed equipment (i.e., hardware failure). In a quality PRA, restoration and repair of loss of off-site power, loss of PCS, loss of diesel generators and loss of DC buses can be credited. Table 8.2-10 of NUREG/CR-4550, Volume 1 provides acceptable values for these events. NUREG-1032 (Ref. 14.7) is also an acceptable source of data for restoration of off-site power. Due to the general lack of acceptable data, restoration and repair of other equipment is not credited in a quality PRA.

The human events selected for evaluation in a quality PRA reflect the actual operating and maintenance practices of the plant. At a minimum, plant walk-throughs, interviews with plant personnel (e.g., training, maintenance, operators, shift supervisor, shift technical advisors), and procedure review are performed in identifying and selecting the human events for a quality PRA. Whenever possible, simulator exercises of the modeled accident sequences are observed to provide important information regarding control room operational practices and crew performance. Similarly, observations of maintenance crew performance is also desirable.

A quality PRA must evaluate both the "diagnosis" and "execution" portion of each post-initiator human event. Diagnosis is usually assumed to include detecting and evaluating a changed or changing condition and then deciding what response is required. Obviously, the complexity can vary, but a diagnosis may entail no more than detecting an indication in the control room and deciding to execute a prescribed response according to symptom-based emergency operating procedures (EOPs). Evaluation of the execution of a human action entails examining the activities to be conducted as indicated by the diagnosis.

In a quality PRA, essentially ***all*** post-initiator human events are assumed to entail a diagnosis phase. Exceptions to evaluating a diagnosis phase include those instances when the diagnosis of a previously modeled human event can be shown to include that for a subsequent event.

Failure to explicitly model and evaluate the execution of a human action is appropriate only when the HRA method being used stipulates that the likelihood of potential execution failures is included in the diagnosis value for certain kinds of events. However, relatively complex actions are not ***a priori*** assumed to be contained within any diagnosis value (e.g., unusual actions performed outside the control room). The application of any HRA method requires the analyst to ensure that the assumptions and characteristics of the method are appropriate for the event being analyzed. Most existing methods provide alternatives for treatment of different types of events.

Screening/Excluding Human Events

There are numerous human events that do not play a "critical" role in either causing, preventing, or mitigating core damage. A screening analysis can be performed in a quality PRA to identify and exclude these events from detailed evaluation.

Human events, such as all pre-initiators, are not excluded from consideration in a quality PRA based on the argument that these events are included in the component hardware data. Many human events (such as miscalibration) occur rarely and are not included in the random failure data. Further, their effects can be subtle in that they impact multiple systems and thus can play a key role in contributing to core damage.

In screening human events, the following criteria are used in a quality PRA:

- A pre-initiator human error that only impacts one system (or train of a system) can be excluded from further consideration if the probability of the human error can be shown to be at least two orders of magnitude lower than the next highest equipment failure probability or unavailability in the associated system.

- Quantitative screening values for pre-initiator human events can be used in the initial accident sequence quantification. The assigned screening values must be high enough to ensure that key human events are not artificially truncated. Therefore, acceptable screening values not less than 1E-2 are used for pre-initiator human events in a quality PRA.

- Quantitative screening values for post-initiator human errors are required in the initial PRA quantification process when the human events are modeled in the event trees as top events or in the fault trees. The screening values assigned are high enough to ensure that the impact of dependencies between events are not underestimated. If screening values are too low and potential dependencies are not considered, important sequences may be truncated and eliminated without their importance being recognized. If screening values are assigned before the initial quantification without any examination of the events and potential dependencies, screening values not less than 0.5 are used in a quality PRA for post-initiator human events (assuming that cut set truncation values around 1E-9/ry are used in the quantification process).

In the final quantification step, if screening values remain for any of the human events, care is taken that this situation does not distort the results. Screening values, by definition, are relatively high probabilities, and when mixed with human events of more realistic values, could erroneously "drive" the results. That is, a sequence could become dominant because it included a human event with a screening value that did not properly represent the actual "reliability" of the operator. In a quality PRA, after the initial quantification, all the human events no in the truncated sequences and cut sets, are quantified with the detailed HRA model.

Evaluating and Quantifying Human Events

In a quality PRA, the actual performance of the operators is reflected in the estimated likelihood of an operator failing to diagnose, perform or properly execute the needed action. Therefore, the quantification of the human events in a quality PRA incorporates plant-specific factors and practices. At a minimum, the following factors are included in a quality PRA:

- the plant "conditions" affecting operator performance including, at a minimum —
  - — the quality (type and frequency) of the operator training, the written procedures, and the administrative controls
  - — the environment (e.g., lighting, heat, radiation) under which the operators work
  - — the accessibility of the equipment requiring manipulation
  - — the necessity, adequacy, and availability of special tools, parts, clothing, etc.
  - — the quality of the human-machine interface

- the time available for the operator to determine and perform the desired action, compared with the time needed to determine and perform the action (In a quality PRA, the available time is accident sequence specific and determined from engineering analyses such as thermal-hydraulic calculations; the needed time is also accident sequence specific and is determined from actual time measurements derived from walk-throughs and simulator observation. The point at which the operators receive relevant indicators is also considered in determining the available time.)

- the expected task demands and likely stress levels for each event

- the potential for additional checks on operator actions (immediate recoveries) and the expected arrival of additional support such as an emergency response team

- dependencies and interfaces between the human events and their relationship to the accident scenario including the following (at a minimum):
  - For pre-initiators, the capability of the operator to impact more than one component, train, or system is considered. (For example, the likelihood of the operator miscalibrating all level and pressure instrumentation simultaneously is considered.)
  - For post-initiators, the human event is evaluated relative to the specific context of the accident progression. Therefore, for different accident sequences, the human event is evaluated for each sequence. Previous human actions and system performance are considered relative to their influence on the human event under consideration. Time dependency is also a factor in the sense that the total available time must be considered across the entire sequence. For example, if most of the total available time is allocated to the first operator action in a sequence, the modeling of the potential success of the remaining actions is impacted.

The following criteria are used in a quality PRA to ensure that no dependencies exists between human events (i.e., the events are truly independent):

- No common "environmental" factors exists (lighting, temperature, etc.).

- No common human-related factors exists (e.g., same/similar procedure, same crew performing multiple calibrations on the same day, etc.).

- All personnel involved in diagnosing or executing the human action or series of human actions are not the same.

Errors made in performance by the original operator can be "recovered" by other plant personnel (e.g., post maintenance verification by a separate operator, role of shift technical advisor, role of emergency response team). In a quality PRA, the total credit for all such "recoveries" does not typically exceed a factor of ten. This limit is based on the uncertainty associated with determining the actual independence of the plant personnel and the ability to precisely quantify human performance, particularly considering all the different uncertainties.

Operators can perform numerous activities to prevent core damage from occurring during an accident. However, the likelihood of these actions can become questionable if too many or unrealistic operator actions are modeled. While all reasonable actions for which time is available can be modeled in a quality PRA, an operator or control room failure in one instance (e.g., failure to follow procedure) has the potential to influence the likelihood of later operator success. Thus, once again, potential dependencies must be considered and, for a given cut set, a total "crew" (both control room and ex-control room operator plus any and all other personnel such as the emergency response team) failure probability of 1E-6 or less is not credited and quantified in a quality PRA.

The above factors are used in determining what data are selected from the various HRA methods in deriving the actual human error probabilities (HEPs). The quantified HEPs are characterized as dictated in the selected HRA method. For example, the Technique for Human Error Rate Prediction characterizes its data as median values with a lognormal distribution. However, the values input into the sequence quantification are most often mean values, therefore, depending on the HRA method being used, conversion to a mean might be required. In a quality PRA, the distribution of an HEP will be checked (and adjusted appropriately) to ensure that the upper bound on the distribution never exceeds 1.0. An acceptable approach used in a quality PRA is the maximum entropy distribution which sets both the upper and lower limits.

An essential aspect in the quantification of the human events is a "sanity" check of the HEPs. In a quality PRA, the analyst reviews the final HEPs relative to each other to check their reasonableness given the plant history and operational practices and experience. For example, the human events wit the relatively higher failure probabilities are generally events involving more complex, difficult activities that are performed under more burdensome, time constrained and stressful circumstances. The human events with the relatively lower failure probabilities are generally events performed under more common, routine, straightforward circumstances.

Integrating HRA Into Sequence Quantification

The human events in a PRA are integrated into the overall model using several methods. Pre-initiator human events are evaluated and included directly in the system fault trees where the process of model quantification correctly accounts for the human error impact on the results. However, post-initiator human errors can be modeled as top events in the accident sequences development (e.g., event trees) or as basic events in the fault trees, and/or may be incorporated directly into the cut sets. However, if post-initiator events are incorporated into the models, care must be taken so that the actual human error probability used in the quantification process addresses dependencies between operator actions, sequence timing, and other factors influencing the human error probability. The attributes of this incorporation are provided in Section 14.3.6.

Documentation of Human Reliability Analysis

The following HRA information is reported in a quality PRA:

- a list or general description of the plant information was used in the HRA
- a list of all human actions evaluated (both pre- and post-initiator)
- a list of all human error probabilities for each human action
- factors used in quantifying the human actions, how they were derived (their bases), and how they were incorporated into the quantification process:
  - time available versus time required
  - dependencies
  - plant-specific performance-shaping factors
  - diagnosis and execution
- sources of data used to quantify human actions
- screening values and their bases
- any assumptions that were made in the human reliability analysis, as well as the bases for the assumptions and their impact on the final results

14.3.6 Accident Sequence Quantification

The objective of the accident sequence quantification task is to quantify the total core damage frequency and its associated uncertainty. The model results include uncertainty analyses and appropriate importance measures and sensitivity analyses, to the extent that these provide additional insights and confidence in the results.

For this task, the following sections provide attributes of selecting the quantification model/code, selecting truncation values, integrating the HRA into the quantification process, estimating uncertainties, computing importance measures and performing sensitivities, and documenting the work.

Selecting the Quantification Model/Code

Several accepted computer codes are available to perform the quantification; however, the computer codes actually used are benchmarked. The computer codes can use the rare event approximation when event probabilities are below 0.1. However, use of the minimal cut set upper bound is always recommended as a minimum to avoid overly pessimistic results. The code is capable of accounting for system successes in addition to system failures in the evaluation of accident sequence cut sets. This can be accomplished using either complimentary logic or the delete term approximation used in many computer programs including the SETS code. In either case, success probabilities of equipment and human actions are used in the computation when the success probability is not close to 1.0.

Initial sequence quantification can be performed using point estimates. The values used for the point estimates are the mean values of the probability distributions for the basic event failure probabilities. As previously indicated, when screening values are used for post-initiator human error probabilities during the initial quantification, they are selected to ensure that no potentially important accident sequence cut sets are eliminated. Cut sets generated from the initial quantification are reviewed to eliminate those that are invalid. Final quantification is performed to replace the post-initiator human screening values with appropriate human error values as discussed subsequently.

Selecting Truncation Values

Truncation is an iterative process of eliminating accident sequences from further consideration, usually based on low frequency of occurrence. This truncation is done to simplify the quantification process and make it less time intensive. Truncation is generally performed at a cut set level during the evaluation of each accident sequence where all cut sets of a frequency less than a sequence-specific value are eliminated. Cut set truncation based on the order of the cut set is not performed in a quality PRA because cut set order is independent of the quantitative significance of the cut set.

The cut set truncation value for each sequence is chosen such that there is at least three-orders of magnitude spread in the retained cut set frequencies both before and after recovery. Sequences with low frequencies can be truncated in either the initial or final quantification process, but the truncation is performed to avoid missing any accident sequences that significantly contribute to the model estimation of total core damage frequency and so that at least 95% of the total core damage frequency and 95% of the early and late release frequencies are expressed in the model results. Also, in a quality PRA, lowering the truncation limit is shown to not significantly increase the model estimation of total core damage and release frequencies. (The increase is less than 1%.)

To avoid discarding important sequences, the impact of truncation is considered both before and after operator recovery actions are applied. The final truncation limits are established through an iterative process demonstrating that the overall model results are not significantly changed and that no important accident sequences are inadvertently eliminated. Typically, in a quality PRA a sequence truncation value that is four orders of magnitude lower than the final core damage frequency (CDF) is used.

Integrating HRA into the Quantification Process

Besides the human error events incorporated directly into the event or fault tree models, events are added during the quantification phase of the analysis to depict the non-recovery probability of proceduralized (or otherwise expected) human actions to mitigate an accident sequence. The number of operator recovery actions added to an accident sequence is limited to "reasonably expected" operator actions. Reasonably expected means that the operator actions are specified in procedures and do not consist of "heroic" actions. Also, as discussed in the previous section, the total credit of post-initiator human actions for a given sequence or cut set is not less than 1E-6 in a quality PRA.

Regardless of the type of human error, care is taken to identify dependencies among multiple human error events that occur in individual cut sets so that the combined human error probability is not optimistically evaluated. This implies that cut set-specific timing and conditional information are used in the calculation and application of post-initiator operator actions and other recovery actions. Such actions are generally not applied at a sequence level.

Estimating Uncertainties

There are two general types of uncertainty. "Parameter uncertainty" results from a lack of knowledge about the failure rates used in the models. "Model uncertainty" occurs when alternative models can be constructed to represent the accident sequence behavior. (This includes concerns about the model completely representing all significant phenomena.)

A quality PRA incorporates parameter uncertainty by propagating the failure rate distributions calculated in the data analysis task through the PRA models. Events in the PRA representing the same component failure with the same failure rate are correlated in the uncertainty analysis. (Correlation can dramatically affect the resulting core damage frequency uncertainty distribution.) To the extent practical, modeling uncertainty is also incorporated into the PRA. This can involve applying weights to different models and propagating the impacts of those models through the entire PRA. An alternative is to perform sensitivity analyses to determine the impact of the different models.

Acceptable methods used in a quality PRA for performing uncertainty analyses include Monte Carlo simulation or the variation known as Latin Hypercube Sampling. Other means of propagating uncertainty (e.g., the method of moments and propagation of discrete probability distributions) are not practical in a quality PRA because of the large size of the required models. The computer codes used for the uncertainty analyses are benchmarked to verify that the results provided are reasonable.

Computing Importance Measures and Performing Sensitivities

The sensitivity of the model results to model boundary conditions and other key assumptions is evaluated using sensitivity analyses to look at key assumptions or parameters both individually and in logical combinations. The combinations analyzed are chosen to account for interactions among the variables affected by the sensitivities are fully accounted for. Areas typically needing evaluation using a sensitivity analysis include modeling assumptions, human error probabilities, common-cause failure probabilities, and safety function success criteria. The results of these sensitivity analyses are needed to provide confidence in the PRA results.

In performing sensitivity analyses, the analyses are not performed by manipulating (requantifying) the "retained" accident sequences and cut sets. The sequences and cut sets that were truncated could potentially be impacted and significantly influence the results (e.g., dominant accident sequences and contributors). Therefore, in a quality PRA, the sensitivity analyses are performed by requantifying the entire PRA model unless it can be shown that only the retained accident sequences and cut sets are impacted.

Importance measure calculations are typically performed in a quality PRA to provide information regarding the contributions of various components and basic events to the model estimation of total core damage frequency. Typical importance measures are Fussel-Vesely, risk achievement, risk reduction, and Birnbaum.

Documentation of the Accident Sequence Quantification

The following information regarding the PRA quantification is documented in a quality PRA:

- a general description of the quantification process including accounting for systems successes, the truncation values used, how recovery and post-initiator human errors were applied, and a description of the computer codes used
- the total plant CDF and contributions from the different initiating events and accident classes
- a list of all the dominant accident sequences and their contributing cut sets (A dominant accident sequence, from a frequency perspective rather than a risk perspective, is one for which the core damage frequency contribution is greater than 1% or the core damage frequency is greater than 1E-8/ry.)
- equipment or human actions that are the key factors in causing the accidents to be non-dominant are listed
- results of all sensitivity studies
- the uncertainty distribution for the total core damage frequency and for each dominant accident sequence
- importance measure results, including at least Fussel-Vesely, risk reduction, and risk achievement
- a list of mutually exclusive events eliminated from the resulting cut sets and their bases for elimination
- a list of all sequences retained after the final quantification, including a brief description of the sequence and its core damage frequency
- records of the actual quantification process such as file manipulations, setting of flags to turn portions of logic either on or off, etc.
- records of the process/results when adding non-recovery terms as part of the final quantification
- records of the cut set review process and any manipulations therein such as eliminating invalid cut sets, requantifying multiple but dependent human errors in the same cut set, etc.

14.4 Attributes of a 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 from the reactor fuel to the environment given the occurrence of an accident that damages the reactor core. To satisfy this objective, a quality Level 2 PRA is comprised of three major parts:

- A quality Level 1 PRA, which provides information regarding the accident sequences to be examined and their frequency. The attributes of performing the analyses associated with this aspect of a PRA are described above, in Section 14.3, and are not discussed further here.
- A structured and comprehensive evaluation of containment performance in response to the accident sequence identified from the Level 1 analysis.
- A quantitative characterization of radiological release to the environment that would result from accident sequences that involve leakage from the containment pressure boundary.

Figure 14.3 illustrates each of these parts and indicates how they relate to each other conceptually. A detailed description of the attributes of conducting the technical analyses associated with each part is provided below.

An *ideal* assessment of containment performance in response to such accidents would be to perform a deterministic calculation with a validated, first-principles model of accident progression. Such a calculation would generate a time-history of loads imposed on the containment pressure boundary. These loads would then be compared against structural performance limits of the containment. If the loads exceed the performance limits, the containment would be expected to fail; conversely, if the performance limits surpass the calculated loads, the containment would be expected to survive. In such an ideal assessment, the overall frequency of accidents resulting in a release to the environment would simply be the frequency of accident sequences in which the calculated containment loads exceed the performance limits.

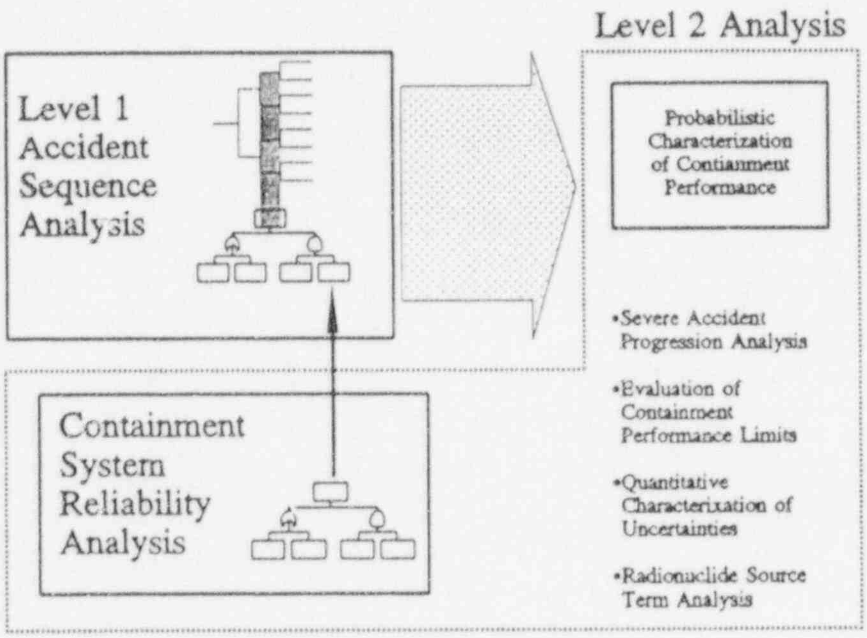

Figure 14.3 Relationship among the major parts of a quality Level 2 PRA.

Unfortunately, neither the current knowledge regarding many aspects of severe accident progression nor (albeit to a lesser extent) the knowledge regarding containment performance limits is sufficiently precise to conduct such an analysis. Rather, in a quality PRA, an assessment of containment performance is performed in a manner that explicitly considers imprecise knowledge of severe accident behavior, the resulting challenges to containment integrity, and the capacity of the containment to withstand various challenges. Therefore, the potential for a release to the environment is typically expressed in terms of the conditional probability of containment failure (or bypass) for the spectrum of accident sequences (determined from Level 1 PRA analysis) that proceed to core damage.

Figure 14.4 indicates how the conditional probability of containment failure is calculated. For each Level 1 core damage accident sequence (frequency, F_i), the probability of the various containment failure modes are calculated. For example, the probability of early containment failure (ef_i), containment bypass (bp_i), late containment failure (lf_i) and no containment failure (nf_i) are determined. For the example shown in Figure 14.4, Accident Sequence 1 completely bypasses the containment and thus the conditional probability of bypass given the occurrence of this accident is unity. These characteristics could result from an accident such as an interfacing system LOCA. Alternatively, Accident Sequence 2 could result in several different containment failure modes or no containment failure. For this accident, the probability of early failure (0.5) could be caused by several mechanisms such as overpressure, shell melt-through and others. Containment bypass (0.1) could be the result of induced steam generator tube rupture (for PWRs only). Whether the containment fails late (0.2) or not at all (0.2) depends on several factors including the operability of containment heat removal systems. Once the probabilities of these containment failure modes has been determined for each accident sequence, the probabilities conditional on total core damage are calculated.

The probability of early containment failure conditional on core damage ($CCFP_{ef}$) is determined by summing ($i=1 \rightarrow n$) the early failure probabilities for all accident sequence weighted by their respective frequencies (F_i). The summation is then divided by the total core damage frequency (CDF).

$$CCFP_{ef} = \frac{\sum_{i=1}^{n} [ef_i][F_i]}{CDF}$$

A similar approach is used to determine the conditional probabilities of bypass accidents, late containment failure and no containment failure.

In addition to estimating the probability of a radiological release to the environment, the Level 2 portion of a quality PRA of a nuclear reactor characterizes the resulting release in terms of magnitude, timing, and other attributes important to an assessment of off-site accident consequences. This information has two purposes. First, it provides a quantitative scale for ranking the relative severity of various accident sequences; secondly, it represents the "source term" for a quantitative evaluation of off-site consequences (i.e., health effects, property damage, etc.), which are estimated in the Level 3 portion of a PRA (refer to Section 14.5).

This section describes the attributes of a quality Level 2 PRA analysis, emphasizing the *scope* and *level of detail* associated with major elements a Level 2 analysis, rather than the specific methods used to assemble a probabilistic model. This approach is deliberately used because several different methods have been used to generate and display the probabilistic aspects of severe accident behavior and containment performance. By far, the most common methods are those that use standard event and/or fault tree logic structures; however, some practitioners use other

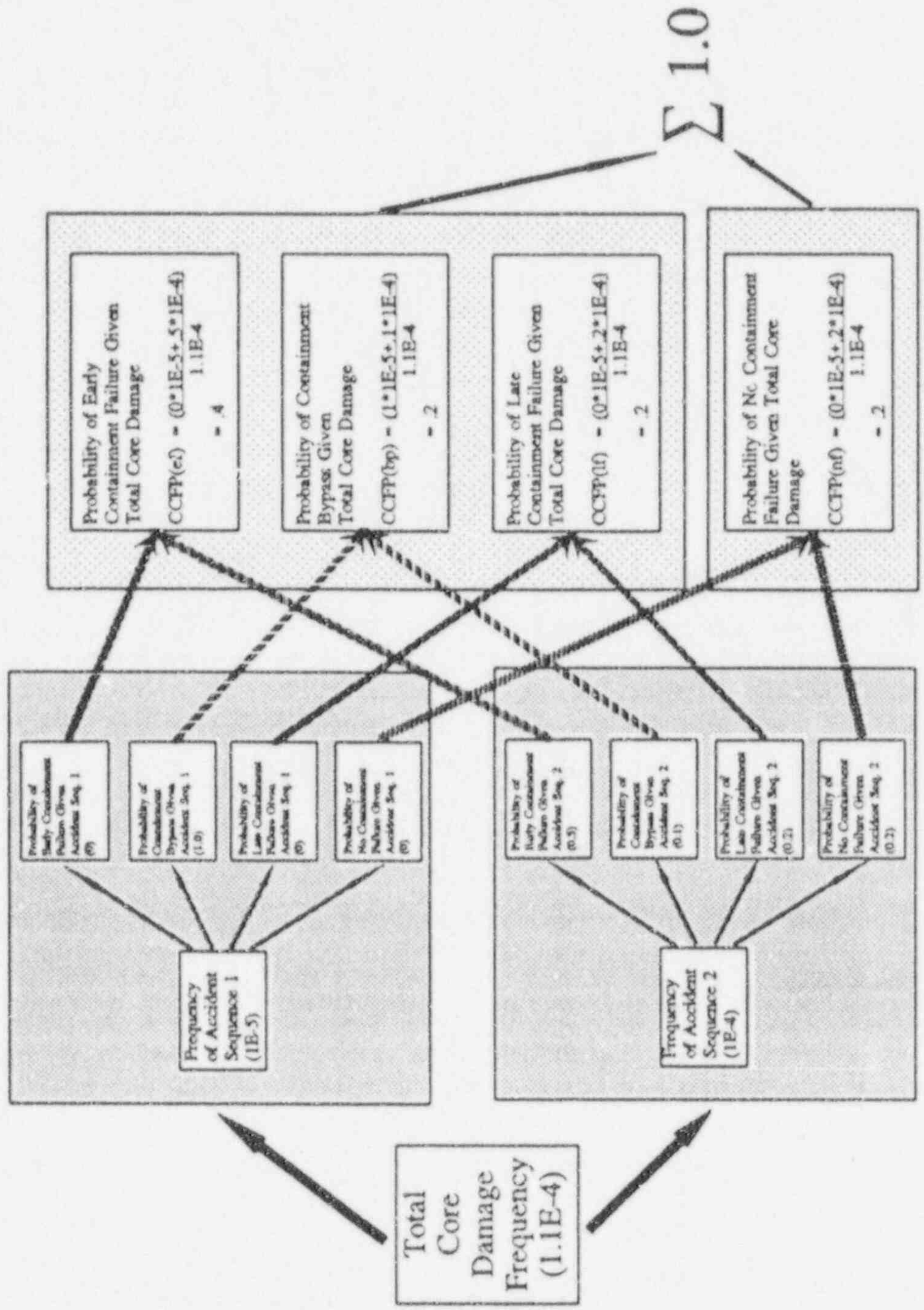

Figure 14.4 Conditional probability of containment failure.

techniques. Further, the specific way in which ostensibly similar logic structures are organized and solved (numerically) can differ substantially from one study to another, primarily as a result of differences in quantification techniques and associated computer software offered by vendors of PRA services. In principle, any of these methods can be used to produce a quality Level 2 PRA ***provided*** that they encompass the ***scope*** and level of detail described below.

As indicated above, the two major products of a quality Level 2 PRA are (1) the conditional probability of containment failure or bypass for accident sequences that proceed to core damage and (2) a characterization of the radiological source term to the environment for each sequence resulting in containment failure or bypass. Although the analyses conducted to generate these products are closely coupled, the characteristics of the analyses used to generate them are best described separately. Hence, the characteristics of a probabilistic evaluation of containment performance are described in Section 14.4.1; characteristics of the accompanying estimates of radionuclide release are described in Section 14.4.2.

14.4.1 Evaluation of Containment Performance

Although the specific analysis tasks within various Level 2 PRAs may be organized in a slightly different manner, the following three critical elements are always developed:

- an assessment of the range of challenges to containment integrity (i.e., determination of failure mechanisms and range of structural loads)

- characterization of the capacity of the containment to withstand challenges (i.e., determination of performance limits)

- a process of organizing and integrating the uncertainties associated with these two evaluations to estimate the ***probability*** that the containment would fail (or be bypassed) for a given accident sequence

Attributes of developing each of these elements are described below.

14.4.1.1 Assessment of Challenges to Containment Integrity

The primary objective of this element of a Level 2 PRA is to characterize the type and severity of challenges to containment integrity that may arise during postulated severe accidents. A quality analysis to determine these characteristics acknowledges the dependence of containment response on details of the accident sequence. Therefore, a critical first step is developing a structured process for defining the specific accident conditions to be examined. Attributes of determining which of the myriad accident sequences generated by Level 1 PRA analysis must be examined via computer simulations, or other methods of severe accident progression analysis are defined in two parts:

(1) attributes of reducing the large number of accident sequence developed for Level 1 PRA analysis to a practical number for detailed Level 2 analysis

(2) attributes of performing and coupling the assessment of containment system performance (i.e., reliability analysis) with Level 1 accident sequence analyses

Defining the Accident Sequences

Because of the diversity and redundancy of safety systems designed to prevent and/or mitigate potential accident conditions in a nuclear plant, multiple failures must occur for an accident to proceed far enough to damage the reactor fuel. The primary purpose of a Level 1 PRA analysis is to identify the specific combinations of system or component failures (i.e., accident sequence cut sets) that would allow core damage to occur.

Unfortunately, the number of cut sets generated by a Level 1 analysis is very large (typically greater than 10,000). It is impractical to evaluate severe accident progression and resulting containment loads for each of these cut sets. As a result, the common practice is to group the Level 1 cut sets into a sufficiently small number of "plant damage states" to allow a practical assessment of the challenges to containment integrity resulting from the full spectrum of accident sequences.

The number of plant damage states produced by this grouping (or "binning") process cannot be established ***a priori***. Rather, a quality Level 2 PRA first defines the attributes of an accident sequence that represent important initial or boundary conditions to the assessment of severe accident progression or containment response or characteristics of system operation that can have an important effect on the resulting environmental source term. Example attributes are shown in Table 14.1.

Table 14.1 Example attributes for grouping accident sequence cut sets.

Attribute	Possible states
Accident Initiator	• Large, Intermediate, or Small LOCAs • Transients • LOCA outside the containment pressure boundary • Steam Generator Tube Rupture
RCS Pressure at the Onset of Core Damage	• High • Low
Status of Emergency Coolant Injection Systems	• Operate in injection mode, but fail upon switchover to recirculation cooling • Fail to operate in injection mode
Status of Steam Generators (PWRs)	• Auxiliary feedwater operates/fails • Secondary isolated/depressurized
Status of Residual Heat Removal Systems	• Operate • Failed
Status of Containment at Onset of Core Damage	• Isolated • Not isolated
Status of Containment Safeguard Systems	• Sprays always operate/fail or are available if demanded • Sprays operate in injection mode, but fail upon switchover to recirculation cooling • Fan coolers always operate/fail or are available if demanded • Containment venting system(s) operate/fail • Hydrogen control system(s) operate/fail

The functional effect of the specific failures represented by the terms in each accident sequence cut set are then mapped into possible plant damage states according to the binning attributes. There is no "unique" list of attributes against which this exercise should be conducted for a quality Level 2 analysis; Table 14.1 simply provides examples, not an exhaustive list. A comprehensive list of attributes for representative PWR and BWR Level 2 analyses can be found in NUREG/CR-4551, Volume 3 (Ref. 14.8) and Volume 4 (Ref. 14.9), respectively. Although many of these attributes can be applied generically across many different reactor/containment designs, special attributes are often necessary to address plant-specific design features (e.g., isolation condenser operation in certain BWRs.) In a quality Level 2 PRA, any characteristic of the plant response to a given initiating event that would influence either subsequent containment response or the resulting radionuclide source term to the environment is represented as an attribute in the plant damage state binning scheme. These characteristics include the following:

- ***The status of systems that have the capacity to inject water to either the reactor vessel or the containment cavity (or drywell pedestal).*** Defining system status simply as "failed" or "operating" is not sufficient in a quality Level 2 analysis. Low-pressure injection systems may be available but not operating at the onset of core damage because they are "dead-headed" (i.e., reactor vessel pressure is above their shutoff head). Such states are distinguished from "failed" low-pressure injection to account for the capability of dead-headed systems to discharge after reactor vessel failure (i.e., providing a mechanism for flooding the reactor cavity).

- ***The status of systems that provide heat removal from the reactor vessel or containment.*** Careful attention is paid to the interactions between such systems and the coolant injection systems. For example, the status properly accounts for limitations in the capability of dual-function systems such as the RHR system in most BWRs (which provides pumping capacity for LPCI and heat removal for suppression pool cooling).

- ***Recoverability of "failed" systems after the onset of core damage.*** Typical recovery actions include restoration of AC power to active components and alignment of nonsafety-grade systems to provide (low-pressure) coolant injection to the reactor vessel or to operate containment sprays. Constraints on recoverability (such as no credit for repair of failed hardware) are defined in a manner that is consistent with recovery analysis in the Level 1 PRA.

- ***The interdependence of various systems for successful operation.*** For example, if successful operation of a low-pressure coolant injection system is necessary to provide adequate suction pressure for successful operation of a high-pressure coolant injection system, failure of the low-pressure system (by any mechanism) automatically renders the high-pressure system unavailable. This information may only be indirectly available in the results of the Level 1 analysis, but is explicitly represented in the plant damage state attributes *if* recovery of the low-pressure system (after the onset of core damage) is modeled.

Several subtle aspects of the mapping of accident sequence cut sets from the Level 1 analysis to plant damage states used as input to a Level 2 analysis are worth noting at this point:

- The ***entire*** core damage frequency generated by the Level 1 accident sequence analysis is carried forward into the Level 2 analysis. The reason for conserving the CDF is to allow capture of the risk contribution from low-frequency, high-consequence accident sequences.

- The mapping is performed at the cut set level, not the accident sequence level. There are several reasons for this level of detail:

- Depending on the level of detail represented in the Level 1 accident sequence event trees, it may be impossible to properly capture the effects of support system failures and other dependencies among the various binning attributes without reviewing the basic events that caused a system failure.

- Recovery of failed systems ***after the onset of core damage*** is considered in the containment performance assessment of a quality Level 2 PRA. For this activity to be modeled correctly, system failures that are "recoverable" are distinguished from failures that are "not recoverable." This information typically lies only within the sequence cut sets. Note that the definition of recoverable is consistent with the recovery analysis performed in the Level 1 PRA.

- To appropriately model human reliability related to operator actions that occur after the onset of core damage, information regarding prior operator performance (i.e., prior to the onset of core damage) is carried forward from the Level 1 analysis. Again, this information typically lies only within sequence cut sets.

- For some accident sequences, the status of all systems may not be determined from the sequence cut sets. For example, if the success criteria for a large break LOCA in a PWR require successful accumulator operation, the large LOCA sequence cut sets involving failure of all accumulators will contain no information about the status of other coolant injection systems. However, realistic resolution of the status of such systems often provides a mechanism for representing accident sequences that are arrested before substantial core damage and radionuclide release occur. In a Level 2 analysis, these systems are not simply assumed to operate as designed. Rather, their failure frequencies are estimated in a manner that preserves relevant support system dependencies. These are then numerically combined with the sequence cut set frequency from the Level 1 analysis.

Assessment of Containment System Performance

The reliability of systems whose primary function is to maintain containment integrity during accident conditions is incorporated into the accident sequence analysis performed during a quality Level 1 PRA. Such systems may include containment isolation, fan coolers, distributed sprays, and hydrogen igniters. An assessment of the reliability of these systems is incorporated into a quality Level 2 analysis to ascertain whether they would operate as designed to mitigate containment response during core damage accidents. The methods, scope, and technical rigor used to evaluate the reliability of these systems are comparable to those used in the Level 1 analysis of other "front-line" systems (refer to Section 14.3.3). Fault tree models (or other techniques) for estimating failure probabilities are developed and linked directly to the accident sequence models from the Level 1 PRA. This linkage is necessary to properly capture the important influence of mutual dependencies between failure mechanisms for containment systems and other systems. Obvious examples include support system dependencies, such as electrical power, component cooling water, and instrument/control air. Other dependencies that need to be represented in a manner consistent with the Level 1 system models are more subtle however, as illustrated by the following examples:

- Indirect failure of containment systems caused by harsh environmental conditions (resulting from failure of a support system) are represented in the assessment of containment system reliability. One important example is failure of reactor or auxiliary building room cooling causing the failure of containment systems because of high ambient temperatures.

- The influence of containment system operation prior to the onset of core damage is accounted for in the evaluation of system operability after the onset of core damage. For example, consider an accident sequence in which containment sprays successfully initiate on an automatic signal early in an accident sequence. If later in the sequence (but prior to the onset of core damage) emergency operating procedures direct reactor operators to terminate containment spray operation to allow realignment of emergency coolant injection systems, the configuration of the containment spray system (and thus its reliability) differ from a sequence in which containment sprays would not have been demanded prior to the onset of core damage.

- The human reliability analysis associated with manual actuation of containment systems (e.g., hydrogen igniters) accounts for operator performance during earlier stages of an accident sequence. This analysis follows the same practices used in the Level 1 analysis as described in Section 14.3.5.

The long-term performance of containment systems is also evaluated although the issues to be considered may differ substantially from those listed above. This evaluation accounts for degradation of the environment within which systems are required to operate as an accident sequence proceeds in time. Examples of factors that may arise after the onset of core damage include:

- loss of NPSH for coolant pumps due to suppression pool heat up in BWRs

- plugging of fan cooler inlet plena as a result of the accumulation of aerosols (generated perhaps as a consequence of core-concrete interactions) in PWRs

- failure of systems with components internal to the containment pressure boundary as a result of high temperatures or pressure associated with hydrogen combustion

In all cases, the assessment of failure probability for containment systems are based on realistic equipment performance limits rather than bounding (design-basis or equipment qualification) criteria.

Evaluation of Severe Accident Progression

The element of a Level 2 PRA that often receives the most 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 the Modular Accident Analysis Program (MAAP) (Ref. 14.10) or MELCOR (Ref. 14.11)] 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 a range of potential 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 a quality Level 2 PRA. There are several reasons for this:

(1) Many of the models embodied in severe accident analysis codes address highly uncertain phenomena. In each case, certain assumptions are made (either by the model developers or the code user) regarding controlling physical processes and the appropriate formulation of models that represent them. In some instances, the importance of these assumptions can be tested via parametric analysis. However, the extent to which the results of any code calculation can be demonstrated to be robust in light of the numerous

uncertainties involved is severely limited by practical constraints of time and resources. Therefore, the assumptions inherent in many code models remain untested.

(2) None of the integral severe accident codes contain models to represent all accident phenomena of interest. For example, models for certain hydrodynamic phenomena such as buoyant plumes, intra-volume natural circulation, and gas-phase stratification, are not represented in most integral computer codes. Similarly, certain severe accident phenomena, such as dynamic fuel-coolant interactions (i.e., steam explosions) and hydrogen detonations, are not represented.

(3) It is simply impractical to perform an integral calculation for all severe accident sequences 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 other prior PRAs or severe accident studies, focused engineering analyses of particular issues, and experimental data. The manner in which each of these sources of information are used in a quality Level 2 PRA is described below.

The following are used to determine the number of plant specific calculations that would be performed using an integral code to support a quality Level 2 PRA:

- At least one integral calculation (addressing the complete time domain of severe accident progression) is performed for each plant damage state. However, this may not be practical depending on the number of plant damage states developed according to the above discussion. At a minimum, calculations are performed to address the dominant accident sequences (i.e., those with the highest contribution to the total core damage frequency). Calculations are also performed to address sequences that are anticipated to result in relatively high radiological releases (e.g., containment bypass scenarios).

- In addition to the calculations of a spectrum of accident sequences described above, several sensitivity calculations are performed to examine the effects of major uncertainties on calculated accident behavior. For example, multiple calculations of a single sequence are performed in which code input parameters are changed to investigate the effects of alternative assumptions regarding the timing of stochastic events (such as operator actions to restore water injection), or the models used to represent uncertain phenomena (such as the size of the opening in containment following over-pressure failure). These calculations provide information that is essential to the quantitative characterization of uncertainty in the Level 2 probabilistic logic models (refer to the discussion of logic model development and assignment of probabilities below).

Table 14.2 lists phenomena that can occur during a core meltdown accident and involve considerable uncertainty. This list was based on information in NUREG-1265 (Ref. 14.12), NUREG/CR-4551 (Ref. 14.13) and other studies. It is recognized that considerable disagreement persists within the technical community regarding the magnitude (and in some cases, the specific source) of uncertainty in several of the phenomena listed in Table 14.2. A major objective of the expert panels assembled as part of the research program that culminated in NUREG-1150 (Ref. 14.14) was to translate the range of technical opinions within the severe accident research community into a quantitative measure of uncertainty in specific technical issues. In a quality Level 2 PRA, the results of this effort are used as guidance for defining the range of values of uncertain modeling parameters to be used in the sensitivity calculations described above.

Table 14.2 Severe accident phenomena.

Phenomena	Characteristics of accident phenomena
Hydrogen generation and combustion	• Enhanced steam generation from melt/debris relocation • Steam starvation caused by degraded fuel assembly flow blockage • Clad ballooning • Recovery of coolant injection systems • Steam/hydrogen distribution within containment • De-inerting due to steam condensation or spray operation
Induced failure of the reactor coolant system pressure boundary	• Natural circulation flow patterns within the reactor vessel upper plenum, hot legs, and steam generators • Creep rupture of hot leg nozzles, pressurizer surge line, and steam generator U-tubes
Debris bed coolability and core-concrete interactions	• Debris spreading/depth on the containment floor • Crust formation at debris bed surface and effects on heat transfer • Debris fragmentation and cooling upon contact with water pools • Steam generation and debris oxidation
Fuel coolant interactions	• Potential for dynamic loads to bounding structures • Hydrogen generation during melt-coolant interaction
Melt/debris ejection following reactor vessel failure	• Melt/debris state and composition in the lower head • Mode of lower head failure • Debris dispersal and heat transfer following high-pressure melt ejection
Shell melt-through failure in Mark I containments	• Melt spreading dynamics • Effects of water • Shell heat transfer and failure mechanism

A fundamental design objective of the integral severe accident analysis codes used to support Level 2 PRA (e.g., MAAP, MELCOR) is that they be fast running. Efficient code operation is necessary to allow sensitivity calculations to be performed within a reasonably short time and with minimal resources. One consequence of this objective, however, is that many complex phenomena are modeled in a relatively simple manner or, in some cases, are not represented at all. Therefore, a quality Level 2 PRA addresses the inherent limitations of integral code calculations in two respects. First, the importance of phenomena not represented by the integral codes are evaluated by some other means (i.e., either application of specialized computational models or experimental investigation). Secondly, the effects of modeling simplification are examined by comparisons with mechanistic code calculations.

