NUREG-1765

Basis Document for Large Early Release Frequency (LERF) Significance Determination Process (SDP)

Inspection Findings that May Affect LERF

U.S. Nuclear Regulatory Commission
Office of Nuclear RegulatoryResearch
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Inspection Findings that May Affect LERF

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ABSTRACT

A significance determination process (SDP) is proposed that assigns risk characterization to inspection findings based on large early release frequency (LERF) considerations. This process is designed to interface directly with the SDP that is based on findings related to those structures, systems, and components (SSCs) that can influence the core damage frequency (CDF). The proposed LERF-based SDP will capture findings for those SSCs that can influence CDF determinations but which can also influence LERF. In addition, the proposed LERF-based SDP approach will address findings related to SSCs that do not influence CDF determinations but which can impact the containment function.

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SECY-99-007a discusses the need for a method of assigning a risk characterization to inspection findings. This risk characterization is necessary so that inspection findings can be correlated with risk-informed plant performance indicators (PIs) during the plant performance assessment process. An attachment to SECY-99-007a describes in detail the staff's efforts to characterize the risk inspection findings that potentially impact at-power operations when the findings involve the initiating event, mitigating system, or barrier cornerstones for the reactor safety strategic performance area. This significance determination process (SDP), discussed in SECY-99-007a, focuses on risk-significant issues that could influence the determination of the change in core damage frequency (ΔCDF) at a nuclear power plant. In the context of the SDP, risk significance is based on the ΔCDF acceptance guidelines in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

A performance issue that leads to an increase in core damage frequency (ΔCDF) larger than 10^{-4} per reactor-year (/ry) is risk-significant and is assigned to the highest risk category (red) in Table 1. Lower frequency ranges have lower risk significance categories in decrements of one order of magnitude (yellow, white, and green).

Table 1 Risk Significance Based on ΔLERF vs. ΔCDF

Frequency Range/ry	SDP Based on ΔCDF	SDP Based on ΔLERF
$\geq 10^{-4}$	red	red
$< 10^{-4} - 10^{-5}$	yellow	red
$< 10^{-5} - 10^{-6}$	white	yellow
$< 10^{-6} - 10^{-7}$	green	white
$<10^{-7}$	green	green

Only core damage (CD) accidents that can lead to large, unmitigated releases from containment before effective evacuation of the nearby population have the potential to cause prompt fatalities. Such accidents generally include unscrubbed releases associated with early containment failure at or shortly after vessel breach, containment bypass events, and loss of containment isolation. The frequency of all accidents of this type is called the large early release frequency (LERF) in Regulatory Guide 1.174. Using this metric leads to the LERF-based risk-significant characterizations in Table 1. It is clear from the risk characterizations in Table 1 that the LERF-based approach is one order of magnitude more stringent than the approach based on core damage frequency (CDF). Therefore, in some circumstance it may be necessary to characterize the risk significance of an inspection finding using the LERF-based approach. It is the purpose of this report to provide the basis for deciding when the LERF-based SDP should be used.

Scope and Limitations

The focus of the LERF-based SDP is on internal events at full power. Issues associated with shutdown risk, emergency preparedness, radiation safety, and safeguards are not addressed.

The approach has a number of built-in assumptions and limitations:

(1) Since this SDP is focused on LERF, i.e., a release large enough and early enough to predict an prompt (or early) fatality, long-term risk measures such as population dose (person-rem) and latent cancer fatalities are not addressed in this report. In addition, slowly developing accident sequences that involve failure of containment heat removal and ultimately progress to containment failure, e.g., TW sequences in boiling water reactors (BWRs), are assumed not to contribute to LERF. It is assumed that effective emergency response actions can be taken for these accident sequences.

(2) LERF determinations depend on the containment design. The attributes and features of containments vary considerably.

(3) LERF determinations are also partly based on published data and analyses of the likelihood of early containment failure during core meltdown accidents. The data is limited and the analyses are subject to uncertainty.

(4) It was conservatively assumed for all PWR interfacing system loss-of-coolant-accidents (ISLOCAs) that the path outside containment is not submerged (i.e., the release is not scrubbed).

(5) It was conservatively assumed for all steam generator tube ruptures (SGTRs) that the secondary side is open so that a path outside containment exists and the release is not scrubbed.

(6) For those findings that impact the containment function, baseline CDFs were assumed in order to simplify the calculation of the change in risk. The baseline CDFs assumed were 10^{-4}/ry for pressurized water reactors (PWRs) and 10^{-5}/ry for BWRs.

(7) The SDP provides guidance on individual findings; combinations or groups of findings are not addressed by the SDP and should be evaluated through an assessment.

(8) It was assumed, conservatively, that a main steam isolation valve (MSIV) leakage rate in excess of 10,000 standard cubic feet per hour (scfh) in BWRs with Mark I and Mark II containments is significant to LERF. Leakage past the MSIVs in a Mark III would be stopped at the safety-grade main steam shutoff valve (MSSV).

Given the above assumptions and limitations and the generic nature of the LERF-based SDP, a utility can use plant-specific arguments to claim that the risk significance of a particular finding at their nuclear power plant is too high. Such claims should be evaluated on a case-by-case basis.

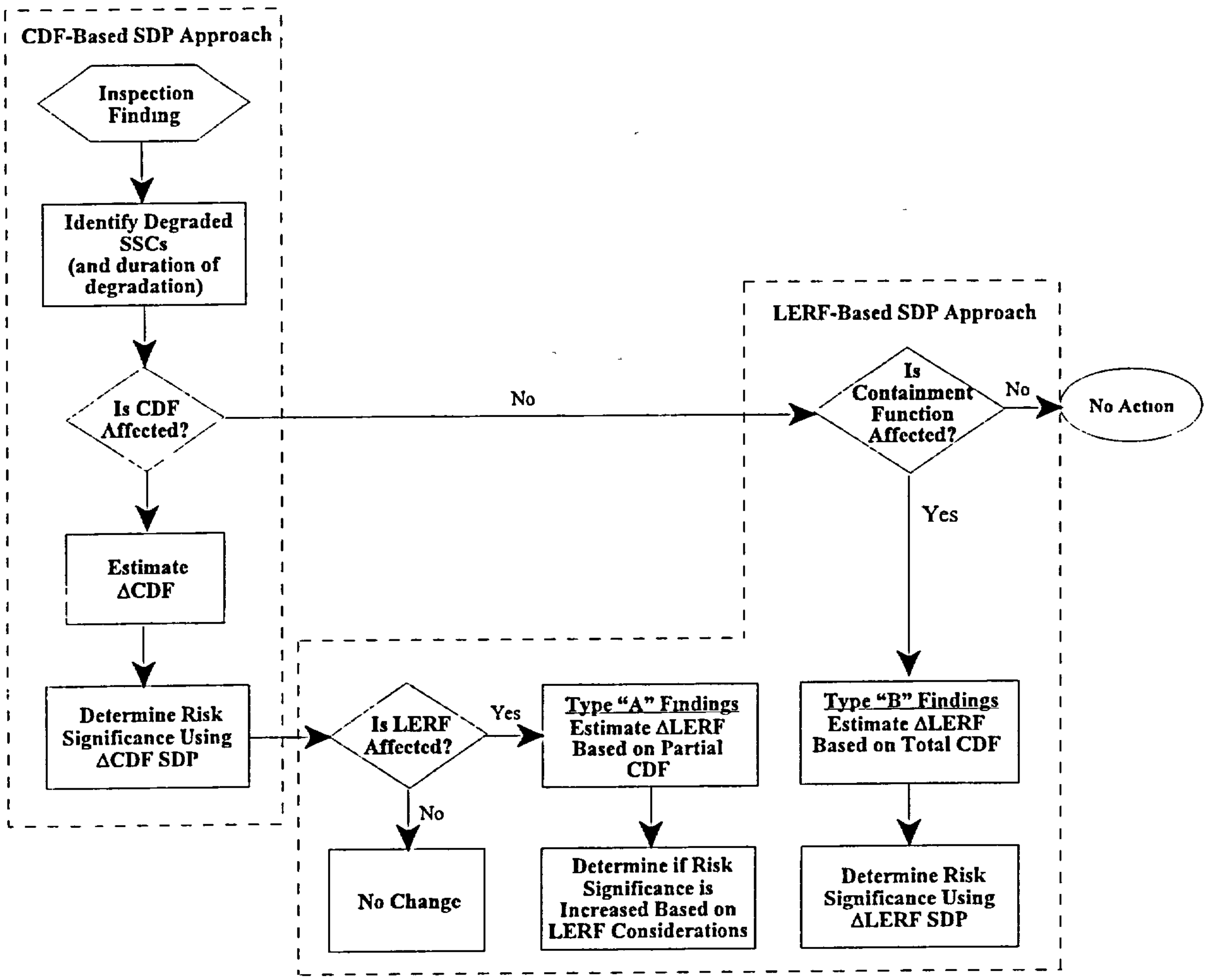

Figure 1 LERF-Based Significance Determination Process

Approach

Figure 1 describes the process flow of typical inspection findings or issues. The process is designed to interface closely with the existing CDF-based SDP. An inspection finding, therefore, will identify the degraded system, structure, or component (SSC) and determine the impact on initiating event, mitigating system, or barrier cornerstones. If the degraded condition is found to influence the likelihood of accidents leading to core damage, then the risk significance of the finding should be determined using the CDF-based SDP. The process assigns the finding to a risk significance category corresponding to one of the colors in the ΔCDF column in Table 1. If the finding does not influence ΔLERF, then the risk category remains the same and the SDP is complete. However, findings that have an impact on scenarios that contribute to LERF, identified in this report as Type A findings, need to be assessed with respect to LERF criteria as discussed below.

It is possible for a finding to be unrelated to those SSCs that are needed to prevent accidents from leading to core damage but to have potentially important implications for the integrity of the containment. Findings of this type have no impact on the determination of the ΔCDF and therefore are not put through the CDF-based SDP. These findings, however, are potentially important to ΔLERF determinations and have to be assigned an appropriate risk category. Findings of this nature are classified as Type B in Figure 1.

Type B findings are therefore fundamentally different from Type A findings. Type A findings are processed through the CDF-based SDP and assigned a significance category, which may be adjusted based on LERF considerations. Type B findings do not undergo the CDF-based SDP and are assigned significance categories based only on LERF considerations

Type A Findings

Some findings that pass through the CDF based SDP affect LERF sequences. A subset of these findings (identified in Table 2) should be examined through the LERF-based SDP. Table 2 is designed to interface with the accident categories (i.e., transients, loss-of-coolant accidents (LOCAs), anticipated transients without scram (ATWS), SGTR, and ISLOCAs) used in the CDF-based SDP. Each finding that affects LERF sequences processed through the CDF-based SDP should therefore be checked against Table 2. If a finding is found to be related to any of the accident sequences or characteristics in Table 2 then the staff should consider changing the risk significance category based on LERF.

The "Factor" in Table 2, relates the frequency range for a particular set of core damage (CD) accidents to the LERF:

$$\Delta\text{LERF} = \text{Factor x } (\Delta\text{CDF affecting LERF sequences})$$

A Factor of 1.0 implies that ΔLERF is equivalent to the ΔCDF for those sequences that affect LERF. In these circumstances the risk significance based on LERF is higher than the CDF-based risk category (refer to Table 1). Therefore the risk significance category should be increased by one order of magnitude for findings of this type. Only a few accident sequences (e.g., SGTR and ISLOCA) have been identified where the Factor = 1.0 (refer to Table 2). For these accidents the containment is completely bypassed and the release is assumed to be unscrubbed.

Table 2 Type A Findings			
LERF Significant Sequences	Containment Type	Factor	Comments
ISLOCA	All PWR Containments	1.0	ΔLERF is equivalent to ISLOCA ΔCDF; therefore, the risk significance (i.e., color assignment) of the finding based on ΔCDF should be increased by one order of magnitude.
ATWS	BWR Mark I	0.3	Candidate for increasing risk significance.
	BWR Mark II	0.4	Candidate for increasing risk significance.
All transients and small-break LOCAs involving high reactor coolant system (RCS) pressure	BWR Mark I[1]	1.0	ΔLERF is equivalent to ΔCDF (high-pressure sequences); therefore, the risk significance (color) of the finding should be increased by one order of magnitude.
All transients and small-break LOCAs involving high RCS pressure and flooded drywell floor at vessel breach (VB)	BWR Mark I[1]	0.6	Candidate for increasing risk significance.
All transients and small-break LOCAs involving high RCS pressure	BWR Mark II[2]	0.3	Candidate for increasing risk significance.
	BWR Mark III[3]	0.2	
Station blackout (SBO) sequences	PWR Ice Condenser	1.0	ΔLERF is equivalent to ΔCDF (SBO).
	BWR Mark III	0.2	Candidate for increasing risk significance.
All transients involving low RCS pressure and dry drywell floor at vessel breach	BWR Mark I[1]	> 0.1	Candidate for increasing risk significance.
	BWR Mark II[1]	> 0.1	Candidate for increasing risk significance.
SGTR	All PWR Containments	1.0	ΔLERF is equivalent to ΔCDF (SGTR); therefore, the risk significance (color) of the finding should be increased by one order of magnitude.

1. Of interest is the increase in the frequency of (1) high-pressure sequences and (2) low-pressure sequences with a dry drywell floor at vessel breach. For the high-pressure sequences that have a flooded drywell floor, the Factor is 0.6 so these sequences are candidates for increasing the risk significance because of LERF considerations.
2. The Mark II containment does not have the liner melt-through issue but high-pressure sequences are predicted to fail containment with a relatively low conditional probability.
3. The Mark III containment is predicted to fail with a relatively high probability during high pressure and SBO core melt sequences, but the suppression pool is expected to remain intact. Thus, the release is scrubbed and the LERF determination is relatively low. As shown in Section 5.1.3, the factor applies to all transients with the RCS at high pressure and to all SBO sequences regardless of whether the RCS is at high or low pressure.

As the Factor decreases, the influence of the accident sequence on the determination of ΔLERF decreases correspondingly. A Factor of 0.1 the implies that the ΔLERF range is one order of magnitude lower than the ΔCDF range. This means that the risk significance is the same for the LERF-based approach as for the CDF-based approach (refer to Table 1). Therefore for Factors equal to or less than 0.1, the risk category obtained using the CDF based SDP is appropriate and should be left unchanged.

When the Factor is between 0.1 and 1.0, judgment is needed to determine if the risk category obtained from the CDF-based SDP needs to be changed. Any decision to change the significance category should consider the limitations and assumptions implicit in the numerical values of the Factors in Table 2 (refer to the previous section).

Type A findings for sequences and containments not listed are not expected to significantly impact LERF.

Type B Findings

Findings that have no impact on the determination of the ΔCDF but are potentially important to ΔLERF determinations are classified as Type B findings.

All of the SSCs associated with maintaining containment integrity were reviewed to determine if their degradation would affect ΔLERF. Only the containment-related SSCs in Table 3 were found to potentially influence ΔLERF. If a finding reveals that the function of any of the SSCs in Table 3 would have been unavailable in the event of core damage, its significance category can be determined from the duration of the degraded condition.

Since the containment function may be compromised in a Type B finding, a Type B finding can potentially affect either all CD accidents or a subset of CD accidents that impact the feature that is compromised. Baseline CDFs were assumed in order to simplify the calculation of the change in risk for this type of finding. The baseline CDFs assumed were 10^{-4}/ry for PWRs and 10^{-5}/ry for BWRs. The assumption of baseline CDFs in the initial screening is a necessary limitation in the light of the relatively wide ranges associated with CDF estimates. The plant-specific CDF should be considered when making the final determination. The risk significance categories in Table 3 were obtained using the following relationship:

$$\Delta\text{LERF} = \text{Factor x Fraction of (relevant) CDF x (multiplier for the duration of degraded condition)}$$

In the above relationship the duration of the degraded condition is a simple multiplier for three periods:

Duration	Multiplier
>30 days	1.0
30–3 days	0.1
<3 days	0.01

Table 3 Type B Findings				
SSC Affected by Findings	Reactor Containment Type	Duration of Condition		
		>30 days	30–3 days	<3 days
Containment Penetration Seals, Isolation Valves, and Purge and Vent lines	BWR Mark I and II[1]	yellow	white	green
	BWR Mark III[1]	white	green	green
	PWR Large Dry[1]	red	yellow	white
	PWR Ice Condenser[1]	red	yellow	white
Suppression Pool Bypass	All BWR Containments[2]	yellow	white	green
MSIV Leakage[3]	BWR Mark I and II[4]	yellow	white	green
Ice Condenser Integrity - Partial Failure of Doors	PWR Ice Condenser	red	yellow	white
Hydrogen Igniters	PWR Ice Condenser	red	yellow	white
	BWR Mark III	white	green	green
Containment Spray	BWR Mark I[5] (Drywell)	yellow	white	green
	BWR Mark II (Drywell)	white	green	green
	BWR Mark III (Wetwell)	white	green	green

1. Leakage from the drywell (containment) to the environment is >200 x L_a for BWRs with Mark I and II containments, >500 x L_a for BWRs with Mark III containments, and >1000 x L_a for PWRs. Leakage from the wetwell to the environment is not a LERF concern.
2. The only sequences of interest are high-pressure sequences. This is because low-pressure sequences would be less energetic, thus resulting in slower transport of fission products which provides more opportunity for removing the fission products before they enter the environment. The release would therefore not be "large" and potentially not "early."
3. Excessive leakage that can impact LERF is defined as a leakage rate >10,000 scfh passed through both the inboard *and* the associated outboard MSIV (PRAB-02-01). An inability to quantify the leakage rate leads to a similar finding.
4. MSIV leakage is only applicable to BWRs with Mark I and II containments. BWRs with Mark III containments have a safety-grade main steam shutoff valve (MSSV). The MSSV is a relatively slow-closing, low-leakage valve. Thus, any leakage past the MSIV in a Mark III plant is stopped at the MSSV.
5. The probability of early containment failure from liner melt-through in a Mark I is negligible for accidents with the RCS at low pressure because it is assumed that the drywell floor is flooded. If a finding implies that the drywell floor will be dry, the risk significance (i.e., color) of the finding should be increased by one order of magnitude for high-pressure sequences.