There are obvious practical benefits to applying or adapting results of completed studies of severe accident progression in other plants to the PRA of interest. If the applicability of such studies can be demonstrated, substantial savings can be achieved by eliminating unnecessary (repetitive) analysis. Application of analyses from studies of similar plants is common in quality Level 2 PRAs. For example, in the NUREG-1150 analysis of Peach Bottom (a BWR 4 with a Mark I containment), results of severe accident progression calculations performed for the Browns Ferry plant (another BWR 4 with a Mark I containment) were used as a surrogate for plant-specific calculations regarding selected aspects of in-vessel core melt progression. However, such analyses can not completely supplant the plant-specific evaluations described above.

The prerequisite for applying results of studies for another plant is a demonstration of similarity in plant design and operational characteristics such that the same results would be generated if plant-specific analyses were performed. Demonstration of similarity involves a direct comparison of key plant design features and, if necessary, scaling analysis. Examples of features to be included in such a comparison are listed in Table 14.3. The effects of differences in these design features is examined, and techniques for adapting or scaling the results of the surrogate analyses developed.

Table 14.3 Example plant design/operational parameters to be compared to demonstrate similarity for use as surrogate analysis [examples for BWR study].

Component	Design characteristics of component
Reactor Core	• Nominal Power • Number of Fuel Assemblies • Number of Fuel Rods per Assembly • Core Mass (UO_2, Cladding, Misc. support structures)
Reactor Vessel	• Inside Diameter • Height • Nominal Operating Pressure • Number of Safety/Pressure Relief Valves • Safety / Relief Valve relief valve design flow rate • Reactor Coolant System Liquid Volume
Containment Drywell	• Total Free Volume • Design Pressure • Nominal Internal Operating Pressure • Atmosphere composition • Reactor Pedestal Floor Area • Drywell Floor Area • Radius from Reactor Centerline to Drywell Wall • Penetration arrangement and construction • Water Capacity before Spill-over into Wetwell • Concrete (floor) composition • Position of Vents/Downcomers to Wetwell Relative to Reactor Centerline and Pedestal
Containment Wetwell	• Total Volume • Suppression Pool Mass • Nominal Suppression Pool Temperature • Number and Size of Vents/Downcomers connecting Drywell and Wetwell • Vent/Downcomer Diameter

In summary, evaluating severe accident progression involves a complex process of plant-specific sensitivity studies using integral codes, mechanistic code calculations, use of prior calculations, experimental data and expert judgement. Examples of this process are given for each of the phenomena in Table 14.2 in the following sections.

Hydrogen Generation and Combustion

After the onset of severe core damage, metals in the core and vessel, predominantly zirconium and to a lesser extent stainless steel, react with steam to form hydrogen. For accidents that progress to vessel failure, additional hydrogen is generated from the interactions between core debris released from the vessel and concrete structures below the vessel (i.e., core-concrete interactions). Depending on the course of the accident progression, various amounts of hydrogen can be produced. Due to its noncondensibility and combustible nature, the hydrogen produced during the accident poses a significant threat to the containment structure in the event of a hydrogen deflagration or detonation event.

Hydrogen phenomena was identified in the NUREG-1150 study as an area where considerable uncertainty existed and, hence, issues associated with hydrogen phenomena were addressed by NUREG-1150 expert panels. The generation of in-vessel hydrogen was addressed by the In-Vessel Expert Panel (Ref. 14.13) whereas issues concerned with the combustion of hydrogen were addressed by the Containment Loads Expert Panel (Ref. 14.15). The distributions developed by these panels represented the current state of knowledge at the time of the

NUREG-1150 study. Since these expert panels explicitly considered the uncertainties associated with key phenomena (and also accounting for uncertainties in the initial and boundary conditions) and developed distributions that characterized these uncertainties, the information from these panels provides a convenient and important framework for assessing uncertainties for this application. To facilitate this discussion, hydrogen phenomena is divided into hydrogen production and hydrogen combustion.

The uncertainty in the amount of hydrogen produced during the in vessel phase of a severe core damage accident was addressed in the NUREG-1150 study by the In-Vessel Expert Panel. Results from this panel are provided in NUREG/CR-4551, Volume 2, Part 1 for both PWRs and BWRs. In that report distributions are provided for the percentage of in-vessel zirconium that is oxidized. For the PWRs, the percentage of zirconium oxidized was analyzed considering two parameters: the RCS pressure and whether accumulator discharge occurs before, during or after core melt. For the BWRs, the two parameters considered were the pressure in the reactor pressure vessel and on whether coolant injection was restored to the vessel during the core degradation process. For both the PWRs and BWRs, the probability distributions are broad, covering nearly the entire range of zirconium oxidation. Some of the key phenomenological uncertainties that lead to these broad distributions included the presences, timing and extent of core blockages that can result in steam starvation in the core region, the timing and temperature during core relocation, the formation of eutectics between the oxides of zirconium and uranium, the extent of stainless steel oxidation in the upper plenum, the strength and duration of natural circulation, and the amount of hydrogen produced during quenching processes.

Clearly, as evident by the NUREG-1150 distributions, there is considerable uncertainty in the amount of zirconium oxidized in vessel and the use of a single number (for example from a MELCOR or MAAP code calculation) is not adequate for a quality PRA. While these codes can all predict the amount of hydrogen produce during the accident, the amounts that they predict often vary since they model the phenomena differently. Similarly, a series of sensitivity evaluations with a single code is usually not sufficient to assess the uncertainties since typically a single code will not include all of the relevant phenomena. Instead, a quality PRA includes distributions such as those developed by the In-Vessel Expert Panel to characterize the uncertainty in the amount of hydrogen generated during the in-vessel phase of the accident.

Uncertainties in hydrogen combustion phenomena were addressed in the NUREG-1150 study by the Containment Loads Expert Panel. For PWRs, hydrogen combustion is a more significant concern in the smaller volume ice condenser containments than it is in the large volume containments. For BWRs, hydrogen combustion[14.4] is

[14.4]Here combustion refers to combustion in the containment. However, following failure of the containment, combustion of hydrogen in the reactor buildings surrounding Mark I and Mark II containments can also be a concern. Combustion in the reactor building surrounding a Mark I plant was addressed by the Containment Loads Expert Panel and is discussed in Section 5.3 of NUREG/CR-4551, Vol. 2, Rev. 1, Part 2.

typically only a concern for plants with Mark III containments since both the Mark I and Mark II containments are inerted during normal operation and past PRAs have considered this method reliable under most accident conditions. Hence, the Containment Loads Panel assessed the combustion phenomena at the Grand Gulf plant (BWR, Mark III) and the Sequoyah plant (PWR, Ice Condenser). Results from this assessment are documented in Sections 5.1 and 5.2 in NUREG/CR-4551, (Ref. 14.17). Information regarding the incorporation of this information into the NUREG-1150 PRAs are provided in NUREG/CR-4551, Volume 5 (Ref. 14.16) for the Sequoyah plant analysis and in NUREG/CR-4551, Vol. 6 (Ref. 14.17) for the Grand Gulf plant analysis. The major elements of the hydrogen combustion phenomena, as modeled in a PRA, are discussed below.

The concentration of hydrogen in the containment, as well as the concentration of oxygen and steam, are important parameters because many aspects of the combustion process depend on them. The combustion limits, both for deflagrations and detonations, depend on these concentrations. Similarly, the frequency of ignition, the frequency of a deflagration to detonation transition (DDT), and the magnitude of any resulting loads also depend on these concentrations. The concentrations depend on the amount of hydrogen generated during the accident, the amounts of air and steam in the containment atmosphere, and on the distribution of these gases in the containment. The amount of hydrogen generated in the vessel was discussed above. The amount of steam in the containment atmosphere and the distribution of gases in the containment depend on characteristics of the containment and the accident sequence. For example, the degree of compartmentalization of the containment, the availability of flow paths between compartments, the type of accident sequence (i.e., LOCA, station blackout, long-term loss of containment heat removal), and the availability and/or effectiveness of containment heat removal systems (such as containment sprays, ice condenser, and vapor suppression pools) can all affect the concentration of these gases in the containment.

Since this information tends to be specific to the plant and the accident sequences being analyzed, relevant deterministic calculations are used to provide guidance when determining the amount of steam in the containment atmosphere and for determining the distribution of gases in the various compartments. Considering these characteristics, the concentrations of hydrogen, oxygen, and steam are determined for each containment volume where combustion is a concern. These concentrations are then used to determine whether a combustible mixture exists. Of particular concern are local areas where hydrogen can accumulate and thereby form a mixture that can potentially detonate. For compartmentalized containments, such as ice condenser containments, there can be considerable uncertainty in these concentrations for the various compartments necessitating the development of uncertainty distributions. A discussion of these uncertainties for an ice condenser containment can be found in Section 5.2 of NUREG/CR-4551, Vol. 2, Part 2, Rev. 1. The calculation of the total concentration of hydrogen in containment takes into account both the hydrogen produced in vessel and ex vessel (through the core-concrete interaction) in cases where the containment does not fail at vessel breach.

Combustible mixtures that form in the containment can be ignited from a number of sources including igniters, AC powered equipment, and hot surfaces. For situations where there are no identifiable ignition sources, such as during a station blackout, it is still possible for a combustible mixture of hydrogen to ignite since ignition requires very little energy. The ignition of hydrogen under this last condition was addressed in NUREG-1150 by the Containment Loads Expert Panel. Results from this panel are provided in Section 5.1 for the Grand Gulf plant (BWR, Mark III) and Section 5.2 for the Sequoyah plant (PWR, Ice Condenser) NUREG/CR-4551, Vol. 2, Part 2. The panel provided distributions that characterized the uncertainty in the ignition frequency for situations where AC power is not available in the containment. For the PWR plant, the ignition frequency was analyzed considering different concentrations of hydrogen and different locations (i.e., dome, upper plenum, ice condenser). For the BWR plant, the ignition frequency was analyzed considering different concentrations of hydrogen and different pressures in the reactor vessel (impacts how the hydrogen is released). The ignition frequency was assessed to increase with an increase in hydrogen concentration. The probability distributions for both plants are broad reflecting considerable uncertainty in this issue. A major source of this uncertainty was the lack of identifiable ignition sources and a lack of experimental results. Ignition sources considered by the experts included static sparks and sparks created when the ice condenser doors bang together.

Quasi-static loads from hydrogen combustion events were assessed in the NUREG-1150 study by the Containment Loads Expert panel for both the Grand Gulf and the Sequoyah plants. Generally, the experts based the peak overpressures on the adiabatic isochoric complete combustion model and then corrected the pressures to account for burn completeness, heat transfer and expansion into non-participating compartments. For the PWR plant, the experts felt that the uncertainty in the peak overpressure was small compared to the uncertainties in the hydrogen concentration and ignition frequencies and, hence, a single estimate of the peak overpressure as a function of hydrogen concentration was provided instead of a probability distribution. These estimates are provided in Section 5.2 in NUREG/CR-4551, Vol. 2, Part 2, Rev. 1. For the BWR plant the uncertainty in the peak overpressure was driven by the uncertainty in the burn completeness (although it was also acknowledged by these experts that the uncertainty in the ignition frequency is a key uncertainty associated with the hydrogen combustion phenomena) and, hence probability distributions were developed. The distributions developed by this panel are provided in Section 5.1 in NUREG/CR-4551, Volume 2, Part 2. The distributions for the peak pressure in the containment depend on the global concentrations of hydrogen and steam in the containment. Two levels of steam were considered: a low steam case and a high steam case. The high steam case includes the situation where the containment was initially steam inert but a combustible atmosphere was created when the containment sprays were restored and steam was condensed.

Since the publication of NUREG-1150, some additional research has been conducted on combustion of hydrogen-air-steam mixtures in condensing environments (Ref. 14.18). In these experiments ignition was provided by thermal igniters. These experimental results provide relevant information that was not available during the NUREG-1150 study and are accounted for when assessing the peak pressure in a rapidly condensing environment with igniters available.

Hydrogen detonations in the Grand Gulf and Sequoyah containments were also addressed by the Containment Loads Expert Panel and are discussed in Sections 5.1 and 5.2 of NUREG/CR-4551, Vol. 2, Part 2, Rev. 1, respectively. The panel assessed the frequency that a deflagration to detonation transition (DDT) would occur. The DDT frequency was analyzed considering different locations within the containment and different concentrations of hydrogen within each location. The probability distributions that characterize the uncertainty in the DDT frequency are broad for both the BWR and the PWR plants. Given that a detonation occurs, the expert panel also assessed the resulting peak impulse. The geometry in the area where the ignition occurs is a key uncertainty that affects the likelihood that a DDT will occur. Similarly, the interaction between the detonation wave and structures is a key uncertainty that affects the peak impulse.

Induced Failure of the RCS Pressure Boundary

In a station blackout accident sequence with failure of secondary heat removal, the failure to remove decay heat while the primary system is at high pressure creates a possibility for a temperature-induced creep rupture of the steam generator (SG) tubes. This phenomenon can potentially lead to a containment bypass event (i.e. opening a direct path out of containment).

The possibility of a temperature-induced rupture of the SG tubes is affected by several factors including the thermal hydraulic conditions at various locations in the primary system, which determine the temperatures (and the time at those temperatures) and the pressures to which the SG tubes are subjected as the accident progresses. Other relevant factors include the effective temperature required for creep rupture failure of the SG tube, and the presence of pre-existing defects in the SG tubes which increase the likelihood of rupture.

Thermal hydraulic code calculations for this accident sequence show that temperature-induced steam generator tube rupture (SGTR) requires the same conditions that lead to induced hot leg failure. In fact, the temperatures in the hot leg, the surge line, and SG tubes are closely coupled. However, based on these calculations, the temperatures in the surge line are higher than the SG tube temperatures because of counter-current flow in the hot leg. Under these circumstances, the surge line, followed by the hot leg, will fail earlier than the SG tubes leading to a temperature-induced LOCA that will depressurize the RCS. If this occurs, a temperature-induced SGTR is unlikely. On the other hand, some experts believe that the presence of pre-existing defects in the SG tubes can change this conclusion and lead to a higher likelihood of SG tube rupture.

In NUREG-1150, this issue was treated in the expert elicitation process. All experts agreed that hot leg failure, including failure of the surge line, was much more likely to occur before a rupture of a steam generator tube. Two experts felt that pre-existing defects in the SG tubes could lead to a higher probability of SG tube rupture. The third expert felt that due to the long time lag between temperatures in the hot leg and the SG tubes, the frequency of temperature-induced SGTR was so small that it could be expressed as a (small) constant value regardless of pre-existing defects.

A conditional probability distribution of temperature-induced SGTR was developed in NUREG-1150 by aggregating the individual distributions provided by three experts. A discussion of the phenomenon and the assignment of the conditional probability distribution of temperature-induced SGTR is contained in NUREG/CR-4551, Vol. 2. This distribution was applied in the accident progression event trees developed for the Zion and for the Surry plants in NUREG-1150. The Zion and Surry reports [NUREG/CR-4551, Vol. 7, (Ref. 14.19) and NUREG/CR-4551, Vol. 3, respectively] can be consulted for information related to how the conditional probability distribution of temperature-induced SGTR should be applied to obtain the split fractions for the containment event tree for this issue.

Debris Bed Coolability and Core-Concrete Interactions

An important issue in virtually all Level 2 PRAs is whether the molten core debris released from the reactor vessel into the reactor cavity is coolable if water is added to the debris, and the impact of the interaction between debris, structure, and water on the containment performance. Debris coolability is an important issue because if the debris is brought to a coolable geometry, the only source for containment pressurization will be the generation of steam from boiloff of the overlying water. This is a slow process and, in the absence of containment heat removal, would result in very late containment failure allowing ample time for remedial actions. Furthermore, a coolable debris geometry would limit basemat penetration.

In addition, if a coolable debris bed is formed in the cavity or pedestal and makeup water is continuously supplied then interactions between the core debris and concrete with be minimized and release of radioactive material from this source would be avoided. Even if molten core-concrete interaction (CCI) occurs, a continuous overlaying pool of water can substantially reduce the release of radioactive material to the containment.

There is, however, a significant likelihood that, even if a water supply is available, the core debris will not be coolable and, therefore, will attack the concrete basemat. If CCI does occur (i.e., the debris bed is not coolable), experimental results indicate that the presence or absence of an overlying water pool does not have much effect on the downward progression of the melt front. The projected consequences of basemat melt-through are, however, minor in comparison with above ground failure of the containment boundary.

The mechanisms that govern debris coolability are conduction heat transfer, shrinkage cracking, gas sparging and melt eruption, and crust failure under the weight of the water. Experimental research (Ref. 14.20) has been carried out to investigate this issue. These tests include the SWISS-1 (Ref. 14.21) and -2, FRAG-3 and -4, (Ref. 14.22) WETCOR-1, (Ref. 14.23) and MACE (Ref. 14.24) series of tests. This experimental information would be considered in a quality PRA when developing distributions for the likelihood of forming a coolable debris bed for a particular plant configuration. The expert panel convened for NUREG-1150 specifically for molten core-concrete interaction issues is an example of how major input parameters for this issue are quantified.

Fuel-Coolant Interactions

If an accident at a nuclear plant leads to sufficient severe core damage result in large-scale changes in core geometry, debris from the damaged core would at some point begin to relocate into the lower plenum of the reactor vessel. If an appreciable amount of water remained in the lower plenum, molten core material falling into it could potentially cause a steam spike and, if sufficiently severe, an explosion. The latter requires sudden disintegration of part of the melted core material into fine particles. The rapid increase in surface area could generate large amounts of steam through a sudden increase of heat transfer, before pressure relief through expansion could take place.

The energy required to "trigger" a steam explosion is greater at high reactor vessel pressures than at low pressure. Also, the efficiency of the explosion is reduced substantially by intervening structures and by the presence of unmelted material relocating into the water pool.

The pressure surge in the pool of water could possibly cause some water and portions of the core and lower core support structures to be accelerated upward as a slug. If this were to impact the vessel head with sufficient energy, it could break the bolts holding the vessel head in place, and could accelerate the head upward as a missile, threatening the containment structure. This mode of early containment failure was identified in the Reactor Safety Study, WASH-1400 (Ref. 14.25), as the alpha mode containment failure.

For an accident leading to a severely damaged core, the probability of a steam explosion causing early containment failure was assumed in WASH-1400 to be between 0.1 and 0.01. In 1985, the first Steam Explosion Review Group (SERG-1) workshop was held to systematically evaluate the alpha-mode failure issue. The experts who participated in that workshop reviewed the then current understanding of the potential for containment failure from in-vessel steam explosion, and reached a nearly unanimous opinion that the probability of alpha-mode failure is less than that used in WASH-1400. NRC-sponsored research carried out since 1985 has played a major role in developing an understanding of the key physical processes involved in energetic fuel coolant interactions (FCI).

In June 1995, the second Steam Explosion Review Group (SERG-2) workshop was held to revisit the alpha-mode failure issue, and to evaluate the current understanding of other FCI issues of potential risk significance, such as shock loading of the lower head and ex-vessel support structures. The estimates of failure probability expressed by SERG-2 experts were generally an order of magnitude lower than the SERG-1 estimate.

Melt/Debris Ejection Following Reactor Vessel Failure

In certain severe accidents, the failure of the RPV can occur while the RCS is at elevated pressure. In these accidents, the expulsion of the molten core debris and blow down of the RCS could lead to a very rapid and efficient heat transfer to the containment atmosphere, possibly accompanied by oxidation reactions and hydrogen combustion that further enhances the energy transfer. These processes, which lead to containment pressurization, are collectively referred to as direct containment heating (DCH). Overpressurization resulting from DCH is a significant containment challenge that can lead to early containment failure.

The NUREG-1150 severe accident risk study was the first systematic attempt to treat DCH from a PRA perspective by integrating sequence probabilities with uncertainties associated with initial/boundary conditions and phenomenological uncertainties associated with predicting containment loads.

Since the completion of the NUREG-1150 study, advances have been made in the ability to predict the probability of containment failure by DCH in PWRs. The NRC has identified DCH as a major issue for resolution in the Revised Severe Accident Research plan and has sponsored analytical and experimental programs for understanding the key physical processes in DCH. An extensive database resulted from scaled counterpart experiments conducted by Sandia National Laboratory (SNL) and Argonne National Laboratory (ANL). This database has allowed the development and validation of simple analytical models for predicting the containment loads. In particular, a two-cell equilibrium model was developed based on insight from the experimental program and has been used in the DCH issue resolution process. The two-cell equilibrium model takes into account the coherence between the entrained debris and the RCS blowdown steam.

The results of a probability assessment of DCH-induced containment failure for the Zion Nuclear Power Plant were published in NUREG/CR-6075 (Ref. 14.26) and its supplement. NUREG/CR-6338 (Ref. 14.27) used the methodology and scenarios described in NUREG/CR-6075 to address the DCH issue for all Westinghouse plants with large volume containments, including 34 plants with large dry containments and seven plants with subatmospheric containments. DCH loads versus strength evaluation were performed in a consistent manner for all plants. The phenomenological modeling was closely tied to the experimental database. Plant-specific analyses were performed, but sequence uncertainties were enveloped by a small number of splinter scenarios without assignment of probabilities. The results of screening calculations reported in NUREG/CR-6338 indicate that only one plant showed a CCFP based on the mean fragility curves greater than 0.001. The CCFP for this one plant was found to be less than 0.01. These results can, therefore, be used for Level 2 PRAs for Westinghouse plants with large volume containments. For BWRs and other PWR plants, the methodology reported in NUREG/CR-6338 for performing load/strength evaluations using the plant-specific input to the two-cell equilibrium model or appropriate containment analysis codes, can be used to provide a PRA-integrated perspective on this issue. For plants with ice-condenser containments, it is believed that the ice chamber in the plant can, to a certain extent, trap dispersing core debris, and provide cooling to moderate the effect of DCH. BWRs with the Mark I and Mark II containments are inerted with nitrogen and therefore hydrogen combustion is not of concern when addressing the DCH issue.

Shell Melt-through Failure in Mark I Containments

The issue of shell failure in Mark I containments concerns the possibility that molten corium from the reactor vessel will come in contact with and then breach the containment shell. Because this shell is the containment pressure boundary, such a breach would constitute containment failure. This phenomenon is characterized by an intricate combination of physical processes associated with the conditions of the melt release (e.g., oxidic or metallic composition and the quantity of release), the melt spreading dynamics, the core-concrete interactions, the shell heat transfer, and the shell failure criteria.

To address the shell melt issue in NUREG-1150, a panel of experts was convened to provide input as to the probability of shell melt for five scenarios: (1) low and medium flow with water, (2) low and medium flow without water, (3) high flow with water, (4) high flow without water and two of three parameters (pressure, fraction of metal, and superheat) high, and (5) high flow without water and two of three parameters (pressure, fraction of metal, and superheat) low. The individual elicitations were then averaged and presented in Table 6-1 of NUREG/CR-4551, Volume 2, Part 2.

In a more recent report, Theofanous et al., published a probabilistic methodology in NUREG/CR-6025 (Ref. 14.28) as an overall systematic approach for addressing the Mark I shell melt-through issue. Details of the approach are contained in the reference and are not recounted here. Because of major uncertainties in the core melt progression or the in-vessel portion of the accident, the authors of NUREG/CR-6025 attempted to envelope the shell melt-through issue by means of two melt release scenarios. Scenario 1 was characterized by a sudden massive release of a significant portion of the release (oxidic), followed by a gradual release of the remaining quantity. This scenario was intended to simulate the behavior where core plate failure was delayed, melt accumulated prior to its release in the lower plenum, and a local lower head failure occurred soon after this release. Scenario 2 was characterized by a gradual release (metallic) over an extended period. This scenario was intended to simulate the behavior dominated by early local failure(s), and the release of core material through such, as they become molten in the lower plenum. Both scenarios were limited to low pressure and were analyzed with and without the drywell flooded.

The above approaches are examples of generating probabilistic information on shell melt-through. A quality Level 2 PRA would investigate plant specific design features, including pedestal door arrangement (and relative alignment of downcomers), drywell floor area and sump volumes, and in particular, the amount of fuel in the reactor and the downcomer entrance height above the drywell floor. The downcomer entrance height affects not only the amount of water attainable on the floor, but more importantly, if the amount of fuel is sufficient that melt can run directly into the downcomer, liner failure is virtually assured. The probabilities of shell melt-through should apply to a steel lined reinforced concrete containment, however, if sufficient technical basis is provided, the effective failure size in the containment structure may be adjusted accordingly (though there should be no credit given for "self-healing" of the containment boundary).

Documentation of Assessment of Challenges to Containment Integrity

In general, sufficient information in the documentation of analyses performed to determine and characterize the challenges to containment integrity is provided that allows an independent analyst to reproduce the results. At a minimum, the following information is documented for a quality PRA:

Documentation of accident sequences assessed:

- a thorough description of the procedure used to group (bin) individual accident sequence cut sets into plant damage states, or other reduced set of accident scenarios for detailed Level 2 analysis
- a listing of the specific attributes or rules used to group cut sets

- a listing and/or computerized data base providing cross reference for all cut sets to plant damage states and vice versa

Documentation of containment system performance assessments:

- a description of information used to develop containment systems' analysis models and link them with other system reliability models (This documentation is prepared in the same manner as that generated in Level 1 analysis of other systems as discussed Section 14.3.3).

Documentation of analyses of severe accident progression:

- a description of plant-specific accident simulation models (e.g., for MAAP or MELCOR)including extensive references to source documentation for input data

- a listing of all computer code calculations performed and used as a basis for quantifying any event in the containment probabilistic logic model including a unique calculation identifier or name, a description of key modeling assumptions or input data used, and a ref :rence to documentation of calculated results (If input and/or output data are archived for quality assurance records or other purposes, an appropriate reference to calculation archive records is also provided.)

- a description of key modeling assumptions selected as the basis for performing "base case" or "best-estimate" calculations of plant response and a description of the technical bases for these assumptions

- a description of plant-specific calculations performed to examine the effects of alternate modeling approaches or assumptions

- if analyses of a surrogate (i.e., 'similar') plant are used as a basis for characterizing any aspect of severe accident progression in the plant being analyzed, references to, or copies of documentation of the original analysis, and a description of the technical basis for assuming applicability of results

- for all other original engineering calculations, a sufficiently complete description of the analysis method, assumptions and calculated results is prepared to accommodate an independent (peer) review

14.4.1.2 Establishing Containment Performance Limits

The objective of this element of a Level 2 PRA is to determine the limits (or capacity) that the containment can withstand given the range and magnitude of the potential challenges. These challenges take many forms, including internal pressure rises (that occur over a sufficiently long time frame that they can be considered "static" in terms of the structural response of the containment), high temperatures, thermo-mechanical erosion of concrete structures, and under some circumstances, localized dynamic loads such as shock waves and internally generated missiles. Realistic estimates for the capacity of the containment structure to withstand these challenges are generated to provide a metric against which the likelihood of containment failure can be estimated.

In a quality Level 2 PRA, the attributes of the analyses necessary to characterize containment performance limits are consistent with those of the containment load analyses against which they will be compared.

- They focus on *plant-specific* containment performance (i.e., application of reference plant analyses is generally inadequate).

- They consider design details of the containment structure such as:
    - containment type (free-standing steel shell; concrete-backed steel shell; pre-stressed, post-tensioned, or reinforced concrete)
    - the full range of penetration sizes, types, and their distribution (equipment and personnel hatches, piping penetrations, electrical penetration assemblies, ventilation penetrations)
    - penetration seal configuration and materials
    - discontinuities in the containment structure (shape transitions, wall anchorage to floors, changes in steel shell or concrete reinforcement)

- They consider interactions between the containment structure and neighboring structures (the reactor vessel and pedestal, auxiliary building(s), and internal walls).

A thorough assessment of containment performance generally begins with a structured process of identifying potential containment failure modes (i.e., mechanisms by which integrity might be violated). This assessment commonly begins by reviewing a list of failure modes identified in PRAs for other plants to determine their applicability to the current design. Such a list was incorporated in the NRC's guidance for performing an IPE (Ref. 14.29). This review is then supplemented by a systematic examination of plant-specific design features and emergency operating procedures to ascertain whether additional, unique failure modes are conceivable. For each plausible failure mode, containment performance analyses are performed using validated structural response models, as well as plant-specific data for structural materials and their properties.

For many containment designs, over-pressure has been found to be a dominant failure mechanism. In a quality Level 2 PRA, the evaluation of ultimate pressure capacity is performed using a plant-specific, finite-element model of the containment pressure boundary including sufficient detail to represent major discontinuities such as those listed above. The influence of time-varying containment atmosphere temperatures is taken into account by performing the calculation for a reasonable range of internal temperatures. To the extent that internal temperatures are anticipated to be elevated for long periods of time (e.g., during the period of aggressive core-concrete interactions), thermal growth and creep rupture of steel containment structures is taken into account.

The characterization of containment performance limits is not simply a matter of defining a threshold load at which the structure "fails." A quality Level 2 PRA attempts to distinguish between structural damage that results in "catastrophic failure" of the containment from damage that results in significant leakage[(14.5)] to the environment. Leakage is often characterized by a smaller opening (i.e., one that may not preclude subsequent increases in containment pressure). Failure to isolate the containment is also considered. It is very important to assess both the location and size of the containment failure because of the implications for the source term calculation, e.g., given the same in-vessel and ex-vessel releases inside containment, a rupture in the drywell of a Mark II containment will result in higher releases to the environment than a leak in the wetwell.

Unfortunately, current models for the response of complex structures to even "simple" loads (such as internal pressure) are not sufficiently robust to allow simultaneous prediction of a failure threshold and resulting failure size. This is particularly true for structures composed of non-homogeneous materials with highly non-linear mechanical properties such as reinforced concrete. As a result, calculations to establish performance limits are supplemented with information from experimental observations of containment failure characteristics and expert judgment. Examples of this process can be found in NUREG-1150.

[14.5]Significant leakage is defined relative to the design basis leakage for the plant. Leakage rates greater than 100 times the design basis have been found risk significant in past studies.

The NUREG-1150 Expert Panel for Structural Response Issues assessed the containment overpressure failure issue for the Peach Bottom, Sequoyah, Surry and Zion plants. The assessments of the expert panel are documented in (Ref. 14.30). Two of these plants have free-standing steel containments and two have reinforced concrete containments. In addition to the distributions the expert panel provided for overpressure failure loads for these containment structures, the panel also provided conditional probabilities for failure location and failure mode (leak, rupture or catastrophic rupture). Both containment types were considered to be vulnerable to the propagation of cracks into ruptures. For a single containment, the panel assessed the conditional probability of multiple failure locations and sizes. For example, six different location/size failures (failure modes) were obtained for overpressure failure for the Peach Bottom containment: 1) wetwell leak or 2) rupture, no suppression pool bypass (discontinuity strains at T-stiffeners), 3) wetwell rupture, suppression pool bypass (membrane failure), 4) drywell leak (bending strain at the downcomers), 5) drywell head leak (gasket failure), and 6) drywell rupture (in main body near penetration due to loss of concrete wall back support).

Failure location and size by dynamic pressure loads and internally generated missiles are also probabilistically examined. The structural response expert panel for NUREG-1150 assessed the size and location of the containment breach by dynamic pressure loads for Grand Gulf (reinforced concrete) and Sequoyah (free-standing steel). Both leaks and ruptures were predicted to occur in the containment response to detonations at Grand Gulf, and ruptures were predicted to occur at Sequoyah. Alpha mode failure (for all NUREG-1150 plants) and steel shell melt-through of a containment wall by direct contact of core debris (for Peach Bottom and Sequoyah) were treated as rupture failures of containment in NUREG-1150.

Basemat melt-through is generally treated as a leak in most Level 2 PRAs because of the protracted times involved as well as the predicted radionuclide retention in the soil. If a bypass of containment, such as an interfacing systems LOCA, is predicted to occur, then its effective size and location (e.g., probability that the break is submerged in water) are also estimated in order to perform the source term calculations.

Documentation of Containment Performance Limits

In general, sufficient information in the documentation of analyses performed to establish quantitative containment performance limits is provided that allows an independent analyst to reproduce the results. At a minimum, the following information is documented for a quality PRA:

- a general description of the containment structure including illustrative figures to indicate the general configuration, penetration types and location, and major construction materials

- a description of the modeling approach used to calculate or otherwise define containment failure criteria

- if computer models are used (e.g., finite element analysis to establish over-pressure failure criteria), a description of the way in which the containment structure is nodalized including a specific discussion of how local discontinuities, such as penetrations, are addressed

- if experimentally-determined failure data are used, a sufficiently detailed description of the experimental conditions to demonstrate applicability of results to plant-specific containment structures

14.4.1.3 Probabilistic Modeling of Containment Performance

One feature that distinguishes a quality Level 2 PRA from other, less comprehensive assessments is the way in which uncertainties are represented in the characterization of containment performance[14.6]. In particular, explicit and quantitative recognition is given to uncertainties in the individual processes and parameters that influence severe accident behavior and attendant containment performance. These uncertainties are then quantitatively integrated by means of a probabilistic logic structure that allows the conditional probability of containment failure to be quantitatively estimated, as well as the ***uncertainty*** in the containment failure probability.

Two elements of such an assessment are described below. First, the characteristics of the logic structure used to organize the various contributors to uncertainty are described. However, the ***major*** distinguishing element of a full-scope approach to characterizing containment performance is the assignment and propagation of ***uncertainty distributions*** for major events in the logic model. The key phrase here is uncertainty distributions (i.e., point estimates of probability are not universally applied to the logic model). Characteristics of these distributions and the manner in which they are used in a typical logic model are described later in this section.

Logic Structures for Tracking Alternative Accident Progressions

The primary function of a "containment event tree," or any other probabilistic model evaluating containment performance, is to provide a structured framework for organizing and ranking the alternative accident progressions that may evolve from a given core damage sequence. In developing this framework, whether it be in the form of an event tree, fault tree or other logic structure, several elements are necessary to allow a rigorous assessment of containment performance:

- Explicit recognition of the important time phases of severe accident progression. Different phenomena may control the nature and intensity of challenges to containment integrity and the release and transport of radionuclides as an accident proceeds in time. The following time frames are of particular interest to a Level 2 analysis:

  — ***After the initiating event, but before the onset of core damage.*** This time period establishes important initial conditions for containment response after core damage begins.

  — ***After the core damage begins, but prior to failure of the reactor vessel lower head.*** This period is characterized by core damage and radionuclide release (from fuel) while core material is confined within the reactor vessel.

  — ***Immediately following reactor vessel failure.*** Prior analysis of containment performance suggests that many of the important challenges to containment integrity occur immediately following reactor vessel failure. These challenges may be short-lived, but often occur only as a direct consequence of the release of molten core materials from the reactor vessel immediately following lower head failure.

[14.6]Uncertainties in the estimation of fission product source terms are also represented in a full-scope Level 2 PRA, however, this topic is discussed in Section 14.4.2.

— *Long-term accident behavior*. Some accident sequences evolve rather slowly and generate relatively benign loads to containment structures early in the accident progression. However, in the absence of some mechanism by which energy generated within the containment can be safely rejected to the environment, these loads may steadily increase to the point of failure in the long-term.

When linked end-to-end, these time frames constitute the outline for most probabilistic containment performance models. Within each time frame, uncertainties in the occurrence or intensity of governing phenomena are systematically evaluated.

- Consistency in the treatment of severe accident events from one time frame to another. Many phenomena may occur during several different time frames of a severe accident. However, certain limitations apply to the composite (integral) contribution of some phenomena over the *entire accident sequence* and these are represented in the formulation of a probabilistic model.

  A good example is hydrogen combustion in a PWR containment. Hydrogen generated during core degradation can be released to the containment over several time periods. However, an important contribution to the uncertainty in containment loads generated by a combustion event is the total mass of hydrogen involved in a particular combustion event. One possibility is that hydrogen released to the containment over the entire in-vessel core damage period is allowed to accumulate without being burned (perhaps) as a result of the absence of a sufficiently strong ignition source. Molten core debris released to the reactor cavity at vessel breach could represent a strong ignition source, which would initiate a large burn (assuming the cavity atmosphere is not steam inert). Because of the mass of hydrogen involved, this combustion event might challenge containment integrity. Another possibility is that while the same total amount of hydrogen is being released to the containment during in-vessel core degradation, a sufficiently strong ignition source exists to cause several small burns to occur *prior to vessel breach*. In this case, the mass of hydrogen remaining in the containment atmosphere at vessel breach would be very small in comparison to the first case, and the likelihood of a significant challenge to containment integrity at that time should be correspondingly lower. Therefore, the logic for evaluating the probability of containment failure associated with a large combustion event occurring at the time of vessel breach is able to distinguish these two cases and preclude the possibility of a large combustion event if hydrogen was consumed during an earlier time frame.

- Recognition of the interdependencies of phenomena. Most severe accident phenomena and associated events require certain initial or boundary conditions to be relevant. For example, a steam explosion can only occur if molten core debris comes in contact with a pool of water. Therefore, it may not be meaningful to consider ex-vessel steam explosions during accident scenarios in which the drywell floor (BWR) or reactor cavity (PWR) is dry at the time of vessel breach. Logic models for evaluating containment performance capture these and many other such interdependencies among severe accident events and phenomena. Explicit representation of these interdependencies provides the mechanism for allowing complete traceability between a particular accident sequence (or plant damage state) and a specific containment failure mode.

Containment Logic Structure Quantification

There are many approaches to transforming the technical information concerning containment loads and performance limits to an estimate of failure probability, but three approaches appear to dominate the literature. In the first (least rigorous) approach, qualitative terms expressing various degrees of uncertainty are translated into quantitative (point estimate) probabilities. For example, terms such as "likely" or "unlikely" are assigned numerical values (such as 0.9 and 0.1). Superlatives, such as "very" likely or "highly" unlikely, are then used to suggest degrees of confidence that a particular event outcome is appropriate. The subjectivity associated with this method is controlled to some extent by developing rigorous guidelines for the amount and quality of information necessary to justify progressively higher confidence levels (i.e., probabilities approaching 1.0 or 0.0). Nonetheless, this method is ***not*** considered an appropriate technique for assigning probabilities to represent the state of knowledge uncertainties[14.7] in a quality PRA. Among its weaknesses, this approach simply produces a point estimate of probability and is not a rigorous technique for developing probability distributions.

The second technique involves a convolution of paired probability density functions. In this technique, probability density functions are developed to represent the distribution of credible values for a parameter of interest (e.g., containment pressure load) and for its corresponding failure criterion (e.g., ultimate pressure capacity). This method is more rigorous than the one described above in the sense that it explicitly represents the uncertainty in each quantity in the probabilistic model. The basis for developing these distributions is the collective set of information generated from plant-specific integral code calculations, corresponding sensitivity calculations, other relevant mechanistic calculations, experimental observations, and expert judgment. The conditional probability of containment failure (for a given accident sequence) is then calculated as the intersection of the two density functions (see Figure 14.5).

While this technique provides an explicit treatment of uncertainty at intermediate stages of the analysis, it still ultimately generates a point estimate for the probability of containment failure caused by a particular mechanism.

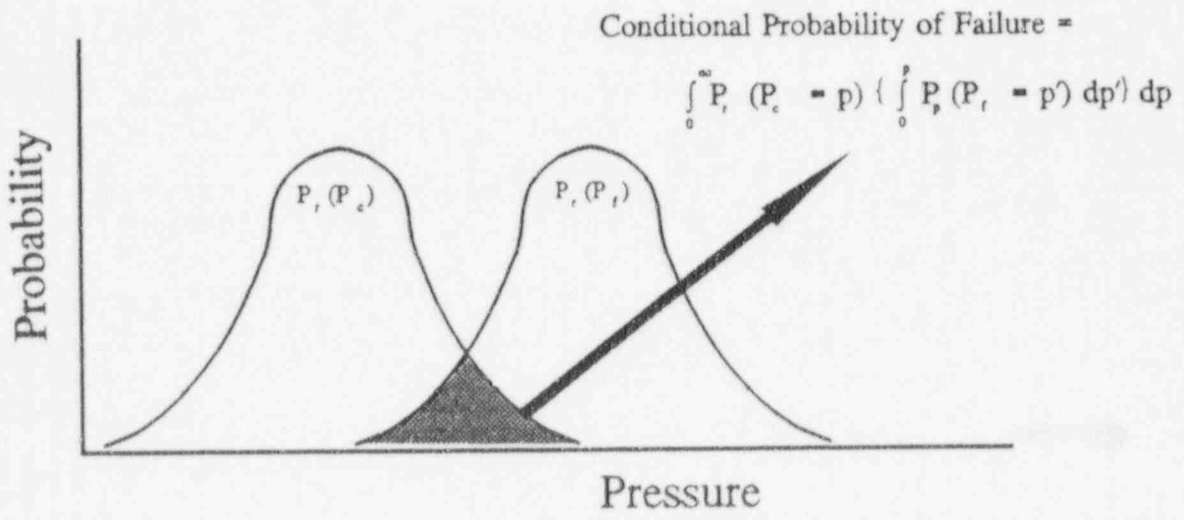

Figure 14.5 Probability density functions for containment peak pressure (P_c) and failure pressure (P_f)

[14.7]Such uncertainties tend to dominate a Level 2 PRA, rather than uncertainty associated with random behavior.

The contributions to (and magnitude of) uncertainty in the final (total) containment failure probability is discarded in the process.