For Type B findings the Factor is a multiplier either on the total CDF or on the fraction of CDF that is relevant to the containment function that is compromised. It is the difference between assuming the complete failure of the SSC and the conditional failure probability of the SSC assumed in the baseline risk estimate multiplied by the fraction of CDF that pertains to the phenomena that impact (or are impacted by) the failure of the SSC under consideration. For example, suppose a PWR with an ice condenser containment is found to have a significant number of inoperable hydrogen igniters. Such a finding implies that the containment is vulnerable to failure from a hydrogen deflagration or detonation in a core damage accident. However, the igniters require AC power to operate and would therefore be unavailable anyway in a station blackout accident. Hence, the risk significance of the finding pertains only to the non-SBO portion of the total CDF. In making a final determination, the staff should consider the plant-specific CDF for the major accident classes and estimate the multiplier more precisely (if the actual duration can be established).

If the Factor is 1.0 and the duration of the degraded condition is >30 days, the implication is that ΔLERF is equivalent to either the total CDF or the fraction of CDF that is relevant to the containment function that is compromised. As the product of the Factor and the relevant fraction of CDF decreases, the influence of the containment SSC on the determination of LERF decreases correspondingly. When this product reaches 0.1, the significance category drops one order of magnitude.

The bases for the significance categories in Table 3 are provided in the body of this report. In general, the data used to address containment performance for Type A findings is derived from the NUREG-1150 studies and supplemented by studies on selected severe accident issues. In contrast, the data used to establish the risk significance of Type B findings is based mainly on the results of the Individual Plant Examination (IPE) program reported in NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," but also includes information from NUREG-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," NUREG/CR-6025, "The Probability of Mark I Containment Failure by Melt Attack of the Liner," and NUREG/CR-4832, "Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (PMIEP)," and the experience of using a draft version of this report. Containment-related findings that are not addressed in this LERF SDP are not expected to be risk-significant with respect to LERF.

FOREWORD

This report provides a composite perspective of those structures, systems, and components (SSCs) whose failure could represent a reasonable likelihood of releasing to the environment a sufficiently large quantity of fission products in an early enough time frame to have the potential to affect a prompt (or early) fatality (LERF). These SSCs are related to containment, containment integrity, and features related to fission product retention or removal. In probabilistic risk assessments (PRAs), this is also referred to as Level II. The conclusions presented in this report have taken into consideration analyses and reports from WASH-1400, source term code package analyses, NUREG-1150, to the Individual Plant Examinations (IPEs), and assessments of current research. Information that was generated over the last 26 years, differences in analysis techniques, level of detail, and opinions have lead to different conclusions in different reports. This report does not represent all points of view, but attempts to present an understanding of the various SSCs, their role in nuclear safety, and the physical processes to produce a reasonable assessment of importance of the SSC to LERF considerations.

This report is a snapshot in time. As our understanding improves from additional research and as our experience with evaluating SSCs in the plant increases, revisions to this report may be appropriate. The approach taken in this report is to be conservative, which will sometimes result in additional analyses. These additional analyses will produce a better understanding, which can result in a change in SSC importance identified in this report.

This report relates only to the potential for a prompt fatality. It does not address other important consequence measures, i.e., latent fatalities or the effects on land or population due to evacuation or relocation. To properly address the risk importance of an SSC, all consequence metrics should be considered. This report does not address the other elements of integrated risk-informed decisionmaking such as defense-in-depth, margins, or uncertainty.

By using the information in this report, NRC staff, including resident inspectors, will be able to be more effective and efficient by focusing resources on risk important containment findings. This is also supports the agency's performance goal to reduce unnecessary burden on stakeholders by de-emphasizing activities in areas of low risk importance. A draft version of this report was incorporated into Inspection Manual Chapter 0609 as Appendix H. We have endeavored to incorporate the comments from the regions and NRR into this final version, which should be used in lieu of the draft version in Appendix H.

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ABBREVIATIONS

AC	alternating current
ANS	American Nuclear Society
AOP	abnormal operating procedures
ATWS	anticipated transient without scram
BNL	Brookhaven National Laboratory
BWR	boiling water reactor
BWROG	BWR Owners' Group
CCFP	conditional containment failure probability
CD	core damage
CDF	core damage frequency
CCI	core-concrete interaction
CFR	*Code of Federal Regulations*
CHRS	containment heat removal system
CPI	Containment Performance Improvement program
C_{PEF}	conditional probability of early containment failure given an accident
C_{PPB}	conditional probability of pool bypass given early containment failure.
CPS	Columbia Power Station
DBA	design basis accident
DCH	direct containment heating
ΔCDF	change in core damage frequency
ΔLERF	change in large early release frequency
EOP	emergency operating procedure
FCI	fuel-coolant interaction
HCOG	Hydrogen Control Owners Group
HPME	high-pressure melt ejection
IPE	individual plant examination
ISLOCA	interfacing system loss-of-coolant accident
L_a	maximum allowable containment leak rate as specified in the plant-specific technical specifications
LERF	large early release frequency
ISP	international standard problem
LOCA	loss-of-coolant accident
MSIV	main steam isolation valve
MSSV	main steam shutoff valve
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboartory
PDS	plant damage state
PI	performance indicator
PRA	probabilistic risk assessment
PWR	pressurized water reactor
RCS	reactor coolant system

RHR	residual heat removal
RWST	refueling water storage tank
ry	reactor-year
SAM	severe accident guidelines
SBO	station blackout
scfh	standard cubic feet per hour
SDP	significance determination process
SGTR	steam generator tube rupture
SGTS	standby gas treatment system
SNL	Sandia National Laboratories
SP	suppression pool
SRV	safety relief valve
SSC	structure, system, or component
WNP-2	Washington Nuclear Project No. 2
WOG	Westinghouse Owners Group

1 INTRODUCTION

1.1 Background

SECY-99-007a discusses the need for establishing a method of assigning a risk characterization to inspection findings. This risk characterization is necessary so that inspection findings can be correlated with risk-informed plant performance indicators (PIs) during the plant performance assessment process. An attachment to SECY-99-007a describes in detail the staff's efforts to characterize the risk inspection findings that potentially impact at-power operations when the findings involve the initiating event, mitigating system, or barrier cornerstones for the reactor safety strategic performance area. The significance determination process (SDP), discussed in the SECY-99-007a focuses on risk-significant issues that could influence the determination of the change in core damage frequency (ΔCDF) at a nuclear power plant. In the context of this SDP, risk significance is based on the CDF acceptance guidelines in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

A performance issue that leads to an increase in core damage frequency (ΔCDF) larger than 10^{-4}/ry is risk-significant and is assigned to the highest risk category (red) in Table 1.1. Lower frequency ranges have lower risk significance categories in decrements of one order of magnitude (yellow, white, and green).

Table 1.1 Risk Significance Based on ΔLERF vs. ΔCDF

Frequency Range/ry	SDP Based on ΔCDF	SDP Based on ΔLERF
$\geq 10^{-4}$	red	red
$< 10^{-4} - 10^{-5}$	yellow	red
$< 10^{-5} - 10^{-6}$	white	yellow
$< 10^{-6} - 10^{-7}$	green	white
$<10^{-7}$	green	green

Only core damage (CD) accidents that can lead to large, unmitigated releases from containment before effective evacuation of the nearby population have the potential to cause prompt fatalities. Such accidents generally include unscrubbed releases associated with early containment failure at or shortly after vessel breach, containment bypass events, and loss of containment isolation. The frequency of all accidents of this type is called the large early release frequency (LERF) and this was used as a surrogate for the prompt fatality quantitative health objective in Regulatory Guide 1.174. Using this metric leads to the LERF-based risk-significant characterizations in Table 1.1. It is clear from the risk characterizations in Table 1.1 that the LERF-based approach is one order of magnitude more stringent than the CDF-based approach. Therefore, in some circumstance it may be necessary to characterize the risk significance of an inspection finding using the LERF-based approach. It is the purpose of this report to provide the basis for deciding when the LERF-based SDP should be used.

1.2 Scope and Limitations

The focus of the LERF-based significance determination process (SDP) is on internal events at full power operation. Issues associated with shutdown risk, emergency preparedness, radiation safety, and safeguards are not addressed.

The approach has a number of built-in assumptions and limitations:

(1) Since this SDP is focused on LERF, i.e., a release large enough and early enough to predict an prompt (or early) fatality, long-term risk measures such as population dose (person-rem) and latent cancer fatalities are not addressed in this report. In addition, slowly developing accident sequences that involve failure of containment heat removal and ultimately progress to containment failure, e.g., TW sequences in BWRs, are assumed not to contribute to LERF. It is assumed that effective emergency response actions can be taken for these accident sequences.

(2) LERF determinations depend on the containment design. The attributes and features of contaiments vary considerably.

(3) LERF determinations are also partly based on published data and analyses of the likelihood of early containment failure during core meltdown accidents. The data is limited and the analyses are subject to uncertainty.

(4) It was conservatively assumed for all PWR interfacing system loss-of-coolant-accidents (ISLOCAs) that the path outside containment is not submerged (i.e., the release is not scrubbed).

(5) It was conservatively assumed for all SGTRs that the secondary side is open so that a path outside containment exists and the release is not scrubbed.

(6) For those findings that impact the containment function, baseline CDFs were assumed in order to simplify the calculation of the change in risk. The baseline CDFs assumed were 10^{-4}/ry for PWRs and 10^{-5}/ry for BWRs.

(7) The SDP provides guidance on individual findings; combinations or groups of findings are not addressed by the SDP and should be evaluated through an assessment.

(8) It was assumed, conservatively, that a main steam isolation valve (MSIV) leakage rate in excess of 10,000 scfh in BWRs with Mark I and Mark II containments is significant to LERF. Leakage past the MSIVs in a Mark III would be stopped at the safety-grade main steam shutoff valve (MSSV).

Given the above assumptions and limitations and the generic nature of the LERF-based SDP, a utility can use plant-specific arguments to claim that the risk significance of a particular finding at their nuclear power plant is too high. Such claims should be evaluated on a case-by-case basis.

1.3 Approach

Depending on the nature of the finding, the LERF determination process follows one of two paths. The first path is closely related to the existing CDF-based SDP. This process identifies findings that need to be examined using the LERF-based SDP. These findings are then be reevaluated to determine if their risk significance needs to be increased based on LERF considerations. The second LERF-based SDP path deals with findings that only affect the containment function. As the containment function may be compromised, a finding of this type can potentially affect all CD accidents. Generic baseline CDFs were assumed in order to simplify the calculation of the change in risk. The generic baseline CDFs assumed were 10^{-4}/ry for PWRs and 10^{-5}/ry for BWRs. The assumed generic baseline CDFs is a necessary limitation in the light of the relatively wide ranges associated with CDF estimates.

1.4 Organization of the Report

The LERF-based SDP approach is described in detail in Chapter 2. The approach is based on the results of extensive severe accident research over the last 26 years. The significance categories were determined by reference to the published data, including the results reported in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," the Containment Performance Improvement (CPI) program,[1] the Individual Plant Examination (IPE) program (NUREG-1560), and numerous other technical reports on experimental data and analyses related to various aspects of severe accident phenomena. The significance categories recommended in Chapter 2 depend on the characteristics of the various containment designs. The bases of the various categories are described separately for each containment design. Chapter 3 addresses BWRs with Mark I containments, Chapter 4 addresses BWR Mark II containments, Chapter 5 addresses BWR Mark III containments, Chapter 6 addresses PWR large-volume and subatmospheric containments,

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1 NUREG/CR-5423, "The Probability of Liner Failure in a Mark I Containment," NUREG/CR-5565, "The Response of BWR Mark II Containments to Station Blackout Severe Accident Sequences," NUREG/CR-5571, "The Response of BWR Mark III Containments to Short Term Station Blackout Severe Accident Sequences," NUREG/CR-5623, "BWR Mark II Ex-Vessel Corium Interaction Analyses."

and Chapter 7 addresses PWR ice condenser containments. This structure involves some repetition to make each chapter self-contained so that an inspector need not refer to other chapters.

2 LERF-BASED SIGNIFICANCE DETERMINATION PROCESS

2.1 Approach

Figure 2.1 shows the process flow of typical inspection findings or issues. The process is designed to interface closely with the existing CDF-based SDP. An inspection finding, therefore, will identify the degraded system, structure, or component (SSC) and assess the impact on initiating event, mitigating system, or barrier cornerstones. If the degraded condition is found to influence the likelihood of accidents leading to core damage, then the risk significance of the finding should be determined using the CDF-based SDP. The process assigns the finding to a risk significance category corresponding to one of the colors in the ΔCDF column in Table 1.1. If the finding does not influence ΔLERF, then the risk category remains the same and the SDP is complete. However, findings that have an impact on scenarios that contribute to LERF, identified in this report as Type A findings, need to be assessed with respect to LERF criteria as discussed below.

It is possible for a finding to be unrelated to those SSCs that are needed to prevent accidents from leading to core damage but to have potentially important implications for the integrity of the containment. Findings of this type have no impact on the determination of the ΔCDF and therefore are not put through the CDF-based SDP. These findings, however, are potentially important to ΔLERF determinations and have to be assigned an appropriate risk category. Findings of this nature are classified as Type B in Figure 2.1.

Type B findings are therefore fundamentally different from Type A findings. Type A findings are processed through the CDF-based SDP and assigned a significance category, which may be adjusted based on LERF considerations. Type B findings do not undergo the CDF-based SDP and are assigned significance categories based only on LERF considerations.

2.2 Type A Findings

A subset of the findings that pass through the CDF-based SDP has been identified (refer to Table 2.1) for further examination using LERF considerations. Table 2.1 is designed to interface with the accident categories (i.e. transients, LOCAs, ATWS, SGTR, and ISLOCAs) used in the CDF-based SDP. Each finding processed through the CDF-based SDP should therefore be compared with the attributes in Table 2.1. If a finding is found to be related to any of the accident sequences or characteristics in this table, then consideration needs to be given to changing the risk significance category based on LERF. Findings of this nature are classified as Type A in Figure 2.1.

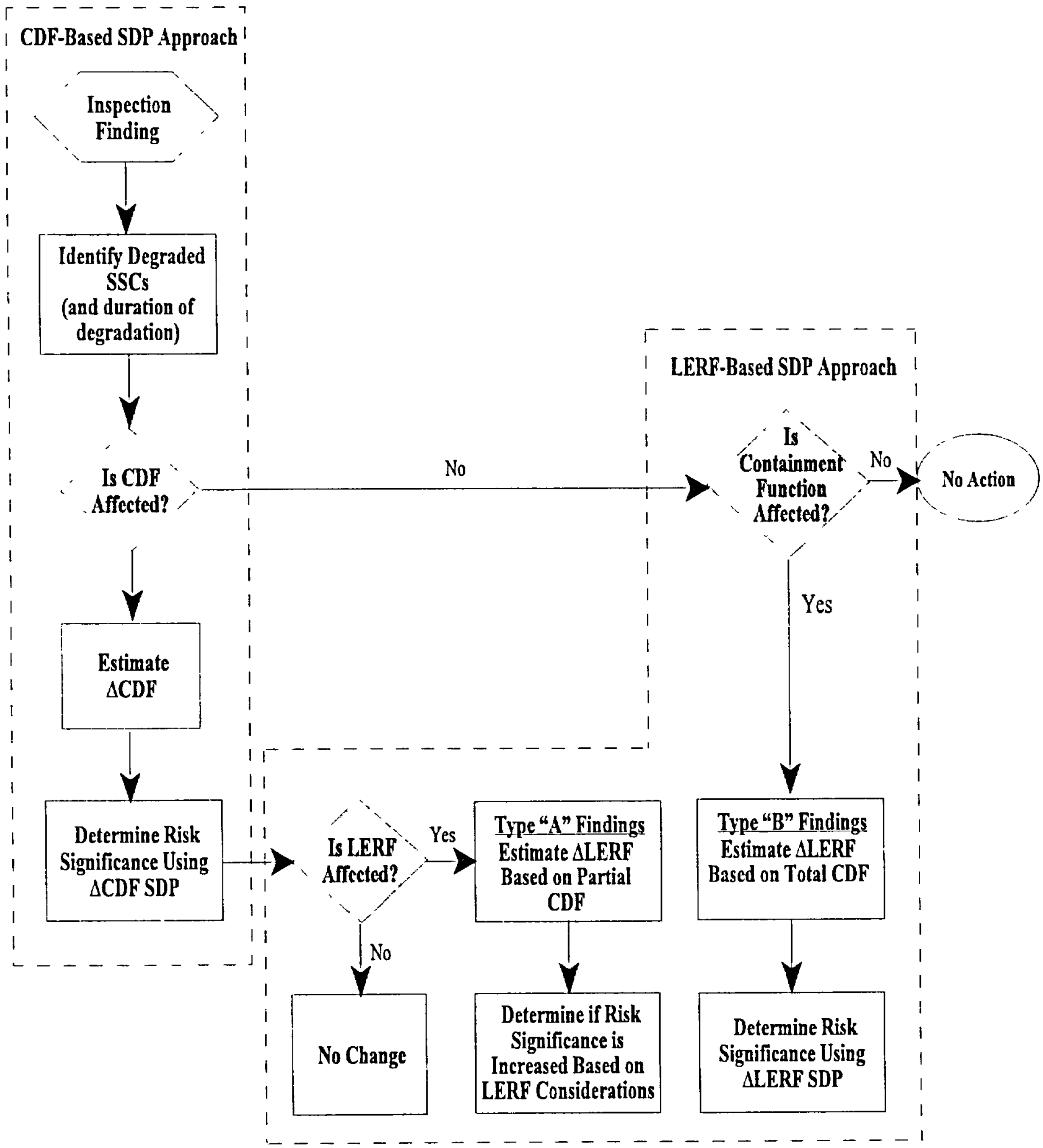

Figure 2.1 LERF-Based Significance Determination Process

Table 2.1 Type A Findings

LERF Significant Sequences	Containment Type	Factor	Comments
ISLOCA	All PWR Containments	1.0	ΔLERF is equivalent to ISLOCA ΔCDF; therefore, the risk significance (color) of the finding based on ΔCDF should be increased by one order of magnitude.
ATWS	BWR Mark I	0.3	Candidate for increasing risk significance.
	BWR Mark II	0.4	Candidate for increasing risk significance.
All transients and small-break LOCAs involving high reactor coolant system (RCS) pressure	BWR Mark I[1]	1.0	ΔLERF is equivalent to ΔCDF (high-pressure sequences); therefore, the risk significance (color) of the finding should be increased by one order of magnitude.
All transients and small-break LOCAs involving high RCS pressure and flooded drywell floor at vessel breach (VB)	BWR Mark I[1]	0.6	Candidate for increasing risk significance.
All transients and small-break LOCAs involving high RCS pressure	BWR Mark II[2]	0.3	Candidate for increasing risk significance.
	BWR Mark III[3]	0.2	
Station blackout (SBO) sequences	PWR Ice Condenser	1.0	ΔLERF is equivalent to ΔCDF (SBO).
	BWR Mark III	0.2	Candidate for increasing risk significance.
All transients involving low RCS pressure and dry drywell floor at vessel breach	BWR Mark I[1]	> 0.1	Candidate for increasing risk significance.
	BWR Mark II[1]	> 0.1	Candidate for increasing risk significance.
SGTR	All PWR Containments	1.0	ΔLERF is equivalent to ΔCDF (SGTR); therefore, the risk significance (color) of the finding should be increased by one order of magnitude.