The third technique involves adding an additional feature to the technique described above. That is, the probability density functions representing uncertainty in each term of the containment performance logic model are propagated throughout the entire model to allow calculation of statistical quantities such as importance measures. One means for accomplishing this objective is the application of Monte Carlo sampling techniques (such as Latin Hypercube). The application of this technique to Level 2 PRA logic models, pioneered in NUREG-1150, (Ref. 14.15) accommodates a large number of uncertain variables. Other techniques have been developed for specialized applications, such as the direct propagation of uncertainty technique developed to assess the probability of containment failure as a result of direct containment heating in a large dry PWR.(Ref. 14.28) However, these other techniques are constrained to a small number of variables and are not currently capable of applications involving the potentially large number of uncertain variables addressed in a quality Level 2 PRA.

Documentation of Probabilistic Modeling of Containment Performance

The following documentation is generated to describe the process by which the conditional probability of containment failure is calculated:

- a listing and description of the structure of the overall logic model used to assemble the probabilistic representation of containment performance (Graphical displays of events trees, fault trees or other logic formats are provided to illustrate the logic hierarchy and event dependencies.)

- a description of the technical basis (with complete references to documentation of original engineering analyses) for the assignment of all probabilities or probability distributions with the logic structure

- a description of the rationale used to assign probability values to phenomena or events involving subjective, expert judgment

- a description of the computer program used to exercise the logic model and calculate final results

14.4.2 Radionuclide Release Characterization

The second, albeit equally important, product 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 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, order-of-magnitude estimates of the size of release for a few representative radionuclide species provide a satisfactory scale for ranking accident severity. In a quality Level 2 PRA, however, the characterization of radionuclide release to the environment provides sufficient information to completely define the "source term" for calculating off-site health and economic consequences for use in a Level 3 PRA. Further, the rigor required of the evaluation of radionuclide release, transport, and deposition directly parallels that used to evaluate containment performance:

- Source term analyses (deterministic computer code calculations) reflect plant-specific features of system design and operation. In particular, the models used to calculate radionuclide source terms faithfully represent plant-specific characteristics such as fuel, control material, and in-core support structure composition and spatial distribution; configuration and deposition areas of primary coolant system and containment structures; reactor cavity (or drywell floor) configuration and concrete composition; and topology of transport pathways from the fuel and/or core debris to the environment.

- Calculations of radionuclide release, transport, and deposition represent sequence-specific variations in primary coolant system and containment characteristics. For example, reactor vessel pressure during in-vessel core melt progression and operation (or failure) of containment safeguard systems such as distributed sprays are represented in manner that directly accounts for their effects on radionuclide release and/or transport. The procedure for organizing the numerous accident sequences generated in a Level 1 PRA into a reasonably small number of groups that exhibit similar radionuclide release characteristics was described in Section 14.4.1.

- Uncertainties in the processes governing radionuclide release, transport, and deposition are quantified. Uncertainties related to radionuclide behavior under severe accident conditions are quantified to characterize uncertainties in the radionuclide source term associated with individual accident sequences. This is achieved in the same way uncertainties for the phenomena governing severe accident progression are used to characterize uncertainty in the probability of containment failure (described above).

The specific manner in which radionuclide source terms are characterized in a quality Level 2 analysis is described first. Attributes of coupling the evaluation of radionuclide release to analyses of severe accident progression for particular sequences are also described. Finally, attributes of addressing uncertainties in radionuclide source terms are described.

Definition of Radionuclide Source Terms

The analysis of health and economic consequences resulting from an accidental release of radionuclides from a nuclear plant (in a Level 3 PRA) requires specification of several parameters (from a Level 2 PRA) that define the environmental source term. Ideally, the following information is developed:

- the time at which a release begins

- the time history of the release of all radioisotopes that contribute to early (deterministic) and late (stochastic) health consequences

- the elevation (above local ground level) at which the release occurs

- the energy with which the release is discharged to the environment

- the size distribution of radioactive material released in the form of an aerosol (i.e., particulate)

As in many other aspects of a comprehensive PRA, it is impractical to generate this information for the full spectrum of accident conditions produced by Level 1 and 2 analyses. To address this constraint, several simplifications are made in a quality Level 2 analysis. In particular, the following assumptions are typically made regarding the radioactive material of interest:

- All isotopes of a single chemical element are released from fuel at the same rate.

- Chemical elements exhibiting similar properties in terms of their measured rate of release from fuel, physical transport by means of fluid advection, and chemical behavior in terms of interactions with other elemental species and bounding structural surfaces can be effectively modeled as one composite radionuclide species. Typically, the specific properties of a single (mass dominant) element are used to represent the properties of all species within a group.

The combination of these two assumptions leads to a radionuclide grouping scheme that reduces the total number of modeled radionuclide species to nine groups, as shown in Table 14.4.

Table 14.4 Radionuclide grouping scheme used in a quality Level 2 PRA.

Group	Representative element	Elements represented by the group	Important isotopes within the group
1	Xe	Xe, Kr	Xe-133, Xe-135, Kr-85, Kr-85M, Kr-87, Kr-88
2	I	I, Br	I-131, I-132, I-133, I-134, I-135
3	Cs	Cs, Rb	Cs-134, Cs-136, Cs-137, Rb-86
4	Te	Te, Sb, Se	Te-127, Te-127M, Te-129, Te-129M, Te-131, Te-132, Sb-127, Sb-129
5	Sr	Sr	Sr-89, Sr-90, Sr-91, Sr-92
6	Ru	Ru, Rh, Co, Mo, Tc, Pd	Ru-103, Ru-105, Ru-106, Rh-105, Co-58, Co-60, Mo-99, Tc-99M
7	La	La, Y, Zr*, No, Nd, Pr, Am, Mc, Sm	La-140, La-141, La-142, Y-90, Y-91, Y-92, Y-93, Zr-95, Zr-97, Nb-95, Nd-147, Pr-143, Am-241, Cm-242, Cn-244
8	Ce	Ce, Np, Pu	Ce-141, Ce-143, Ce-144, Np-239, Pu-238, Pu-239, Pu-240, Pu-241
9	Ba	Ba	Ba-139, Ba-140
*Radionuclide Zirconium (not the structural metal)			

Although the species listed above are released from fuel in their elemental form, it is firmly established that many species quickly combine with other elements to form compounds as they migrate away from the point of release. The formation of these compounds and the associated change in the physio-chemical properties of individual radionuclide groups are taken into account in the analysis of radionuclide transport and deposition. In particular, volatile radionuclides species, such as iodine and cesium, may be transported in more than one chemical form - each with different properties that affect their transport.

Chemical forms of these radionuclide groups represented in the source term analysis of a full-scope PRA include:

Radionuclide Group	Chemical forms for transport
I	I_2, CH_3I, HI [vapor] CsI [aerosol]
Cs	CsOH, CsI [aerosol]

A second simplification in the characterization of radionuclide release involves the treatment of time-dependence. Temporal variations in radionuclide release are calculated as a natural product of deterministic source term calculations. However, in a Level 2 PRA these variations are reduced to a series of discrete periods of radiological release, each of which is described by a starting time, a duration, a (constant) release rate, and a release energy. For example, results of an integral severe accident/source term code calculation might suggest the radiological release rate shown as the solid line in Figure 14.6. The continuous release rate is simplified to represent major characteristics or the release history such as an early, short-lived, large release rate immediately following containment failure (sometimes referred to as the "puff release"), followed by two longer periods of a sustained release. The specific characteristics of these discrete release periods may vary from one accident sequence (or plant damage state) to another, but the timing characteristics (i.e., start time and duration) are the same for each radionuclide group (i.e., only the release rate varies from one group to another for a given release period). The total number of release periods is typically small (i.e., 3 or 4) and represents distinct periods of severe accident progression. For example, the following time periods may be represented:

Very early	[containment leakage prior to containment failure]
Puff release	[immediately following containment failure]
Early	[relatively large release rate period accompanying containment depressurization following breach of the containment pressure boundary]
Late	[long-term, low release rate after containment depressurization]

Note that the above time periods are for illustrative purposes only; others are developed, as necessary, to suit the specific results of a plant-specific assessment.

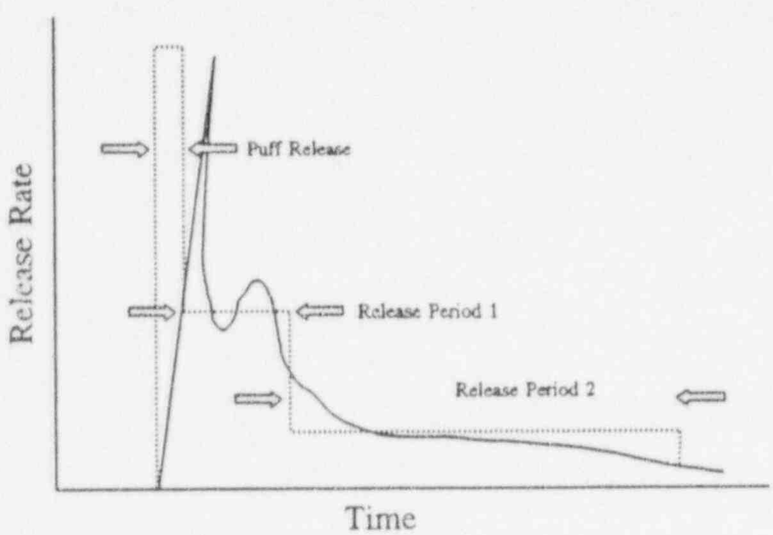

Figure 14.6 Example of simplified radionuclide release rates.

Coupling Source Term and Severe Accident Progression Analyses

The number of unique severe accident sequences represented in a Level 2 PRA can be exceedingly large. Comprehensive, probabilistic consideration of the numerous uncertainties in severe accident progression can easily expand a single accident sequence (or plant damage state) from the Level 1 systems analysis into a large number of alternative severe accident progressions. A radionuclide source term must be estimated for each of these accident progressions. Clearly, it is impractical to perform that many deterministic source term calculations.

A common practice in many Level 2 PRAs (although insufficient for a quality PRA) is to reduce the analysis burden by grouping the alternative severe accident progressions into "source term bins" or "release categories." This grouping process is analogous to the one used at the interface between the Level 1 and Level 2 analyses to group accident sequence cut sets into plant damage states. The principal objective of the source term grouping (or binning) exercise is to reduce the number of specific severe accident scenarios for which deterministic source term calculations must be performed to a practical value. A structured process similar to the one described in Section 14.4.1 (related to the assessment of accident sequences addressed in a quality Level 2 PRA) is typically followed to accomplish the grouping. Characteristics of severe accident behavior and containment performance that have a controlling influence on the magnitude and timing of radionuclide release to the environment are used to group (or bin) the alternative accident progressions into appropriate release categories. A deterministic source term calculation is then performed for a single accident progression within each release category (typically the highest frequency) to represent the entire group.

As indicated above, this approach is inadequate for a quality Level 2 analysis because the radionuclide source term for any given severe accident progression cannot be calculated with certainty. The influence of uncertainties related to the myriad processes governing radionuclide release from fuel, transport through the primary coolant system and containment, and deposition on intervening structures is significant and must be quantified with a similar level of rigor afforded to severe accident progression uncertainties. Examples of these uncertainties were given in Section 14.4.1. Further, a quality Level 2 PRA is performed in a manner that allows the relative contribution of individual parameter uncertainties to the overall uncertainty in risk to be calculated directly (i.e., via rank regression or some other statistically rigorous manner). This requires a probabilistic modeling process that combines the uncertainty distributions associated with the evaluation of accident frequency, severe accident progression, containment performance, and radionuclide source terms in an integrated, consistent fashion.

In performing this integrated uncertainty analysis, special care must be taken to ensure consistency between uncertain parameters associated with radionuclide release, transport, and deposition and other aspects of accident behavior. In particular, the analysis must account for important correlations between the behavior of radionuclides and the other characteristics of severe accident progression. For example:

- The magnitude of radionuclide release from fuel is known to be influenced by the magnitude of Zircaloy (clad) oxidation. Therefore, the distributions of plausible values for the release fraction of various radionuclides are correlated to the distribution of values for the fraction of clad oxidized in-vessel.

- In the NUREG-1150 assessments, uncertainty in the retention efficiency of aerosols transported through the primary coolant system was found to depend strongly on primary coolant system pressure during in-vessel melt progression. Higher retention efficiencies were attributed to sequences involving low coolant system pressure than those involving high pressure.

These and other similar relationships are described in the experts' determination of source term issues in NUREG/CR-4551 (Ref. 14.31).

Treatment of Source Term Uncertainties

Results of the Level 2 PRAs described in NUREG-1150 indicate that uncertainties associated with processes governing radionuclide release from fuel; transport through the primary coolant system, secondary coolant system (if applicable), and containment; and deposition on bounding structures can be a major contributor to the uncertainty in some measures of risk. For example, uncertainties in the magnitude of radionuclide release from fuel during in-vessel melt progression, and uncertainties in the amount of retention on the shell (secondary) side of steam generators were found to be among the largest contributors to the overall uncertainty in early fatality risk associated with steam generator tube rupture events (a significant contributor to the core damage frequency in some PWRs). Similarly, uncertainties in processes such as radionuclide release during core-concrete interactions and late release of iodine initially captured by pressure suppression pools were found to be important contributors to various risk measures in BWRs.

Uncertainties in the processes specifically related to radionuclide source term assessment are, therefore, represented in a quality Level 2 PRA. A systematic process and calculation tools to accommodate source term uncertainties into the overall evaluation of severe accident risks were developed for the Level 2 PRAs described in NUREG-1150. A detailed description of this process and the associated tools is not provided here; the reader is referred to NUREG/CR-4551, Vol. 2, Part 4 (Ref. 14.34), NUREG-1335 Appendix A (Ref. 14.31), and NUREG/CR-5360 (Ref. 14.32) for additional information on these topics.

Table 14.5 summarizes the areas in which key uncertainties are addressed in a quality Level 2 analysis. These key uncertainties are derived, in part, from the results of the NUREG-1150 analyses, as well as more recent statements of key source term uncertainties published by the NRC for light-water reactor licensing purposes (Ref. 14.33).

Table 14.5 Areas of key radionuclide source term uncertainties represented in a quality Level 2 PRA.

Magnitude of radionuclide release from fuel during core damage and material relocation in-vessel (primarily for volatile and semi-volatile radionuclide species).
Chemical form of iodine for transport and deposition.
Retention efficiency during transport through the primary and secondary coolant systems (particularly for long release pathways).
Magnitude of radionuclide release from fuel (primarily refractory metals) and non-radioactive aerosol generation during core-concrete interactions.
Decontamination efficiency radionuclide flow streams passing through pools of water (BWR suppression pools and PWR containment sumps).
Late revaporization and release of iodine initially captured in water pools.
Capture and retention efficiency of aerosols in containment and secondary enclosure buildings.

For shutdown accidents, it may be necessary to address additional phenomena that are unique or more important during modes of operation other than full power, steady state. For example, configurations where air can enter the reactor vessel (such as when the vessel head has been removed for refueling) have been postulated to cause an enhanced release of certain radionuclides. The effect that air ingression has on the source term in such configurations is assessed and, if important, included in the Level 2 model. When deterministic codes are used to estimate the source term, it is important to account for all of the relevant phenomena (even when the code does not explicitly include models for all of the phenomena). When a model is not available for certain important phenomena, it is not acceptable to simple ignore the phenomena. Instead, alternative methods are used, such as consulting different code calculations, using specialized codes, or assessing relevant experimental results.

When estimating consequences in the PRA, it is also important to accurately represent the timing of the release. Past studies have shown that the number of early fatalities can be particularly sensitive to when the release occurs relative to when the general public is being evaluated. Hence, it is also important that the approach used to estimate the source term properly accounts for timing characteristics of the release.

Documentation of Radiological Source Term Characterization

In general, sufficient information of the documentation of analyses performed to characterize radiological source terms is provided that allows an independent analyst to reproduce the results. At a minimum, the following information is documented for a quality PRA:

- a summary of all computer code calculations used as the basis for estimating plant-specific source terms for selected accident sequences

- a description of modeling methods used to perform plant-specific source term calculations including a description of the method by which source terms are assigned to accident sequences for which computer code (i.e., MAAP or MELCOR) calculations were not performed

- if analyses of a surrogate (i.e., "similar") plant are used as a basis for characterizing any aspect of radionuclide release, transport, or deposition in the plant being analyzed, references to, or copies of documentation of the original analysis, and a description of the technical basis for assuming applicability of results

- a description of the method by which uncertainties in source terms are addressed

- for all other original engineering calculations, a sufficiently complete description of the analysis method, assumptions and calculated results is prepared to accommodate an independent (peer) review

14.5 Attributes of a Level 3 PRA

Analyses performed as part of the Level 3 portion of a quality PRA consist of two major elements:

- accident consequence analysis
- computation of risk by integrating the results of Level 1, 2, and 3 analyses

Attributes of a quality analysis in each of these areas are described in Sections 14.5.1 and 14.5.2, respectively.

14.5.1 Accident Consequence Analysis

The consequences of an accidental release of radioactive material from a nuclear power plant can be expressed in several forms including impacts on human health, the environment, or economics. The consequence measures of interest to a Level 3 PRA for a nuclear power plant focus on impacts on human health. These impacts are estimated both in societal terms and in terms of the most-exposed individual. A quality PRA develops the following specific measures of accident consequence:

- early fatalities (societal)
- early injuries (societal)
- latent cancer fatalities (societal)
- far-field population dose (i.e., within a 50 mile radius of the plant)
- near-field population dose (i.e., with a 10 mile radius of the plant)
- individual early fatality risk (i.e., risk to the average individual within 1 mile of the site boundary)
- individual latent cancer risk (i.e., risk to the average individual within 10 miles of the plant).

There are two computer codes that are currently in use that incorporate current, state-of-the-art models for estimating the consequences of postulated radiological releases: the MELCOR Accident Consequence Code System (MACCS) and COSYMA. The MACCS computer code (Ref. 14.34) is recommended by the NRC for use in nuclear power plant Level 3 PRAs. This code was used in the analyses reported in NUREG-1150. MACCS is the successor to the CRAC-2 computer code (Ref. 14.35), which is no longer endorsed by the NRC as a current model for estimating consequences of postulated radiological accidents. The COSYMA computer code (Ref. 14.36) is an alternative state-of-the-art model being developed by the European Communities, and it is primarily used by European organizations conducting PRAs.

Performing consequence calculations with MACCS (or COSYMA) requires a substantial amount of supporting information. Atmospheric dispersion models require the specification of local meteorology and terrain; deposition models require information regarding frequency and intensity of precipitation; dose and health effects models require information regarding local demographics and land use (i.e., crops grown, dairy activity). In a quality, full-scope evaluation of accident consequences, this information represents current, site-specific conditions.

To assess the effects of variable weather on consequences, the complete transport, deposition, and dose calculation is repeated numerous time for each specified radionuclide source term. That is, for each specific wind direction (typically 1 of 16 points on a 360° wind rose), the consequence calculation is performed for a full spectrum of local weather conditions. The assumed wind direction determines the population over which the plume passes (based on local demographic data). Each calculation contains information about how the wind direction, wind stability, and precipitation changes from hour to hour. Effects of local evacuation or sheltering (as required by the plant-specific emergency plan) are also represented. Ideally, the starting time for evacuation would be based on accident-specific conditions. However, because each source term for which accident consequences are calculated actually represents several different accident sequences (each with different timing characteristics), it is acceptable to specify a single, representative evacuation start time for a given source term. It is not recommended that complete evacuation is assumed to occur; rather, 0.5 percent of the local population is assumed to remain in place during the entire release period.

It is not necessary to directly quantify and incorporate uncertainties in models for atmospheric dispersions, deposition, and health effects in the calculations of accident consequences. Although these uncertainties are generally acknowledged to be substantial, they are not currently included in quality Level 3 PRAs.

14.5.2 Computation of Risk

The final step in a quality Level 3 PRA is the integration of results from all previous analyses to compute individual measures of risk. The severe accident progression and the radionuclide source term analyses conducted in the Level 2 portion of the PRA, as well as the consequence analysis conducted in the Level 3 portion of the PRA, were performed on a conditional basis. That is, the evaluations of alternative severe accident progressions, resulting source terms, and consequences were performed without regard to the absolute or relative frequency of the postulated accidents. The final computation of risk is the process by which each of these portions of the accident analysis are linked together in a self-consistent and statistically rigorous manner.

The metric for judging the rigor of the process is the ability to demonstrate traceability from a specific accident sequence through the relative likelihood of alternative severe accident progressions and measures of attendant containment performance (i.e., early versus late failure) and ultimately to the distribution of radionuclide source terms and accident consequences. This traceability is evident in both directions (i.e., from an accident sequence to a distribution of consequences) and from a specific level of accident consequences back to the radionuclide source terms, containment performance measures, or accident sequences that contribute to that consequence level.

14.5.3 Documentation of a Level 3 PRA

In general, sufficient information of the documentation of analyses performed to estimate consequences associated with the accidental release of radioactive material to the environment is provided that allows an independent analyst to reproduce the results. At a minimum, the following information is documented for a quality PRA:

- a description of the site-specific data used to perform all consequence calculations (e.g., meteorology, demographics, land-use, etc.)

- a description of modeling methods used to assign consequences to individual accident sequences represented in the probabilistic logic model including a description of the method by which the full spectrum of severe accident source terms, generated as part of the uncertainty analysis, are linked to a limited number of actual consequence calculations

- a description of the computational process used to integrate the entire PRA model (Level 1 through Level 3)

- a summary of all calculated results including frequency distributions for each risk measure

14.6 Attributes of Internal Flooding Analysis

A quality PRA covering internal flooding analysis uses much of the same processes and meets the same attributes provided under the discussion of full-power internal events (Section 14.3). However, internal flooding analysis requires significant work to define and screen the most important flood sources and possible scenarios for further evaluation.

The major tasks associated with the Level 1 portion of an internal flooding analysis include:

- Flood source and propagation pathway identification and screening

- Flood scenario identification and screening

- Flooding model development and quantification

While analogous to the initiating event identification and exclusion portions of the full-power internal events PRA, the first two tasks require consideration of different plant characteristics with particular emphasis on the spatial aspects of the plant's design. Consideration of structures, barriers, drainage designs, and different failure modes (e.g., water submersion of equipment and water spray on electrical equipment) are examples of aspects of the plant that are considered in the internal flood analysis that are not necessarily addressed in the internal events analysis. After the flooding scenarios have been screened for detailed quantification, the third flooding task follows much of the modeling and quantification aspects already carried out in the internal events analysis with relatively minor modification.

In a quality PRA, three scoping attributes are met to better ensure completeness of the analysis. First, a quality PRA not only considers floods as initiating events, but includes the possibility of flooding as a subsequent event to some other initiator. Second, a quality PRA considers both water and steam sources, including splashing and condensation effects. Additionally, a quality PRA examines flooding induced by both equipment failure as well as human-induced events (such as failure to properly isolate a potential flood source before performing maintenance). Attributes associated with the unique aspects of the internal flood analysis (compared to the internal events analysis) are addressed below.

As with the Level 1 portion of the internal flooding analysis, the Level 2 internal flooding analysis considers the impact of the flooding sources on the core damage mitigating systems and the containment. The attributes for performing and documenting the Level 2 portion of the analysis includes the same considerations as discussed in Section 14.4 and for the internal events analysis discussed in this section. For example, the potential for an internal flood resulting in the failure of containment isolation valves to close, failure of containment spray systems, or failure of gas treatment systems all are evaluated. Although it is not expected that an internal flood would present new accident phenomena or a new challenge to the containment integrity, a thorough review is performed to substantiate this position.

An internal flood accident is not expected to present any conditions that would change the consequence assessment generated for the internal events analysis other than potential differences in the source terms being released. Thus the discussion in Section 14.5 is also applicable for an internal flood analysis. However, it is verified that there are no impacts on the consequence assessment.

14.6.1 Identification and Screening of Flood Sources, Propagation Pathways, and Flood Scenarios

These tasks are performed together in a somewhat iterative nature as there are numerous interactions between the tasks. In a sense, the flood scenarios to be investigated are a product of the flood source and propagation identification process. Thus they are discussed together here. The objectives of these tasks include identifying (a) flood zones, (b) potential sources of internal flooding, (c) locations in the plant potentially affected by the flood sources, (d) the manner in which the flood can propagate from one location to another, (e) the equipment potentially affected by the resulting flood scenarios that are of possible interest (including refinement of those scenarios based on consideration of failure scenarios that can lead to core damage), and (f) appropriate screening of the above as a practical resource-saving measure.

NUREG/CR-4832, Volume 10 (Ref. 14.37) provides an acceptable source used in a quality PRA for performing the specific steps to achieve the above objectives. These specific steps are not reproduced here. However, certain overriding attributes of quality are met in performing this analysis, as highlighted below.

All substantial water and steam sources are addressed and carefully screened. As a minimum, possible sources include piping, valves, pumps, tanks, heat exchangers, room coolers, chillers, fire systems, relief valves, and potentially large bodies of water in the plant (such as the suppression pool in BWRs and the spent fuel pool). Any qualitative arguments used to screen or otherwise eliminate flood sources (e.g., small size, location arguments, effects are similar and greater for another flood source, etc.) are well documented and based on sound engineering principles and judgment. While probabilistic arguments can be used at this stage, such arguments are limiting and meet the initiating event exclusion principles provided in the discussion of internal events (Section 14.3). Both leakage and rupture failure modes are considered, as well as the potential for human-induced flooding.

Sources and locations of concern and particularly the identification of propagation pathways are supported by actual walkdowns of the plant. The flood zone definitions considers the existence of barriers and drains that can confine the flood to an area. Propagation pathways from one flood zone to another consider stairways, doorways, hatches, floor and wall penetrations and cracks, drain lines, HVAC ducts, piping/conduits, etc., as well as the potential failure of barriers to propagation (e.g., normally closed door failing open once the flood water reaches a certain height behind the door). Any assumptions or other judgments used to define and screen out possible locations and pathways are documented and based on analyses, calculations, or sound engineering judgment. Similarly, if isolating the flood or low flow arguments are used to screen out flood sources and/or pathways, the rules used to make such arguments are clearly stated and based on sound analysis and judgment. Isolation arguments consider methods of detection, access and available means to isolate or otherwise mitigate the flood source, and the time to carry out appropriate actions. With regard to determining possible flow rates, the analyst considers whether forced flow (such as from an active pump) or passive flow rates are expected.

The above information leads to the formulation of possible flood scenarios that need to be considered. These scenarios are more completely defined by considering what and how equipment is affected in the context of the possible accident sequences that can lead to core damage (as indicated by the internal events analysis). It is therefore important that the possible flood-induced failure modes (i.e., susceptibility) of equipment be considered in addition to the random equipment failures covered in the internal events analysis. Any guidance used in the flooding analysis with regard to the failure modes to be considered are clearly defined and have a reasonable basis. For instance, electrical equipment (buses; motor control centers (MCCs); batteries; inverters; motors for valves, pumps, and fans; etc.) if submerged, significantly splashed, or exposed to a high steam environment are assumed to short-out and therefore be unable to operate, at least during the screening steps conducted in the analysis. Mechanical equipment may only be considered to fail under special circumstances such as when HVAC ducting is flooded and fails because of the water weight, and so on. Screening of potential accident sequences on the basis of what equipment is or is not affected, as well as consideration of the above failure modes are clearly identified and supported.

14.6.2 Flooding Model Development and Quantification

The modeling of the resulting unscreened scenarios follow and use the modeling performed for the internal events analysis. Hence, many of the same sequence models (typically event trees) and system failure models (fault trees) are used but with modification. The model accounts for possible combinations of flood-induced as well as random failures of equipment, and recognizes the potential for new or more severe performance shaping factors when considering human failure probabilities and possible recovery actions. Data values will need to be changed accordingly. As previously stated, consideration is given to internal floods as initiating events, as well as floods that occur during or as a result of some other transient.

The quantification portion of the analysis follows the same attributes provided in Section 14.3. If following the initial quantification, it is judged that any bounding assessments remaining from the earlier phases of the analysis are leading to unacceptable results, refinements are made to provide a more realistic "best estimate" analysis. Any such refinements are clearly documented and peer reviewed for appropriateness.

14.6.3 Documentation of Internal Flooding Model Development and Quantification

The process of identifying flood sources, flood pathways, flood scenarios, and their screening, and internal flood model development and quantification are documented in a quality PRA. At a minimum, the following information is documented:

- a definition of the flood zones used in the analysis and the reason for eliminating any of these areas from further analysis
- a list of all flood sources considered in the analysis and any rules used to eliminate these sources
- a discussion on the propagation pathways between flood zones and any assumptions, calculations, or other bases for eliminating any of these propagation pathways
- a listing of accident mitigating equipment located in each flood zone not screened from further analysis
- a list of any assumptions concerning the impacts of submergence, spray, temperature, or other flood-induced effects on equipment operability
- a discussion of how the internal event analysis models were modified for the internal flooding analysis
- a list of the flood frequencies and component failure probabilities from flood effects and their bases
- a discussion of any calculations or other analyses used to refine the flooding evaluation

14.7 Attributes of a Peer Review

A peer review is an essential part of a quality PRA. The peer review checks the reasonableness of the key inputs, models, and results, and therefore, provides an "independent" assurance that a quality PRA has been performed. A quality peer review includes two major parts (which are discussed in detail below) and include the following:

- Review team and qualifications, and
- Review process.

14.7.1 Review Team Composition and Qualifications

For a quality peer review, the peer review is composed of a team rather than an individual, because the basic tasks in the analyses generally involve expertise in multiple disciplines. For the PRA peer review, experts will be needed in the following areas: systems analysis, data analysis, human reliability analysis (HRA), severe accident phenomena (if Level 2 analysis performed for submittal), source term (if Level 2 analysis performed for submittal), consequence modeling (if Level 3 analysis performed for submittal).

Each peer reviewer generally has at least 5 years of direct experience covering multiple plants in performing the task that the reviewer is assigned to review. This experience consists of knowledge of typical inputs, assumptions, methods and techniques, models, scope, level of detail, and form of results for the assigned review area. The reviewers also have a general familiarity with the plant design being analyzed. At least one member has a good knowledge of the specific plant and its operation.

A quality peer review is intended to be an "independent" peer review, and as such would not involve personnel or individuals who participated in the PRA. An unbiased review is also ensured by selecting team members who are independent of the organization that performed the PRA. Similarly, it is not appropriate for other utilities to perform the peer review.

14.7.2 Review Process

The peer review is conducted through group meetings (versus isolated reviewers), and the meetings are held at the plant so that additional information can be obtained as needed. A single member acts as the team leader providing overall coordination and control of the review process.

All members, prior to the review meeting, are provided sufficient information to gain an understanding of the PRA (e.g., accident sequence development) and any supporting information and analyses (e.g., plant-specific room heatup calculations). The documentation to be reviewed is provided to the peer reviewers in sufficient time to allow adequate review. The plant analysts that performed the analyses being reviewed should be available during the peer review meeting to answer specific questions that are not adequately addressed in the documentation.

Although not necessary, conducting an ongoing peer review in parallel with performing the PRA (versus waiting until after the PRA is completed) is the most efficient. In this way, any deficiencies found during a specific phase of the work is identified and corrected in a timely and resourceful manner. PRA peer reviews are best utilized if they are timed to correspond to the major phases of the work: major modeling phase, data base construction and initial quantification phase, and the finalization and documentation phase. Reviewing the initial PRA construction is particularly useful because it provides early feedback so that subsequent phases of the work do not get too far advanced without a review of the previous phase(s) of the analysis.

The first goal of the PRA peer review is to examine the inputs, techniques, and analyses for the PRA. In performing the review, attention is given to the completeness, accuracy, and the information, so that the PRA reflects the as-built, as-operated plant. The PRA is based on information obtained from walkdowns, controlled documentation concerning the plant design and operation, involvement of plant staff, and a "freeze date" for the analysis (including any updates). To provide assurance that the approaches were generally applied appropriately, the peer reviewers are expected to spot check selected portions of the analyses. For those portions, an in-depth review is performed. The specific goals of the peer review panels are:

- to determine the validity of the input information sources, assumptions, models, data, and analyses forming the basis for the proposed change, and

- to determine the validity of the results obtained in the analyses and the corresponding conclusions related to the proposed change.

It is expected that problems of varying importance will be uncovered during the peer reviews. The peer review reports those problems that are significant enough to change the conclusions and results of the PRA. The peer review examines the PRA inputs to determine whether the PRA has met the attributes for an acceptable PRA that are outlined in Sections 14.2-14.6. The review consists of an overall evaluation of the task area being reviewed coupled with sampling of very specific aspects of the task using a very detailed review approach. The inputs to the PRA that are examined by the peer review team to determine whether they meet the attributes in Sections 14.2-14.6 are discussed in the following paragraphs, followed by a discussion of the outputs that are examined.

Level 1 Modeling

The items to be examined for the overall examination are discussed first. In each case, the areas should be addressed relative to the specific attributes discussed in Sections 14.2-14.3 and Section 14.6 for an acceptable PRA. To ensure a quality Level 1 PRA, the peer review focusses on the following items:

- The initiating events included in the PRA are reviewed to assess the completeness of the initiators considered, whether the basis for excluding any initiators is adequate, and to determine the reasonableness of the initiator frequencies used in the PRA.

- The reviewers consider whether the success criteria for each initiator is reasonable, and determine if there is an adequate basis for any success criteria that is not typical for the type of plant being reviewed.

- The accident sequence models (e.g., event trees (functional or systemic) are scanned to determine whether the logic is reasonable.

- The modeling of systems are reviewed to determine whether the failures considered are comprehensive. Operability during accident and harsh environments (e.g., trip points for reactor core isolation cooling system) are considered, as well as the completeness of the failure modes (e.g., failure to start, run).

- The system dependency matrix is reviewed to assess whether dependencies are appropriately considered in the PRA.

- The operator actions that are included in the PRA, the failure probabilities for the actions, and the basis for excluding actions from the analysis are reviewed to determine the completeness of the analysis and the reasonableness of the probabilities calculated for each operator action.

- While the peer review is not expected to provide a detailed review of all failure frequencies/probabilities used in the PRA, the methods used for determining the failure frequencies/probabilities (including common cause treatment) are examined, the adequacy of data sources are assessed, the failure frequencies/probabilities (including common cause values), and the uncertainties are scanned for reasonableness.

- The adequacy of the quantification method, including the screening criteria are addressed.

To supplement the reviews listed above, the independent peer reviewers also perform detailed spot checks of selected accident sequence models (e.g., event trees), systems models (e.g., fault trees), and the associated quantification. The reviewers are also expected to spot check the documentation of plant walkdowns.

Level 2/3 Modeling

The review team will need to evaluate the adequacy of the Level 2/3 analyses relative to the specific attributes discussed in Sections 14.4-14.6. To ensure a quality Level 2/3 PRA, the peer review focusses on the following items:

- The event trees (or equivalent system models) are reviewed to determine whether the treatment of severe accident phenomena is comprehensive for the plant under consideration. The treatment of systems and phenomena are reviewed, including the basis for probabilities, to determine if the attributes provided in Section 14.4.1 were followed.

- The source term and consequence modeling and inputs are reviewed to determine whether they satisfy the attributes for a quality PRA that are provided in Section 14.4.2.

- The process used to bin results for the Level 2/3 analysis are checked (e.g., if plant damage states, accident progression bins, or source term groups are created) to ensure that the grouping maintains the separate effects of the key factors affecting the results. The actual mechanics of the binning are examined for selected cases to determine whether the calculations were correctly performed.

Review of PRA Results

In addition to reviewing the inputs to the PRA, the peer review team also provides an independent evaluation of the sensibility of the results. The review focuses on the appropriateness of the identified dominant accident sequences, and when a full Level 2/3 analysis is performed, the containment failure modes, releases and consequences. The review also considers whether the aspects of the plant design, operation, and maintenance that are found to contribute most to risk in the PRA are reasonable. The results examined are:

- The top cut sets (limited to ~300 per sequence) are scanned, looking for unreasonable combinations of events.

- The sequence CDFs calculated before and after crediting recovery actions are scanned for reasonableness.

- The frequencies of accident progression pathways as grouped for source term calculations, the frequencies and magnitudes of source terms, the individual early and latent fatality frequencies, and the uncertainty characterizations for these frequencies are assessed for reasonableness.

14.7.3 Documentation

The documentation from the peer reviews include descriptions of the peer review process and findings, and the responses to the peer review findings.

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15. COMPARISON OF INDIVIDUAL PLANT EXAMINATIONS TO A QUALITY PROBABILISTIC RISK ASSESSMENT

15.1 Introduction

In Generic Letter (GL) 88-20 (Ref. 15.1) the U.S. Nuclear Regulatory Commission (NRC) identified three acceptable approaches that licensees could use to perform an Individual Plant Examination (IPE) for their plants. One method consists of a Level 1 probabilistic risk assessment (PRA) plus a containment performance analysis (essentially a Level 2 PRA). All of the licensees eventually elected this method and performed PRAs for their IPEs. These will provide the basis for increased use of PRA in risk-informed regulation. The acceptability of the use of these PRAs in future risk-informed regulation depends significantly on their quality.

Chapter 6 of Part 1 summarizes the key perspectives regarding the IPEs with respect to risk-informed regulation. Section 6.2 (of Chapter 6, Part 1) summarizes the characteristics that comprise a quality PRA. Chapter 14 provides a more in-depth discussion of these characteristics addressing in detail the attributes of each element of a quality PRA. To provide some perspective on the quality of the IPEs, this chapter presents a general comparison of the IPEs to the attributes of a quality PRA presented in Chapter 14 of this document.

GL 88-20 did not provide specific guidance, along the lines presented in Chapter 14 of this document, on how to perform the Level 1 PRA, since the methods are generally well established and documented in various PRA procedure guides. Instead, GL 88-20 indicated that PRAs performed according to the procedures described in NUREG/CR-2300 (Ref. 15.2), NUREG/CR-2815 (Ref. 15.3), or NUREG/CR-4550 (Ref. 15.4) were adequate as IPEs. These procedure guides generally reflect the attributes of a Level 1 PRA presented in Chapter 14 of this document. However, it was recognized that a wide range of views exist concerning the phenomena (and their probabilities) associated with core melting, release of molten core material to the containment, and subsequent containment performance. Thus, general guidance for conducting the containment performance analysis was provided in Appendix 1 to GL 88-20. Some significant variations exist between that guidance and the attributes listed in Chapter 14 of this document.

The NRC also provided guidance on the format and content of the licensees' IPE submittals in NUREG-1335 (Ref. 15.5). Unlike the attributes for documenting a quality PRA, licensees were not required to submit all information documenting the IPE. Furthermore, as with GL 88-20, NUREG-1335 did not provide specific guidance for performing the Level 1 PRA, with the exceptions of the guidance inferred in the response to comments and questions provided in Appendix C to NUREG-1335. Further guidance on how to perform the containment performance assessment was provided in Appendix A to NUREG-1335.

A detailed comparison of the IPEs to the attributes listed in Chapter 14 of this document has not been performed. However, the staff reviewed each licensee's IPE submittal to determine if the licensee's IPE met the intent of GL 88-20. The staff's review did not, however, verify or validate the PRAs. For example, the staff's IPE review did not include verifying and validating supporting calculations. Instead, the NRC staff's review focused on the following items:

- the "completeness" of the PRA with regard to inclusion of the necessary elements of a PRA and whether the methods used are acceptable to the staff
- the "reasonableness" of the assumptions, boundary conditions, and data used in the PRA
- the "reasonableness" of the results, given the design and operation of the plant

In addition to the staff's review of the individual IPEs, significant insights into the general quality of the PRAs has been obtained as a result of this study. This chapter documents these insights for the Level 1 analyses, the containment performance assessment (i.e., the Level 2 assessments), and the Level 3 assessments performed by a few licensees. However, it is emphasized that because the attributes of a quality PRA delineated in Chapter 14 have evolved over time and were not available at the time the IPEs were performed, some shortcomings are to be expected particularly in the containment performance assessment. Variations between the attributes in Chapter 14 of this document and the guidance specified in GL 88-20 and NUREG-1335 are highlighted to place identified shortcomings in perspective. In addition, the identified shortcomings are not applicable to every IPE. Thus, the adequacy of an individual IPE for use in future risk-informed applications can only be determined by performing a more detailed review of the PRA.

15.2 Comparison of IPEs to a Quality Level 1 PRA

In order to provide perspectives regarding the general level of completeness and technical quality of the IPEs, the information in the IPE submittals has been compared to the attributes of a quality Level 1 PRA defined in Chapter 14. A comparison is presented for each of the major Level 1 PRA tasks. Although characteristics from each IPE are reviewed as part of this comparison, the staff made no attempt to compare individual IPEs against the characteristics listed in Chapter 14. Instead, the staff compared the IPEs as a group. Some IPEs are obviously better than others in certain areas; for others, adequate documentation was not always provided to allow a full comparison. The resulting perspectives, in turn, can be valuable in determining the potential role of the IPEs in future risk-informed regulation. The Level 1 analysis performed in the IPEs generally compare well to the quality PRA characteristics listed in Chapter 14 of this document.

15.2.1 IPE Initiating Event Analysis Comparison to a Quality PRA

With the exception of some guidelines for documentation, GL 88-20 and NUREG-1335 did not provide specific guidance for the initiating event analysis task of the IPEs. The potential accident initiating events at a nuclear power plant are well understood and have been characterized in PRA procedure guides, PRAs and PRA-related studies, and other documentation (e.g., updated safety analysis reports). Thus, the initiating event analyses performed in the IPEs generally compare well to the attributes for identifying, excluding, grouping, and documenting initiating events as discussed in Chapter 14 of this document. More specific details are provided below.