1. Of interest is the increase in the frequency of (1) high-pressure sequences and (2) low-pressure sequences with a dry drywell floor at vessel breach. For the high-pressure sequences that have a flooded drywell floor, the Factor is 0.6 so these sequences are candidates for increasing the risk significance because of LERF considerations.
2. The Mark II containment does not have the liner melt-through issue but high-pressure sequences are predicted to fail containment with a relatively low conditional probability.
3. The Mark III containment is predicted to fail with a relatively high probability during high-pressure and SBO core melt sequences, but the suppression pool is expected to remain intact. Thus, the release is scrubbed and the LERF determination is relatively low. As shown in Section 5.1.3, the factor applies to all transients with the RCS at high pressure and to all SBO sequences regardless of whether the RCS is at high or low pressure.

The "Factor" in Table 2.1 relates the frequency range for a particular set of core damage (CD) accidents to the LERF:

$$\Delta\text{LERF} = \text{Factor x } (\Delta\text{CDF affecting LERF sequences})$$

A Factor of 1.0 implies that ΔLERF is equivalent to the ΔCDF for those sequences that affect LERF. In these circumstances the risk significance based on LERF is higher than the CDF-based risk category (refer to Table 1.1). Therefore, the risk significance category should be increased by one order of magnitude for findings of this type. Only a few accident sequences (e.g., SGTR and ISLOCA) have been identified where the Factor = 1.0 (refer to Table 2.1). For these accidents the containment is completely bypassed and the release is assumed to be unscrubbed.

As the Factor decreases, the influence of the accident sequence on the determination of ΔLERF decreases correspondingly. A Factor of 0.1 implies that the ΔLERF range is one order of magnitude lower than the ΔCDF range. This means that the risk significance is the same for the LERF-based approach as for the CDF-based approach (refer to Table 1.1). Therefore, for Factors equal to or less than 0.1, the risk category obtained using the CDF-based SDP is appropriate and should be left unchanged.

For situations where the Factor is between 0.1 and 1.0 judgment is needed to determine if the risk category obtained from the CDF-based SDP needs to be changed. Any decision to change the significance category should take into consideration the limitations and assumptions (refer to Section 1.2) implicit in the numerical value of the Factors selected for Table 2.1.

The bases for the Factors in Table 2.1 are discussed by containment type. Chapter 3 addresses BWRs with Mark I containments, Chapter 4 addresses BWR Mark II containments, Chapter 5 addresses BWR Mark III containments, Chapter 6 addresses PWR large-volume and subatmospheric containments, and Chapter 7 addresses PWR ice condenser containments.

2.3 Type B Findings

Findings that have no impact on the determination of the ΔCDF but are potentially important to ΔLERF determinations are classified as Type B findings. The following SSCs (associated with maintaining containment integrity) were reviewed to determine if their degradation would affect ΔLERF:

1. Containment penetration seals,
2. Containment isolation valves (including main steam isolation valves),
3. Containment vent and purge systems (including lines, valves, pumps, and filters and excluding filters on the control room vents),
4. Containment sprays,
5. Containment flooding system(s),
6. Hydrogen igniters (for hydrogen control),
7. Suppression pool (SP) systems important to SP integrity (e.g., vacuum breakers),

8. Suppression pool cooling (an operating mode of the residual heat removal (RHR) system),
9. Fan coolers,
10. Ice condenser baskets, and
11. Ice condenser doors.

The characteristics of the severe accidents that contribute to LERF and those SSCs important to maintaining containment integrity are plant specific and depend upon the containment design (i.e., large volume as compared to the various pressure suppression designs). These issues will, therefore, be discussed in the context of six containment designs: BWRs with Mark I, II, or III containments; PWRs with large dry and subatmospheric containments; and PWRs with ice condenser containments. Only those containment-related SSCs included in Table 2.2 were found to potentially influence ΔLERF. If a finding reveals that the function of any of the SSCs in Table 2.2 would have been unavailable or degraded in the event of core damage, its significance category can be determined from the duration of the degraded condition.

Since the containment function may be compromised in a Type B finding, a Type B finding can potentially affect either all CD accidents or a subset of CD accidents that impact the feature that is compromised. Baseline CDFs were assumed in order to simplify the calculation of the change in risk for this type of finding. The baseline CDFs assumed were 10^{-4}/ry for PWRs and 10^{-5}/ry for BWRs. The assumption of baseline CDFs in the initial screening is a necessary limitation in the light of the relatively wide ranges associated with CDF estimates. The plant-specific CDF should be considered when making the final determination. The risk significance categories in Table 2.2 were obtained using the following relationship:

$$\Delta \text{LERF} = \text{Factor x Fraction of (relevant) CDF x (multiplier for the duration of degraded condition)}$$

In the above relationship the duration of the degraded condition is a simple multiplier for one of three periods:

Duration	Multiplier
>30 days	1.0
30–3 days	0.1
<3 days	0.01

For Type B findings the Factor is a multiplier either on the total CDF or on the fraction of CDF that is relevant to the containment function that is compromised. It is the difference between assuming complete failure of the SSC and the conditional failure probability assumed in the baseline risk estimate multiplied by the fraction of CDF that pertains to the phenomena that impact (or are impacted by) the failure of the SSC under consideration.

Table 2.2 Type B Findings				
SSC Affected by Findings	Reactor Containment Type	Duration of Condition		
		>30 days	30–3 days	<3 days
Containment Penetration Seals, Isolation Valves, and Purge and Vent lines	BWR Mark I and II[1]	yellow	white	green
	BWR Mark III[1]	white	green	green
	PWR Large Dry[1]	red	yellow	white
	PWR Ice Condenser[1]	red	yellow	white
Suppression Pool Bypass	All BWR Containments[2]	yellow	white	green
MSIV Leakage[3]	BWR Mark I and II[4]	yellow	white	green
Ice Condenser Integrity - Partial Failure of Doors	PWR Ice Condenser	red	yellow	white
Hydrogen Igniters	PWR Ice Condenser	red	yellow	white
	BWR Mark III	white	green	green
Containment Spray	BWR Mark I[5] (Drywell)	yellow	white	green
	BWR Mark II (Drywell)	white	green	green
	BWR Mark III (Wetwell)	white	green	green

1. Leakage from the drywell (containment) to the environment is >200 x L_a for BWRs with Mark I and II containments, >500 x L_a for BWRs with Mark III containments, and >1000 x L_a for PWRs. Leakage from the wetwell to the environment is not a LERF concern.
2. The only sequences of interest are high-pressure sequences. This is because low-pressure sequences would be less energetic, thus resulting in slower transport of fission products which provides more opportunity for removing the fission products before they enter the environment. The release would therefore not be "large" and potentially not "early."
3. Excessive leakage that can impact LERF is defined as a leakage rate > 10,000 scfh passed through both the inboard *and* the associated outboard MSIV (PRAB-02-01). An inability to quantify the leakage rate leads to a similar finding.
4. MSIV leakage is only applicable to BWRs with Mark I and II containments. BWRs with Mark III containments have a safety-grade main steam shutoff valve (MSSV). The MSSV is a relatively slow-closing, low leakage valve. Thus, any leakage past the MSIV in a Mark III plant is stopped at the MSSV.
5. The probability of early containment failure from liner melt-through in a Mark I is negligible for accidents with the RCS at low pressure because it is assumed that the drywell floor is flooded. If a finding implies that the drywell floor will be dry, the risk significance (i.e., color) of the finding should be increased by one order of magnitude for high-pressure sequences.

For example, suppose a PWR with an ice condenser containment is found to have a significant number of inoperable hydrogen igniters. Such a finding implies that the containment is vulnerable to failure from a hydrogen deflagration or detonation in a core damage accident. However, the igniters require AC power to operate and would therefore be unavailable anyway in a station blackout accident. Hence, the risk significance of the finding pertains only to the non-SBO portion of the total CDF. In making a final determination, the staff should consider the plant-specific CDF for the major accident classes and estimate more precisely the multiplier (if the actual duration can be established).

If the Factor is 1.0 and the duration of the degraded condition is >30 days, the implication is that ΔLERF is equivalent to either the total CDF or the fraction of CDF that is relevant to the containment function that is compromised. As the product of the Factor and the relevant fraction of CDF decreases, the influence of the containment SSC on the determination of LERF decreases correspondingly. When this product reaches 0.1, the significance category drops one order of magnitude.

The bases for the significance categories in Table 2.2 are provided by containment type, as follows. Chapter 3 addresses BWRs with Mark I containments, Chapter 4 addresses BWRs with Mark II containments, Chapter 5 addresses BWRs with Mark III containments, Chapter 6 addresses PWRs with large-volume and subatmospheric containments, and Chapter 7 addresses PWRs with ice condenser containments.

In general, the data used to address containment performance for Type A findings is derived from the NUREG-1150 studies, supplemented by studies on selected severe accident issues. In contrast, the data used to establish the risk significance of Type B findings is based mainly on the results of the IPE program reported in NUREG-1560, but also includes information from NUREG-6595, NUREG/CR-6025, and NUREG/CR-4832, and the experience of using a draft version of this report.

3 BOILING WATER REACTORS WITH MARK I CONTAINMENTS

This chapter presents the technical bases for the risk significance categories recommended for BWRs with Mark I containments in Chapter 2. This containment design relies on water in the suppression pool to condense steam and to scrub fission products released from the reactor coolant system (RCS). Mark I and II containments have a smaller volume and a higher design pressure than Mark III containments and are inerted to prevent combustion of non-condensable gases released during the accident (e.g., hydrogen deflagration or detonation). As shown by the results of the Individual Plant Examination (IPE) program, BWRs have generally low core damage frequencies, on the average an order of magnitude lower than PWRs since they have multiple ways of supplying water to the core following an initiating event. However, the relatively smaller volumes of some BWR Mark I containments generally result in a higher conditional probability of containment failure given the occurrence of core damage. The Mark I containment accident sequences that contribute to LERF involve both early containment failure and bypass of the suppression pool. For those accident sequences leading to releases that pass through the suppression pool, most of the fission products will be retained in the pool; hence, these releases will not be large. Thus, containment failures involving failures of the wetwell airspace alone will not be contributors to LERF.

This chapter follows the format of previous chapters by discussing Type A and Type B findings separately.

3.1 Type A Findings

Type A findings are associated with accidents that have been assessed using the CDF-based SDP but may influence the determination of LERF. Each of the accident classes impacted by the CDF-based SDP therefore has to be evaluated in terms of its influence on LERF. This section describes the technical bases for the "Factors" recommended in Section 2.2 for Type A findings for BWRs with Mark I containments.

3.1.1 ISLOCA

An important insight from NUREG-1150, the IPE program, and NUREG/CR-5124, "Interfacing Systems LOCA: Boiling Water Reactors," is that these accident sequences are not significant contributors to LERF for any of the BWR containment designs due to their low frequency of occurrence. In addition, the release path is tortuous and significant fission product holdup and scrubbing are expected in the compartments and buildings along the path of release to the environment. Hence ISLOCA is not likely to be a contributor to LERF in BWR containments.

3.1.2 ATWS

Another important insight from the IPE program and numerous published probabilistic risk assessments (PRAs) is that ATWS accident sequences are significant contributors to LERF for BWRs with Mark I containment designs. The energy input to containment from these accident sequences cannot be removed by the normal containment heat removal systems (CHRSs). The resulting rapid pressure rise may cause the containment to fail before or shortly after core damage. If the suppression pool is bypassed, the result might be a large release. The results of past PRAs and IPEs were reviewed to provide guidance on how important this failure mode is to BWRs. The IPE report (NUREG-1560, Volume 2) indicates that the significance of ATWS events in the various IPEs depends on plant-specific features (such as the ability of pumps to work with saturated water) and on assumptions about the power level, when the event occurs in the fuel cycle, and the effectiveness of operator response. For Mark I containments, the conditional probability of early containment failure and suppression pool bypass due to ATWS sequences was estimated. The results of this survey are expressed by the following equation:

$$\text{Factor} = C_{PEF} * C_{PPB}$$

where: Factor is the multiplier on the core damage frequency,
C_{PEF} is the conditional probability of early containment failure given ATWS, and
C_{PPB} is the conditional probability of pool bypass given early containment failure.

In this case, based on data from the IPE program, the C_{PEF} is 0.6 and C_{PPB} is 0.5. This results in a Factor of 0.3 for ATWS sequences.

The above Factor indicates that the staff should consider increasing the risk significance of findings put through the CDF-based SDP for an ATWS in a BWR with a Mark I containment.

3.1.3 Transients

This class of accidents includes a wide range of transient-initiated events, including station blackout (SBO) scenarios. Published PRAs and the IPE results were reviewed to determine if any of the attributes of these accidents might influence LERF determinations.

The Mark I containment design relies on the water in the suppression pool to mitigate the consequences of design basis accidents, such as loss-of-coolant accidents (LOCAs). Not all releases that are scrubbed by the suppression pool are large and, therefore, not all contribute to LERF. Only those early accident sequences where the release from the reactor coolant system bypasses the suppression pool are likely to contribute to LERF. These sequences potentially include (1) failures of the containment due to fuel-coolant interactions (FCIs), (2) containment (drywell liner) meltthrough (from reactor vessel failure with no water on the drywell floor), and (3) high-pressure melt ejection sequences (which include vessel blowdown forces and direct heating of the containment wall and containment atmosphere) resulting in over pressure failure of containment, usually from drywell head seal failure (bypass) as the result of lifting of the drywell head. (Mark I containment atmospheres, however, are inert during operation and therefore hydrogen combustion (deflagration and detonation) is not possible and not a contributor to containment failure.) In all of these cases, the release path to the environment is tortuous and significant fission product holdup and scrubbing are expected to occur. However, no credit for any fission product holdup or scrubbing has been taken in this SDP.

Fuel-coolant interactions have been identified as phenomena that could potentially challenge containment. In particular, an in-vessel FCI was postulated in NUREG-1150 as having the potential to fail the reactor vessel head by "launching" it into the drywell head with enough force to fail containment (referred to as the "alpha failure mode"). Mark I containments are more vulnerable to this vessel failure mode because of the close proximity of the drywell head to the reactor vessel head, but all reactors may have some vulnerability. WASH-1400, "Reactor Safety Study — An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," considered the best-estimate failure probability to be less than 10^{-2}. Since WASH-1400, much work has been done to investigate the potential for an alpha mode failure. A steam explosion review group held a workshop (documented in NUREG/CR-1524) to discuss this issue. The conclusion of this workshop was that the alpha mode failure probability ranged from 10^{-3} to 10^{-5} and that this failure mode was physically unreasonable and "highly unlikely." Therefore, the alpha mode failure is considered unlikely and is a very small contributor to LERF.

Another potential concern is an ex-vessel FCI, which can occur after the core melts through the reactor vessel and contacts water. Experiments have been performed in both the FARO facility (20 experiments) and the KROTOS facility (11 experiments). The experiments used prototypic oxidic melts, with various amounts of metal and at various pressures. None of the experiments resulted in any explosive behavior (NEA/CSNI/R(97)26). Other experiments were performed in the ZREX facility (28 experiments). Of these experiments only those experiments which were subjected to external triggers resulted in explosions (NEA/CSNI/R(97)26, page 598). Furthermore, the ZREX experiments which resulted in "explosive interactions involved extensive production of hydrogen; the explosive energetics, in terms of the mechanical energy output, were very small compared to the available thermal and chemical energy. Apparently, the chemical energy release was not effectively converted into mechanical work" (NEA/CSNI/R(97)26, page 604). Based on the large number of experiments that have been performed without any steam explosions, the probability of an external FCI that fails containment early is considered small and consequently not important for LERF.

The conditional probability of containment failure at vessel breach for Mark I plants was reported in NUREG-1150 and in numerous IPE submittals and PRAs to be strongly influenced by two factors: (1) whether the reactor coolant system is at high or low pressure and (2) for low-pressure sequences, whether water is available for ex-vessel cooling of debris on the drywell floor. The RCS being at high pressure at vessel failure has important implications for the pressure loads on the containment structure. (Water on the floor of the drywell will reduce the chances and consequences of steel containment (liner) meltthrough. The issue of liner meltthrough has received significant attention since the publication of NUREG-1150).

The results of NUREG-1150 and other PRAs are summarized in NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events." The results indicate that if the RCS is at high pressure, the conditional probability of containment failure is 1.0 if there is no water on the drywell floor and 0.6 if the drywell floor is flooded. The conversion Factor for Type A findings is therefore 1.0 for that fraction of the transient accident class that has a high RCS pressure at the time of vessel breach during core meltdown provided the drywell floor is dry. Thus, if a finding related to any transient with high RCS pressure is processed through the

CDF-based SDP, the risk significance (i.e. color assignment) should be increased by one order of magnitude for CDFs equal to 10^{-7}/ry or greater up to the color assignment of red. The conversion Factor for Type A findings is 0.6 for that fraction of the transient accident class that has a high RCS pressure at the time of vessel breach during core meltdown and a flooded drywell floor. Thus, if a finding related to any transient with high RCS pressure and a flooded drywell floor is processed through the CDF-based SDP, the staff should consider increasing the risk significance because of LERF considerations.