The modeling of all general plant transient events (i.e., plant transients that have historically occurred at many plants such as turbine trips and loss of feedwater events) appears to be complete for almost all of the plants. This completeness should not be surprising since these transients have occurred at most plants, are well documented, and have been modeled in all PRAs before the issuance of GL 88-20. What was not clear from many of the IPE submittals was the grouping of these general transients into transient initiating event groups. Some variation in this grouping can occur as a result of differences in plant design or operation. Most of the IPE submittals did not provide sufficient information to determine how this grouping was performed and to verify if this grouping was performed according to the attributes of a quality PRA. For example, the treatment of manual shutdowns in the IPEs varied. Some IPEs clearly indicated that controlled shutdowns were treated as separate initiating events while others indicated that they were grouped with other transient events. Some included planned shutdowns, such as for refueling outages, while others only included unplanned shutdowns resulting from abnormal occurrences. However, for the majority of the plants, it was not clear from the submittals if manual shutdowns were included in the analysis or how they were included.

All of the licensees modeled loss-of-coolant accidents (LOCAs) of various sizes (typically small, medium, and large) determined primarily by the requirements of mitigating systems. However, some variation in the modeling of LOCA sizes occurred and impacted the reported results. For example, some IPEs for boiling water reactors (BWRs) only modeled small and large breaks, while others reported that intermediate LOCAs (which require vessel depressurization for low-pressure coolant injection) were dominant LOCA scenarios. Very small LOCAs such as recirculation pump seal LOCAs were considered in many IPEs, but instrumentation line breaks were generally not explicitly mentioned. An unmitigatable reactor vessel rupture was generally given cursory treatment in the IPEs (if it was discussed at all) because of the expected low frequency of occurrence.

Some licensees also differentiated the LOCAs by location (e.g., steam line versus water line breaks in BWRs, and inside containment versus outside containment). However, the majority of the BWR IPEs combined steam and water line breaks, while breaks outside containment (with the exception of interfacing system LOCAs) were widely ignored. Interfacing system LOCAs were analyzed in the vast majority of the IPEs primarily because of their potential for bypassing the containment.

The modeling of support system transient initiators is more variable than other initiating events. Most licensees presented justifications for modeling or eliminating various support system initiators consistent with a quality PRA. However, some licensees chose to include a support system as an initiator because it could not easily be determined if a reactor scram would actually occur as a result of the loss of the system. 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., no calculations were 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.

15.2.2 IPE Accident Sequence Analysis Comparison to a Quality PRA

With the exception of some documentation guidelines, no specific guidance was provided in GL 88-20 or NUREG-1335 for the accident sequence analysis portion of the IPE. As a result, various acceptable methods for a quality PRA were used. These methods involved either event trees (the most common method used in the IPEs) or event sequence diagrams. The use of event trees was somewhat variable ranging from functional to systemic trees. In addition, some licensees chose to use support system fault trees to determine support states, while others chose the fault tree linking process to account for support system interactions. Most plants used separate event tree models to delineate the events leading to core damage and to model the subsequent containment response. The core damage sequences were generally binned manually into plant damage states that served as input into the containment response analysis. However, some licensees used plant response trees that asked the operability of mitigating systems both before and after core damage for the purpose of directly identifying plant damage states.

The key attributes for the accident sequence analysis portion of a quality PRA involve establishing the success criteria and delineating the accident progression for each accident initiating event. A general comparison of the accident progression represented in the IPE models to the attributes of a quality PRA, including the establishment of success criteria, is discussed below. In general, the accident sequence analyses performed in the majority of the IPEs have the attributes of a quality PRA as described in Chapter 14.

One factor that can influence both the success criteria and the accident progression is the definition of core damage, which varied substantially in the IPEs from definitions involving vessel level to definitions involving fuel cladding temperature or oxidation (the parameter used in the core damage definition in Chapter 14). To a large extent, many of these definitions could be somewhat equivalent and thus meet the core damage definition used in Chapter 14 (i.e., high fuel cladding temperatures and oxidation could be expected when the vessel is uncovered to a certain level). For the majority of situations, the core damage definition did not impact the IPE modeling. However, there were some cases when the core damage definition influenced the success criteria. For example, a licensee who assumed that core damage would occur when the vessel level reaches the top of the active fuel did not credit systems that could not prevent partial core uncovery but would eventually refill the core (e.g., the use of control rod drive system in some BWRs). Licensees who assumed core damage would occur when the vessel level is much lower would credit such systems.

The use of realistic success criteria with well-established bases is an attribute of a quality PRA. The success criteria used in the IPEs for mitigating systems responding to the various initiating events generally is more realistic than assumed in the analyses documented in the licensees' final safety analysis reports (FSARs). However, a small fraction of the licensees chose to use the limiting FSAR criteria in their analyses. The bases for emergency core cooling system and alternative coolant injection system success criteria are generally well established in the IPEs through references to existing thermal-hydraulic analyses or other PRAs (including the NUREG-1150 studies). In addition, some licensees performed plant-specific calculations to determine whether systems, such as the control rod drive system in BWRs, could be used early to mitigate transients. However, in a few cases, system-level codes such as Modular Accident Analysis Code (MAAP) (Ref. 15.6) were used to determine thermal-hydraulic success and event timing. The use of such codes for these purposes in a quality PRA is questionable due to the simplistic nature of these codes for assessing success criteria requiring a high level of accuracy in the results.

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. Finally, it should be noted that 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 (such as using a realistic definition for core damage instead of the 2200°F hot channel criteria in the FSAR), another licensee may have elected to not perform the necessary analysis, and therefore, not credit the system(s) in the success criteria.

The accident progression modeling in the IPEs is generally within the attributes set in Chapter 14 for a quality PRA with the notable exception of the treatment of phenomenological impacts on system operability. Phenomenological impacts on system operability were not always rigorously analyzed or included in the accident progression. Some illustrative examples are provided below. Protective trips of systems resulting during an accident sequence did not always appear to be included in the analysis. For example, steam line detection trips of the reactor core isolation cooling system in BWRs was identified as occurring during station blackout sequences (heating ventilating and air conditioning (HVAC) systems would be inoperable) in some IPEs but not in others. Whether this modeling difference was caused by design differences or oversights was not clear in some cases. The loss of emergency core cooling system pump net positive suction head following loss of suppression pool cooling was identified in most BWRs. However, clogging of pump strainers (significant especially during LOCAs) was widely ignored in most of the IPEs. Finally, the failure of components outside the containment following containment failure (or venting in BWRs) as a result of expected harsh environments was generally considered in a qualitative manner rather than through a rigorous quantitative process (see section 15.2.3 for further discussion on this topic).

15.2.3 IPE Systems Analysis Comparison to a Quality PRA

With the exception that the IPE must represent the as-built and as-operated plant, no specific guidance was provided in GL 88-20 or NUREG-1335 for performing the systems analysis portion of the IPEs. However, the guidance for the systems analysis in a quality PRA presented in Chapter 14 includes establishing the failure modes to include in the system model, defining criteria for excluding failure modes, and guidance for modeling system dependencies. The systems analysis reporting requests in NUREG-1335 are less stringent than the documentation for a quality PRA. In fact, the licensees were not requested to provide the system models as part of their IPE submittals. As a result, the individual system models (i.e., fault trees) have not been reviewed. However, the general system modeling guidelines could be gleaned from many of the IPEs. The results of a general comparison of the IPE submittal discussion on the systems analysis to both NUREG-1335 and the attributes of a quality PRA is provided below. The results of this comparison tend to indicate that the IPE system analyses generally meet the attributes of a quality PRA.

NUREG-1335 recommended that a plant walkthrough be performed to verify that the as-built plant was being modeled in the IPE. The need for system walk-throughs is also an attribute for a quality PRA. Some of the licensees indicated in their submittals that plant walk-throughs were performed. Other licensees indicated that part of their internal review process involved verification that the system models represent the current system configuration. Thus, for many of the plants, some verification that the system models represented the then current system configuration was provided. However, for some plants, no such verification was evident from the IPE submittal.

Because the IPE submittals do not usually provide the system models, the NRC staff could not thoroughly assess the adequacy of these models as part of their audit of the IPE submittals. It was not always evident from the IPE submittal when detailed or simplified fault trees were used to model a system, and thus the adequacy of the system modeling could not be fully ascertained. 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 consistent with those modeled in a quality PRA and included hardware failures, testing and maintenance outages, human errors, and common cause failures. Passive component failures such as pipe breaks were generally ignored in the IPEs, which is consistent with the characteristics of a quality PRA as described in Chapter 14.

Of particular concern is the treatment of common cause failures in the IPEs. The actual components considered for common cause failures vary among the IPEs with most of the licensees only modeling common cause failures for components that are required to change state. The common cause database, referenced in Chapter 14 as being the most current, also includes passive component common cause failures (e.g., heat exchangers and strainers). Thus, the types of components requiring common cause treatment was not generally as complete in the IPEs as described by the attributes in Chapter 14. As is common PRA practice and consistent with a quality PRA, most licensees only treated common cause failures within a system; a few modeled failures across systems.

The system dependencies on support systems such as electrical power and cooling water systems were generally documented using dependency matrices as requested by NUREG-1335. However, the level of detail included in the dependency matrices varied, making it difficult in some cases to identify which systems or trains of a system were affected by a failure of a particular electrical bus or other supporting subsystem. In addition, the basis for eliminating some dependencies (primarily HVAC dependencies), in some cases, was qualitative and made apparently without calculational support.

Finally, many of the IPEs addressed the operability of some systems under adverse conditions or environments. NUREG-1335 addresses the issue of equipment survivability in the response to questions and comments provided in Appendix C. The NRC staff indicated that credit for equipment operability during severe accidents be based upon some evidence of the capacity of the equipment to operate in the expected harsh environments. This could involve extrapolation of equipment qualification data. This criteria is in agreement with that presented in Chapter 14 for a quality PRA. For the most part, the issue of equipment survivability was addressed qualitatively rather than quantitatively. The location of equipment with respect to expected harsh environments was often used to determine if a system survived. On the one hand, a system in the turbine building was generally considered to continue to operate following containment failure since harsh environments would not be expected in that area. Equipment in the reactor building, on the other hand was often assumed to fail since harsh environments could be expected in that area. Quantitative analyses were generally limited to analyzing the impacts of loss of HVAC. Room-heatup calculations were performed by some licensees to determine if and when equipment in certain areas would fail upon loss of HVAC. Without a thorough review of the system design and modeling, the staff cannot confirm that equipment survivability was comprehensively examined.

15.2.4 IPE Data Analysis Comparison to a Quality PRA

The guidance in NUREG-1335 concerning data analysis implied that plant-specific data should be used whenever possible for initiating events and for important components. NUREG-1335 further states that plant-specific data should be used when there is several years of operating experience but only when statistically meaningful data exists. The method for performing the plant-specific data analysis (e.g., classical or Bayesian methods) was not specified. In contrast, the data analysis attributes for a quality PRA presented in Chapter 14 indicates that plant-specific data should be used when possible for all initiating events and system components. Bayesian updating is the preferred method for combining plant-specific component failure data with generic failure data. However, initiating event frequencies and component outages (associated with testing or maintenance) should reflect only plant experience (i.e., Bayesian updating should not be performed). A general comparison of the IPEs to these two sets of attributes would tend to indicate that the data analysis portion of the IPEs may be one area of weakness in the Level 1 IPE analyses. Some of these weaknesses are highlighted below.

The plant-specific data process was not documented in the IPE submittals. Therefore, it is not possible to determine if the process of selecting data sources and using that data was performed according to the attributes described in Chapter 14. However, it is clear from the submittals that the data analysis performed in the IPEs is highly variable. This is most evident from a review of the component hardware failure data analysis. Most licensees used a combination of plant-specific and generic data to quantify their IPE system models. Some licensees performed Bayesian updates of generic data while others directly used plant-specific and generic data. While most of the licensees used at least some plant-specific data, some methods are being used that are inappropriate. Examples include counting zero failures as equal to one-third of a failure in estimating the failure rate and incorrect application of the Bayesian updating.

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. Although the generic data sources used in most IPEs are from the nuclear industry, a large variability has been observed in the failure rates for some components (e.g., generic data for failure to start for turbine-driven pumps ranging from 5E-3 to 9E-2).

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 as a result of this type of examination of the common cause database. 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 also needs to consider possible undiscovered failures that are relevant to that particular plant.

15.2.5 IPE Human Reliability Analysis Comparison to a Quality PRA

In the response to questions in Appendix C to NUREG-1335, the NRC staff indicated that human reliability analysis (HRA) methodology is not mature, and therefore, would not allow the establishment of specific guidance on performing the HRAs needed for the IPEs. Although a particular HRA method is not endorsed in Chapter 14 as being appropriate for a quality PRA, the attributes of an acceptable methodology are identified. A general comparison of the HRAs performed in the IPEs to the attributes of a quality PRA is summarized below.

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 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. One particular application of an HRA methodology as been determined to be inconsistent with the intent of GL 88-20 and the attributes defined in Chapter 14. Additional assumptions made regarding the applications of the method were invalid and, consequently, the method as applied did not produce consistently reasonable results. The potential impact of methodological affects on HRA results in the IPEs will be addressed in an upcoming NUREG/CR report.

As far as completeness in modeling human errors, most IPEs include both pre-initiator and response and recovery post-initiator human errors. Pre-initiator errors are often given screening values and subsequently screened out or determined to be insignificant. Some licensees only considered failure to restore components after maintenance and dismissed calibration errors; others included both types of errors. 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 rarely occur and, therefore, are not represented in the random failure data.

Most licensees included post-initiator operator actions in their IPEs, but consideration of 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. The guidance in NUREG-1335 Appendix C and Chapter 14 indicates that recovery actions that are not proceduralized should generally not be credited. In general, the licensees did not credit such recovery actions in the IPEs. However, a substantial number of IPEs did credit repair of failed components. Although plant-specific procedures are generally used in the HRA evaluations, the use of simulator exercises and operator talk-throughs is somewhat variable.

Several concerns were identified in the quantification of post-initiator operator actions in the IPEs. First, it was noted in some IPEs that the time available to perform the actions may not have always been correct. Specifically, the time at which the operator would be cued to an abnormal situation requiring action was not always correctly determined. Second, in many cases average performance shaping factors were used rather than situation-specific parameters. Thus, a plant-specific HRA was not always the end product of the IPE. Third, dependencies between human error events were not always handled correctly. This was particularly true for plants that used low screening values for the human errors during preliminary quantification of the IPE models.

15.2.6 IPE Accident Sequence Quantification Comparison to a Quality PRA

Specific guidance on performing the accident sequence quantification portion of the IPEs was not provided in either GL 88-20 or NUREG-1335. However, the method by which the accident sequences were quantified, including the computer programs used was asked to be included in the submittal. Sensitivity studies and an uncertainty analysis was not requested as part of the IPE process. Substantial guidance was provided in NUREG-1335 for reporting the results of the analysis. In particular, screening criteria were provided to determine which accident sequences to report to the NRC staff. For each of the selected sequences that use these screening criteria, NUREG-1335 specified that a discussion of the accident progression, specific assumptions, essential equipment subjected to environmental conditions beyond the design bases, and applicable recovery actions be included in the IPE submittal. Any sequences that fall below the screening criteria because its frequency drops by more than an order of magnitude as a result of credit taken for human recovery actions were to be identified. In addition, a list of major contributors to those accident sequences was to be provided along with a list of any identified vulnerabilities.

The quantification guidance for a quality PRA provided in Chapter 14 goes beyond that specified in NUREG-1335 and includes guidance on selecting truncation values, incorporating recovery actions, estimating uncertainty, and performing sensitivity studies. In addition, the documentation attributes encompass those listed above plus additional items concerning the entire quantification process (e.g., flag files, list of mutually exclusive events, and the results of uncertainty and sensitivity calculations).

A comparison of the IPE quantification process to the NUREG-1335 and quality PRA attributes is difficult because the details of the quantification process used in the IPEs are not always available. However, several general conclusions can be reached. First, the IPE submittals generally met the reporting guidelines presented in NUREG-1335. Most of the licensees discussed the methods and computer codes used in the quantification process. Because these methods and codes are adequate to treat the Boolean reduction process, it is expected that the core damage sequences are appropriately generated.

Second, the results of the IPE analysis were generally reported according to the guidelines listed in NUREG-1335 with one major exception. Many of the licensees failed to list those sequences eliminated as a result of application of human recovery actions. This failure in reporting made it difficult to ascertain if recovery actions were being applied according to the attributes for a quality PRA.

Third, 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% of the core damage frequency (CDF) would have been captured as described by both NUREG-1335 and the attributes of a quality PRA. Finally, many of the licensees went beyond the requests of NUREG-1335 by performing sensitivity studies and/or uncertainty analyses as part of their IPEs.

15.3 Comparison of IPEs to a Quality Level 2 PRA

GL 88-20 requested licensees to perform Level 2 analyses for their IPEs to search for plant-specific vulnerabilities to severe accidents. In addition, Supplement 1 of GL 88-20 (Ref. 15.7) provides insights of technical information obtained from completion of the Containment Performance Information Program for BWR Mark I containments, and Supplement 3 of GL 88-20 (Ref. 15.8) provided insights for BWR Mark II, Mark III, and pressurized water reactor (PWR) containments. The insights would be of benefit to licensees for searching for plant-specific vulnerabilities.

Section 2.2 of NUREG-1335 provides general guidance on the reporting of the Level 2 analyses in IPEs. In addition, Appendix A of NUREG-1335 specifies an approach to the Level 2 portion of an IPE, with guidance on plant familiarization, sequence grouping, determination of containment failure modes, development of containment event trees (CETs), determination of containment challenges and time of failure, determination of source term magnitudes, quantification of CETs, and performing sensitivity studies.

A quality full-scope Level 2 analysis includes an assessment of the full range of credible challenges to containment integrity (i.e., determination of plausible failure mechanisms and corresponding structural challenges), a characterization of the capacity of the containment to withstand various challenges (i.e., determination of performance limits), and an 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.

Since IPEs provide the foundation for the future use of IPEs/PRAs in risk-informed regulation, the quality of the Level 2 analyses in IPEs needs to be examined. This examination requires a comparison of Level 2 analyses between what was requested by the staff and what were submitted by licensees, and a comparison of IPEs to a quality Level 2 PRA. The comparisons are made for each key task activity associated with the PRA elements and tasks of a quality Level 2 PRA. These comparisons are discussed below in light of the scope of GL 88-20 and its supplements.

15.3.1 Assessment of Credible Challenges to Containment Integrity

15.3.1.1 Determination of Accident Sequences to be Assessed

Appendix A of NUREG-1335 gives an example of plant damage state (PDS) bin characteristics which can be used to consolidate the larger number of core damage sequences into a smaller number of PDSs. The intent is that all accidents within a particular PDS can be treated as a group for assessing accident progression, containment response, and radionuclide release.

Most licensees adequately treated the interface between the Level 1 and the Level 2 portions of the IPEs. For these IPEs, the licensees could track the effect on the Level 2 results caused by changes in (or dependencies from) the Level 1 analysis.

A large majority of IPEs did use the PDS methodology and provided descriptions of binning characteristics for their PDSs. However, cut set carryover appears to be difficult if not prohibited by the interface methodology. Many IPEs did not develop PDSs. However, there was one logic structure that started from initiating events to containment failure and radionuclide releases. This should assure the maximum coupling from the Level 1 analysis to the Level 2 analysis, including complete dependency and cut set carryover.

For most IPEs, conservation of CDF in going from the Level 1 analysis to the Level 2 analysis was upheld, and all Level 1 sequences are coupled to the Level 2 analysis. For some IPEs, the CDF used in the Level 2 analysis was a subset of the CDF calculated at the end of the Level 1 analysis. Thus, there is the possibility that certain initiating events (originally not important but possibly important in certain PRA licensing applications) could get lost and not be part of the Level 2 analysis.

A quality PRA would group cut sets on the basis of accident characteristics that influence either subsequent containment performance or resulting radionuclide source term. Many IPEs have met the attributes of a quality PRA in grouping cut sets on the basis of these accident characteristics.

15.3.1.2 Containment System Performance Analysis

NUREG-1335 noted that the coupling of the Level 1 analysis to the Level 2 is through the binning of the multitude of Level 1 sequences into a few groups of damage states with similar Level 2 characteristics, namely, timing of important events or operability of key features. Further, all Level 1 to Level 2 sequence interfaces need to be concisely documented, and the adopted binning needs to be justified.

Since many IPEs provided good descriptions concerning the attributes for PDSs, it was easy to follow which dominant core damage sequences, generated from the Level 1 analysis, were grouped into which PDS. For some IPEs, there was a lack of sufficient descriptions in the IPEs on which accident sequences were binned into which PDSs.

A large number of IPEs explicitly defined early and late containment failures. However, timing of events in the accident progression, and times to containment failure relative to core damage were not sufficiently described in some IPEs. Since GL 88-20 recommended at least 24 hours for the mission time, the mission times for IPEs varied from 24 hours to 48 hours. For many IPEs that stopped their accident progression analyses at 24 hours, some of these sequences could have the potential to contribute significantly to late containment failures. For example, some IPEs that indicated no containment failure at 24 hours might have a failed containment with radiological release if the analyses were carried out to 48 hours. Similarly, for some IPEs that went to 48 hours, carrying some sequences beyond 48 hours might have contributed to late containment failure. For example, for some PWR accident sequences, the containment would fail by overpressurization but not before 48 hours.

For many IPEs that used PDSs, the licensees provided sufficient documentation to describe how the mapping of CDFs into PDSs was performed at the cut set level. For some IPEs that used PDSs, there was insufficient information to determine whether the mapping of CDF into PDSs were performed at the cut set level, not at the accident sequence level. Therefore, it is not clear that dependencies and recovery actions on the cut set level were properly accounted for.

A quality PRA performs PDS binning or grouping at the cut set level. It appears that many IPEs met the attributes of a quality PRA in performing PDS binning or grouping at the cut set level.

15.3.1.3 Severe Accident Progression Analysis

Most IPEs seem to carry the entire core damage frequency into the Level 2 analysis. However, methods, scope, and technical rigor of the Level 2 analysis in most IPEs are not comparable to those applied in the Level 1 analysis. This is because the methodology for the Level 2 analysis is not as well developed and understood as that of the Level 1 analysis.

Many IPEs have linked containment systems models directly with accident sequence models from the Level 1 to account for mutual dependencies. Some IPEs took advantage of the software used available to link the Level 2 to the Level 1 analysis data. In some IPEs, containment systems were explicitly modeled; in other IPEs, it was not possible to discern the actual state of the containment systems regarding their operability and availability. A large majority of IPEs did not take credit for non-safety containment systems. For example, many PWR IPEs did not take credit in the analyses for fan coolers and cross-ties to containment spray via alternate systems.

Since many IPEs did not provide enough description regarding the environmental impact of long-term performance of containment safeguard systems during severe accidents, it is not clear whether the environmental impact on containment systems were examined in the IPEs. Nevertheless, some IPEs examined the effects of steam and aerosol on some vital equipment. Prolonged high-temperature effects on containment penetration elastomer seal materials were generally considered in IPEs.

In general, the MAAP code was the most popular code that was used to characterize accident timing and containment response. By varying the magnitude of MAAP input parameters, sensitivity analyses were performed to quantify the effects of varying assumptions. A majority of IPEs performed sensitivity analyses founded on the Electric Power Research Institute (EPRI) recommendation list; however, there is generally not enough information in IPEs to explain why some EPRI recommended sensitivity analyses were not performed. A few licensees performed more credible sensitivity analyses without solely relying on the EPRI's recommended list.

A majority of IPEs used analyses from surrogate plants (very often the NUREG-1150 (Ref. 15.9) plants) to substitute for their plant-specific analyses. Some IPEs provided brief descriptions to explain why the analyses from surrogate plants were applicable to their plants. However, sensitivities addressing the differences were not performed in many cases.

Most IPEs have mentioned the appropriate accident phenomena in their IPEs. However, some potentially important phenomena, such as direct containment heating, were dismissed from further consideration without adequate justification. The important phenomena were often treated in an "absolute" manner without any consideration of uncertainty.

A number of IPEs addressed phenomenological uncertainties qualitatively by using technical position papers generated by Fauske and Associates, which included a literature search as part of the basis for the positions taken. The plant-specific technical papers seem to have been developed from a set of generic technical papers with minor modifications.

The severe accident progression analysis in a quality Level 2 PRA would include the following considerations:

- The entire core damage frequency is carried forward into the Level 2 analysis.
- Methods, scope, and technical rigor are comparable to that applied in the Level 1 analysis.
- Containment systems models are linked directly with accident sequence models from the Level 1 to fully account for mutual dependencies.
- Long-term performance of containment safeguard systems accounts for degradation of environments within which the system must operate during 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 must be evaluated by some other means.

Many IPEs do not meet this attribute of a quality Level 2 analysis in performing accident progression analysis.

15.3.2 Characterization of Containment Performance Limits

15.3.2.1 Engineering Analyses of Structural Capacity to Withstand Loads

Appendix 1 of GL 88-20 noted the use of existing structural analyses to determine the ultimate pressure capability of the containment (i.e., the quasi-static internal pressure resulting in containment failure). These should be modified to account for any unique aspects that could substantially modify the range of possible failure pressures.

Many IPEs used outside contractors in performing containment structural analyses. Some IPEs used computer codes and performed the containment structural analyses in-house. However, a few licensees did not use outside contractors, did not perform any in-house code calculations for containment structure, and used existing analyses of similar plants and extrapolated the results to their plants.

Many IPEs used the estimate for hydrogen concentration resulting from 100% metal-water reaction and performed bounding analyses (homogeneous adiabatic assumptions) to estimate the load on containment resulting from a single global deflagration of hydrogen. Some IPEs used the hydrogen concentrations from MAAP code runs and performed analyses on containment loading resulting from hydrogen combustion. However, most IPEs did not examine the effects of global or local hydrogen detonations on containment structure.

A quality PRA would require engineering analyses of performance limits to account for plant-specific design features, such as discontinuities in the containment shell, penetrations, and interactions with neighboring structures. A number of IPEs appeared to have met the attributes of a quality PRA in performing containment structural analyses.

15.3.2.2 Systematic Search for Plant-Specific Containment Failure Modes

Appendix 1 of GL 88-20 noted that licensees identify the most probable list of potential containment failure mechanisms applicable to the plant under consideration (e.g., Table 7-1, NUREG/CR-2300). Furthermore, NUREG-1335 provided a list of potential failure modes and mechanisms (direct bypass, failure to isolate, vapor explosions, overpressurization, combustion processes, core-concrete interaction, blowdown forces, melt-through, and thermal attack of containment penetrations).

A majority of licensees performed an adequate job in analyzing their containments to determine the different containment failure modes resulting from overpressure, overtemperature, and core debris contact. However, many licensees did not provide information to explain why other containment failure modes were dismissed for their plants. It is not clear for many IPEs whether the effects of localized design features were captured in the analyses. A few IPEs examined their plant layout and systems in detail and identified plant-specific containment failure modes. In response to the findings from the Containment Performance Improvements program, some licensees that have large, dry PWR containments have performed walkdowns of containment to search for locations where local hydrogen pocketing might occur.

A quality PRA would require evaluating ultimate pressure capacity using a finite-element model of sufficient detail to capture the effects of localized design features and performing engineering analyses to characterize likely failure sizes for each postulated failure mode. Many IPEs did not meet the attributes of a quality PRA in searching for plant-specific containment failure modes.

15.3.3 Probabilistic Characterization of Containment Performance

15.3.3.1 Development of a Probabilistic Logic Structure to Trace Severe Accident Progression

Appendix 1 to GL 88-20 noted that system/human response should be realistically integrated with phenomenological aspects into simplified, but realistic CETs for the plant being examined. Allowance should be made for the probability of recovery or other accident management procedures (particularly for long-term responses). The quantification of the CETs should both clearly take into account the expected progression of the accident and aim to envelop phenomenological behavior (i.e., account for uncertainties).

Most licensees developed large or small CETs to characterize containment responses to core damage sequences. A few IPEs used plant response trees to integrate Level 1 analysis with the Level 2 analysis and, therefore, PDSs were not needed.

For those that used large CETs, the licensees examined the CETs in the NUREG-1150 studies to select the top event nodes appropriate for their plant-specific CETs. While none of these IPEs had the level of detail and complexity of the NUREG-1150 CET structure, many had a robust structure, with sufficient top events (and coupling to logic trees that defined the top events) to allow for a thorough understanding of the progression of the severe accidents to containment failure. The CETs in these IPEs had a good mix of "phenomenological," "system" and, if important, "operator action" top events.

For those that used small CETs, the licensees used simplistic CETs with a few top events. Very often these top events were system-oriented, and the containment phenomenological issues and operator actions were not explicitly addressed in the CETs. However, for some IPEs that used the small CET approach, more detailed decision/decomposition trees representing the phenomena were presented.

For several licensees that relied on their contractors' technical position papers, the CETs in their IPEs had a limited number of top event nodes because the effects of important severe accident phenomena were dismissed in the technical position papers and hence were not in the CETs. As a result, no early containment failures were expected to result from severe accident phenomena including steam explosions, direct containment heating, vessel thrust forces, thermal attack on containment penetrations, and hydrogen combustion.

In general, many IPEs did not consider the impact of operator performance on the accident progression for PWRs. For BWRs, the necessary operator actions were generally recognized although not adequately quantified or justified.

A substantial number of IPEs used fault tree analyses to estimate containment isolation failure probability. However, it is not clear whether common-mode failures have been considered in the fault tree analyses.

A quality Level 2 PRA would use logic structures to explicitly recognize important time phases in severe accident progression, and consistently address the interdependence of accident phenomena to create a time-line of accident progression for each accident sequence.

Many IPEs did not meet the attributes of a quality PRA in producing logic structures to adequately treat the accident phenomena and human interactions. Since their CET structures were incomplete or too simplistic, serious limitations exist as to their usefulness in regulatory applications.

15.3.3.2 Assignment of Probabilities to Uncertain Parameters and Events in the Logic Model

Appendix 1 to GL 88-20 states that CET quantification should include consideration of uncertainties. NUREG-1335 stated that a well-structured sensitivity study ought to determine what has the largest effect on the likelihood or time of containment failure and the magnitude of the source term without calculating the uncertainties explicitly. Table A.5 of NUREG-1335 provides a table of parameters for sensitivity study.

Many licensees selected the split fractions for their CETs using the values from the NUREG-1150 study. A few licensees used the fault tree analyses to estimate the split fractions for CETs. Some IPEs assigned zeros and ones to the split fractions for their CETs. In general, the values assigned to split fractions of CETs seemed arbitrary and not justified. Many IPEs varied the magnitude of split fractions in sensitivity studies to consider the impact of uncertainties.

A quality PRA would address uncertainty in the quantification of events and phenomena through the assignment of probability density functions to individual events (i.e., logic model provides statistical mechanism for combining the distributions in a consistent manner). In general, IPEs do not meet the attributes of a quality Level 2 PRA in addressing uncertainty of phenomena because probability density functions were not used in IPEs to quantify uncertainties of phenomena.

15.3.4 Characterization of Radionuclide Release

15.3.4.1 Source Term Definition

The proper characterization of the source term includes analyzing a representative number of radionuclide release categories and determining the quantity, timing, and duration of the releases. NUREG-1335 noted that the timing, magnitude, and characteristics of accidental radionuclide releases can be determined after the timing and mode of containment failure or bypass were determined and a description of the various potential radionuclide paths are obtained. For many IPEs, the licensees provided a sufficiently complete representation of the accident source terms. However, a number of IPEs contained an incomplete representation of the source terms; for example, only cesium iodine (CsI) is considered for early releases in some IPEs.

A quality PRA would define radiological source term(s) for each accident sequence and failure mode in the Level 2 model. Many IPEs do not meet the attributes of a quality PRA in defining their radiological source terms.

15.3.4.2 Coupling of Source Term and Severe Accident Progression Analysis

NUREG-1335 noted that determination of source terms may be done by the selection of source terms for similar sequences that have been identified for a similar plant or by code calculation. If a large number of source term calculations are combined into a set of release categories, the rationale for the process should be provided.

Some IPEs used results from plant-specific MAAP runs to estimate source terms for each dominant accident progression. A few IPEs did not use the MAAP code but some other code to estimate source terms. Some IPEs used results of analyses for a reference plant to estimate source terms. One IPE directly applied the pertinent source terms from the NUREG-1150 study. For many IPEs, there was insufficient information to determine how representative source terms were obtained from the accident progression analyzed.

A quality PRA would quantify uncertainties in the radiological source term for a given accident sequence to identify the distribution of plausible environmental releases, identify correlations between uncertain parameters associated with radionuclide release, transport, and deposition and with other aspects of severe accident behavior and treat them consistently, and use sufficient information to characterize source terms to calculate environmental consequences in a Level 3 PRA. In general, the IPEs do not meet the attributes of a quality PRA in quantifying uncertainties in the source terms.

15.4 Comparison of IPEs to a Quality Level 3 PRA

Since one of the four objectives of the IPE program was to gain a more quantitative understanding of the overall probabilities of core damage and radionuclide releases, GL 88-20 did not ask licensees to assess the consequences (the impacts on human health, the environment, and economics) associated with accidental releases. Therefore, most licensees did not elect to perform Level 3 analyses for their IPEs. A few IPEs provided descriptions of input data and assumptions to be used in a Level 3 analysis. For example, one licensee used the Calculations of Reactor Accident Consequences Version 2 Code (CRAC2) (Ref. 15.10) to perform consequence calculations. The input data for CRAC2 includes meteorological data, deposition rates, radiation pathways, evacuation and sheltering, health physics data, and population distribution. The limited number of Level 3 analyses reported in the IPEs have not been reviewed by the staff.

The consequence analysis of a quality Level 3 PRA would require collecting demographic and weather-related data (i.e., data on local meteorology and terrain, site demographics, and local land use representing current, plant-specific conditions), appropriate source terms from the Level 2 analysis, and performing consequence calculations for the full range of early and late health effects. The consequence calculations would address variability in weather.

It has not been determined whether the small number of Level 3 analyses performed by licensees are consistent with the attributes of a quality Level 3 PRA.

15.5 Comparison of IPEs to a Quality Internal Flooding Analysis

In GL 88-20, the NRC staff indicated that the IPEs should include an examination of internal flooding events. Other external events such as internal fires and seismic events were deferred to allow the NRC staff to identify which events should be analyzed, to potentially develop simplified methods for analyzing these events, and to coordinate ongoing programs at the NRC and within the industry. NUREG-1335 did not provide any specific guidance on performing the internal flooding analysis other than that the results of the analysis be included in the submittal. More specific guidance on a quality internal flooding analysis is provided in Chapter 14 of this document. This guidance includes discussion on identifying and screening flood sources, identifying flood propagation pathways, and including flood-induced failure mechanisms.

A general comparison of the internal flooding analyses reported in the IPEs to the guidance criteria provided for a quality PRA was performed. Since the internal flooding analysis uses the same processes and models used in the internal events analysis, this comparison focused on the modeling pertinent to the internal flood analysis. However, the conclusions reached on the general acceptability of the internal event models, as addressed above in Section 15.2, are also applicable to the internal flooding analyses. In general, the IPE flood analyses contain most of the attributes of a quality PRA. Some highlights of this comparison are provided below.

The majority of the IPEs appeared to have comprehensively considered possible internal flooding sources consistent with a quality PRA. The sources considered included tank ruptures, pipe breaks, and seal failures. 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 potential for human-induced flooding was included in the IPE analyses.

In general, the modeling of flood scenarios in the IPEs appeared to be consistent with a quality PRA. 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 included 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 analyses.

The flood zone definitions appeared from the limited information provided in the IPE submittals to consider the existence of barriers such as walls, doors, and curbing. In addition, consideration of drains that would limit a flood to a particular area was also generally considered in the flood zone definitions. Flood propagation pathways from one flood zone to another through stairwells, drains, hatches, and doors were also generally considered in the IPEs. The failure of barriers to propagation was not a significant factor in the IPE analyses. Finally, isolation of floods was often included in the IPE evaluations. However, the methods of detecting a flood and the time available for performing such isolations were not always provided with a firm basis.

15.6 Comparison of IPEs to a Quality Peer Review Process

In GL 88-20, the NRC requested that each utility perform "...an independent in-house review to ensure the accuracy of the documentation packages and to validate both the IPE process and its results." The purpose of the in-house review, as discussed in NUREG-1335, was twofold. First, the combination of persons performing and reviewing the IPE would give the utilities a group of PRA-experienced personnel to use the IPE in other applications. Second, the in-house review would provide quality assurance and control to the IPE process. Independence of the quality assurance and review team is an essential feature in this quality control aspect. The NRC staff indicated that use of staff from an adjacent unit or outside contractors would help achieve some level of independence. However, there was no requirement for using outside contractors in the review process.

The attributes in Chapter 14 indicate that an external peer review is an essential aspect of a quality PRA. A quality external peer review addresses two major areas:

(1) review team composition and qualifications
(2) review process

The attributes of peer review describes a team of individuals with direct experience in ***performing all elements*** of a PRA. Knowledge of the techniques, models, methods, assumptions, scope, and other attributes associated with each element and task of a PRA is necessary in a quality peer review. With this expertise, the team can determine whether the proper methods were selected and implemented. In addition, the peer review is intended to be "independent" to ensure that an inherent "bias" is not introduced into the peer review process. Therefore, individuals who performed the PRA and other utility personnel are excluded from the peer review team.

The peer review is intended to ensure that the analysis is complete and the results are reasonable considering the design and operation of the plant. The peer review is not intended to perform a detailed verification of all inputs and analyses. Therefore, the peer reviewers are expected to spot check selected portions of the analyses in conjunction with the goals of the peer review.

Most of the licensees performed an in-house review of their IPEs. This process generally involved a review of the information used in the modeling process by utility personnel unfamiliar with PRA (e.g., system engineers, operators, and plant management personnel). This level of review provided assurance that the as-built, as-operated plant was being modeled, but generally could not address the content or the accuracy of the PRA models. To alleviate this shortcoming, many licensees chose to use external PRA consultants to review their IPE models. The extent of these external reviews was generally not documented in the IPE submittals and thus whether the review process meets the attributes described in Chapter 14 is largely unknown at this time.

REFERENCES FOR CHAPTER 15

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15.2 Hickman, J.W., "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," NUREG/CR-2300, Vols. 1 and 2, American Nuclear Society and Institute of Electrical and Electronic Engineers, January 1983.

15.3 M. McCann, *et al.*, "Probabilistic Safety Analysis Procedure Guide," NUREG/CR-2815, Vols. 1 and 2, Revision 1, Brookhaven National Laboratory, August 1985.

15.4 Erickson, D.M., Jr. (editor), *et al.*, "Analysis of Core Damage Frequency: Internal Events Methodology," NUREG/CR-4550, Volume 1, Revision 1, Sandia National Laboratory, January 1990.

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

15.6 EPRI,"MAAP4-Modular Accident Analysis Program for LWR Power Plants," RP 3131-02, Vols 1-4, 1994.

15.7 USNRC, "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities," Generic Letter 88-20, Supplement 1, August 29, 1989.

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

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

15.10 L.T. Ritchie, et al., "CRAC2 Model Description," NUREG/CR-2442, SAND82-0342, Sandia National Laboratories, 1984.

PART 5

ADDITIONAL IPE PERSPECTIVES

16. SAFETY GOAL IMPLICATIONS

Chapter 7 of Part 1 summarizes the key perspectives on several additional items. Section 7.1 (of Chapter 7, Part 1) summarizes the Individual Plant Examination (IPE) results relative to the current risk level of U.S. plant as compared with the Commission's Safety Goals.[16.1] Chapter 16 provides a more in-depth discussion including the approach adopted to infer how the IPE results were compared to the quantitative health objectives.

16.1 Background

The Safety Goal Policy Statement (Ref. 16.1) established two qualitative safety goals which are supported by two quantitative health objectives (QHOs):

> "The risk to an average individual 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.
>
> 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%) of the sum of cancer fatality risks resulting from all other causes."

For purposes of comparison with the results of probabilistic risk assessments, these QHOs have been translated into two numerical objectives, as follows:

(1) ***Prompt fatalities QHO***: The individual risk of a prompt fatality from all "other accidents to which members of the U.S. population are generally exposed," such as fatal automobile accidents, etc. is about 5E-4 per reactor-year (ry). One-tenth of one percent of this figure implies that the individual risk of prompt fatality from a reactor accident should be less than 5E-7/ry. The "vicinity" of a nuclear power plant is understood to be a distance extending to 1 mile from the reactor. The "average" individual risk is determined by dividing the number of prompt or early fatalities (societal risk) to 1 mile due to all accidents, weighted by the frequency of each accident, by the total population to 1 mile and summing over all accidents.

(2) ***Cancer fatalities QHO***: "The sum of cancer fatality risks resulting from all other causes" is taken to be the cancer fatality rate in the U.S. which is about 1 in 500 or 2E-3/ry. One-tenth of one percent of this implies that the risk of cancer to the population in the area near a nuclear power plant due to its operation should be limited to 2E-6/ry. The "area" is understood to be an annulus of 10 mile radius centered on the plant. The cancer risk is also determined on the basis of an 'average individual', i.e. by evaluating the number of latent cancers (societal risk) due to all accidents to a distance of 10 miles from the plant, weighted by the frequency of the accidents, dividing by the total population to 10 miles, and summing over all accidents.

When comparing the IPE results with the above QHOs, it should be noted that the scope of the IPE program is limited to accidents initiated by internal events (excluding internal fires) during full power operation only. Therefore, the risk estimates inferred from the IPE results may reflect only a fraction of the total risk of operating the plant.

[16.1] The safety goal objectives are targets for generic regulatory requirements, and not requirements, standards, or criteria for individual licensing decisions.

The results of other probabilistic risk assessments (PRAs) that include external events (and internal fire) and other modes of operation (e.g., low power, shutdown) indicate risk levels comparable with those obtained for internal events during full power operation.

The issuance of the Safety Goal Policy Statement was followed by the Integration Plan for Closure of Severe Accident Issues (Ref. 16.2). The Integration Plan has several elements, one of which is the IPEs. However, the licensees, in responding to Generic Letter (GL) 88-20 (Ref. 16.3), are not requested to calculate off-site health effects (although some did, only two compared their results to the QHOs), and thus the IPE results, in general, cannot be directly compared against the above health objectives.

During the safety goal policy implementation, several subsidiary objectives were discussed (Ref. 16.4). In particular, core damage frequency (CDF), conditional containment failure probability (CCFP), and a "large" release frequency were proposed. The concept of large release frequency was subsequently dropped, but the CDF and CCFP objectives have been used in a number of applications. As most IPEs performed the equivalent of a Level 2 PRA, the numerical values of these subsidiary objectives can be compared directly to the IPE results.

16.2 Subsidiary Objectives

A total CDF of 1E-4/ry and a total CCFP of 0.1 have been proposed (Ref. 16.4) as numerical objectives. The CCFP[16.2] is usually defined as the frequency of early containment failure and bypass divided by the CDF. The numerical values of these objectives can be obtained directly from the IPE results and are presented in Figure 16.1 (core damage frequency) and Figure 16.2 (conditional containment failure probability). Figure 16.2 presents the CCFP values for early structural failure of the containment and containment bypass accidents separately.

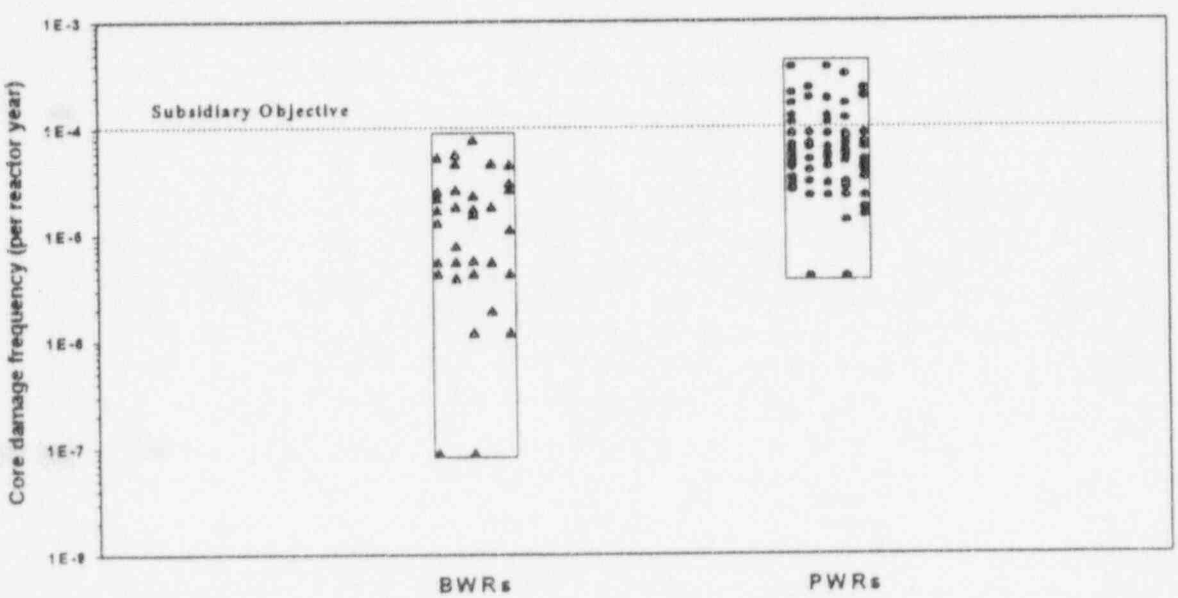

Figure 16.1 Core damage frequency for BWR and PWR IPEs.

[16.2]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 on the definition and estimation of conditional failure probability.

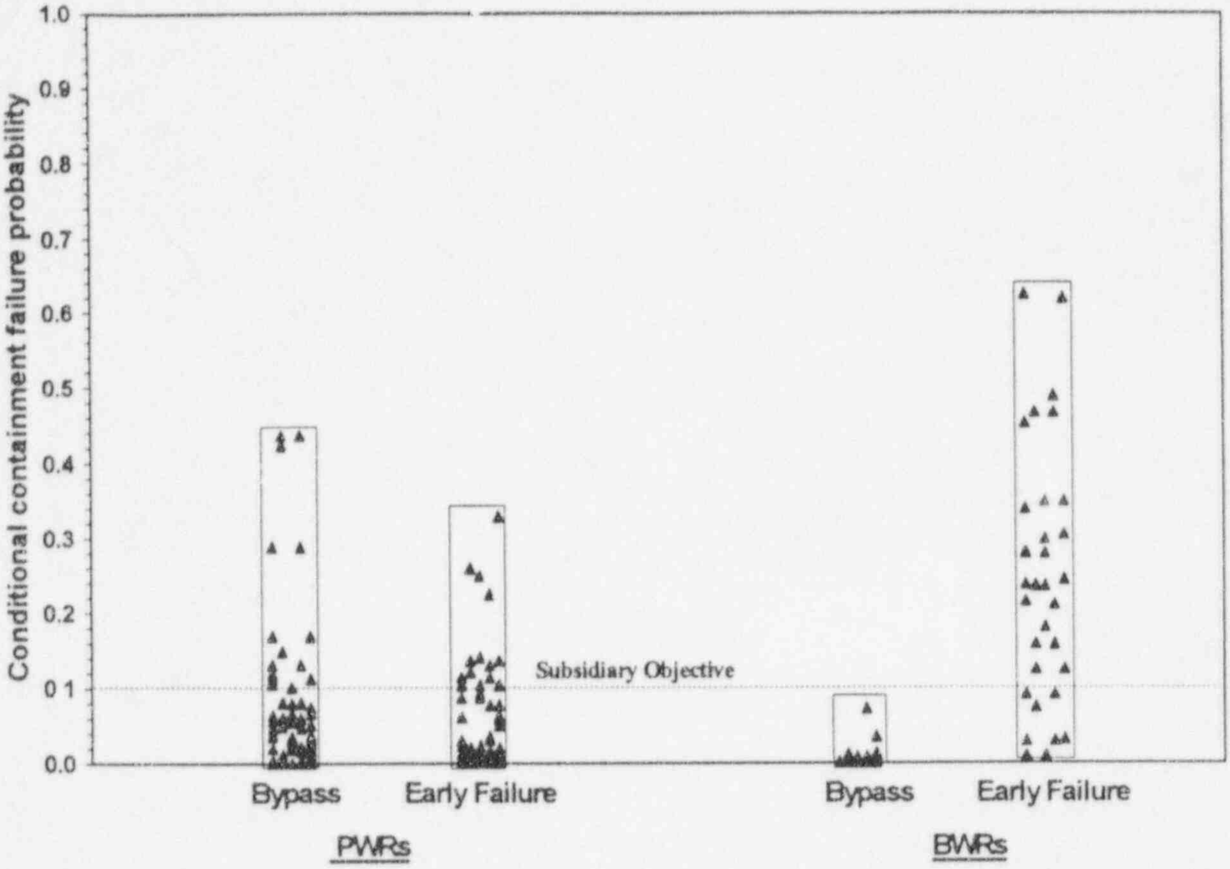

Figure 16.2 Conditional containment bypass and early failure probabilities for all BWRs and PWRs.

The CDFs for all of the boiling water reactors (BWRs) and most of pressurized water reactors (PWR) populations fall below 1E-4/ry. Nine licensees representing 15 PWR units reported CDFs above 1E-4/ry (refer to Chapter 11 for a discussion on the factors influencing these results). Most of the CCFPs for bypass and early containment failure for PWRs are below 0.1. 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. These results were not unexpected given the characteristics of BWR pressure suppression containments and PWR large volume containments (refer to Chapter 12 for a detailed discussion on containment performance during severe accidents).

16.3 Comparison With the Quantitative Health Objectives

Since several plants do exceed the subsidiary objectives, the QHOs are compared against the IPE results in order to determine how well the population of plants meet the Safety Goals. However, as noted above, in general, off-site consequences are not calculated in the IPEs. Off-site health effects are reported, though, for five plants in NUREG-1150 (Ref. 16.5). The NUREG-1150 results can then be used to extrapolate the IPE results to risk estimates. The approach adopted is a two-step process, as shown in Figure 16.3.

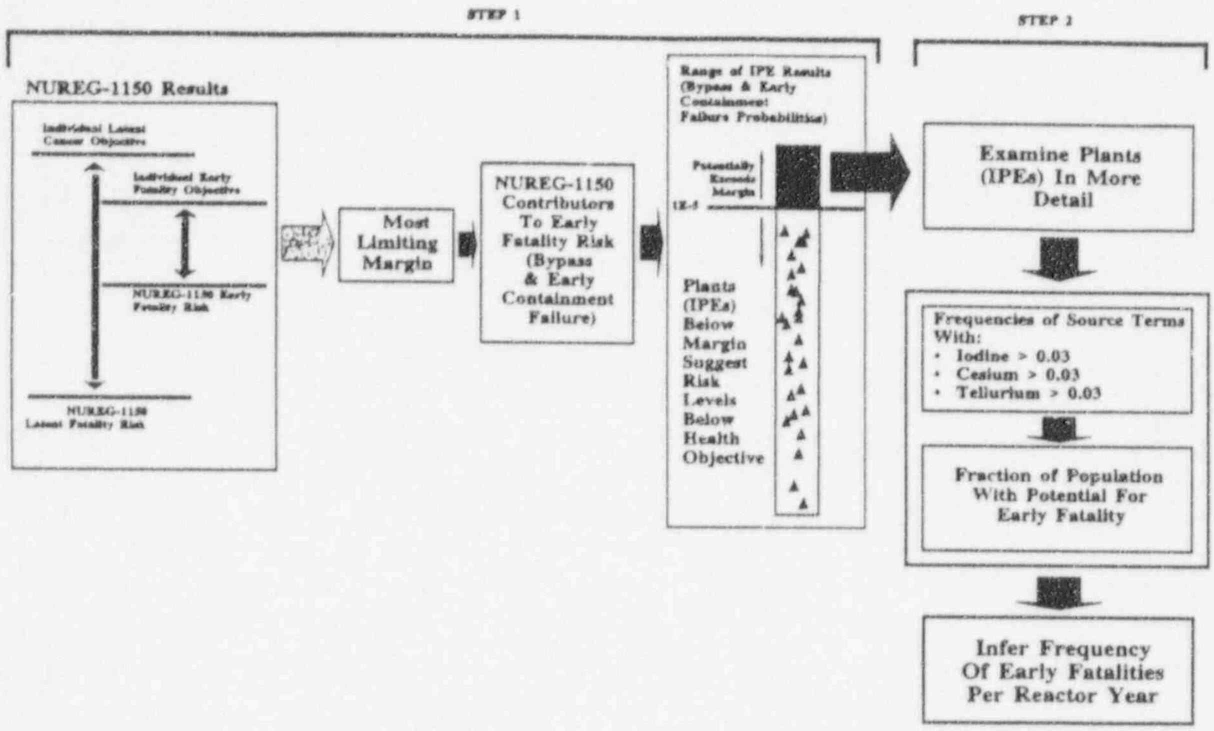

Figure 16.3 Approach for comparing IPE results to quantitative health objectives.

(1) In the first step, a relatively simple approach is used. The key containment failure modes that contribute to off-site health risks were identified in NUREG-1150 for several plants. The frequencies of key containment failure modes (such as early structural failure and bypass) as reported in the IPE submittals are then compared to the NUREG-1150 results. This comparison is used to infer to how the IPE results might compare to the health objectives of the safety goals.

(2) In the second step, those plants that have relatively high frequencies for the key containment failure modes are examined in more detail to more accurately assess the risk estimates.

One of the objectives 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. Several risk measures were calculated, including individual fatality risks for comparison with the QHOs. This comparison is reproduced from NUREG-1150 in Figure 16.4.

The results reported in NUREG-1150 (refer to Figure 16.4) indicate that the early fatality and latent cancer fatality risk estimates for all of the five plants studied are below QHOs. Based on 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. Individual latent cancer fatality risk would have to increase (for Sequoyah or Zion) by over 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.

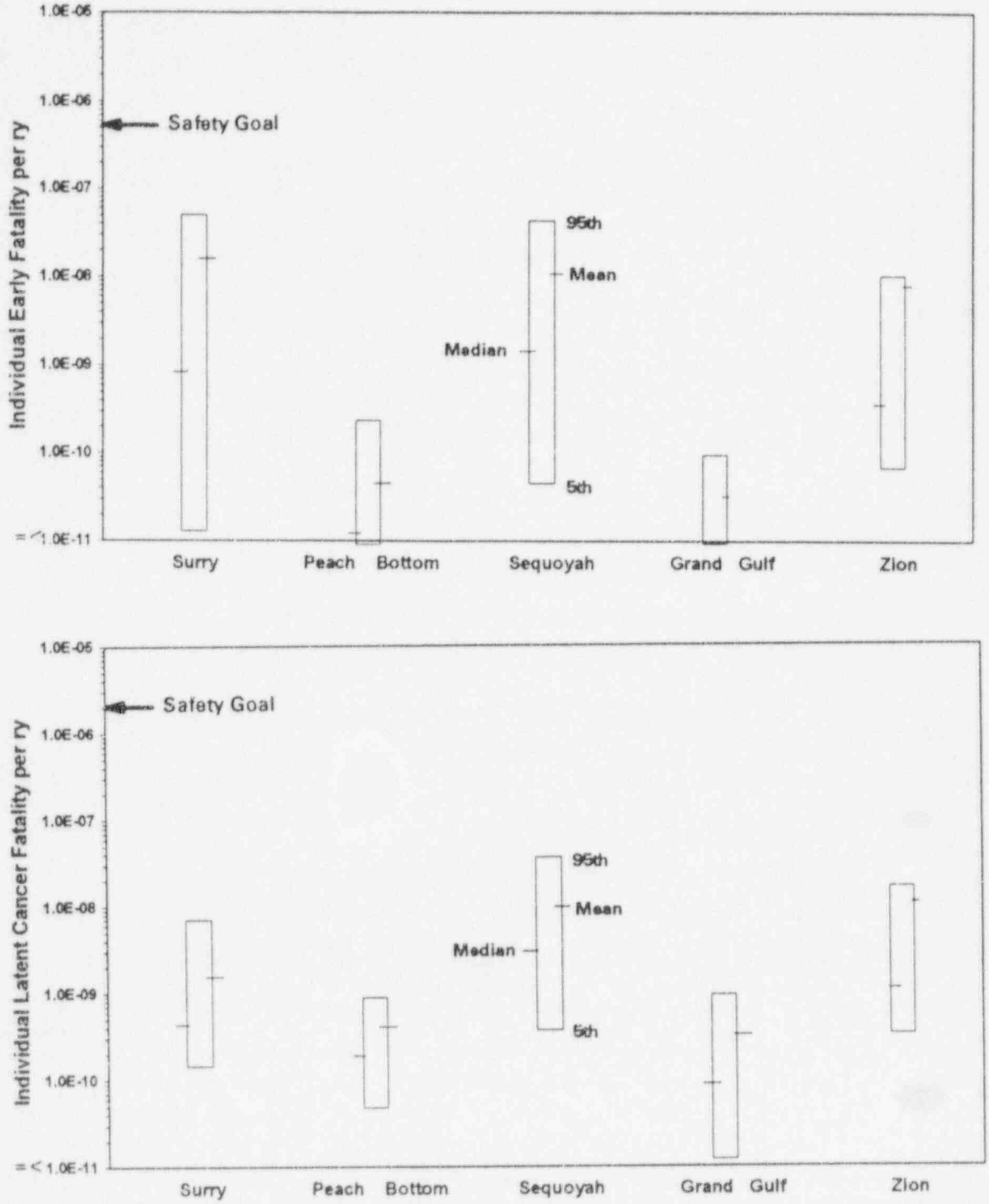

Figure 16.4 Comparison of individual early fatality and latent cancer fatality risk at the NUREG-1150 Plants (internal initiators).

NUREG-1150 reports the contributions of the various containment failure modes to mean early fatality risk and population dose (which is proportional to latent cancer fatality risk) for the five plants. The percentage contribution of the containment failure modes to these off-site health consequence measures is given in Tables 16.1 and 16.2 below:

Table 16.1 Principal contributors to early fatality risk.*

Containment failure modes	Grand Gulf	Peach Bottom	Sequoyah	Surry	Zion
Early Containment Failure	86%	89%	38%	7%	93%
Late Containment Failure	12%	3%	<1%	<1%	-
Containment Vented	2%	8%	-	-	-
Containment Bypass	-	-	62%	93%	7%
No Containment Failure	-	-	-	-	-

*Reproduced from supporting documentation (Ref. 16.6) to NUREG-1150.

Table 16.2 Principal contributors to population dose.*

Containment failure modes	Grand Gulf	Peach Bottom	Sequoyah	Surry	Zion
Early Containment Failure	61%	84%	35%	5%	77%
Late Containment Failure	28%	5%	22%	17%	-
Containment Vented	4%	10%	-	-	-
Containment Bypass	-	-	43%	78%	23%
No Containment Failure	7%	1%	-	-	-

*Reproduced from supporting documentation (Ref. 16.6) to NUREG-1150.

From an inspection of the above tables, it is clear that accidents that result in containment bypass or early containment failure are dominant contributors to both early fatality and latent cancer fatality risk. Consequently, if one compares the frequencies of early containment failure or bypass reported in the various IPE submittals with the NUREG-1150 results, one can draw conclusions regarding how the population of plants might compare with the NRC quantitative health objectives. The early containment failure and bypass frequencies reported in each of the IPEs are compared with the NUREG-1150 frequencies in Figure 16.5. 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.

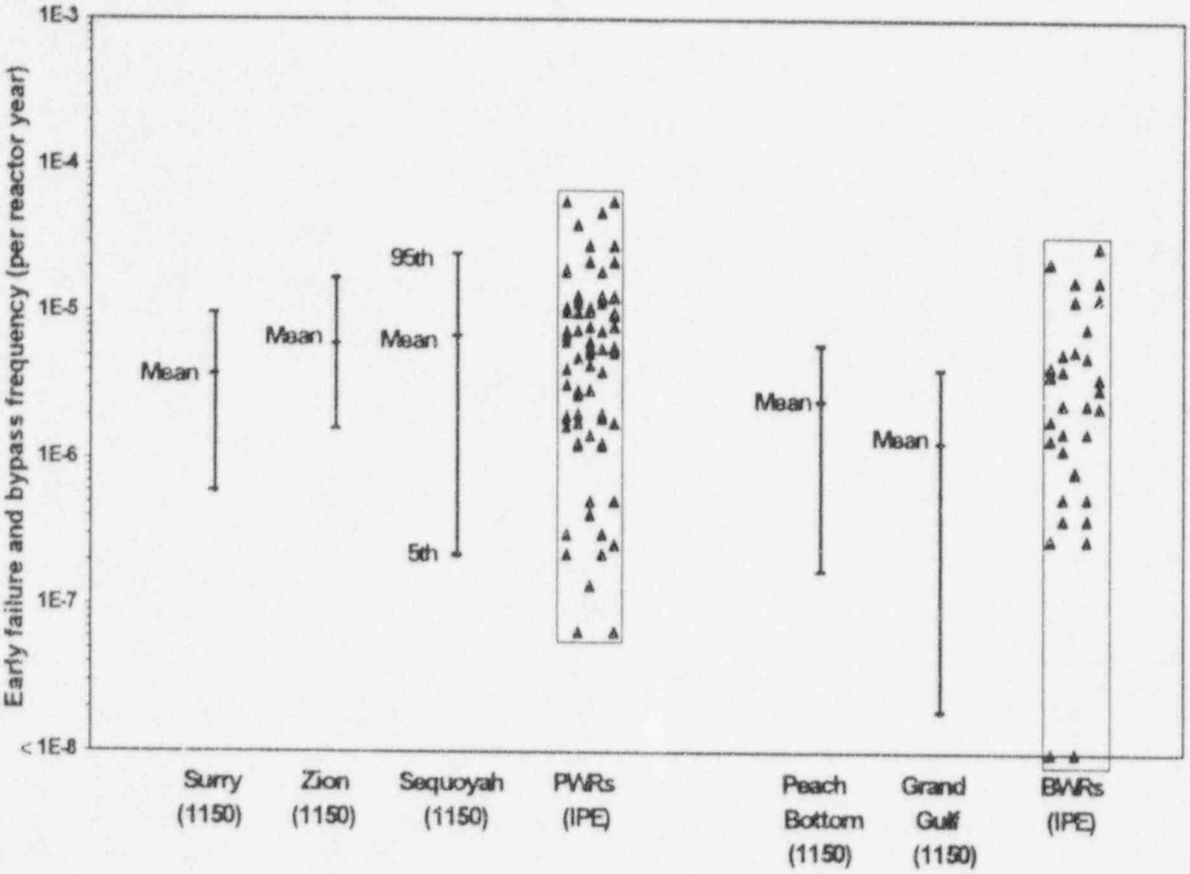

Figure 16.5 Comparison of NUREG-1150 and IPEs early failure and bypass frequency for PWRs and BWRs.

A comparison of the frequencies of early containment failure or bypass given in Figure 16.5 indicates that most of the IPE results are similar to or lower than the NUREG-1150 results. This implies (based on the reported IPE results) that most plants have similar risk levels to those reported in NUREG-1150. However, there are a number of plants that have frequencies of early containment failure or bypass that are an order of magnitude higher than the Surry results. If the Surry results are extrapolated 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 those plants with higher frequencies of early containment failure and bypass could approach this health objective. However, scaling risk estimates by using the ratios of containment failure and bypass frequencies could be 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. For these reasons, the higher frequencies of early containment failure and bypass for the IPE results in Figure 16.5 may not translate into equivalent increases in the early health risk. Therefore, those plants with frequencies of early containment failure and bypass higher than 1E-5/ry are examined in more detail. As shown in Figure 16.6 this resulted in 79 units being screened out as candidates for approaching the early fatality QHO.

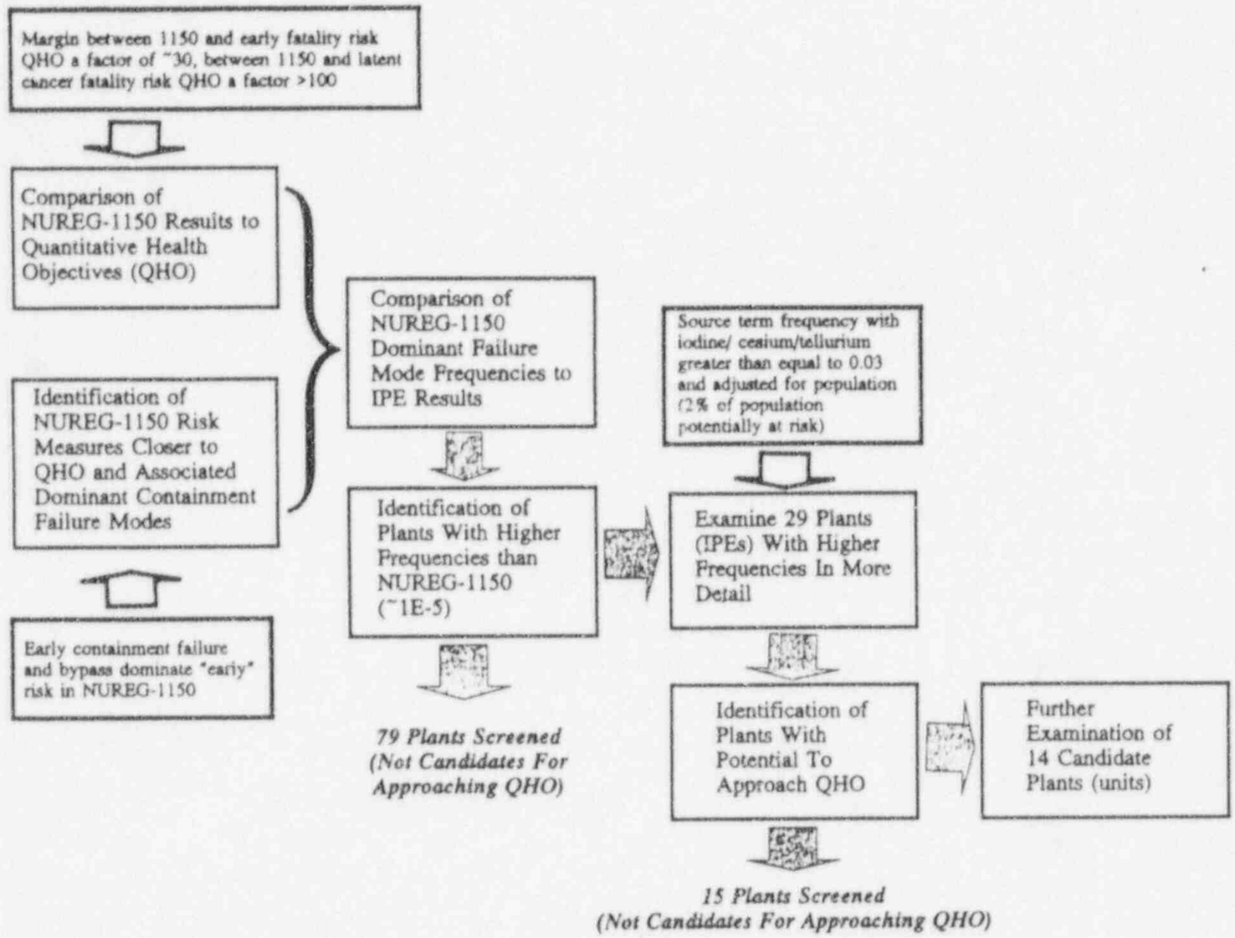

Figure 16.6 Process for comparing IPE results to quantitative health objectives.

Eighteen IPE licensees (representing 29 units) reported frequencies of early containment failure and bypass higher than 1E-5 per reactor year. Each of these submittals is reviewed to determine the frequencies of source terms that would be expected to give rise to early fatalities (EFs) within 1 mile of the plant. One licensee (LaSalle 1&2) submitted a Level 3 IPE and was eliminated from further consideration based on the reported results. The individual risk of early fatality (IREF) embodied in the safety goal is given by:

$$\text{IREF} = \sum_i (\text{Frequency of Release i}) * \frac{\text{Number of EF to 1 mile from Release i}}{\sum \text{Population to 1 mile}} \tag{1}$$

On the reasonable assumption that the only major contribution to early fatality is from sequences leading to either early containment failure or bypass (ECF/B), the above equation can be rewritten as:

$$\text{IREF} = \sum_i (\text{Frequency of ECF/B})_i * \frac{\text{Number of EF to 1 mile from }(\text{ECF/B})_i}{\sum \text{Population to 1 mile}} \tag{2}$$

In general, the IPEs contain information which bins the frequencies of the plant damage states (PDS) into several containment failure modes, including early failures and bypass. Individual source terms or release classes (RCs) are then identified under the particular failure mode category with a conditional frequency (CF) of occurrence, i.e., given the occurrence of the PDS. Equation (2) can then be written as:

$$\text{IREF} = \sum_i (\text{Frequency of PDS})_i * \sum_j \text{CF (RCEF/B)}_{ij} * \frac{\text{Number of EF to 1 mile from (RCEF/B)}_{ij}}{\sum \text{Population to 1 mile}} \quad (3)$$

where $\sum_j$ sums over early failure and bypass release classes $(\text{RCEF/B})_{ij}$ belonging to a particular $(\text{PDS})_i$

$\sum_i$ sums over all plant damage states.

Generally, a Level 3 probabilistic consequence assessment code, such as MACCS (Ref. 16.7), is used to calculate the number of early fatalities to 1 mile at a particular site, taking into account site-specific meteorological data, the timing and energy of the release, the warning time before evacuation can begin, the evacuation speed, and the population distribution.

Absent such a consequence calculation, some approximations have to be made in order to permit a rough estimation of IREF from the information provided. The information that is available from the submittals is the frequencies of the plant damage states, the binning of the containment failure modes (early failure, bypass, late failure, etc.) for each PDS, the conditional frequency of each release class under each PDS and containment failure mode (conditional on the occurrence of the PDS), and the release fractions of various radionuclide groups belonging to each release class.

The following approximations are based partly on the calculations performed and the results obtained in the Large Release Study (Ref. 16.8):

1. All release classes belonging to the early failure/bypass containment failure modes are assumed to have a release timing shorter than the statutory warning time for start of evacuation, i.e., the release is assumed to occur before evacuation can begin.

2. Source terms associated with the early failure/bypass release classes which have release fractions of the volatile/semi-volatile radionuclides (iodine (I), cesium (Cs), tellurium (Te)) greater than about 0.03 can potentially give rise to early fatalities within 1 mile of the plant.[16.3]

[16.3]Most plants in the U.S. have been licensed with exclusion zone boundaries extending from 0.25 mile to 0.4 mile; with respect to the prompt fatality QHO, the population at risk lies in an annulus extending from the exclusion zone boundary to 1 mile. In the Large Release Study, (Ref. 16.8), many calculations were performed to determine the conditions under which an off-site early fatality could occur as a function of the fraction of the core inventory released of different radionuclide groups. The I, Cs, Te $\geq$0.03 threshold is based on a spectrum of these calculations. This threshold should not be confused with the 0.1 threshold for fractional volatile radionuclide releases discussed in Chapter 12. The 0.1 threshold was used by several licensees for the purpose of determining the frequencies of significant radionuclide release. The use of this definition of a significant release in Chapter 12 does not imply that source terms with lower radionuclide releases would not result in early fatalities.

3. For most of the neutral to stable weathers, the plume will extend laterally to about 1/3 of a 22.5° angular sector out to 1 mile. In other words, approximately 1/3 of a sector's population out to 1 mile can potentially suffer early fatalities given the occurrence of an early failure/bypass source term with I, Cs, Te release fraction ≥ 0.03 and assuming that the release starts before evacuation can occur.

4. As a first approximation, it is assumed that the population is uniformly distributed over all 16 angular sectors. Then, given the occurrence of an early failure/bypass release class $(RCEF/B)_{ij}$ (belonging to $(PDS)_i$) with I, Cs, Te release fractions ≥ 0.03,

$$\frac{\text{Number of EF to 1 mile from } (RCEF/B)_{ij}}{\sum \text{Population to 1 mile}} \approx \frac{1}{3} * \frac{1}{16} \approx \frac{1}{50} \approx 0.02 \tag{4}$$

This simplifies the IREF calculation to:

$$IREF = \sum_i (\text{Frequency of PDS})_i * \sum_j \text{Conditional Frequency of } (RCEF/B)_{ij} * 0.02 \tag{5}$$

where the $\sum_j$ extends over those early failure/bypass release class source terms which have I, Cs, Te fractions ≥ 0.03.

After this second step, 14 units (refer to Figure 16.6) are 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 per year). Of these 14 units, two are BWRs with Mark I containments. For these units (Browns Ferry and Hope Creek), shell melt-through is the primary cause of early containment failure. This failure mode was identified as a Mark I specific failure mode before the issuance of GL 88-20 and has been addressed separately through experimental research and analysis.

Eight of the other units are PWRs with large dry containments and relatively high CDFs and CCFPs. The relatively high CCFP (approximately 0.3) for one plant (Palisades) is caused by 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 (approximately 0.1) than Palisades and driven by assumptions regarding early over pressurization failures associated partly with direct containment heating. This failure mode was also identified before the issuance of GL 88-20 and is also being addressed by research activities. Two other units, Ginna and Haddam Neck, have relatively high frequencies of containment bypass.

The remaining four units are PWRs with subatmospheric containments. The CCFP probability for two of these units (Beaver Valley 1&2) is also relatively high (approximately 0.3) and isolation failure was 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 early fatality risk from internal initiators at Surry in the NUREG-1150 study, the mean frequency of bypass in NUREG-1150 was estimated to be more than one order of magnitude smaller compared to the IPE result. Hence, the mean of the NUREG-1150 distribution of the individual risk of early fatality lies well below the early fatality QHO).

While these preliminary estimates of early fatality risk based on the IPE results are approximate, they do point to the need to examine site-specific characteristics at individual plants more carefully. For example, the conditional consequences of early failures may be more severe at sites such as Beaver Valley which have a relatively higher population density as opposed to, say, Ginna with a much lower population in the plant vicinity. In addition, site meteorology and the size of the plant (i.e., the radionuclide inventory) can also influence off-site consequences. In summary, a comparison of the IPE results with the NUREG-1150 study, shows that most of the IPE results, with a few exceptions, are likely to meet the QHOs.

16.4 Summary

The IPE results, which reflect only accidents initiated by internal events at full power, imply risk levels below the individual latent cancer fatality health objective. The IPE results also suggest that risk levels at most plants are below the individual early fatality health objective, with a few possible exceptions. Although 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.

REFERENCES FOR CHAPTER 16

16.1 USNRC, "Safety Goals for the Operation of Nuclear Power Plants; Policy Statement," *Federal Register*, Vol. 51, p. 30028, August 21, 1986.

16.2 USNRC, "Integration Plan for Closure of Severe Accident Issues," SECY-88-147, May 25, 1988.

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

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

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

16.6 R.J. Breeding, et al., "Evaluation of Severe Accident Risks: Surry Unit 1," NUREG/CR-4551, Vol. 3, Revision 1, SAND86-1309, Sandia National Laboratories, October 1990.

A.C. Payne, Jr., et al., "Evaluation of Severe Accident Risk: Peach Bottom Unit 2," NUREG/CR-4551, Vol. 4, Revision 1, SAND86-1309, Sandia National Laboratories, December 1990.

J.J. Gregory, et al., "Evaluation of Severe Accident Risk: Sequoyah Unit 1," NUREG/CR-4551, Vol. 5, Revision 1, SAND86-1309, Sandia National Laboratories, December 1990.

T.D. Brown, et al., "Evaluation of Severe Accident Risk: Grand Gulf Unit 1," NUREG/CR-4551, Vol. 6, Revision 1, SAND86-1309, Sandia National Laboratories, December 1990.

C.K. Park, et al., "Evaluation of Severe Accident Risk: Zion Unit 1," NUREG/CR-4551, Vol. 7, Revision 1 BNL-NUREG-52029, Brookhaven National Laboratory, March 1993.

16.7 USNRC, "MELCOR Accident Consequence Code System (MACCS)," NUREG/CR-4691 (SAND86-1562), Vols. 1-3, February 1990.

16.8 USNRC, "Calculations in Support of a Potential Definition of Large Release," NUREG/CR-6094, May 1994.

17. IMPACT OF STATION BLACKOUT RULE ON CORE DAMAGE FREQUENCIES

Chapter 7 of Part 1 summarizes the key perspectives on several additional items. Section 7.2 (of Chapter 7, Part 1) summarizes 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 (SBO). Chapter 17 provides a more in-depth discussion including the details on the approach used to address the impact of the station blackout rule, and the factors affecting the station blackout core damage frequency (CDF).

17.1 Background of the Station Blackout Rule

An SBO is defined as the total loss of alternating current (AC) electric power to the essential and nonessential switchgear buses at a nuclear power plant. An SBO results when both offsite power and onsite emergency AC power are unavailable (Ref. 17.1). If AC power is not recovered in time, an SBO can result in core damage since most nuclear reactors require AC support for decay heat removal. Based on station blackout studies (Refs. 17.1 and 17.2) that highlight these concerns, the Nuclear Regulatory Commission (NRC) developed the station blackout rule (SBOR) (Ref. 17.3) to reduce the risk of severe accidents from SBO, with the goal of limiting the average contribution to core damage from SBO to about 1E-5/ry (Ref. 17.4) and (Ref. 17.5). Based on the SBOR, all licensees and applicants are required to assess the capability of their plants to maintain adequate cooling and appropriate containment integrity during an SBO. In support of the SBOR, the NRC developed Regulatory Guide 1.155 (Ref. 17.6), which describes a method acceptable to the NRC for maintaining a high level of reliability for emergency diesel generators (EDGs), developing procedures and training to cope with and recover from an SBO, and establishing plant-specific durations for withstanding a station blackout. Application of the method resulted in the determination that plants could withstand SBO for 2 to 16 hours, depending on the plants' specific design and site-related characteristics (Ref. 17.7). This method provided plant units with several alternatives for complying with the SBOR (e.g., adding EDGs as alternate AC (AAC)).

Even though all nuclear plants in the United States have committed to the SBOR and many have implemented it, its impact has not been evaluated. Consequently, the SBOR has been evaluated as a part of the Individual Plant Examination (IPE) Program: Perspective On Reactor Safety and Plant Performance. The primary objective of the SBOR evaluation is to use the SBO information in the nuclear industry's IPEs and the NRC SBOR documents referred to above to gauge the stress of the SBOR in reducing CDF. The SBOR evaluation also provides perspectives on SBO at light water reactors (LWRs) by identifying and characterizing the important variables that contribute to the variation in SBO CDF results.

17.2 Technical Approach

This section of the report describes the technical approach used in the SBOR evaluation. The approach consists of four steps described in Sections 17.2.1 through 17.2.4.

17.2.1 IPE Preliminary Review

The first step is to review 75 IPE submittals (108 plant units) and various responses to Requests for Additional Information (RAIs) to determine how the submittals incorporate the SBOR. The IPE submittals examined and the supporting information provided by the NRC on the SBOR coping methods is presented in Table 17.1.

Table 17.1 Station blackout coping methods for the plants reviewed

Type of AAC/coping	Type/no. of units	Plant's name
Excess capacity EDG	PWR-14	Beaver Valley 1&2, Braidwood 1&2, Diablo Canyon 1&2, Milestone 2, Byron 1&2, Turkey Point 3&4, St. Lucie 1&2, Prairie Island
	BWR-5	Brunswick 1&2, Limerick 1&2, and Millstone 1
Excess redundancy EDG	PWR-6	Farley 1&2, Zion 1&2 and South Texas 1&2
	BWR-2	Hatch 1&2
Using HPCS EDG (AAC assist)	BWR-2	Clinton and Perry
Non-class 1E two diesel generators (DGS) or a large DG	PWR-8	ANO 1&2, Calvert Cliffs 1&2, North Anna 1&2, Surry 1&2
	BWR-4	Dresden 2&3, Quad Cities 1&2
Non-class 1E DGS or DGS with limited connection capability	PWR-3	Davis Besse, Milestone 3 and Kewaunee
	BWR-1	Pilgrim
Adjacent unit's non-class 1E DG	PWR-1	Three Mile Island 1
Non-class 1E combustion turbine generator (CTG)	BWR-2	Fermi 2, Oyster Creek
Non-class 1E gas turbine generator	PWR-6	Indian Point 2, Palo Verde 1,2&3, and Point Beach 1&2
Using hydro generator	BWR-3	Peach Bottom 2&3 and Vermont Yankee
Using appendix R DGs	PWR-10	Catawba 1&2, Maine Yankee, McGuire 1&2, and Robinson 2, Oconee 1,2&3
Using class 1E batteries only	**Type/no. of units**	**Plant's name**
Load shedding required	PWR-14	DC Cook 1&2, Palisades, Salem 1&2, San Onofre 2&3, Seabrook 1, Sequoyah 1&2, Summer, Crystal River, Fort Calhoun, Summer, Waterford 3
	BWR-14	Monticello, Duane Arnold, Fitzpatrick, Haddam Neck, Big Rock Point, Grand Gulf, LaSalle 1&2, Susquehanna 1&2, WNP 2, and Nine Mile Point 1&2
No load shedding required	PWR-9	Calloway, Haddam Neck, Harris, Vogtle 1&2, Wolf Creek, Ginna, Comanche Peak 1&2
	BWR-3	Cooper, River Bend, and Hope Creek

The initial analysis examines the IPE submittals from several viewpoints: (1) whether the SBOR is addressed in the submittal; (2) whether the SBOR is credited[17.1] in the submittal; and (3) whether the impact of the SBOR in terms of reduction in total or SBO CDF is known. Results of the total CDFs, SBO CDFs, percent SBO CDF contributions, reduction in total CDFs, reduction in SBO CDFs, and reduction in percent SBO CDF contributions are also gathered (when available) during this step.

17.2.2 Categorization of Plant Submittals

The second step is to categorize and group the IPE submittals. For the 75 IPE submittals examined, plant units can be grouped into the following categories:

Category (a) The SBOR coping is addressed and credited in the IPE submittal, and the impact of the SBOR in terms of reduction in total CDF (i.e., total CDF before the rule minus the total CDF after the rule) is known. The reduction of the SBO CDF may be unknown but the SBO CDF after crediting the SBOR is known. For example, in the Fermi 2 submittal, the coping method (using a non-class 1E gas turbine generator cross-tie) is credited (total CDF = 6E-6/ry). The total CDF before the SBOR was implemented was approximately 8E-6/ry; however, the reduction in the SBO CDF is unknown.

Since one of the primary functions of batteries is to supply direct current (DC) power during a loss of offsite power, it is assumed that the SBOR has no impact for units that met the SBOR using existing battery capacity. Therefore, a separate, credited, subcategory (a1) has been created to incorporate those units that met the SBOR using existing battery capacity.

Category (b) The SBOR coping method is addressed in the submittal and credited, but the impact (reduction in CDF) of the SBOR is unknown. (Note: SBO CDF after crediting SBOR is known but not before.) For example, in the Clinton submittal, the coping method (high pressure core spray (HPCS) EDG) is credited, resulting in a final SBO CDF of 1E-5/ry. However, the submittals do not list what the total CDF would be if the SBOR coping method had not been credited.