NUREG/CR-5423 documented an analysis performed to evaluate the probability of containment meltthrough for sequences where the vessel fails at low pressure. This analysis considered oxidic and metallic pours with and without water on the drywell floor. The conditional containment failure probability of the containment from corium attack is shown in the table below.

Table 3.1 Conditional Containment Meltthrough Probabilities

	Oxidic Melt	Metallic Melt
Without Water	1.0	.63
With Water	6×10^{-5}	$\leq 1.2 \times 10^{-4}$

A follow-on report, NUREG/CR-6025, stated that "values below 1×10^{-3} indicate a 'physically unreasonable' expectation." This analysis was subjected to intensive peer review, as a result of which several parameters were revised. These reviews resulted in a revision of the conditional containment failure probability for the oxidic melt with water. The revised probability was determined by two different methods to be either 2×10^{-3} or 3×10^{-3}. NUREG/CR-6025 found that "both clearly translate to a 'physically unreasonable' event." It should be noted that while creep rupture of the containment steel is credible without water, the effect was found to be localized (NUREG/CR-6025, page 5-40). Two important considerations could impact the magnitude of the release under dry conditions: (1) as the corium cools, it may plug the hole in the containment after failing the steel containment, and (2) the release path created by the creep rupture is tortuous due to the narrow clearance between the steel containment and the concrete wall with insulation filling the space above the failure location. The tortuosity of the release path promotes fission product holdup and retention. Most particulates are retained (not released to the environment) but all of the noble gases that exit containment are released to the environment. It is, thus, reasonably conservative to assume that a significant fraction of the postulated events (vessel failure with no water on the drywell floor), perhaps as much as one-half, would not result in a large release. However, because the fraction of postulated events that result in a large release is > 0.1, these findings are candidates for increasing the risk significance.

The results of these liner meltthrough studies indicate that if the RCS is depressurized, the conditional probability is very small that the Mark I containment will fail provided the drywell floor is flooded. The conversion Factor for Type A findings is therefore << 0.1 for that fraction of the transient accident class that has a low RCS pressure during core meltdown and a flooded drywell floor. The risk significance determined by the CDF-based SDP for this set of transients therefore appears to be appropriate and need not be changed because of LERF considerations.

If the RCS is at low pressure and a finding indicates that the drywell floor will be dry, the conditional probability of early failure caused by liner meltthrough is > 0.1. Such findings are candidates for increasing the risk significance (i.e., color). One way of supplying water to the drywell floor is by the drywell spray system. Therefore, Type B findings related to the operability of this spray system have important implications for this failure mode. Such findings are discussed in Section 3.2.2.

3.1.4 LOCAs

This class of accidents includes events initiated by a wide range of break sizes, which result in significantly different RCS responses. Published PRAs and the IPE results were reviewed to determine if any of the attributes of these accidents might influence LERF determinations. The RCS pressure during core meltdown was found to have the largest influence on LERF determinations. Thus, the fraction of accidents initiated by LOCAs that result in the highest RCS pressure (i.e. small break LOCAs) needs to be assessed in terms of LERF considerations.

The staff should consider increasing the risk significance of findings related to small-break LOCAs processed through the CDF-based SDP on the basis of LERF considerations. Small-break LOCA findings that would result in high RCS pressure at vessel breach should therefore be combined with transients with the RCS at high pressure and treated in exactly the same way (refer to Section 3.1.3).

3.2 Type B Findings

Type B findings are associated with SSCs that do not impact the CDF determination and, therefore, have been assessed using the CDF-based SDP. These findings, however, are potentially important to LERF determinations and do have to be allocated an appropriate risk category. This section presents the technical bases for the risk significance categories recommended for Type B findings for BWRs with Mark I containments in Section 2.3.

3.2.1 Containment Penetration Seals, Isolation Valves, and Purge and Vent Lines

An important insight from the IPE program and other published PRAs is that containment leakage and loss of containment isolation accident sequences do not significantly contribute to the LERF for plants with Mark I containments. Mark I containments are operated with inert atmospheres. A gross failure of a penetration seal, isolation valve, or venting (e.g., the vent and purge valves) is identified by the failure to inert containment or the loss of the inert containment atmosphere. The risk significance of the failure depends on its location and size. For failures involving the drywell pressure boundary, fission product released directly into the drywell is not scrubbed in the suppression pool. This primarily impacts LOCAs and the ex-vessel phase of other severe accidents.

If the breach in the drywell pressure boundary results in a leakage[2] to the environment greater than 200 x L_a it can constitute a large early release. Drywell sprays, if available, reduce the amount of the release. Data generated in the IPE program and reported in published PRAs suggests that for BWRs with Mark I containments, on average, about one-third of the core damage frequency consists of early containment failure sequences and about a third of the early containment failure sequences are large releases. Hence on average, about 0.1 of the core damage frequency in BWRs with Mark I containments constitutes LERF. Thus if a finding implies the existence of a breach in the drywell pressure boundary that would result in a drywell leakage rate > 200 x L_a the large release probability of 0.1 increases essentially to 1.0. The conversion factor for Type B findings is, therefore, approximately (1.0-0.1) = 0.9 for findings of this type. This assumption neglects the effect of pool scrubbing for those sequences in which the in-vessel release passes through the suppression pool. The risk significance can be determined by using the relationship given in Section 2.3 and assuming a total CDF of 10^{-5}/ry for BWRs:

ΔLERF = 0.9 x 10^{-5} x (multiplier based on duration of degraded condition)

Using the multipliers given in Section 2.3 for each of the three (degraded condition) durations the following three ΔLERFs and the corresponding risk significance categories are obtained:

Duration	ΔLERF	Significance Category
> 30 days	9 x10^{-6}	yellow
30–3 days	9 x10^{-7}	white
< 3 days	9 x10^{-8}	green

If a finding identifies a degraded condition that involves a breach of the drywell pressure boundary that can potentially result in a leakage rate in excess of 200 x L_a and the duration of the degraded condition is also determined, one of the significance categories given above can be assigned to the finding.

2 Several studies, including NUREG/CR-4330, "Review of Light Water Reactor Regulatory Requirements," NUREG-1493, "Performance-Based Containment Leak-Test Program," and NUREG/CR-6418, "Risk Importance of Containment and Related ESF System Performance Requirements," have been performed to determine the risk significance of various levels of containment leakage. While the results vary by plant and containment type, a containment leak rate of about 100 volume percent per day appears to constitute an approximate threshold beyond which the release may become significant to LERF. Design basis leakage from containment is determined by regulatory requirements to assure the containment leakage will be below the maximum allowable leak rate (denoted as L_a) set by Title 10 of the *Code of Federal Regulations* Part 100 dose limits that is incorporated in the plant technical specifications. Typical values of L_a are 0.1 containment volume percent per day for PWRs and 0.5 volume percent per day for Mark I and Mark II BWRs, and 0.2 volume percent per day for Mark III BWRs. Thus a LERF significant leakage rate from containment would be a rate greater than or equal to about 1000 L_a for PWRs, 200 L_a for Mark I and II BWRs, and 500 L_a for Mark III BWRs. The 100 volume percent per day leakage rate is approximately equivalent to a hole size in containment of 2.5 – 3 inches in diameter for PWRs with large dry containments, 2 inches for PWRs with ice condenser containments, 1 inch for BWRs with Mark I and II containments, and 2.5 inches for BWRs with Mark III containments (Palla, 2001).

For failures involving the wetwell pressure boundary, any leakage from the wetwell atmosphere has been scrubbed by the suppression pool and does not contribute to LERF by virtue of not being a "large" release to the environment as long as there is no suppression pool bypass (drywell atmosphere to wetwell atmosphere leakage). Therefore wetwell leakage is not addressed in this SDP.

Another possibility is that a penetration valve seal may appear to be intact and fully functional, but may contain a flaw that would fail to prevent leakage under full-pressure conditions. The presence of the two metal components in such close proximity, held apart only by a seal, limits the size of the opening to a small hole, the size of which is related to the actual dimensions of the valve. The seal reduces the theoretical size of the opening and thereby limits the amount of material released past the valve. Furthermore, redundant containment isolation valves are required. If the second valve seal is functional, there is no release. A release would require the failure of both seals in a single line. A similar failure of the second seal is unlikely. If a similar failure did occur, the release would be substantially reduced (by increased holdup time, plateout, agglomeration and gravitational settling, and impaction, and by reduced flow as the result of reduced pressure across the seals). It has been suggested that such small leakage pathways may actually become plugged by particulates, thereby terminating the release. Therefore, such failures are not expected and are not addressed in this SDP.