Category (c) The SBOR coping method is addressed in the submittal but not credited; however the impact of the SBOR was evaluated separately after the results were finalized. For example, in the Arkansas Nuclear One (ANO 1) submittal, the coping method was not credited before the CDF results were finalized. However, a separate sensitivity study in the submittal states that "ANO 1 has committed to install an AAC. The benefit of this AAC is a approximately 40% reduction in total CDF."

Category (d) The SBOR coping method is addressed in the IPE submittal but not credited, and the impact of the SBOR is unknown. For example, in the Beaver Valley IPE submittal, the SBOR coping method (cross-tie using excess capacity EDG) is stated clearly as not being credited.

Category (e) The SBOR coping method may or may not be addressed in the submittal (and may or may not be credited). For example, from a review of the Haddam Neck IPE, it is unclear if the SBOR coping method (load shedding batteries) is addressed or credited.

[17.1]When the SBOR coping method, as defined in the information provided by the NRC, is clearly used in the IPE model to evaluate the plant unit's CDF, it is considered "credited".

Category (f) The SBOR coping method is directly or indirectly referred to in the IPE submittal, but the information needed to characterize its effect is insufficient, inconsistent, or incomplete. For example, the SBOR coping method for Indian Point 2 is only partially credited in the IPE. Therefore, it was difficult to determine the full impact of the SBOR. For the 75 submittals, the plant units in each category is presented in Table 17.2.

Table 17.2 Plant Units in Each Category

Category	Plant Units
a	Davis Besse*, Fermi 2, and Kewaunee
a1	Cooper, Hope Creek, Callaway, Comanche Peak 1&2, River Bend, Shearon Harris 1, Vogtle 1 & 2, and Wolf Creek
b	Big Rock Point, Braidwood 1&2, Byron 1&2, Catawba 1&2, Clinton, Crystal River 3, Duane Arnold, Farley 1&2, Fitzpatrick, Hatch 1&2, H.B. Robinson 2, Milestone 2, Nine Mile Point 1&2, Oconee 1,2,&3, Peach Bottom 2&3, Perry 1, Pilgrim 1, Prairie Island 1&2, San Onofre 2&3, Sequoyah 1&2, St. Lucie 1&2, TMI 1, Turkey Point 3&4, Vermont Yankee, WNP 2, and Zion 1&2
c	Arkansas 1&2, Calvert Cliffs 1&2*, Diablo Canyon 1&2, Monticello, Palo Verde 1,2,&3, and Point Beach 1&2
d	Beaver Valley 1&2, Brunswick 1&2, D.C. Cook 1&2, Indian Point 3, Limerick 1&2, Milestone 1&3, South Texas 1&2, Salem 1&2, and Surry 1&2
e	Dresden 2&3, Fort Calhoun 1*, Ginna, Grand Gulf, Haddam Neck, Lasalle 1, North Anna 1&2, Quad Cities 1&2, Seabrook 1, Summer, Susquehanna 1&2, Waterford 3, and Watts Bar 1&2
f	Browns Ferry 2, Indian Point 2, McGuire 1&2, Oyster Creek, and Palisades

* Indicates that SBO CDF was not available for these four plant units. Therefore, the results in Chapter 17 do not include these four units.

17.2.3 Evaluations of the Impact of the Station Blackout Rule

The third step is to use information provided in categories (a) through (d) to obtain insights on the impact of the SBOR. The information in categories (e) through (f) was not further analyzed.

First the information in categories (a) and (c) is analyzed. Note that the plant units in these two categories provide estimates of reduction in total CDF that result or will result from implementing the SBOR. The analysis begins with comparisons of the total CDF with and without the rule. Estimates of the mean reduction are then used to provide perspectives on the potential impact of the rule, assuming that a significant portion of the reduction in total CDF gained from the rule is due to a reduction in the SBO CDF. Finally, the total CDFs obtained with the different SBOR coping methods are compared.

Next the information in groups (a), (a1), and (b) is analyzed. Note that the plant units in these three categories credit the SBOR and provide the SBO CDF after the rule is implemented. The average SBO CDF for the plant units in these three categories is compared to the rule's goal of 1E-5/ry to see whether plants implementing the rule have attained the goal. Next, the reasons for significant SBO CDF variations from the goal are explored.

Finally, the information in groups (c) and (d) is analyzed. Note that plant units in these two categories do not credit the SBOR. The average SBO CDF for the plant units in these two categories is compared to the rule's goal. This comparison is made to provide the NRC with insights for evaluating how far, without implementing the rule, these plant units are from approaching the SBOR's goal of 1E-5/ry. The reasons for significant SBO CDF variations from the goal are also explored.

17.2.4 Identification and Investigation of the Factors Affecting Station Blackout

In the fourth and final step of the approach, the factors that contribute to the variability in the SBO CDF results are identified when sufficient details are provided in the submittals. A limited regression analysis is then performed to investigate the relative contribution of each variable to the overall SBO CDF. While information cannot be obtained to model all of the variables, several key variables are examined. These variables affect all units and are directly relevant to the SBOR. The regression technique provides two statistics that indicate the influence of the variables in a regression. The first is the coefficient of determination, R^2. The R^2 value provides an indication of the portion of the variation in the output value (e.g., the SBO CDF) that is accounted for by regressing on a set of input variables (e.g., frequency of a loss of offsite power or battery lifetime). For example, an R^2 of 0.99 indicates that the variables in a model account for 99% of the output variation. The second is called the normalized T statistic. The normalized T statistic for each variable measures how much that variable, relative to the other variables, contributes to the variation in the output. Those factors that are not modeled in the regression are discussed qualitatively in Section 17.3.2. Some of the factors identified are not discussed because there is insufficient detailed information.

17.3 Results of the Evaluation of the Impact of the Station Blackout Rule

This section contains the results of the evaluation of the SBOR for the plant units in categories (a) through (d). The results of the impact of the SBOR are addressed in Section 17.3.1, while the results of the investigation of the variables that contribute to variations in SBO CDFs are discussed in Section 17.3.2.

17.3.1 Results of the Impact of the Station Blackout Rule

17.3.1.1 CDF Reduction Results Due to the Station Blackout Rule

The 10 submittals (15 plant units) in categories (a) and (c) for which total CDF reductions[17.2] are available are summarized in Table 17.3. The table shows that the plants include both BWRs and PWRs, use various SBOR coping methods, and cover a spectrum of SBOR characteristics. For this group of plants, when the rule is implemented the average value of total CDF is reduced by 2E-5/ry.

[17.2] Note: not reduction in SBO CDF; only one plant, Calvert Cliffs 1&2, reported this value.

Table 17.3 Plants that report reductions in total CDF due to the SBOR

Plant name	Plant type	Category	SBOR coping method	SBOR characteristics[a]
Arkansas 1	PWR	c	Adding DG and new cross-tie	4, 0.95, P1, C, I2, ESW1, SW2
Arkansas 2	PWR	c	Adding DG and new cross-tie	4, 0.95, P1, C, I2, ESW1, SW2
Calvert Cliffs 1&2[b]	PWR	c	Adding an EDG and a non-Class 1E DG	4, 0.975, P2, I2, C, ESW4, SW2
Davis-Besse	PWR	a	Using non-Class 1E DG	4, 0.95, C, P1, I2, ESW2, SW2
Diablo Canyon 1&2	PWR	c	Adding an EDG	4, 0.95, D, P1, I2, ESW1, SW1
Kewaunee	PWR	a	Using Technical Support Center (TSC) EDG and installing connection between TSC EDG and 480 V safety bus	4, 0.95, C, P1, I2, ESW2, SW2
Palo Verde 1,2&3	PWR	c	Adding 2 non-Class 1E Gas Turbine Generators (GTGs)	4, 0.95, C, P1, I2, ESW2, SW1
Point Beach 1&2	PWR	c	Modifying non-Class 1E GTG	8, 0.975, D, P2, I2, ESW4, SW2
Fermi 2	BWR	a	Using non-Class 1E GTG	4, 0.95, B, P2, I2, ESW1, SW3
Monticello	BWR	c	Load shedding battery	4, 0.95, C, P1, I2, ESW1, SW2

[a] The SBOR characteristics are defined in (Ref. 17.5, Ref. 17.6).
[b] Reported the reduction in both the SBO and total CDF. The total CDF is based on pre-closeout CDF of 3.2E-4/ry

The potential reduction in total CDF from the SBOR (i.e., categories (a) and (c)) is shown in Figure 17.1 for the LWRs, the BWRs, and the PWRs. The results show average reductions of 2E-5/ry, 3E-6/ry,and 3E-5/ry for the LWRs, BWRs, and the PWRs, respectively. The percent reduction in total CDF that results from implementing the rule is depicted in Figure 17.2 for the LWRs, the BWRs, and the PWRs. The results show average percent reductions of approximately 25%, approximately 20%, and approximately 25% for the LWRs, the BWRs, and the PWRs, respectively. For these plant units, both the average reduction and the average percent reduction are greater for PWRs than for BWRs. However, there are not enough BWRs (only 2 plant units, Fermi 2 and Monticello) to provide meaningful statistics for a comparison of the averages for BWRs and PWRs.

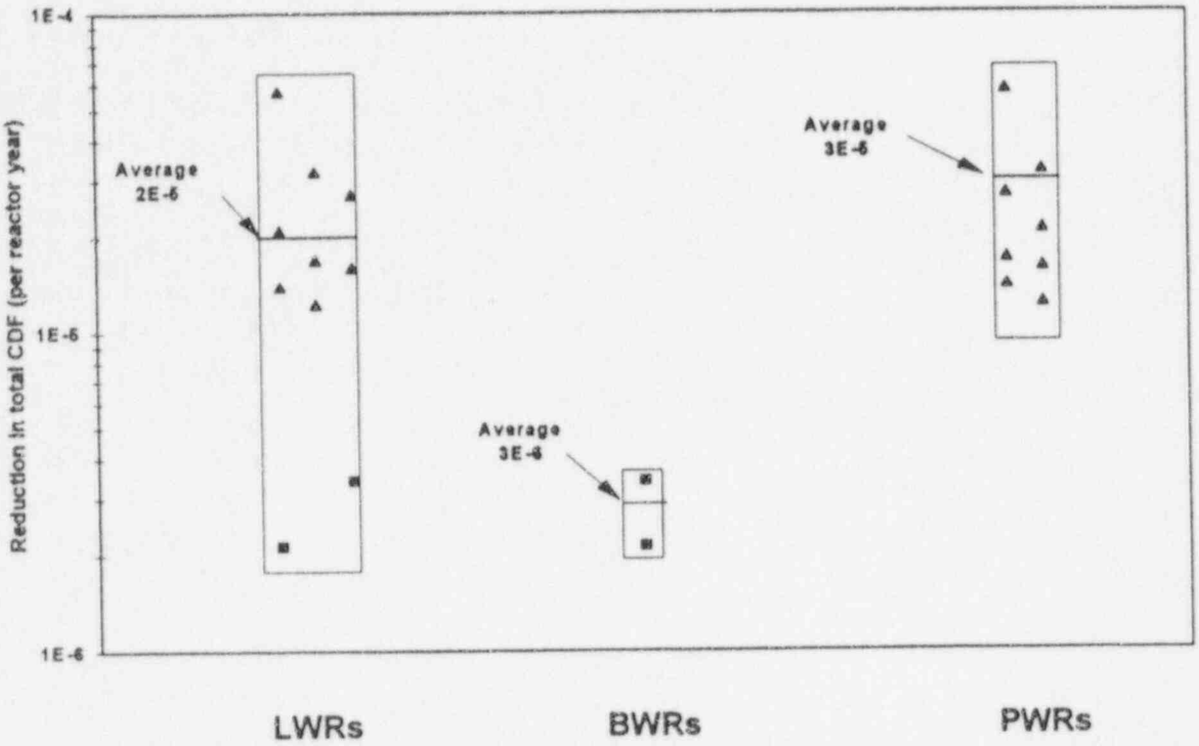

Figure 17.1 Potential reduction in total CDF from implementing the Station Blackout Rule.

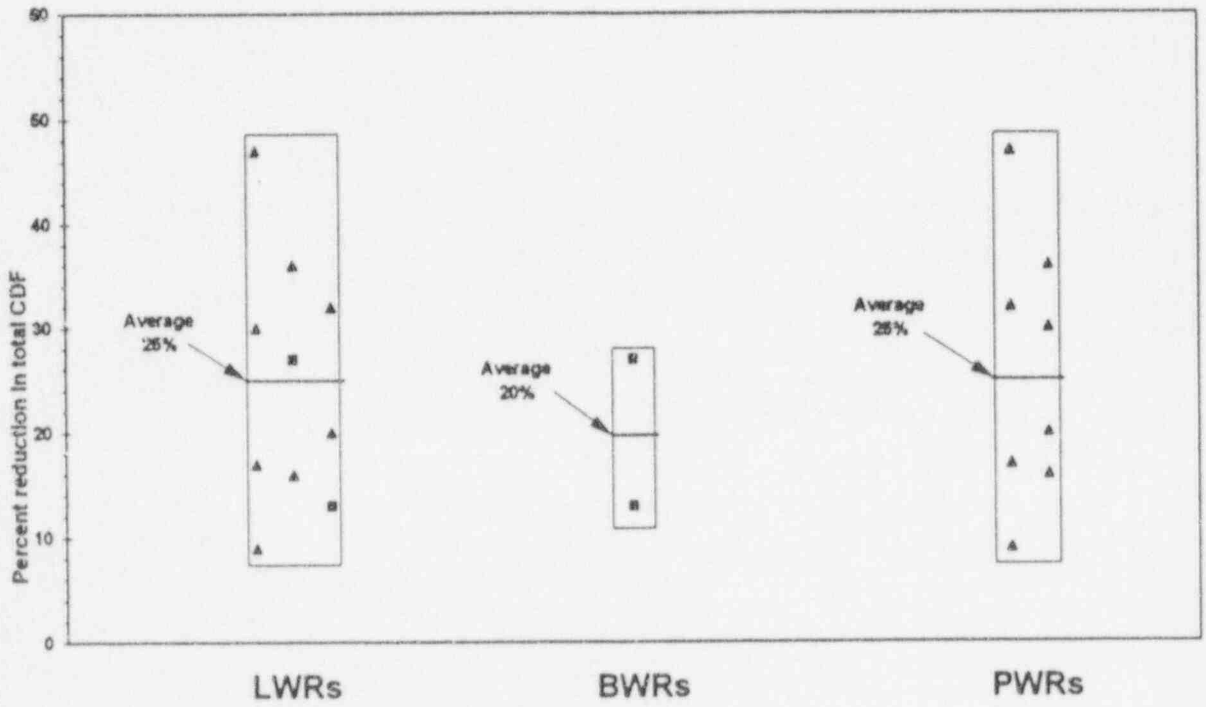

Figure 17.2 Potential percent reduction in total CDF from implementing the Station Blackout Rule.

Although the reduction in SBO CDF is expected to vary depending on the specific unit and SBOR coping method, a significant portion of the reported reduction in total CDF is also expected to be from a reduction in SBO CDF. This is because implementation of the SBOR usually only affects sequences initiated by a loss of offsite power, and most of these sequences go to SBO (average about 70% for the units in this evaluation). Therefore, even though how much of the reduction in total CDF comes from blackout rather than from non-blackout sequences (i.e., sequences initiated by a LOSP but that do not go to blackout) cannot be derived from information available, it is expected that the reduction of SBO CDF from implementing the rule is significant. For the 10 plant units that meet the SBOR using existing battery capacity (i.e., plants in category (a1)), the SBOR has no impact. Further comparisons of the reduction and percent reduction in total CDF by SBO coping methods are depicted in Figures 17.3 and 17.4, respectively. The results show some variations among the different coping methods. However, there are not enough units to provide meaningful statistics for any one method.

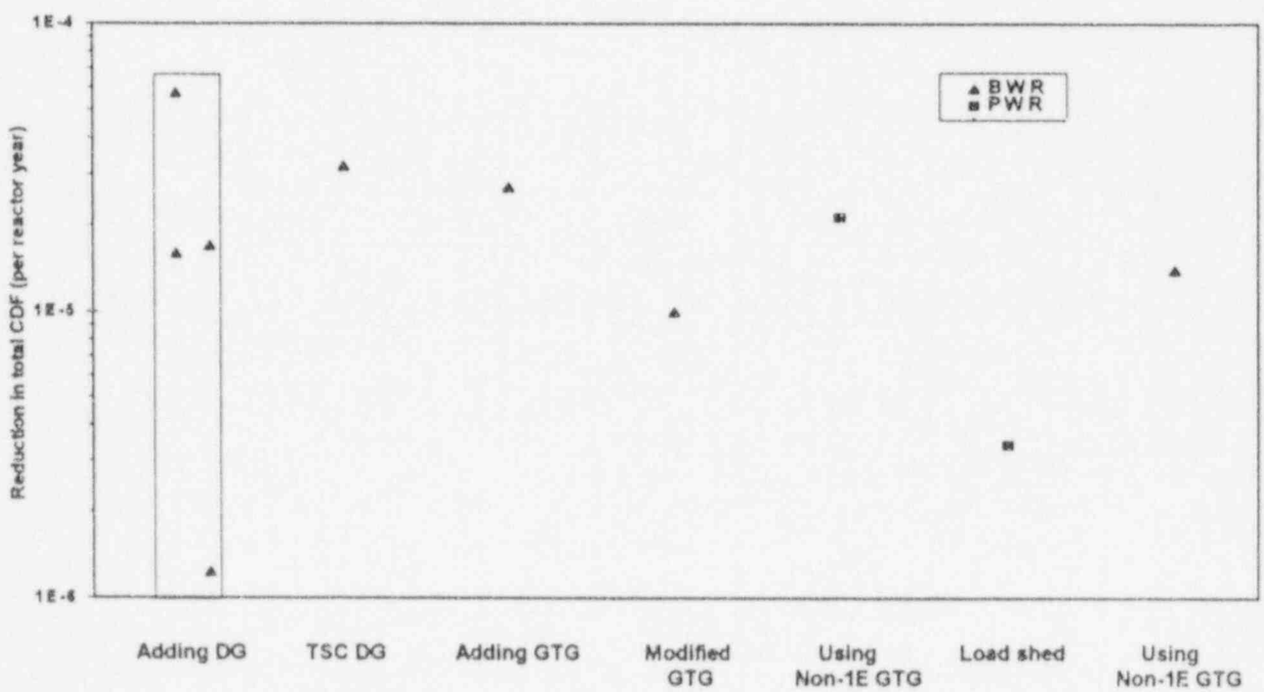

Figure 17.3 Potential reduction in total CDF by coping method.

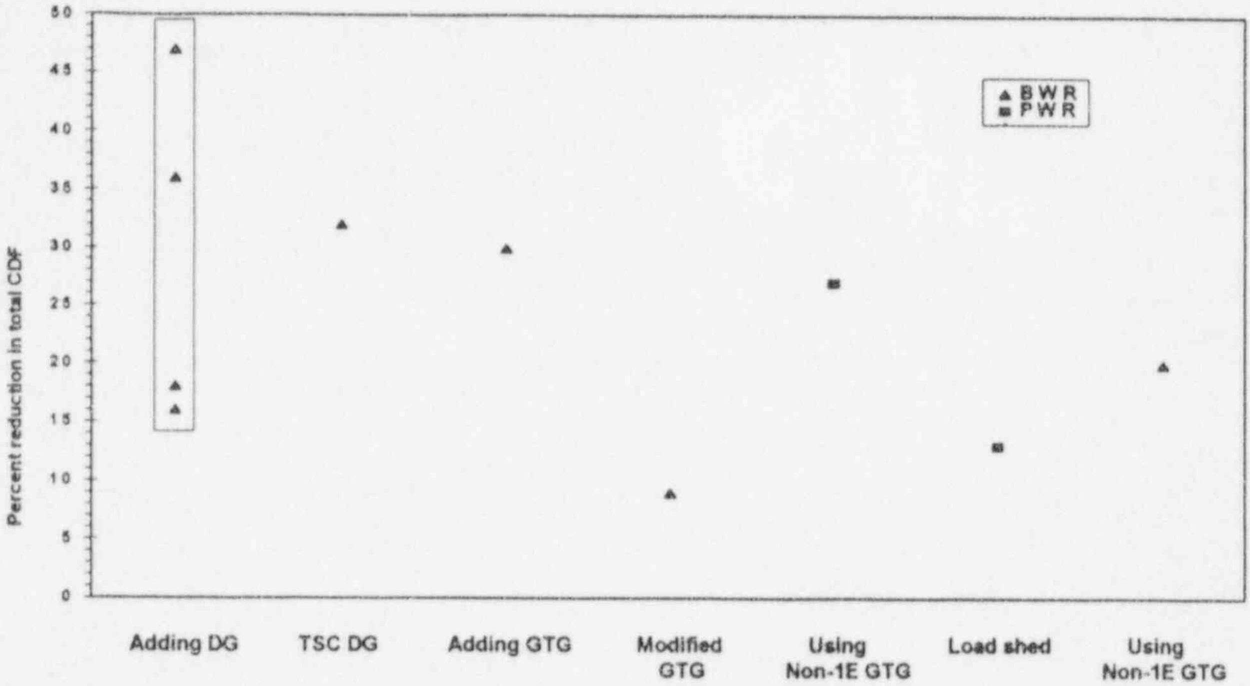

Figure 17.4 Potential percent reduction in total CDF by coping method.

17.3.1.2 Results for Plant Units That Credit the Station Blackout Rule

The SBO CDF results for the 53 plant units that credit the SBOR (i.e., units in categories (a), (a1), and (b)) are presented in Figure 17.5 for light water reactors (LWRs), pressurized water reactors (PWRs), and boiling water reactors (BWRs). The SBO CDFs for this group of units vary from negligible to approximately 5E-5/ry with means of 9E-6/ry, 7E-6/ry, and 1E-5/ry for the LWRs, the BWRs, and the PWRs, respectively.

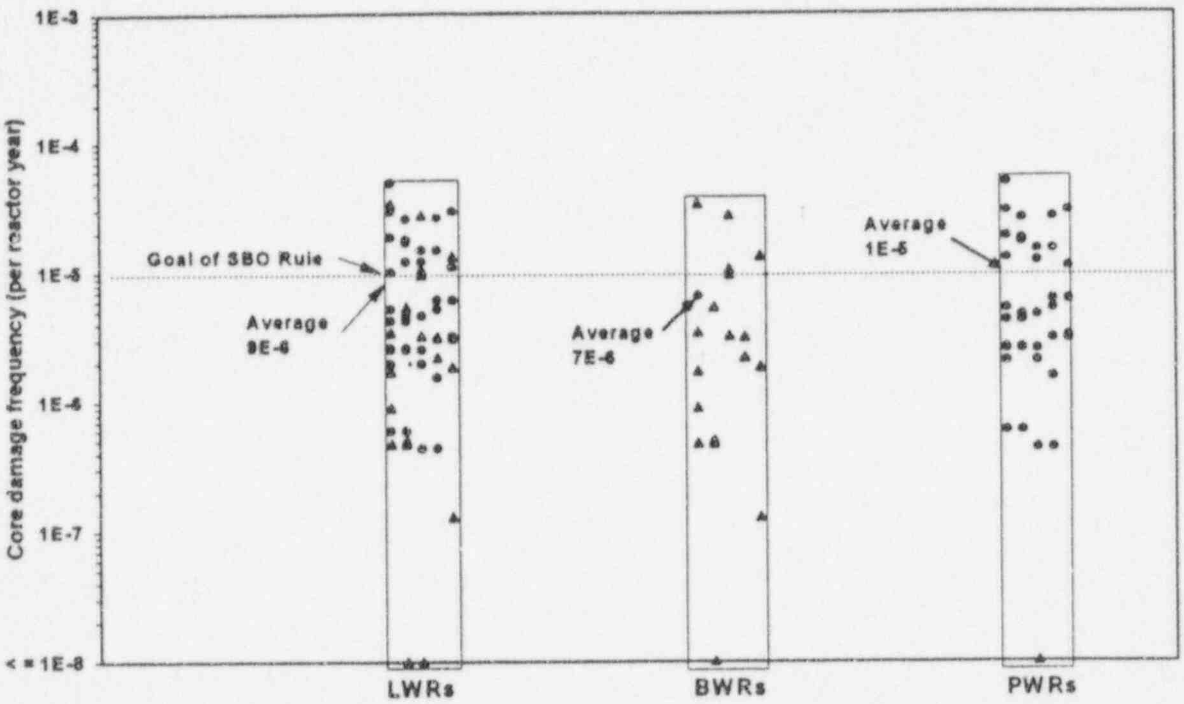

Figure 17.5 SBO CDF results for plants that credit the Station Blackout Rule.

A comparison of the average SBO CDF for the 53 plant units (9E-6/ry) to the SBOR goal of 1E-5/ry shows that the values are in close agreement. The average SBO CDF is lower than the SBOR goal for the BWRs and comparable to the SBOR goal for the PWRs. The variability in the SBO CDFs is large. Some plant units that credit the rule have SBO CDFs close to two orders of magnitude (100 times) lower than the goal while others report SBO CDFs close to three times that goal.

Table 17.4 summarizes some of the key variables for the three submittals (four plant units) with the lowest SBO CDFs. In this group of units, Pilgrim has the lowest SBO CDF (negligible) and attributes the low value to the following facts: the plant uses an additional non-Class 1E EDG as an AAC; the EDGs do not depend on room cooling in a loss of power; and the EDGs have a very high reliability. In addition the unit's batteries have a long lifetime (12 hours). Fermi 2 has the second lowest SBO CDF (1E-7/ry). This plant unit has four redundant and independent EDGs, with intradivisional cross-tie capability, and only two of the EDGs are required for a safe reactor shutdown (i.e., the unit is in emergency AC (EAC) group B, described as having better than typical redundant and independent EAC sources for safe shutdown equipment (Refs. 17.5 and 17.6)). This configuration is backed up by a gas turbine generator (GTG) which is used as an alternate AC (AAC) power source to satisfy the SBOR. For

Fermi 2, the reported reduction in total CDF from using the GTG is approximately 25%. The unit also has a low loss of offsite power (LOSP) initiating event frequency (0.012/ry). Catawba units 1&2 have the third lowest SBO CDF (6E-7/ry) and attribute the low SBO CDF to an additional Appendix R EDG which is used as an AAC.

Table 17.4 Variables contributing to low SBO CDFs for plants that credited the SBOR

Plant name	Plant type	SBO CDF (/ry)	LOSP frequency (/ry)	No. of EDGs/no. required for safe shutdown	AAC	Battery lifetime (hours)	RCP seal LOCA	Other considerations
Pilgrim	BWR	Negligible	0.142	2/1	Non-Class 1E EDG	12 hours	NA	EAC group C Lack of EDG dependency on room cooling High EDG reliability
Fermi 2	PWR	1E-7	.012	4/2	GTG	4 hours	NA	Intradivisional cross-tie capability EAC group B
Catawba 1&2 (each unit)	PWR	6E-7	0.035	2/1	Appendix R EDG	NA	NA	EAC group C

The common denominator that drives SBO CDFs low for all four plant units is the availability of an extra power generator (GTG or EDG). The plant units with the two lowest SBO CDFs also have additional EDG-related capabilities such as the ability to cross-tie between divisions and the lack of EDG dependence on room cooling in a loss of offsite power.

Table 17.5 summarizes some of the key variables for the three submittals (four plant units) with the highest SBO. In this group of units, Summer has the highest SBO CDF (approximately 5E-5/ry). Summer has a high LOSP initiating event frequency (0.073) and low EDG redundancy (i.e., two of three EDGs for safe plant shutdown following a LOSP). Vogtle units 1&2 have the second highest SBO CDF (approximately 3E-5/ry). Do the units conservatively assume that the condensate storage tank (CST) will provide auxiliary feedwater (AFW) suction supply for only 8 hours, instead of 24 hours, which is its capacity limit. The plant units also have a short battery lifetime and low EDG redundancy (i.e., one of two EDGs for safe plant shutdown following a LOSP). Cooper has the third highest SBO CDF (approximately 3E-5/ry). Cooper has a short battery depletion time (4 hours) and thus short time in which to recover offsite power (increasing the probability of not recovering power). This unit also has a low EDG redundancy.

Table 17.5 Variables contributing to high SBO CDFs for plants that credit the SBOR

Plant name	Plant type	SBO CDF (/ry)	LOSP frequency (/ry)	No. of EDGs/no. for safe shutdown	AAC	Battery lifetime (hours)	RCP seal LOCA	Other considerations
Cooper	BWR	3E-5	0.035	2/1	NA	4 hours	NA	Short time in which to recover offsite powers EAC group C
Vogtle 1&2 (each unit)	PWR	3E-5	0.04 (single unit)	4/1 of 2 for safe shutdown of each unit	NA	4 hours	Contributes about 10% of the SBO CDF	Conservatively assumed a limited CST inventory EAC group C
Summer	PWR	5E-5	0.073)	2/1	NA	4 hours	NA	EAC group C

One common denominator that accounts for the high SBO CDFs for all three IPE submittals (i.e., Cooper, Summer, and Vogtle 1&2) is the low EDG redundancy for achieving and maintaining a safe plant shutdown following a LOSP. Also, the plant units with the two highest SBO CDFs both have a short battery life. However, other factors (e.g., a short time in which to recover offsite power) unique to each plant unit also contribute to the high value.

The results of the percent SBO contribution to the total CDF for the 53 plant units that credit the SBOR are shown in Figure 17.6 for LWRs, BWRS, and PWRs. The percent SBO contribution for this group of plant units varies from a low of 0% to a high of 90%, with means of approximately 20%, 30%, and 15% for the LWRs, the BWRs, and the PWRs, respectively. These results show that the percent SBO contribution can be significant even after crediting the rule. These results also show that some plant units that credit the SBOR have negligible SBO contributions while others have very high percent SBO contributions (90%).

Table 17.6 summarizes some of the key variables for the three submittals (four plant units) with the lowest SBO CDFs. In this group of units, Pilgrim has the lowest percent SBO contribution (0%) and attributes the low value to the plant's use of an additional non-Class 1E EDG as an AAC, the EDGs' independence of room cooling in a loss of power, and the very high reliability of the EDGs. In addition the unit's batteries have a long lifetime (12 hours). Catawba units 1&2 have the second lowest percent SBO contribution (approximately 1%) and attribute the low SBO to an additional Appendix R EDG which is used as an AAC. Fermi 2 has the third lowest percent SBO contribution (approximately 2.0%). This plant has four redundant and independent EDGs, with intradivisional cross-tie capability, of which only two of the EDGs are required for a safe reactor shutdown. This configuration is backed up by a GTG which is used as an alternate AC (AAC) to satisfy the SBOR.

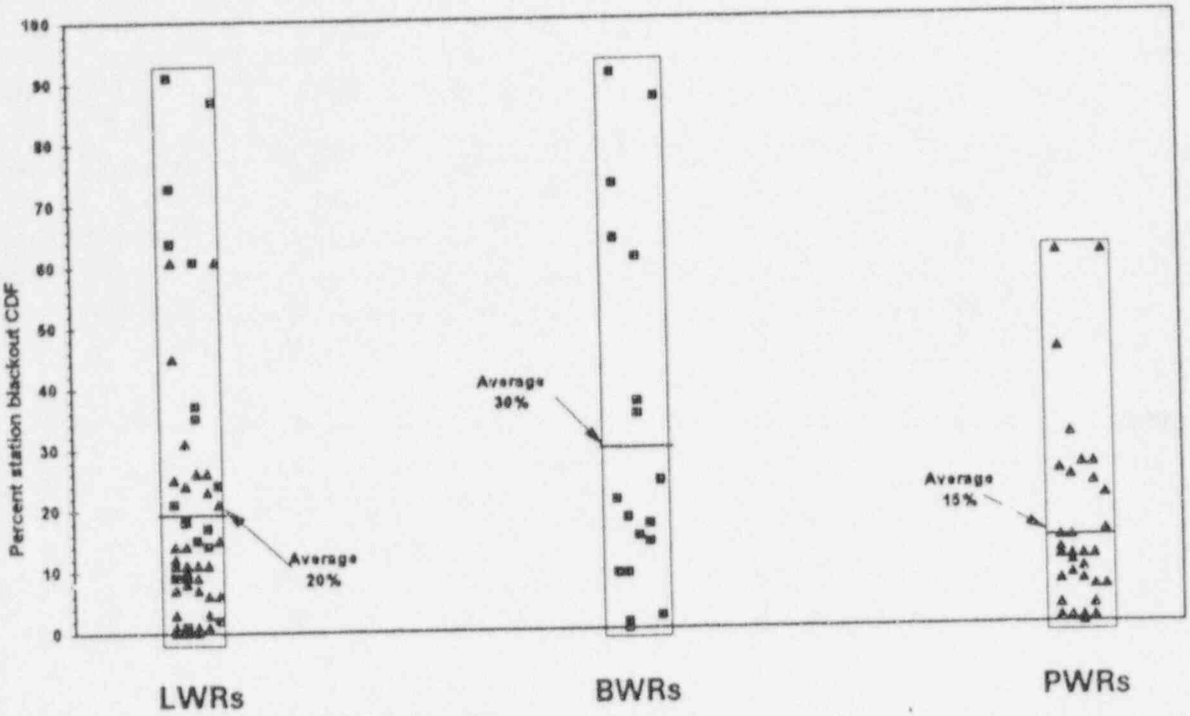

Figure 17.6 Percent SBO CDF results for plants that credit the Station Blackout Rule.

Table 17.6 Variables contributing to low percent contributions for plants that credit the SBOR.

Plant name	Plant type	% SBO contribution	LOSP frequency (/ry)	No. of EDGs/no. required for safe shutdown	AAC	Battery lifetime (hours)	RCP seal LOCA	Other considerations
Pilgrim	BWR	0%	0.142	2/1	Non-Class 1E EDG	12 hours	NA	EAC group C No EDG dependency on room cooling High EDG reliability
Catawba 1&2 (each unit)	PWR	1%	0.035	2/1	Appendix R EDG	NA	NA	EAC group C
Fermi 2	PWR	2%	0.012	4/2	GTG	4 hours	NA	Intradivisional cross-tie capability EAC group B

The common denominator that drives percent SBO contribution low for all three submittals (i.e., Pilgrim, Fermi 2, and Catawba 1&2) is the availability of an extra power generator (GTG or EDG).

Table 17.7 summarizes some of the key variables affecting the three submittals (three plant units) with the highest percent SBO contributions (approximately 90%, 90%, and 65%). A close examination of these three units (Fitzpatrick, River Bend, and Nine Mile Point 1) reveals that their SBO CDFs are lower than or comparable to the SBOR goal of 1E-5/ry. Therefore, the high percent SBO contributions are due to low values for the total CDFs.

Table 17.7 Variables contributing to high percent contributions for plants that credit the SBOR

Plant name	Plant type	% SBO contribution	LOSP frequency (/ry)	No. of EDGs/no. for safe shutdown	AAC	Battery lifetime (hours)	RCP seal LOCA	Other considerations
Fitzpatrick	BWR	90%	0.057	4/1	NA	8 hours with load shedding	NA	EAC group B
River Bend	BWR	90%	0.035	2/1	NA	4 hours	NA	EAC group C
Nine Mile Point 1	BWR	65%	0.05	2/1	NA	8 hours with load shedding	NA	EAC group C

17.3.1.3 Results for Plants That Do Not Credit the Station Blackout Rule

The SBO CDF results for the 27 plant units that do not credit the rule (i.e., plant units in categories (c) and (d)) are depicted in Figure 17.7 for LWRs, BWRS, and PWRs, respectively. The SBO CDFs for this group of plant units vary from a low of 1E-7/ry to a high of approximately 7E-5/ry, with means of 1E-5/ry, 9E-6/ry, and 2E-5/ry for the LWRs, the BWRs, and the PWRs, respectively.

A comparison of the average SBO CDF for 27 plant units (1E-5/ry) to the SBOR goal shows that the average SBO CDF is comparable to the SBOR goal. The average SBO CDF is slightly lower than the goal for BWRs and higher than the goal for PWRs. There is a large variability in the SBO CDFs, with some units having SBO CDFs two orders of magnitude lower than the SBOR goal and other having SBO CDFs close to an order of magnitude lower than the SBOR goal and others having SBO CDFs close to an order of magnitude higher than the SBOR goal.

Table 17.8 summarizes some of the key variables affecting the three submittals (five plant units) with the lowest SBO CDFs. Limerick units 1&2 have the lowest reported SBO CDF (approximately 1E-7/ry) and attribute the low value to the fact that the units have four "better than typical" redundant and independent EDGs per unit (i.e., EAC group B). In addition, the plant units have a long battery lifetime with load shedding (8 hours).

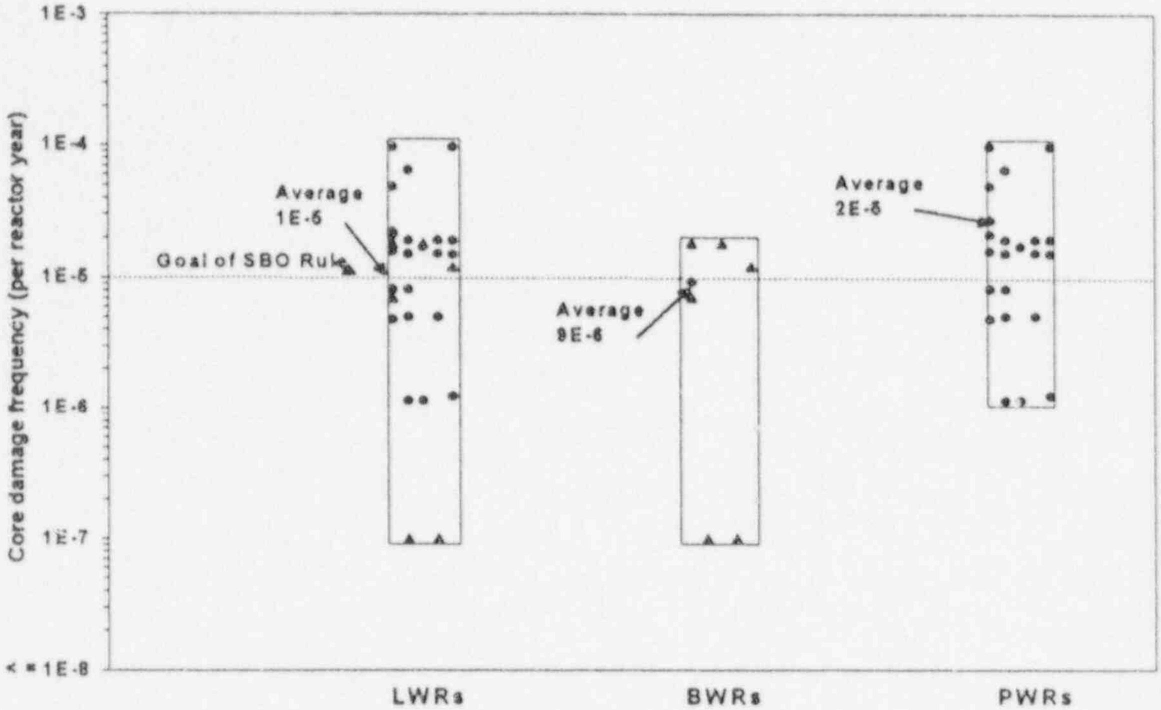

Figure 17.7 SBO CDF results for plants that do not credit the Station Blackout Rule.

Table 17.8 Variables contributing to low SBO CDFs for plants that do not credit the SBOR

Plant name	Plant type	SBO CDF (/ry)	LOSP frequency (/ry)	No. of EDGs/no. required for safe shutdown	AAC	Battery lifetime (hours)	RCP seal LOCA	Other considerations
Limerick 1&2	BWR	1E-7	0.059	4 per unit/ 2 of 4 for shutdown	Excess capacity EDG	8 hours	NA	EAC group B
D.C. Cook 1&2	PWR	1E-6	0.040	2/1	NA	Not available	NA	EAC group C
ANO 2	PWR	1E-6	0.058	2/1 of 2	Adding EDG and new cross-tie	8 hours	Not a significant contributor to SBO sequences	EAC group C Low failure probability for restoring offsite power Manual feedwater control following battery depletion

D.C. Cook units 1&2 have the next lowest SBO CDF and attribute the low value (approximately 1E-6/ry) to an extremely reliable grid which greatly influences the event frequencies for a LOSP (0.04/ry) and SBO (4E-5/ry). Arkansas Nuclear 2 (ANO 2) has the third lowest SBO CDF (approximately 1E-6/ry). For this plant unit reactor coolant pump (RCP) seal loss of coolant accidents (LOCAs) do not contribute significantly to the SBO sequences. This is because this Combustion Engineering (CE) plant unit uses Byron Jackson pumps and does not consider the pumps susceptible to significant leakage during SBO. Also, the plant's submittal uses lower values for failure to restore offsite power (for a specific elapsed time) than other CE plants, credits a long battery lifetime (8 hours), and credits operator action to manually control AFW following battery depletion. For this plant unit, the SBO sequences are about 50% short-term blackout sequences (LOSP with early failure of all feedwater) and 50% long-term blackout sequences (LOSP with initial operation of turbine-driven feedwater but subsequent failure due to battery depletion).

While no common denominator drives the SBO CDFs low for all three submittals, three units have a long battery life. However, other factors (e.g., manual feedwater control following battery depletion) unique to each unit also contribute to the high value.

Table 17.9 summarizes some of the key variables for the two submittals (two plant units) with the highest SBO CDFs. Beaver Valley units 1&2 have the highest and the next highest SBO CDFs (approximately 7E-5/ry and 5E-5/ry) and attribute the high values to the high likelihood of early seal LOCA following SBO. All units also have a very short battery life (2 hours).

Table 17.9 Variables contributing to high SBO CDF for plants that do not credit the SBOR

Plant name	Plant type	SBO CDF (/ry)	LOSP frequency (/ry)	No. of EDGs/no. required for safe shutdown	AAC	Battery lifetime (hours)	RCP seal LOCA	Other considerations
Beaver Valley 1	PWR	7E-5	0.066	4/1 of 2 for shutting down each unit	Excess capacity EDG	[illegible] hours	High probability of a reactor coolant pump seal LOCA	EAC group C
Beaver Valley 2	PWR	5E-5	0.074	4/1 of 2 for shutting down each unit	Excess capacity EDG	2 hours	High probability of a reactor coolant pump seal LOCA	EAC group C

The results of the percent SBO contribution to the total CDF for the 27 plant units that do not credit the rule are shown in Figure 17.8 for LWRs, PWRS, and BWRs, respectively. The percent SBO contribution for this group of plant units varies from a low of 2% to a high of 65%, with means of approximately 25%, 40%, and 20% for the LWRs, the BWRs, and the PWRs, respectively. These results indicate that the percent SBO contribution for LWRs can be significant. The results also show that some plant units like D.C. Cook units 1&2 have very low percent SBO contributions (2%) even before crediting the SBOR. Others like Brunswick units 1&2 have high percent SBO contributions (65%).

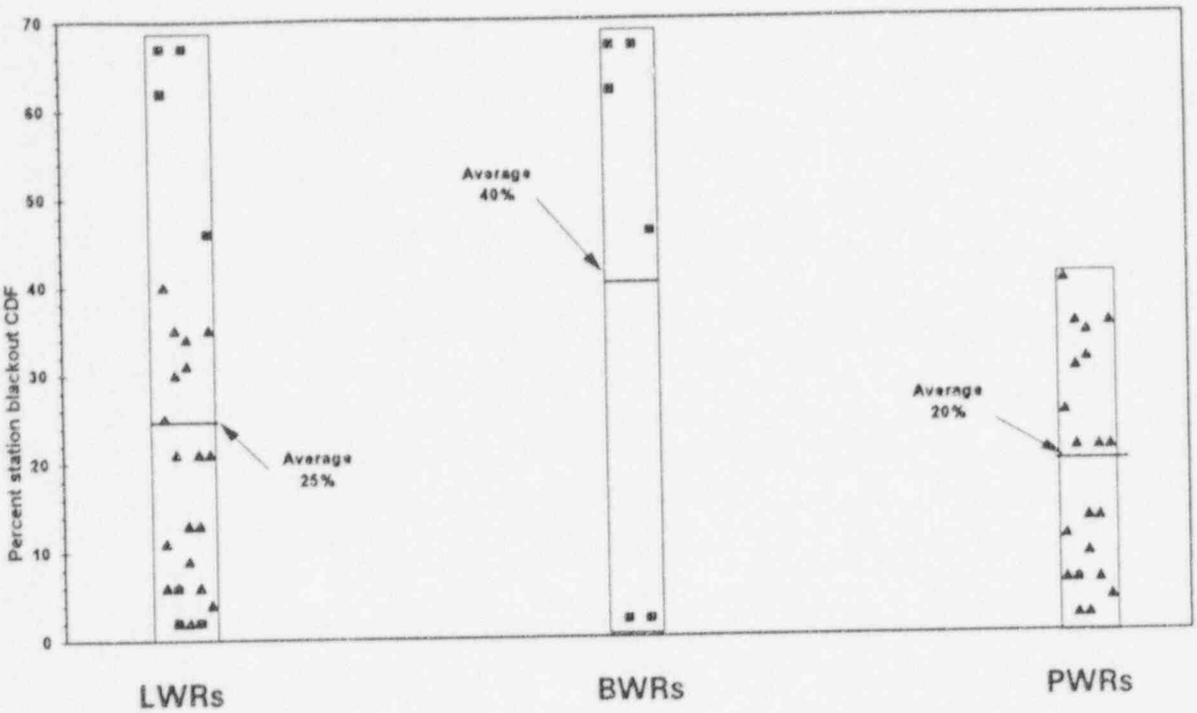

Figure 17.8 Percent SBO CDF results for plants that do not credit the Station Blackout Rule.

Table 17.10 summarizes some of the key variables affecting the three submittals (five plant units) with the lowest percent SBO contributions. (Note that these are the same units with the lowest SBO CDFs.) D.C. Cook units 1&2 have the lowest percent SBO contribution and attribute the low value (approximately 1%) to an extremely reliable grid which greatly influences the event frequencies for a LOSP and SBO (1.4E-5/ry). Limerick units 1&2 have the next lowest reported percent SBO contribution (approximately 2%) and attribute the low value to the fact that the units have four "better than typical" redundant and independent EDGs per unit (i.e., EAC group B). In addition, the units have a long battery lifetime with load shedding (8 hours). Arkansas Nuclear 2 has the third lowest percent SBO contribution (approximately 4%). For this plant unit, RCP seal LOCAs do not contribute significantly to the SBO sequences. This is because this CE plant unit uses Byron Jackson pumps and does not consider the pumps susceptible to significant leakage during SBO. Also, the unit uses lower values for failure to restore offsite power (for a specific elapsed time) than other CE units, credits a long battery lifetime (8 hours), and credits operator action to manually control of feedwater following battery depletion. For this unit, the SBO sequences are about 50% short-term blackout sequences (LOSP with early failure of all feedwater) and 50% long-term blackout sequences (LOSP with initial operation of turbine-driven feedwater but subsequent failure due to battery depletion).

Table 17.10 Variables contributing to low percent SBO contribution for plants that do not credit the SBOR

Plant name	Plant type	% SBO contribution	LOSP frequency (/ry)	No. of EDGs/no. required for safe shutdown	AAC	Battery lifetime (hours)	RCP seal LOCA	Other considerations
D.C. Cook 1&2	PWR	2%	0.040	2/1	NA	Not available	NA	EAC group C
Limerick 1&2	BWR	2%	0.059	4 per unit/ 2 of 4 for shutdown	Excess capacity EDG	8 hours	NA	EAC group B
ANO 2	PWR	4%	0.058	2/1 of 2	Adding EDG and new cross-tie	8 hours	Not a significant contributor to SBO sequences	EAC group C Low probability for restoring offsite power Manual feedwater control following battery depletion

While no common denominator drives the SBO CDFs low for all three submittals (i.e., D.C. Cook units 1&2, Limerick units 1&2, and ANO 2), three of the units (Limerick units 1&2 and ANO 2) have a long battery life. However, other factors (e.g., manual feedwater control following battery depletion) also contribute.

Table 17.11 summarizes some of the key variables for the two submittals (three plant units) with the highest percent SBO contributions (approximately 65% and 45%). An examination of these three plant units (i.e., Brunswick 1&2 and Monticello) reveals that their SBO CDFs (approximately 2E-5/ry and 1E-6/ry, respectively) are comparable to the SBOR goal. Therefore, the very high percent SBO CDF contributions are due to low total CDFs.

Table 17.11 Variables contributing to high percent contributions for plants that do not credit the SBOR

Plant name	Plant type	% SBO contribution	LOSP frequency (/ry)	No. of EDGs/no. required for safe shutdown	AAC	Battery lifetime (hours)	RCP seal LOCA	Other considerations
Brunswick 1&2 (Each unit)	BWR	65%	0.074	4/1 of 2 for safe shutdown of each unit	Excess capacity EDG	2 hours	NA	EAC group C
Monticello	BWR	44%	0.079	2 / 1	NA	4 hours	NA	EAC group C

17.3.1.4 Result of the Comparisons of Plants That Credit SBOR Versus Plants That Do Not Credit the SBOR

A final comparison of the SBO CDF results for the 53 units that credit the SBOR versus the 27 units that do not credit the rule is shown in Figure 17.9 for LWRs, BWRS, and PWRs. A similar comparison for the percent SBO contribution to the total CDF is shown in Figure 17.10 for LWRs, BWRs, and PWRs. The results indicate that on average the SBO CDF for the plant units crediting the SBOR is lower than that for the plant units not crediting the rule. The results are reversed for percent SBO contribution. This is partly because a large percent of SBO contribution does not necessarily correspond to a large SBO CDF.

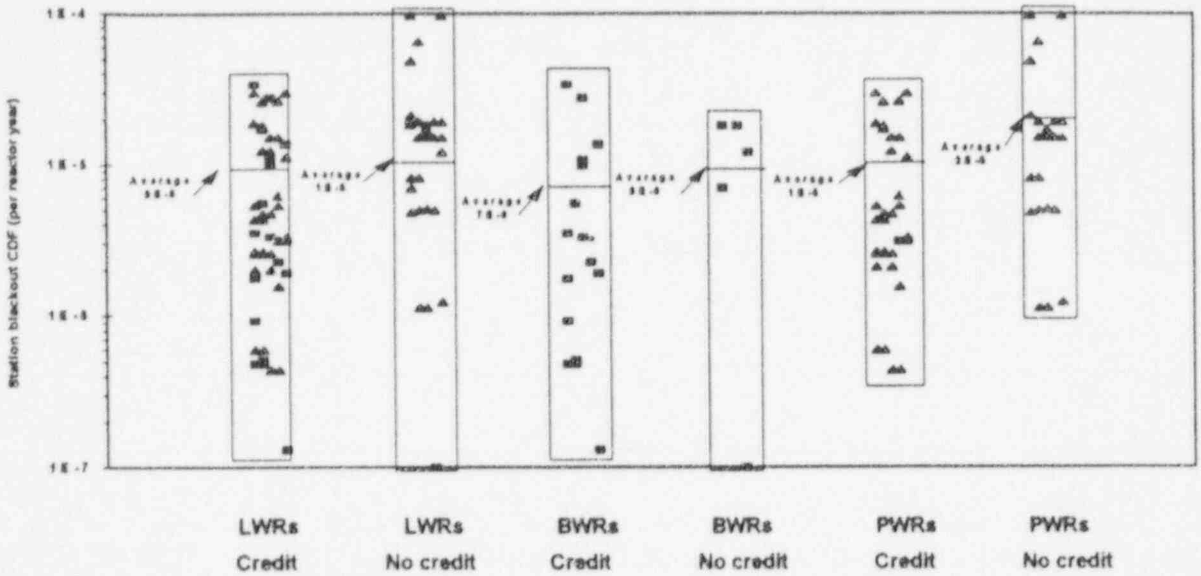

Figure 17.9 Comparison of the SBO CDF results for plants that credit the Station Blackout Rule and for those that do not.

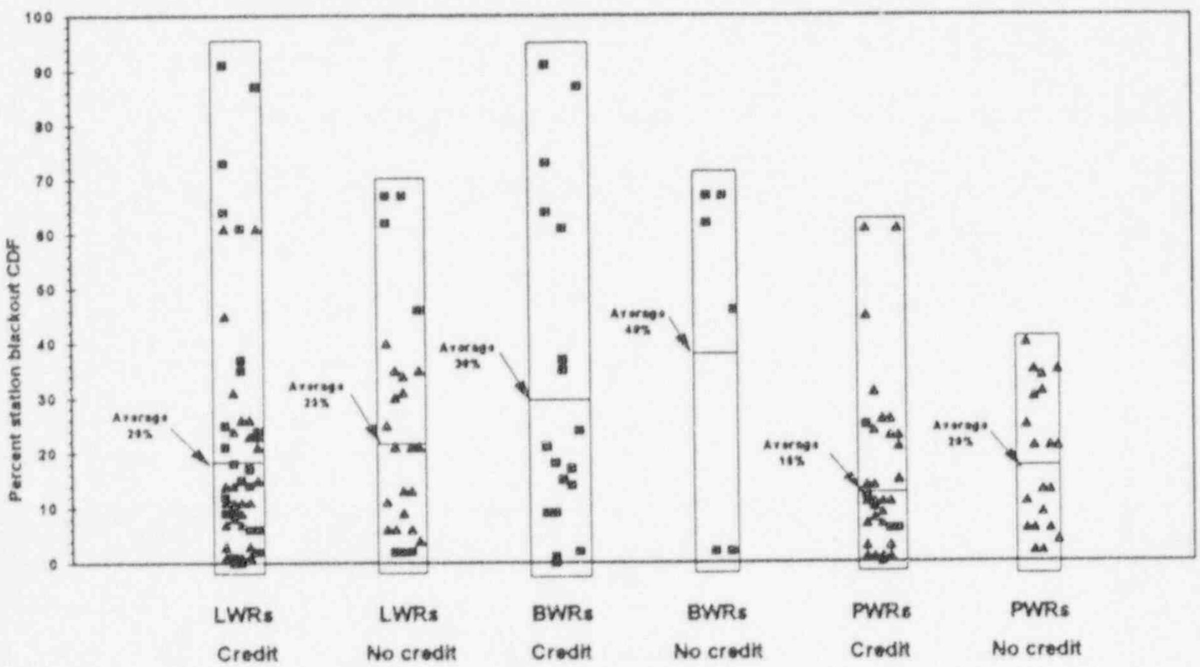

Figure 17.10 Comparison of the percent SBO CDF results for plants that credit the Station Blackout Rule and for those that do not.

17.3.2 Results of the Investigation of the Variables Affecting SBO CDF

As reported in Section 17.3.1, a station blackout can be a significant contributor to the total CDF at many plants. In addition, the variability of the estimated SBO CDF and percent SBO contribution is large. An attempt was made to identify the variables that contribute to the variation in SBO CDF and determine their relative contribution to the overall SBO CDF variation. This section reports the findings.

A review of the IPE submittals identified the following factors as possible contributors to the variation in SBO CDF results for LWRs:

- frequency of an LOSP
- probability of recovering offsite power
- credit for recovery of EDGs
- common cause modeling of EDGs
- EDG support system failures (e.g., standby service water)
- ability to cross-tie power busses within a unit and across units
- number of diesel generators; use of AAC power sources (e.g., gas turbines, appendix R EDGs, hydro power, etc.)
- battery lifetimes (with and without load shedding procedures)
- EAC group configuration (A, B, C, and D defined in Refs. 17.5 and 17.6)
- plant weather characteristics (extremely severe weather (ESW) and severe weather (SW) groups defined in Refs. 17.5 and 17.6)
- independence of offsite power systems (I) groups defined in Refs. 17.5 and 17.6
- core cooling systems that are independent of AC power (e.g., diesel-driven firewater system, turbine-driven AFW pump at PWRs, isolation condensers (ICs), high pressure coolant injection (HPCI), reactor core isolation cooling system (RCIC))
- unique systems (e.g., some PWRs have a standby facility for cooling the reactor coolant pump seals while some BWRs have installed a safe shutdown makeup pump to provide injection to the vessel at high pressure)
- unique events (e.g., vulnerability of reactor coolant pump seal to leakage (seal LOCAs) at PWRs and BWRs with isolation condensers)
- human actions (e.g., load shed to preserve DC power during SBO)

- the volume of water available for AFW in PWRs (e.g., condensate storage tank) and the ability to refill if the tanks are depleted and

- the configuration of support systems such as heating, ventilation, air conditioning (HVAC) or service water

This list does not include all of the possible factors that affect SBO, but these are the factors believed to be most important.

To determine the relative contribution of these variables to the "overall" SBO CDF variation, a regression analysis was performed for BWRs and PWRs. While information could not be obtained to model all of the variables, several key variables were examined. These key variables affect all plants and are directly relevant to station blackout. They include:

- frequency of a LOSP (IE)
- battery lifetimes (BATT)
- number of diesel generators (# EDGs)
- EAC group configuration (A, B, C, and D)
- plant weather characteristics (ESW and SW groups)
- independence of offsite power systems (I) groups and
- seal LOCAs (S-LOCA, considered at PWRs only)

Sufficient information for regressing on these key variables was available for 15 BWR submittals (17 plant units) and 18 PWR submittals (21 plant units).

The result of regression (R^2 = 80%) indicates that the six variables modeled are successful in accounting for most of the observed variation in SBO CDFs for the 15 BWR submittals. The relative contribution of each variable (i.e., normalized *T* statistic) to the SBO CDF variations is depicted in Figure 17.11. The results show that, in the presence of the other variables, the largest contribution to the SBO CDF variations is from the plant's EAC configuration. The EAC configuration is important to SBO CDF because it is dictated by the number of EDGs available at a plant versus the number that is required to achieve and maintain safe plant shutdown following a loss of offsite power. Therefore, the degree of EAC system redundancy is a critical determinant to the potential of core damage following a station blackout. The contributions from the initiating event frequency, the independence of offsite power system configurations and the number of EDGs are comparable, with slightly lower contributions from the other variables. The results of regressing on the SBO CDFs for these 15 BWR submittals indicate that, overall, the variation in SBO CDFs is driven by a combination of various factors, of which EAC configuration is most important.
For the 18 PWR submittals, the results (R^2 = 70%) indicate that the seven variables modeled are successful in accounting for a large portion of the variation in SBO CDF results. The relative contribution of each variable to the SBO CDF variations is depicted in Figure 17.12.

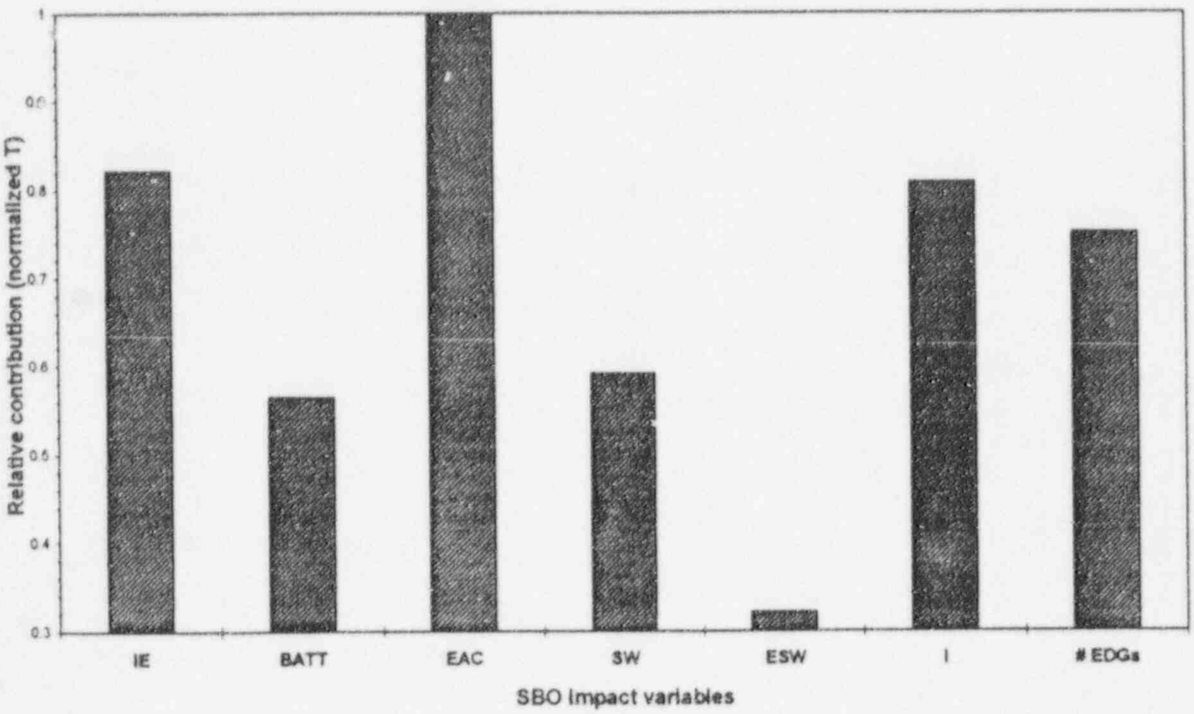

Figure 17.11 Variable contribution results for BWRs.

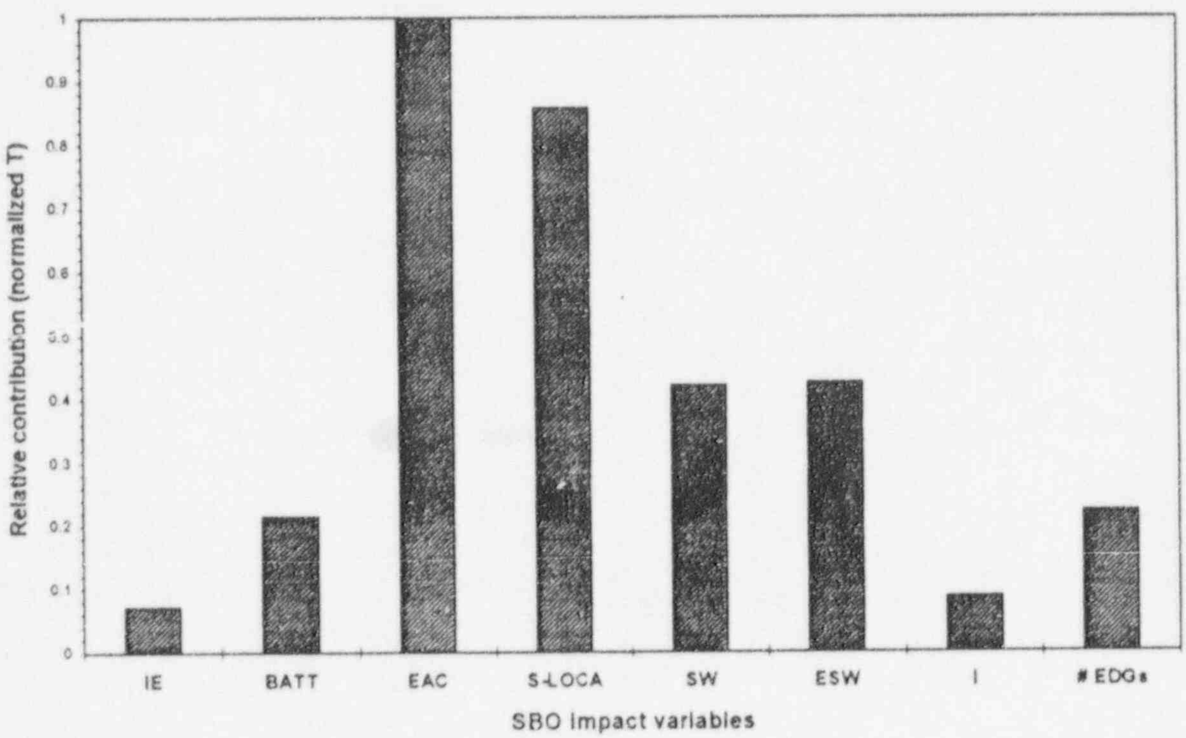

Figure 17.12 Variable contribution results for PWRs.

The results shown in Figure 17.12 indicate that, in the presence of the other variables, the plant's EAC configuration and seal LOCAs contribute most to the variation in SBO CDFs, with lesser contributions from the other variables. The importance of the EAC group configuration to SBO is discussed above. The importance of the seal LOCAs at PWRs is now addressed.

At PWRs, a SBO can lead to a total loss of seal cooling, once it occurs, leads to loss of coolant through the failed seals. There is no system to provide sufficient makeup at a PWR during SBO. Consequently, a seal LOCA is an important contributor to station blackout at many PWRs. The importance of seal LOCAs varies under different circumstances. In a few IPEs, loss of component cooling water (CCW) cooling to the RCP motors with failure of operator action to promptly trip the RCPs is an important contributor to the seal LOCA CDF. With loss of cooling to the RCP motor bearings, a vibration-induced seal LOCA can occur. Some of the licensees have assumed that prompt operator action to trip the RCPs prevents this scenario from being an important contributor. In some IPE submittals for plants that do not require CCW for cooling safety injection pumps that can mitigate a seal LOCA, it is concluded that the probability of both failure of operators to trip the RCPs and failure of injection to mitigate the resulting seal LOCA renders this scenario unimportant.

The IPE submittals indicate that the following modeling assumptions affect the likelihood of a seal LOCA:

- time available for operator action to trip running RCP following loss of RCP motor cooling
- likelihood of operator action to trip RCP following loss of motor cooling
- operator action to restore seal cooling to a tripped RCP, including backup methods for providing cooling for charging pumps
- time required for tripped RCP without seal cooling to develop a seal LOCA
- magnitude of the leak from a seal LOCA and
- credit for depressurization to reduce likelihood of a seal LOCA and leakage rate from a seal LOCA

The results of regressing on the SBO CDFs for these 18 PWR submittals indicate that, overall, the variation in SBO CDFs is driven by a combination of various factors, of which EAC configuration and seal LOCAs are the most important.

A further examination of the IPE submittals reveals that for those plant units with significantly lower SBO CDFs than other plant units, the plant design features that tend to reduce the significance of SBO are the number of EDGs (at least four), battery depletion time (6 hours or more), and a reliable injection source or core heat removal source that is not dependent on AC power (e.g., AFW, HPCI, and RCIC). With regard to the number of EDGs, those plant units that have a backup EDG or gas turbine generator and the ability to cross-tie to other units or between divisions tend to have low SBO CDFs. Credit for operator actions to (1) extend the time available to recover offsite power or EDGs, (2) align firewater for injection, (3) load shed batteries, or (4) manually control AFW following battery depletion also tends to reduce SBO CDF.

For plant units that have higher SBO CDFs compared to other plant units, factors such as only having a short time in which to recover offsite power, a high likelihood of an RCP seal LOCA following a station blackout, a short battery lifetime (4 hours or less), a low EDG redundancy for achieving and maintaining a safe plant shutdown following a LOSP (i.e., one of two EDGs) and a limited supply of condensate suction for AFW contribute to the large values.

These results, when combined with those from the simple regression, indicate that overall no one factor but rather a combination of various design features, site characteristics, and/or modeling assumptions/techniques tends to drive the variation in SBO CDF results.

17.4 Summary and Conclusions on the Impact of the Station Blackout Rule

A four-step technical approach was developed and applied to examine the impact of the station blackout rule, based on submittals of the nuclear industry's IPEs. In the following paragraphs, the results of the examination are summarized and compared to other SBO studies, and a conclusion is drawn.

In the first step of the approach, a preliminary screening analysis of 75 IPE submittals was done to determine how the rule was incorporated into the IPE submittals. Numerical results of the plant units' total CDFs, SBO CDFs, percent SBO CDF contributions, reduction in total CDFs, reduction in SBO CDFs, and reduction in percent SBO CDF contributions were also gathered, whenever available, during this first step. In the second step, the IPE submittals were grouped into seven categories, depending on whether the rule is credited, whether the impact of the rule in terms of reduction in CDF is known, or whether the information to evaluate the impact of the rule is insufficient, inconsistent, or nonexistent. In the third step, in the categories where the SBOR is credited, not credited, and where the impact of the rule in terms of reduction in CDF is known, information from the IPE submittals was used to provide insights into the impact of the station blackout rule. Where the information to categorize the impact of the rule is insufficient, inconsistent, or nonexistent, no further analysis was done. In the fourth and final step, the factors that contribute to the variation in SBO CDF results were identified when sufficient details were provided in the IPE submittals. A regressional analysis was then performed to investigate the relative contribution of each variable to the overall SBO CDF variation. While information could not be obtained to model all of the variables, several key variables were examined. These key variables affect all plants and are directly relevant to the SBOR. The most important insights gained from evaluating the impact of the SBOR are as follows:

- A limited analysis of 10 plant submittals (15 plant units) indicates that the potential reduction and percent reduction in total CDFs from implementing the SBOR can be significant (means of approximately 2E-5/ry and approximately 25%, respectively). Assuming that a substantial portion of the reduction in total CDF from implementing the rule can be attributed to SBO, the potential reduction in SBO CDF is significant. Note that there are some plant units that met the SBOR using existing battery capacity. For this group of plant units, the rule had no impact.

- The evaluation of 53 plant units for which the SBOR was credited indicates a large variability in both the SBO CDF and percent SBO contribution (negligible to approximately 3E-5/ry and 0% to a high of approximately 90%, respectively). However, on average, the estimate of SBO CDF (9E-6/ry) is comparable to the SBOR goal of 1E-5/ry. The large variability in the results arises from the fact that some plant units that credit the rule had SBO CDFs close to two orders of magnitude lower than the goal while others had SBO CDFs close to three times the goal. The plant units with the lowest SBO CDFs and percent contributions attributed the low values not only to reductions from crediting the rule but also to other factors, such as using lower than typical values for diesel generator failure probabilities or having diesels that do not depend on room cooling. For plant units that have high SBO CDFs compared to the SBOR goal and high percent SBO contributions, factors such as a short battery lifetime and thus a short time in which to recover offsite power are important.

- The evaluation of 27 plant units for which the SBOR is not credited also indicates a large variability in both the SBO CDFs and percent SBO contributions (1E-7/ry to approximately 7E-5/ry and approximately 2% to approximately 65%, respectively). In addition, the estimated mean SBO CDF (1E-5/ry) is comparable to the SBOR goal. Some units had SBO CDFs two orders of magnitude lower than the SBOR goal while others have SBO CDFs close to an order of magnitude higher than the SBOR goal. The plant units with the lowest SBO CDFs and percent contributions attribute the low values to existing unit configurations, such as having four "better than typical" redundant and independent EDGs per unit or having an extremely reliable grid. Others attribute the low value to using lower than typical values for failure to restore offsite power and diesel generator failure probabilities. For plant units that have high SBO CDFs compared to the SBOR goal and high percent SBO contributions, the large values are attributed to factors such as high frequency of loss of offsite power or high probability of seal LOCAs.

- The results of a simple regression of the SBO CDFs for 15 BWR submittals indicate that, overall, the variation in SBO CDFs is driven by a combination of various factors (i.e., frequency of a LOSP, battery lifetimes, number of diesel generators, EAC configuration, plant weather characteristics, independence of offsite power systems, and seal LOCAs at PWRs only), of which EAC configuration appears to be the most important. For the 18 PWR submittals, the results of regressing on the SBO CDFs indicate that, overall, the variation in SBO CDFs is driven by a combination of various factors, of which EAC configuration and seal LOCAs appear to be the most important.

- The average reduction in total CDF for 10 IPE submittals that implemented the SBOR is estimated at 2E-5/ry (a significant portion of which is due to the reduction in SBO CDF). This result is consistent with the mean reduction in total CDF (approximately 3E-5/ry) from a backfit analysis of the rule (Ref. 17.7). Another comparison shows that the average SBO CDF results for the plant units considered in this study (approximately 2E-5/ry) are comparable to a "typical" estimate (order of 1E-5/ry) from an evaluation of station blackout accidents at nuclear power plants (Ref. 17.1). The large variability in the SBO CDF results for the units (negligible to approximately 7E-5/ry) is also consistent with the variability in the SBO CDF results (1E-6/ry to 1E-4/ry) from Ref. 17.1.

In conclusion, the nuclear industry's IPE submittals indicate that the impact of implementing the SBOR is a measurable reduction in total CDF (an estimated mean of 2E-5/ry), a significant portion of which is due to the reduction in SBO CDF. However, there are exceptions. For example, in cases where plant units use existing battery capacity to cope with station blackouts, the rule has no impact on the SBO CDFs. Some reduction in CDF can be expected from implementing the SBOR, it can be deduced that the SBOR is one of the reasons why plant units that credit it have, on average, a lower SBO CDF than plant units that do not credit it. However, the SBO CDF is also affected by a combination of factors (ie., design features, site characteristics, and/or modeling assumptions and techniques) unique to each plant unit, and these factors also contribute to this difference. In general, the results from this study appear to be consistent with those from other station blackout studies.

REFERENCES

17.1 USNRC, "Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to the Unresolved Safety Issue A-44," NUREG-1032, June 1988.

17.2 USNRC, "Station Blackout Accident Analyses (Part of NRC Task Action Plan A-44)," NUREG/CR-3226, May 1983

17.3 USNRC, "Loss of All Alternating Current Power," Code of Federal Regulations, Title 10, Section 50.63, July 1988.

17.4 USNRC, "Station Blackout (10 CFR Part 50)," Federal Register, Vol. 53, No. 119, June 1988, p. 23211.

17.5 Nuclear Management and Resources Council, "Guide-lines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," NUMARC-8700, November 1987.

17.6 USNRC, "Station Blackout," Regulatory Guide 1.155, August 1988.

17.7 USNRC, "Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout Accident," NUREG-1109, June 1988.

18. COMPARISON WITH NUREG-1150 PERSPECTIVES

Chapter 7 of Part 1 summarizes the key perspectives on several additional items. Section 7.3 (of Chapter 7, Part 1) summarizes the results of the Individual Plant Examinations (IPEs) compared with the perspectives gained from NUREG-1150[18.1]. Chapter 18 provides a more in-depth discussion including more detail on a numerical comparison of the results and the underlying reasons for the observed difference in the core damage frequency (CDF) analyses and containment performance assessments, and contrasts the perspectives derived from the NUREG-1150 study with those drawn from the reported IPE results.

18.1 Background

In 1990 the U.S. Nuclear Regulatory Commission (NRC) published NUREG-1150 assessing the risks for five U.S. nuclear power plants:

- Unit 1 of Surry Power Station, a Westinghouse-designed three-loop pressurized water reactor (PWR) in a subatmospheric containment building (also evaluated in WASH-1400).[18.2]

- Unit 1 of the Zion Nuclear Plant, a Westinghouse-designed four-loop PWR in a large dry containment;

- Unit 1 of the Sequoyah Nuclear Power Plant, a Westinghouse-designed four-loop PWR in an ice condenser containment building;

- Unit 2 of the Peach Bottom Atomic Power Station, a General Electric-designed boiling water reactor (BWR) in a Mark I containment building (this BWR 4 reactor was also evaluated in WASH-1400);

- Unit 1 of the Grand Gulf Nuclear Station, a General Electric-designed BWR 6 reactor in a Mark III containment building.

While the NUREG-1150 plants represent a spectrum of designs, they do not cover all vendors and do not represent a large enough sample to be considered representative of the industry.

On August 8, 1985, the NRC issued a Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants[18.3] that introduced the Commission's plan to address severe accident issues for existing commercial nuclear power plants. In this Policy Statement, the Commission addressed its plan to formulate an approach for a systematic safety examination of existing plants to study particular accident vulnerabilities and desirable cost-effective changes to ensure that there is no undue risk to public health and safety. To implement this plan, NRC issued Generic Letter 88-20[18.4] in November 1988 requesting that all licensees perform an IPE to identify any plant-specific

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

[18.2]USNRC, "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.

[18.3]USNRC, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," Federal Register, Vol. 50, No. 153, p. 32138, August 8, 1986.

[18.4]USNRC, "Individual Plant Examination for Severe Accident Vulnerabilities-10CFR§50.54(f)," Generic Letter No. 88-20, November 23, 1988.

vulnerabilities to severe accidents, and to report the results to the Commission. Most licensees performed a Level 1 and 2 probabilistic risk assessment (PRA) in response to the generic letter and reported CDF and accident progression results to the Commission. The CDF and accident progression results reported in NUREG-1150 are compared below with IPE results for comparable reactor and containment designs. The perspectives obtained from NUREG-1150 and the IPEs are also compared.

18.2 Core Damage Frequency Results

Figure 18.1 shows the NUREG-1150 CDF results compared to the IPE results. In the NUREG-1150 analyses, an uncertainty distribution was calculated for each of the five plants. For those five plants, the lower and upper extremities of the bars in Figure 18.1 represent the 5th and 95th percentiles of the distributions, with the mean value of each distribution also shown. More detailed descriptions of the shapes of the distributions can be found in NUREG-1150. As can be seen in Figure 18.1, the NUREG-1150 CDF results fall within the range of the IPE CDF results.

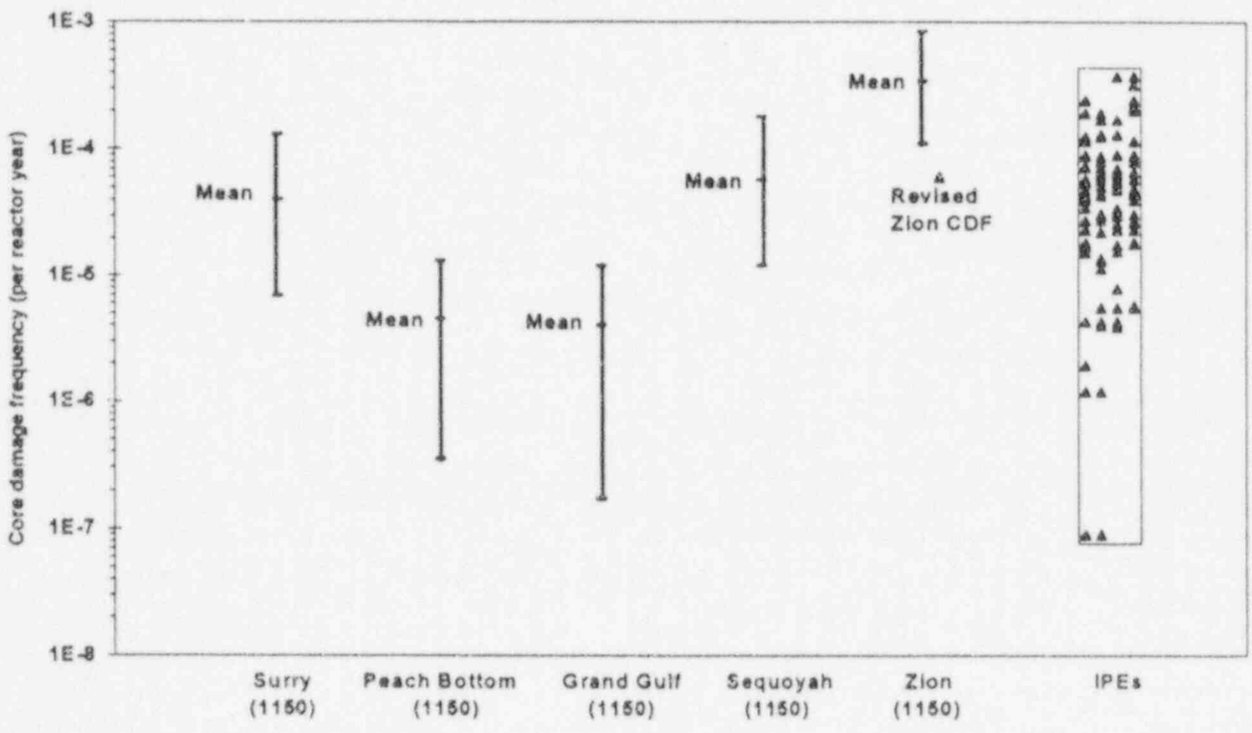

Figure 18.1 Comparison of NUREG-1150 and IPE CDFs.

Figure 18.1 shows that the range between the 5th and 95th percentiles covers from one to two orders of magnitude for the five NUREG-1150 plants. When comparing the NUREG-1150 results to the IPE results, the reader should remember that the IPE results reflect a mix of means and point estimates. Uncertainty distributions are not shown here for the IPEs, as they are not available in many cases. However, when uncertainties have been provided, their range is consistent with the NUREG-1150 distributions. The uncertainty ranges do make it clear that undue weight should not be put on small differences among the results. Based upon Figure 18.1, the overall NUREG-1150 and IPE results are reasonably consistent.

Figure 18.2 shows the results broken down for PWRs and BWRs. The mean CDFs for Surry, Sequoyah, and Zion fall within the range of PWR IPE values. Note that two values are presented for Zion in NUREG-1150, with the second, lower value reflecting some plant changes as of October 1990. Likewise, the Peach Bottom and Grand Gulf mean CDFs fall within the range of BWR IPE values. Figure 18.3 further shows the NUREG-1150 results compared to particular IPE plant groupings. In each case the differences are within the range expected, given that there are many plant-specific design differences and PRA modeling differences.

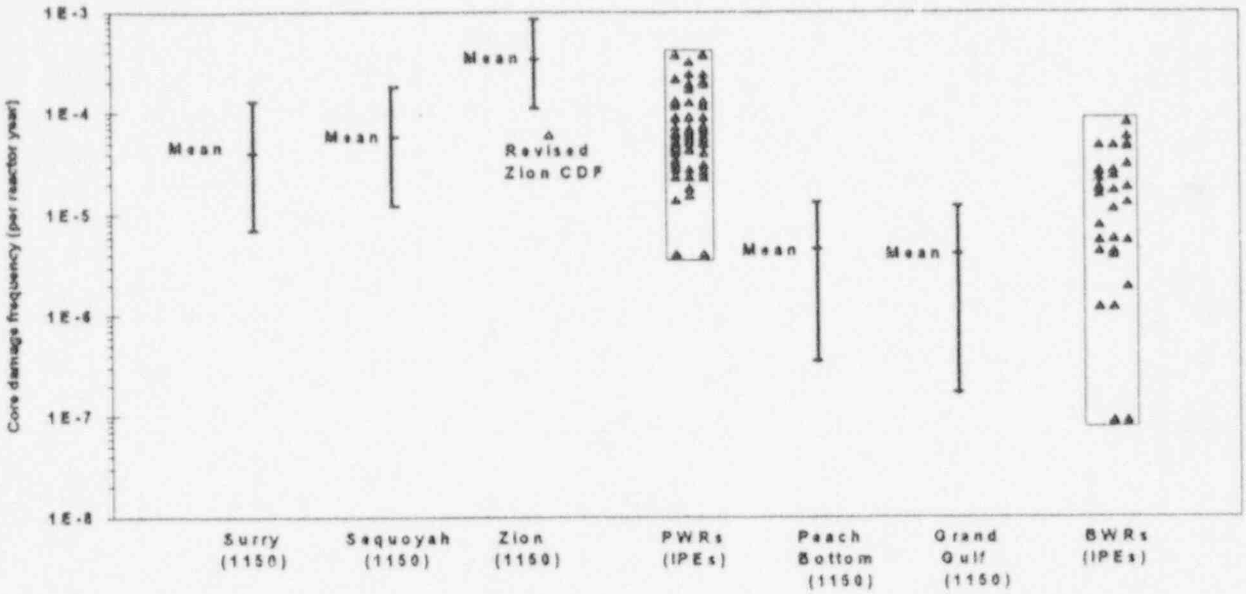

Figure 18.2 Comparison of NUREG-1150 and IPE CDFs for PWRs and BWRs.

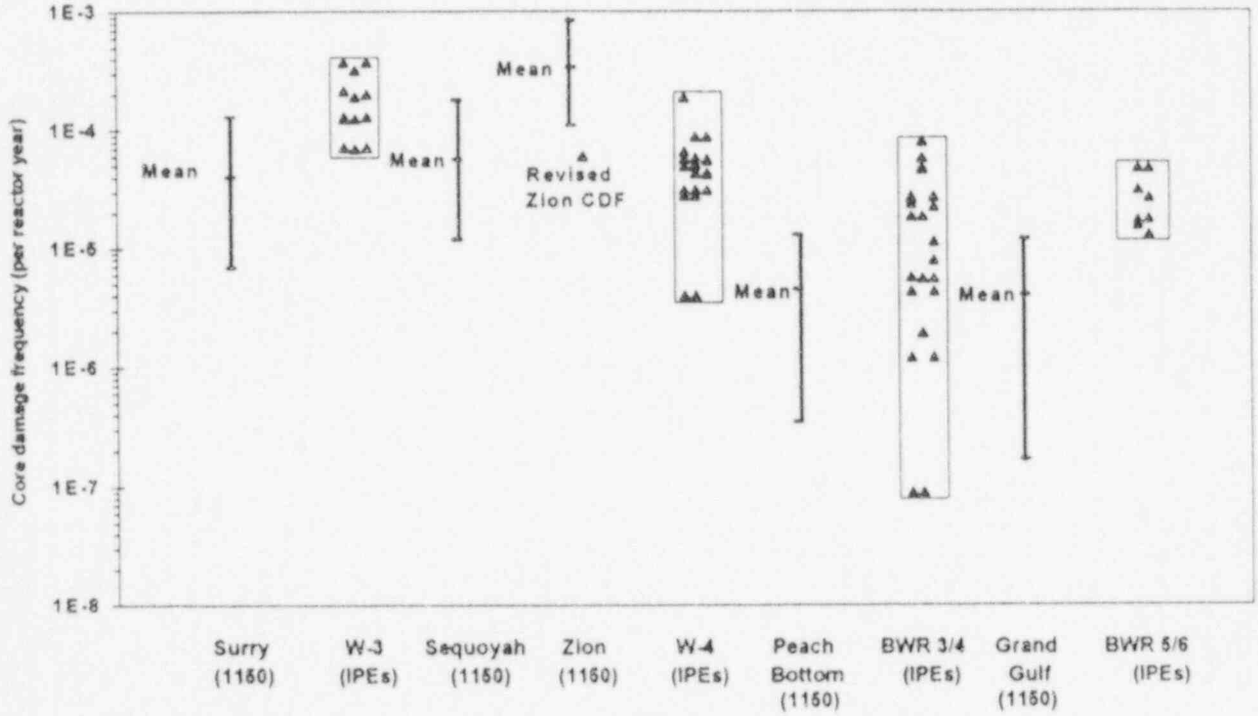

Figure 18.3 Comparison of NUREG-1150 and IPE CDFs for selected plant types.

Both NUREG-1150 and the IPEs have shown that the relative contributions of accident sequences to the CDF are plant specific. Therefore, the accident sequence which dominates in one plant may not be dominant in another. However, Figures 18.4 and 18.5 show that the mix of contributors is consistent with the results found in NUREG-1150. That is, for the PWRs, station blackout, transients and loss-of-coolant accidents (LOCAs) tend to be important contributors, while for the BWRs, station blackout and transients tend to be the most important, with lesser contributions from anticipated transients without scram and LOCAs. Internal flooding was examined only for the Surry and Peach Bottom plants in NUREG-1150 and was not found to be important. Internal flooding is a significant contributor for a few of the IPEs. At a more detailed level, the specific failures leading to core damage sequences are plant specific and cannot be easily compared between NUREG-1150 and the IPEs.

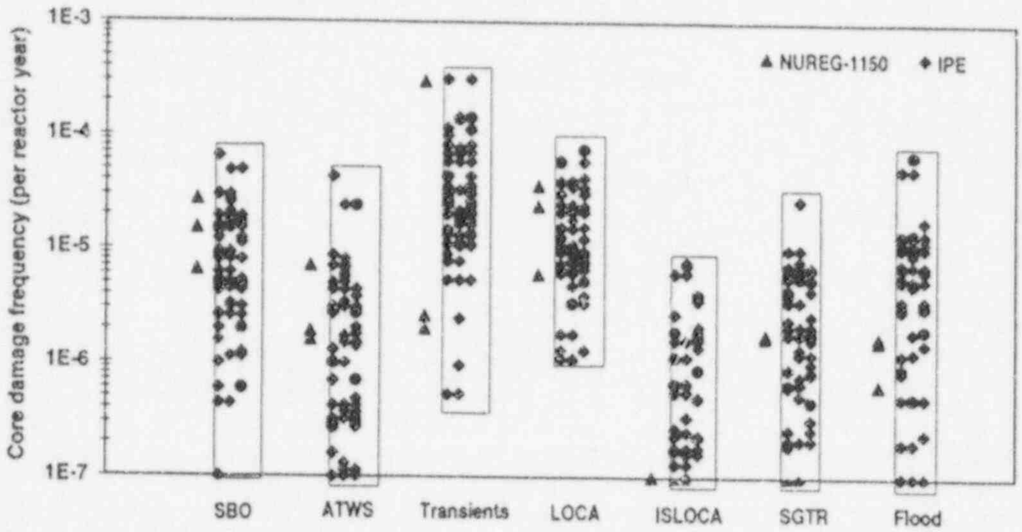

Figure 18.4 Comparison of NUREG-1150 and IPE PWR dominant contributors.

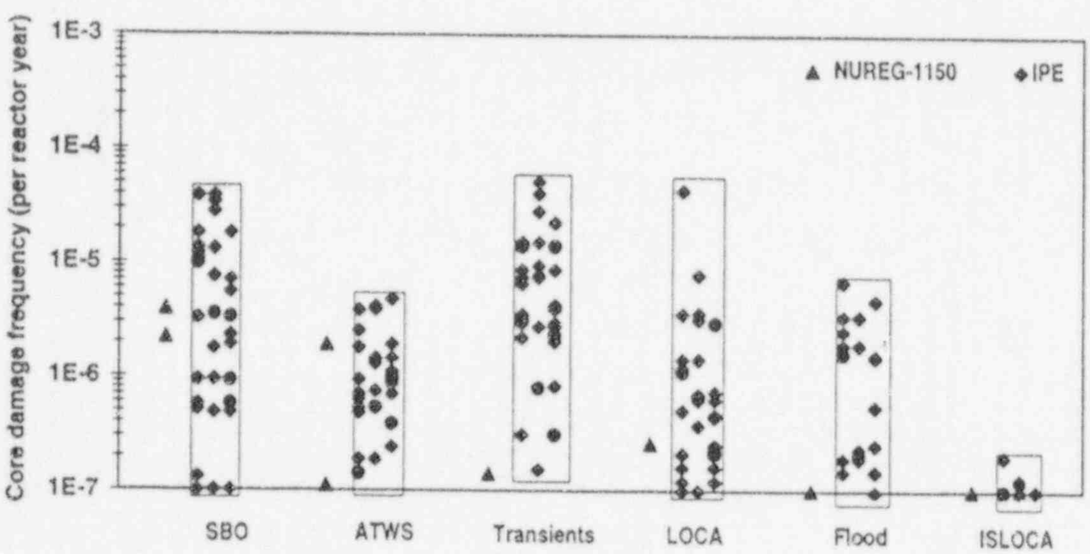

Figure 18.5 Comparison of NUREG-1150 and IPE BWR dominant contributors.

18.3 Accident Progression Results

After evaluating the accident sequences leading to core damage and calculating the CDF, both NUREG-1150 and the IPEs evaluated the ability of the containments to prevent the release of radioactivity. The five containment types included in the NUREG-1150 study are compared with IPE results for similar containment designs. The only exception is the IPE results for BWRs with Mark II containments. This containment design was not included in NUREG-1150 and therefore could not be compared with the IPE results for Mark II containments.

Figures 18.6 and 18.7 compare the NUREG-1150 probability of early containment failure conditional on core damage with the IPE results. These probabilities include events that cause structural failure of containment and also isolation failure. Accidents that cause containment bypass (such as interfacing systems LOCA, and steam generator tube ruptures) are not included in these figures. However, the frequency of these events is combined with the frequency of early containment failure for the results presented in Figures 18.8 and 18.9.

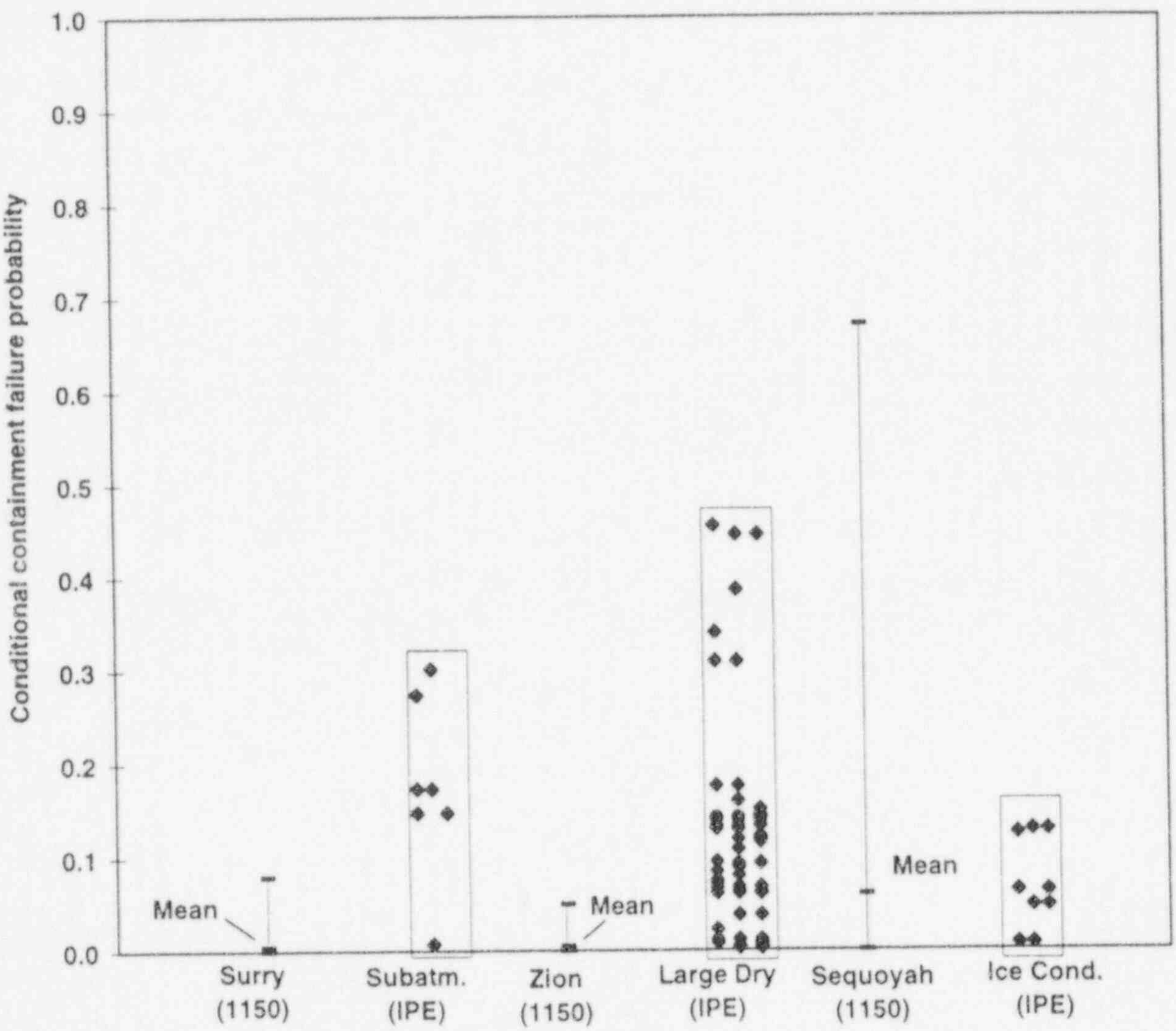

Figure 18.6 Comparison of NUREG-1150 and IPE conditional probabilities for early containment failure for PWR plants.

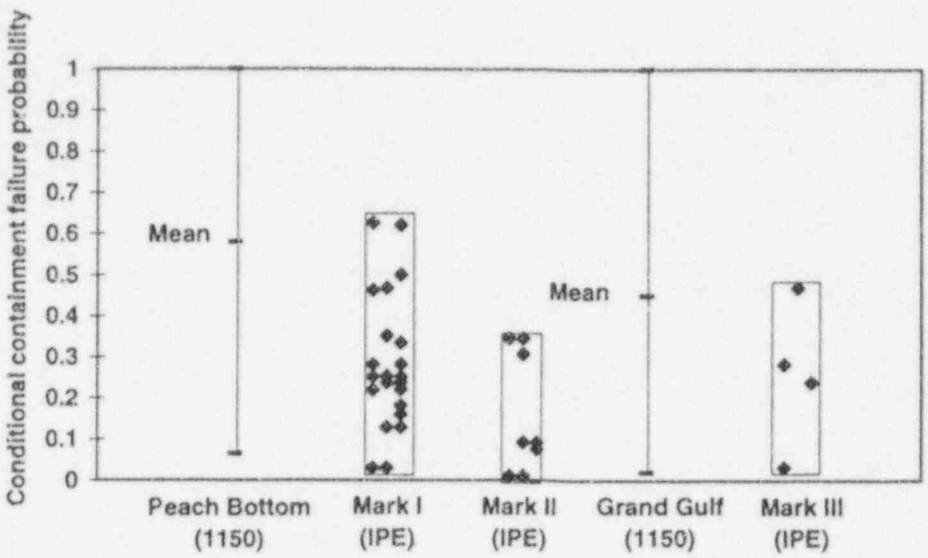

Figure 18.7 Comparison of NUREG-1150 and IPE conditional probabilities for early containment failure for BWR plants.

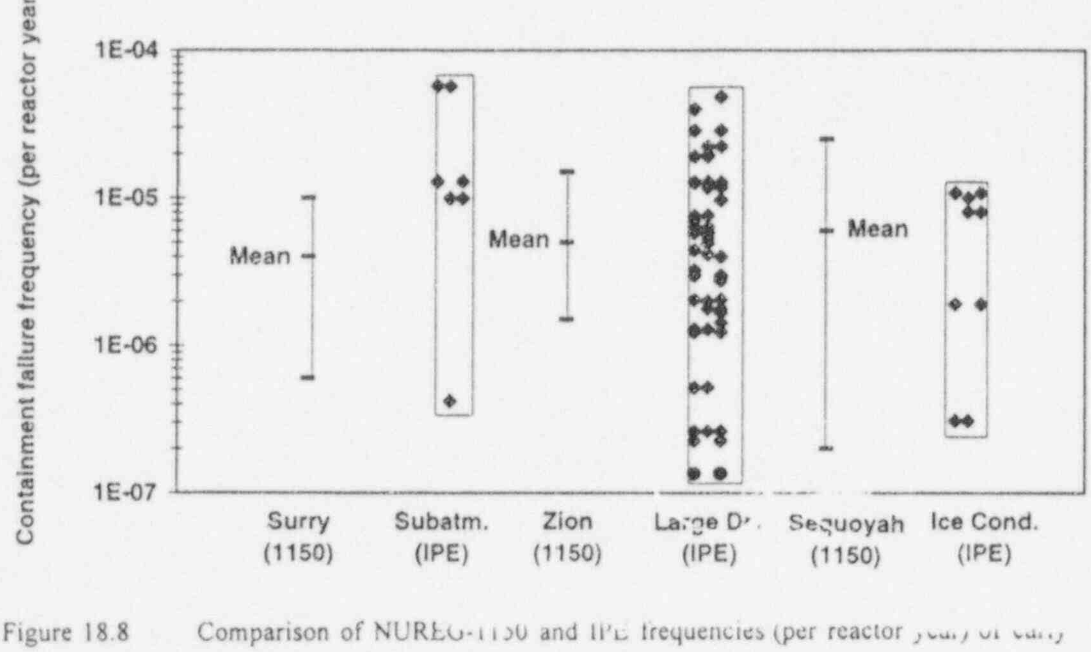

Figure 18.8 Comparison of NUREG-1150 and IPE frequencies (per reactor year) of early containment failure or bypass for PWR plants.

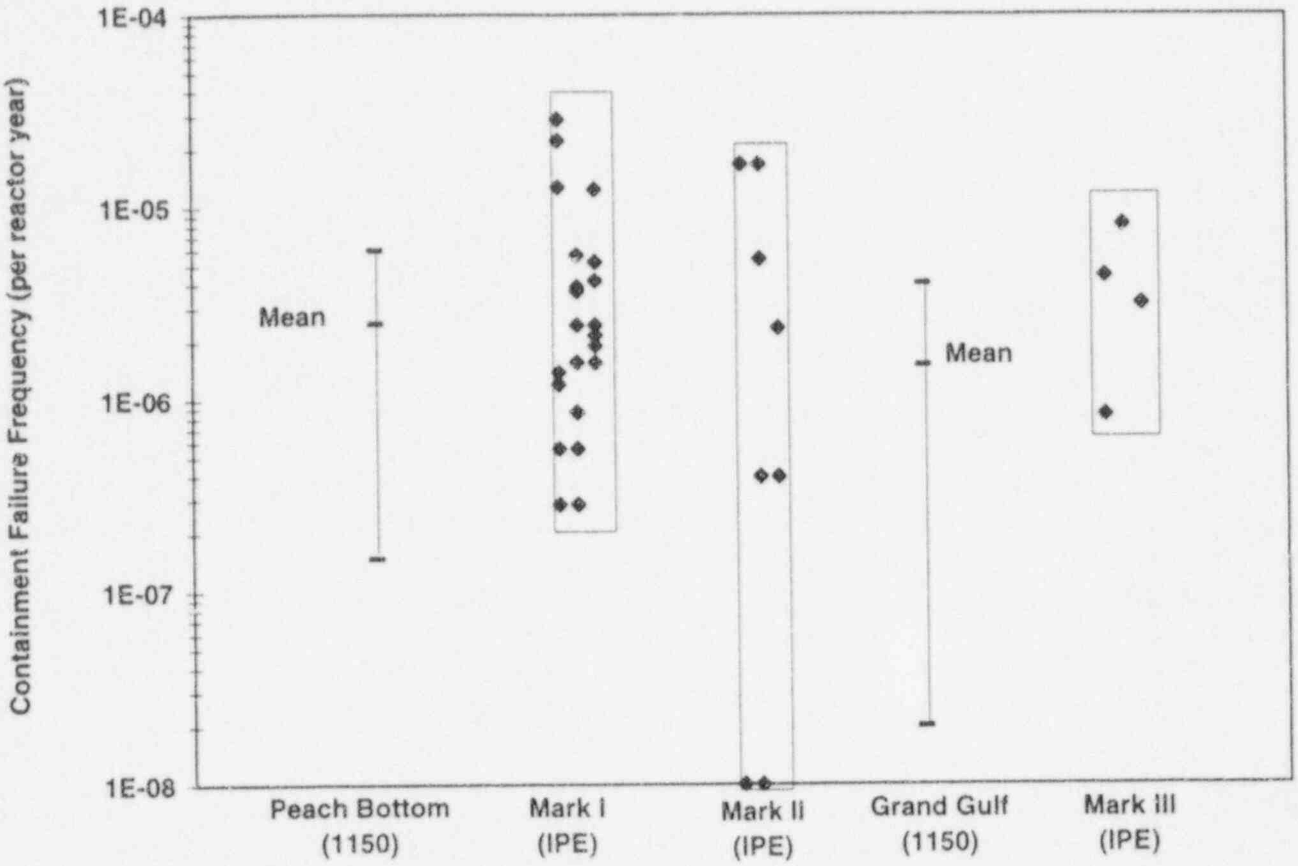

Figure 18.9 Comparison of NUREG-1150 and IPE frequencies (per reactor year) of early containment failure or bypass for BWR plants.

In the NUREG-1150 analyses, uncertainty distributions were calculated for the containment failure and bypass estimates. Therefore, the 5th and 95th percentiles and the means of the distribution are shown in the figures for the five NUREG-1150 plants. As can be seen in the figures, the NUREG-1150 results (mean values) fall within the range of the IPE results for each containment type. However, the ranges between the 5th and 95th percentiles for some of the NUREG-1150 results are quite wide (from two to three orders of magnitude) and span the IPE results.

Figures 18.6 and 18.7 compare the NUREG-1150 results for the conditional probability of an early failure for each containment type evaluated in NUREG-1150 with the IPE results. NUREG-1150 found that the conditional probability of early failure is significantly lower for PWRs with large volume and subatmospheric containments than for PWRs and BWRs with pressure suppression containments. This trend is also apparent for the BWR IPE results but not for the PWR ice condenser containments. The conditional probability of early failure for large volume containments in the IPEs varies from less than 0.01 to 0.3 with an average value of 0.05 and varies for subatmospheric containments from less than 0.01 to 0.25 with an average of 0.1. These results can be compared with the conditional probabilities of early containment failure for ice condenser containments, which vary from less than 0.01 to only to 0.05 with an average value of 0.01, and for BWR containments, which vary from about 0.01 to 0.6 with an average of about 0.3.

The IPE results do indicate that BWR containments generally have higher conditional probabilities for early failure than for PWR plants but there is significant variability in the results. For example, several IPEs with Mark I containments have extremely low early failure probabilities. These low probabilities are in some cases caused by modeling assumptions (neglecting shell melt-through) or by plant specific features (sumps or curbs that physically prevent shell melt-through). Conversely there are PWRs with large dry containments that have relatively high-early failure probabilities. These higher probabilities are also caused by modeling assumptions (e.g., containment failure pressure) and by plant-specific features (e.g., the sump and recirculation piping in the Palisades plant) not found in other large dry containment designs. The greater variation in the IPE results was expected because many more plants, using a wider range of modeling assumptions, were analyzed than in the NUREG-1150 study.

Figures 18.8 and 18.9 provide a comparison of the NUREG-1150 and IPE results for the frequency (per reactor year) of a severe accident with early containment failure or bypass. The NUREG-1150 results show that the absolute frequencies of early containment failure or bypass for the BWR designs analyzed are similar (the means are within 1E-6/ry to 6E-6/ry) to the absolute frequencies for the PWRs. The reason for this result is that the core damage frequencies were found to be lower for the BWRs than for the PWRs studied in NUREG-1150. Therefore, the higher conditional probabilities of early failure for BWRs were compensated for by the lower CDF. The average IPE frequencies are similar to the NUREG-1150 values but individual plant results vary significantly. The frequencies of early failure and bypass for the PWR IPEs varied from about 7E-8/ry to 6E-5/ry with an average of 9E-6/ry, and for the BWR IPEs varied from about 3E-7/ry to 3E-5/ry with an average of 6E-6/ry.

In general, the events that contribute to the IPE frequencies in Figures 18.8 and 18.9 are similar to those that contributed in the NUREG-1150 study. For example, direct containment heating is an important failure mode for PWRs with large dry and subatmospheric containments, hydrogen combustion is important for PWR ice condensers and BWR Mark III containments, and shell melt-through is important for BWR Mark I containments. In addition, accidents that bypass containment are significant contributors to the frequencies shown in Figure 18.8 for some PWR plants. However, both NUREG-1150 and the IPEs show that the relative contributions of the various containment failure modes vary from plant to plant and depend on plant-specific features and modeling assumptions.

18.4 Core Damage Frequency Perspectives

NUREG-1150 provided general perspectives for BWRs and PWRs based on the five plants included in that study. The perspectives derived from NUREG-1150 are compared with those obtained from the IPE results below. The approach adopted is to summarize a NUREG-1150 perspective and then determine whether the IPE results support those findings.

NUREG-1150 Perspective: BWRs tend to have lower CDFs than PWRs

BWRs tend to have internal event core damage frequency distributions that are lower than those of PWRs, although this finding is less pronounced for the IPEs. Figure 18.2 shows that there is overlap between the PWR and BWR CDFs in the IPEs. There are several reasons why the BWR CDFs tend to be lower than the PWR CDFs. The LOCA sequences, often significant in the PWR core damage frequencies, are usually minor contributors in the case of the BWRs. This is not surprising, since most BWRs have many more systems than PWRs for injecting water directly into the reactor coolant system to provide makeup during LOCAs or transient-induced LOCAs involving stuck-open relief valves. Further, BWRs can more easily depressurize to use low-pressure systems. PWRs have highly reliable emergency core cooling systems (ECCS), but with less redundancy and diversity than BWRs. PWRs generally have one high-pressure and one low-pressure ECCS (both multitrain), plus a set of accumulators. For many types of transient events, BWRs also tend to have more systems that can provide decay heat removal than do PWRs.

PWRs usually have somewhat higher station blackout frequencies because, unlike BWRs, if alternating current (AC) power is lost they have no systems to inject directly into the reactor coolant system to provide makeup in the case of system leakage from stuck-open relief valves or seal leaks.

NUREG-1150 Perspective: Support systems are crucial to the CDF

For both BWRs and PWRs, the reliability of the support systems is quite important in determining the CDF. These systems include electric power, service water, instrument air, heaving, ventilating, and air conditioning, and other systems that support the front-line emergency core cooling systems. Because the design of these support systems varies considerably among plants, caution must be exercised when making statements about generic classes of plants, such as PWR versus BWR. Both types of plants have sufficient redundancy and diversity to make multiple independent failures unlikely. Support system failures introduce dependencies among the systems and thus can become dominant for both types of plants. For example, the interdependencies introduced by the support systems can override the higher redundancy of the BWR ECCS. Several of the perspectives below result from support system vulnerabilities.

NUREG-1150 Perspective: Operator recovery actions significantly reduce the CDFs

Recovery actions range from simple actions such as manually opening a valve to complex actions such as providing injection from non-safety systems. The NUREG-1150 PRAs and the IPEs have taken extensive credit for operator recovery actions. Improvements in emergency operating procedures over the past several years are responsible for increased reliability of the operators and identification of additional actions that they can take. In addition, improvements in human reliability analysis methods have led to more realistic assessments of the potential for operator success. The move to symptom-based procedures during the 1980's and improved methods for training have been very positive steps. While there is considerable variability among the plants concerning the effectiveness of particular recovery actions, the overall impact is positive.

NUREG-1150 Perspective: Properly designed cross-ties between systems can substantially decrease the core damage frequency

Many plants can cross-tie at least a few important systems. Cross-ties allow failures within systems to be circumvented. The cross-ties can be fairly simple connections among parallel trains of a system or complex connections among different units of a multi-unit site. Cross-ties typically involve systems such as electric power, auxiliary feedwater, service water, and various water storage tanks. In particular, the availability of electrical cross-ties have lowered the CDF at many plants. Since there is a potential for incorrect cross-connecting, proper administrative control is very important.

NUREG-1150 Perspective: Station blackout is important at both PWRs and BWRs

Station blackout events are usually important at both PWRs and BWRs. On average, station blackout accidents contribute a higher percentage of the core damage frequency for the BWRs. However, on an absolute scale, station blackout tends to occur more frequently at the PWRs than at the BWRs. To some extent this is because design differences between BWRs and PWRs lead to different susceptibilities. For example, in station blackout accidents, Westinghouse PWRs are potentially vulnerable to reactor coolant pump seal LOCAs following loss of seal cooling, leading to loss of inventory with no method for providing makeup. BWRs, on the other hand, have at least one injection system that does not require AC power. While these differences are important, it would be incorrect to imply that they are the only considerations that drive the variations in the core damage frequency. Probably more important is the electric power system design at each plant (including emergency power sources and their support system requirements), which is largely independent of the plant type.

NUREG-1150 Perspective: Containment venting can reduce CDF at BWRs

The response of containment is often a key in determining the core damage frequency for BWRs, unlike most PWRs. For example, in Mark I containments, there are a number of ways in which containment conditions can affect coolant injection systems. High-pressure in containment can lead to closure of primary system relief valves, thus failing low-pressure injection systems, and can also lead to failure of steam-driven high-pressure injection systems due to high-turbine exhaust back pressure. High-suppression pool temperatures can also lead to the failure of systems that are recirculating water from the suppression pool to the reactor coolant system. If the containment ultimately fails, certain systems can fail because net positive suction head in the suppression pool is lost and also because the harsh steam environment in the reactor building can lead to failure of equipment there. Venting the containment can reduce or eliminate many of the above concerns. NUREG-1150 examined the effect of containment venting on the core damage frequency at Peach Bottom, which has a Mark I containment. Assuming the containment venting system was not available, the point estimate of the core damage frequency increased by a factor of approximately 3. Many of the BWR IPEs have taken credit for containment venting.

NUREG-1150 Perspective: Loss of service water or component cooling water can be dominant at PWRs

The NUREG-1150 Zion analysis and many of the PWR IPEs found that component cooling water is needed for operation of the charging and high-pressure safety injection pumps. Loss of component cooling water (or loss of service water, which will also render component cooling water inoperative) will result in loss of these high-pressure systems. This can further lead to loss of cooling to reactor coolant pump seals, resulting in leakage from the reactor coolant system without the capability to inject high-pressure makeup into the primary system. Thus, loss of component cooling water or service water can both cause a small LOCA and disable the systems needed to mitigate it. Seal leakage tends to be more of an issue for Westinghouse plants, although the problem is not confined solely to those plants. New seals and alternative methods of cooling, which are already being implemented at some plants, will considerably reduce the likelihood of significant leakage.

NUREG-1150 Perspective: Feed and bleed cooling is an important safety strategy at many PWRs

Feed and bleed cooling substantially reduces the CDF at many PWRs. It represents an alternative method for decay heat removal in transients involving a total loss of feedwater. Successful feed and bleed cooling requires either opening at least one power operated relief valve (PORV) or, at some plants, using high-pressure pumps that can lift the safety relief valves. Therefore, it is important to keep PORVs unblocked at many PWRs. At some plants, chronic problems with PORV leakage leads to operation with the associated block valves closed. Opening these block valves requires operator action and is prevented by loss of power or hardware failures in certain scenarios. This reduces the availability of decay heat removal via feed and bleed cooling.

Some plants without PORVs have developed the capability to feed and bleed using other valves, such as low-temperature overpressure protection valves. The CDFs for these plants have been reduced as a result of this capability. In general, those plants with the most reliable feed and bleed capability have lower transient and small LOCA frequencies than those with less reliable feed and bleed capability.

NUREG-1150 Perspective: Switchover to recirculation is important to LOCA CDFs at PWRs

There is substantial variation among the PWR LOCA CDFs. A significant part of this variation is driven by plant-specific aspects of the switchover to recirculation in the later stages of a LOCA. For example, NUREG-1150 found higher LOCA CDFs for Sequoyah than the other plants due to three factors: (1) a low-containment spray setpoint that resulted in early spray actuation for small LOCAs (the sprays take suction from the refueling water storage tank, which is also supplying makeup to the reactor coolant system), (2) a relatively small refueling water storage tank, and (3) manual switchover to recirculation. In the IPE submittals, switchover to recirculation is often a dominant contributor for those plants without automatic switchover. Those plants with automatic switchover tend to have lower LOCA CDFs than the other plants. The specific contribution of switchover failures varies from plant to plant, depending on the degree of automation and the complexity of the switchover operation, including the piggybacking of high-pressure systems onto low-pressure systems.

18.5 Accident Progression Perspectives

A comparison of the NUREG-1150 accident progression perspectives to those identified in the IPE submittals is presented in this section. A similar approach to that used for CDF perspective comparison is adopted in this section. First the NUREG-1150 perspective is summarized and then it is determined whether or not the IPE results support the finding.

NUREG-1150 Perspective: Large dry and subatmospheric containments are highly likely to maintain integrity during a severe accident

The NUREG-1150 results for the Zion and Surry Plants indicate that large dry and subatmospheric containment designs appear to be quite robust in their ability to contain severe accident loads. This study shows a high-likelihood of maintaining integrity throughout the early phases of severe accidents in which the potential for large release of radionuclides is greatest. The predicted likelihood of early containment failure in the Zion (large volume containment) plant and the Surry (subatmospheric containment) plant in NUREG-1150 is quite small (mean value of about 0.01). The principal mechanisms leading to these failures are loads resulting from high-pressure melt ejection in accident sequences with high-reactor coolant system (RCS) pressures (at time of vessel breach) and in-vessel steam explosions in sequences with low RCS pressure at vessel breach. However, the uncertainties in describing the magnitude of severe accident loads at vessel breach for pressurized scenarios and the likelihood of depressurization prior to lower head failure are large. The principal reason that the probability of early containment failure from loads at vessel breach is so small in the NUREG-1150 Surry and Zion analyses is that the reactor coolant system is not likely to be at high-pressure when vessel melt-through occurs. Some of the mechanisms that were found to be effective in depressurizing the vessel are hot leg or surge line failure at elevated temperature, failure of a reactor coolant pump seal, or a stuck-open relief valve.

Generally, the IPE results for large dry and subatmospheric containments indicate probabilities of early containment failure that are higher than those calculated in NUREG-1150. The IPE average early failure probabilities are 0.05 for large dry containments and 0.1 for subatmospheric containments with some individual plant results above 0.3. The higher probabilities of early containment failure for the IPEs are caused in some cases by plant-specific features and in other cases by modeling assumptions. For example, the highest early failure probability (0.3) in Figure 18.6 is for the Palisades plant, reflecting a containment failure mode that is unique to Palisades. The postulated failure mode assumes that molten core debris from the reactor cavity flows into the sump and subsequently into recirculation piping. The debris is assumed to melt through the pipe wall and enter the auxiliary building. This failure mode was not identified in other PWRs with large volume or subatmospheric containments.

NUREG-1150 Perspective: The likelihood of early containment failure is higher for ice condenser designs than for large dry and subatmospheric designs

The NUREG-1150 results for the Sequoyah plant indicate that the likelihood of early failure during a severe accident for the Sequoyah plant is higher (0.07) than for the large dry and subatmospheric designs, but is less than for the BWRs analyzed. Early failure is primarily associated with loads introduced at the time of vessel breach. Containment rupture from high-overpressure loads at the time of vessel breach is likely to result in significant damage to the containment wall and subsequent bypass of the ice bed. The IPE results indicate that in general ice condenser containments have lower probabilities for early failure than large dry and subatmospheric designs. The average of the IPE early failure probabilities is 0.02 for ice condenser containments compared with 0.05 for large-dry and subatmospheric designs. Although the differences in the average values for the various designs are less than a factor of 5, the IPE results for ice condenser containments are lower than might have been expected based on NUREG-1150. All of the failures modes (direct containment heating, hydrogen combustion, in-vessel steam explosions, and direct contract of the core debris with the container wall) found important in the NUREG-1150 Sequoyah analysis were considered in the IPEs but judged to have lower probabilities. The primary reason for these lower probabilities was a more optimistic set of modeling assumptions, although accident sequence and plant-specific features (flooded cavity, ice remaining, etc.) also influenced the results.

NUREG-1150 Perspective: There is a substantial likelihood for 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 found to be the dominant failure mechanism in the NUREG-1150 Peach Bottom (Mark I containment) analysis. However, at the time Generic Letter 88-20 was issued, there was considerable uncertainty regarding the likelihood of failure of the drywell as a result of this mechanism. The utilities were therefore given the option of not addressing it in their IPEs. However, most utilities did include consideration of shell melt-through. For those utilities that did consider this failure mechanism, a significant potential was found for early drywell failure (as shown in Figure 18.7). In those submittals that did not consider shell melt-through, the potential for early containment failure was generally found to be quite low. In some IPEs, this failure mechanism was eliminated because of plant-specific features (such as large sumps or the presence of curbs).

NUREG-1150 Perspective: Venting can eliminate some sequences that would otherwise result in gradual overpressure failure of Mark I containments

The principal benefit of wetwell venting indicated by the NUREG-1150 Peach Bottom analysis is in the reduction of the core damage frequency. In many BWR IPEs the CDF was also reduced by taking credit for containment venting. Although NUREG-1150 found that venting is not effective in eliminating some early drywell failure mechanisms, venting could eliminate other sequences that would otherwise have resulted in gradual overpressure failure of the containment. Therefore, venting was found to lower the likelihood of late containment failure in NUREG-1150. In general, most IPE results for Mark I containments also include late containment venting as a way of preventing late containment failure. However, in at least one submittal, early drywell venting is a dominant venting mode. In such plants early drywell venting accompanies a containment flooding procedure.

NUREG-1150 Perspective: Hydrogen deflagration is the principal mechanism for early containment failure in BWR Mark III containments

In NUREG-1150 the Grand Gulf containment was predicted to fail at or before vessel breach in a substantial fraction (0.4) of severe accident sequences. Hydrogen deflagration was found to be the principal mechanism for early containment failure in NUREG-1150. The IPE results indicate that energetic events at the time of vessel breach, including hydrogen combustion, are the principal causes of early containment failure. However, the conditional probabilities of early failure in the IPEs (ranging from less than 0.01 to approximately 0.3) were less than the NUREG-1150 value (0.4). The IPEs also found energetic events at the time the core debris penetrates the reactor vessel as important contributors to the probabilities of early failure. However, these events were again judged to have lower probabilities in the IPEs than in NUREG-1150. The differences seem to be largely driven by modeling assumptions (magnitude of pressure loading, failure pressure of containment, etc.) and by the high-reliability associated with the hydrogen ignition systems in the IPEs.

NUREG-1150 Perspective: Venting was not found effective in preventing containment failure for accident scenarios involving core damage in Mark III containments

In the NUREG-1150 Grand Gulf study accidents involving station blackout were found to dominate the core damage frequency. Containment venting was considered unlikely for station blackout accidentsand therefore it was not found to be particularly effective at Grand Gulf. However, in the Grand Gulf IPE venting of the primary system using the main steam isolation valves was a significant contributor to the probability of loss of early containment integrity. In other IPEs venting was also found to be a significant contributor to loss of containment integrity but late in an accident sequence.

NUREG-1150 Perspective: If core damage is arrested in-vessel, the likelihood of containment failure is small for all containment types

In a significant fraction of core damage scenarios, NUREG-1150 found the potential for the arrest of core degradation within the reactor vessel as the result of recovery procedures (such as in the Three Mile Island Unit 2 accident). The likelihood of containment failure is very small in these scenarios. The potential for arrest of in-vessel core degradation was not considered for all of the IPE submittals. However, for those IPEs that did consider this effect, the impact on the accident progression results was significant. The IPE results also indicate that if the core is retained in the reactor vessel, the likelihood of containment failure is very small.

NUREG-1150 Perspective: Containment bypass events represent a large fraction of high-consequence accidents for PWR containments

NUREG-1150 results indicate that containment bypass sequences (severe accidents initiated by steam generator tube ruptures, tube ruptures induced by hot circulating gases, or interfacing systems LOCAs) represent a substantial fraction of high-consequence accidents. The absolute frequency of these types of failure was, however, found to be small (about 5E-6/ry) in NUREG-1150. The IPE results also found that these types of events have a relatively low-frequency (the average of the PWR IPE results is very similar to the NUREG-1150 values). Bypass events were also found to be significant contributors to the frequency of early containment failure and bypass in the IPE submittals. In fact, some of the highest frequencies (1E-5/ry) in Figure 18.8 are dominated by bypass accident frequencies.

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 (also referred to as a Level 2 PRA)

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 (also referred to as a Level 1 PRA)

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 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

Common mode failure — a single failure that affects two or more components at the same time

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: $P(A \mid B) = P(A \cap B)/P(B)$

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

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 — minimum combination of a set of events (e.g., initiating event and component failures) that, if they occur, will result in the onset of core damage

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 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, causes, 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 demand or 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 — analysis based on probabilities, and mathematical manipulation of those probabilities, (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

Level of detail — different levels of logic modeling used in a PRA; (a failure event in a fault tree analysis can address various levels of detail, depending on how much useful information is available concerning the contributors to the failure event)

Low contributor — an accident class that has a minor impact (on the order of a few percent) on the total core damage frequency or a containment failure mechanism having a minor impact on the total radionuclide frequency

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)

Plant damage state — a set of accident sequences from the level 1 analysis grouped together because their characteristics relevant to the subsequent progression are similar. The Plant Damage States constitute the interface between the level 1 and level 2 analysis of a PRA.

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 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 equipment back to operable status

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)

Release class — a set of accident progression sequences grouped together because they lead to similar radionuclide releases and for which a single representative release calculation can be performed.

Release fraction — the fraction of the total inventory of a radionuclide in the reactor core at the start of the accident which is released to the environment.

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

Station Blackout Rule — Rule requering that nuclear power plant units perform an analysis to establish a method to cope with a station blackout for a specified duration without core damage core damage occuring (coping method)

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

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11. ABSTRACT (200 words or less)

This report provides perspectives gained by reviewing 75 Individual Plant Examination (IPE) submittals pertaining to 108 nuclear power plant units. IPEs are probabilistic analysis that estimate the core damage frequency (CDF) and containment performance for accidents initiated by internal events (including internal flooding, but excluding internal fire). The IPE submittals were reviewed to gain perspectives in three major areas: (1) improvements made to individual plants as a result of their IPEs and the collective result 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) the quality of the IPEs with respect to their potential role in risk-informed regulation. These perspectives were 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. These results were assessed in relation to the design and operational characteristics of the various reactor and containment types, and by comparing the IPEs to attributes of a quality probabilistic risk assessment. Methods data, boundary conditions, and assumptions used in the IPEs were considered in understanding the differences and similarities observed among the various types of plants.

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