3.2.2 Containment Sprays

Using the drywell sprays to flood the drywell floor during core meltdown accidents in Mark I containments is an important strategy for preventing steel containment (liner) meltthrough and, hence, lowering the likelihood of LERF (refer to Section 3.1.3). This section deals with findings related to the operability of the drywell spray system in Mark I containments. It is noted in Section 3.1 that in addition to drywell flooding, the pressure in the reactor vessel at the time of vessel failure also has a significant impact on the likelihood of LERF. Data generated in the IPE program and reported in published PRAs suggests that for BWRs the ratio of high to low pressure sequences is about 50:50. If a finding implies that the drywell floor could not be flooded (using any available means such as residual heat removal (RHR) or diesel-driven fire protection system water pumps), then the large release probability of 0.6 could approach 1.0 for high-pressure scenarios and > 0.1 to 1.0 for low-pressure scenarios (refer to Section 3.1.3). The conversion Factor for Type B findings is therefore approximately $0.5(1.0 - 0.6) + 0.5(1.0 - >0.1) = < 0.7$ for findings of this type. This assumes that all liner meltthrough failures result in LERF and also neglects the effects of pool scrubbing for those sequences in which the in-vessel release passes through the suppression pool and the effects of fission product retention in the tortuous release pathway. Consideration of pool scrubbing and fission product retention would eliminate these sequences as not LERF significant. The early decontamination factor for particulates (fission products important to LERF) would be well over 100 (typical early pool decontamination factors are in the 10,000 range up to 100,000). The risk significance can be determined by using the relationship given in Section 2.3 and assuming a total CDF of 10^{-5}/ry for BWRs:

$$\Delta\text{LERF} = < 0.7 \times 10^{-5} \times \text{(multiplier based on duration of degraded condition)}$$

Using the multipliers given in Section 2.3 for each of the three (degraded condition) durations, the following three ΔLERFs and the corresponding risk significance categories are obtained:

Duration	ΔLERF	Significance Category
> 30 days	$< 7 \times 10^{-6}$	yellow
30–3 days	$< 7 \times 10^{-7}$	white
< 3 days	$< 7 \times 10^{-8}$	green

If a finding identifies a degraded condition that could lead to an inability to flood the drywell floor and the duration of the degraded condition is also determined, the finding can be assigned to one of the significance categories given above.

3.2.3 Suppression Pool Cooling

Heat removal is needed to maintain the suppression pool temperature within its operating limits. Failure of the cooling system could eventually result in containment failure, which could result in inadequate net positive suction head (NPSH) for the reactor heat removal (RHR) water pumps. Failure of these RHR pumps leads to loss of coolant to the core, dryout, meltdown, and, eventually, reactor vessel failure. This particular sequence, however, typically takes about 30 hours before core degradation. Thus loss of suppression pool cooling is unlikely to influence LERF. Therefore findings related to loss of suppression pool cooling should be processed through the CDF-based SDP to assign a risk significance category.

Another potential concern is plugging of the suction strainers and the resulting loss of pool cooling or reactor vessel injection. Resolution of this problem is documented in Nuclear Regulatory Commission (NRC) Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors," Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," and Regulatory Guide 1.82, Revision 2, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," published in May 1996. Therefore, findings related to potential plugging of the strainers are not anticipated and are not expected to influence LERF. However, any such findings should be processed through the CDF-based SDP to be assigned a significance category.

3.2.4 Fan Coolers

Fan coolers normally operate all of the time. They are designed to maintain the containment atmosphere within the plant technical specification operating limits. They are not capable of removing large amounts of heat deposited into containment over short periods, for example, by a pipe break or vessel failure. The fan coolers have minimal capability to remove fission products and no credit is given for any such removal. Therefore, the unavailability of the fan coolers may result in operational problems but will not significantly affect the timing or quantity of fission products released to the environment as a result of a severe core damage accident. Thus, the fan coolers are not an important factor in LERF determinations.

3.2.5 Isolation Condensers

A bypass mechanism, namely induced failure of the emergency condenser tubes, was reported in NUREG-1560 to be a contributor to early loss of containment integrity in one IPE for a BWR with a Mark I containment. (As reported in NUREG-1560, three other BWR Mark 1 plants also use isolation or emergency condensers but did not consider this potential failure mode in their IPE submittals.) This failure mode is similar to the induced failure of steam generator tubes in PWRs. The conditional containment failure probability (CCFP) of this event was found in the IPE to be relatively low (~ 3% of the significant early release) when compared to the probability of early structural failure. In addition, the accident sequence has to be one that involves the RCS remaining at high pressure. Therefore, since it is assumed that high-pressure core melt sequences have a CCFP of 1.0, the risk significance of this failure mode is already subsumed under the category of high-RCS-pressure scenarios.

3.2.6 Containment Flooding Systems

Containment flooding in BWRs with Mark I containments is a long-term strategy designed to cool the core in-vessel and prevent reactor vessel failure. This strategy does not influence LERF determinations. However, flooding the drywell floor does have important implications for mitigating some failure modes such as liner meltthrough in Mark I containments. Any system, e.g., containment drywell sprays, that can inject water into the Mark I drywell affects the probability of containment meltthrough and the magnitude of the release. Once the core has relocated into the bottom of the vessel or failed the bottom head, one potential source of water is the control rod drive pumps. The issue of flooding the drywell is addressed in Section 3.2.2, which deals with drywell spray operation. Flooding by other methods is not addressed in this SDP.

3.2.7 Suppression Pool Bypass

Suppression pool bypass is a failure of a component or structure that would limit the ability of the suppression pool to perform its intended safety function. An example of such a finding is a large drywell vent line open. The bypass has to be shown to prevent most of the flow from passing through the suppression pool, e.g., drywell leakage is not suppression pool bypass (in the context of this report) because most of the flow would (or could) pass through the suppression pool, preventing a large release.

As noted above for BWRs with Mark I containments, on average, about one-third of the core damage frequency consists of early containment failure sequences and about a third of the early containment failure sequences lead to large releases. Hence, on average, about 0.1 of the core damage frequency in BWRs with Mark I containments consists of LERF. If a finding on an SSC important to suppression pool integrity implies that suppression pool bypass can occur, the large release probability 0.1 increases to about 0.3 (the total *early* containment failure frequency) because all of the early containment failure sequences are unscrubbed. In addition, if the suppression pool is bypassed, accidents that would have failed the containment late in the accident sequence can now potentially fail it early due to a lack of heat removal by the pool. This occurs for only those bypass events that are caused by failures of the vacuum breakers in the wetwell airspace. However, even if these potential failure modes are taken into account, they do not change the ΔLERF significance category given below.

The conversion Factor for Type B findings is therefore 0.3 - 0.1 = 0.2 for suppression pool bypass findings and potentially higher if some late failures occur early. The risk significance can be determined by using the relationship given in Section 2.3 and assuming a total CDF of 10^{-5}/ry for BWRs:

$$\Delta LERF = 0.2 \times 10^{-5} \times \text{(multiplier based on duration of degraded condition)}$$

Using the multipliers given in Section 2.3 for each of the three (degraded condition) durations, the following three ΔLERFs and the corresponding risk significance categories are obtained:

Duration	ΔLERF	Significance Category
> 30 days	2×10^{-6}	yellow
30–3 days	2×10^{-7}	white
< 3 days	2×10^{-8}	green

If a finding identifies a degraded condition that could lead to suppression pool bypass and the duration of the degraded condition is also determined, the finding can be assigned to one of the significance categories above.

3.2.8 Main Steam Isolation Valve Leakage

Accidents that involve excessive leakage of the main steam isolation valves (MSIVs) can have a release path that bypasses the containment. Although excessive MSIV leakage has not been demonstrated to be a risk-significant accident in terms of contribution to LERF, it remains a potential bypass mechanism in risk assessments of BWR Mark I containments and should be considered in the SDP process. Core melt accidents involving excessive MSIV leakage with the reactor coolant system at high pressure may have some similarity to induced steam generator tube ruptures in PWRs. Excessive leakage (potentially leading to early health effects) is defined as a leak rate greater than 10,000 scfh (standard cubic feet per hour) passed through both the inboard and the associated outboard MSIV (PRAB-02-01) or the inability to quantify the leakage rate.

Consequently, if a finding reveals excessive MSIV leakage (as defined above), its significance category can be determined by assuming a bypass probability of 1.0 for all high-pressure core melt accidents and from the duration of the degraded condition. The conversion Factor for Type B findings, then, is the difference between the conditional probability of 1.0 and the original conditional probabilities (i.e., 1.0 if the drywell is dry and 0.6 if it is flooded) of a large early release for high-pressure scenarios (which constitute ~0.5 of the total CDF-based on IPE data). Assuming that the ratio of dry to flooded drywell sequences is about 50:50 based on IPE data, the conversion Factor for Type B findings is, therefore, approximately 0.5[0.5(1.0 - 1.0) + 0.5(1.0 - 0.6)] = 0.1 for findings of this type. The risk significance can be determined by using the relationship given in Section 2.3 and assuming a total CDF of 10^{-5}/ry for BWRs:

$$\Delta\text{LERF} = 0.1 \times 10^{-5} \times \text{(multiplier based on duration of degraded condition)}$$

Using the multipliers given in Section 2.3 for each of the three (degraded condition) durations, the following three ΔLERFs and the corresponding risk significance categories are obtained:

Duration	ΔLERF	Significance Category
> 30 days	1×10^{-6}	yellow
30–3 days	1×10^{-7}	white
< 3 days	1×10^{-8}	green

If a finding identifies a degraded condition that could lead to excessive MSIV leakage and the duration of the degraded condition is also determined, the finding can be assigned to one of the significance categories above.

3.2.9 Filtration Systems

Filtration systems, e.g., the standby gas treatment system (SGTS), remove particulates and, in some cases, condition the air. These systems are used extensively outside containment and require AC power to operate. Thus, these systems are not functional in any accident scenario which results in the loss of AC power, i.e., station blackout sequences. Accident sequences which are important for LERF considerations are those which release a substantial portion of the reactor core radionuclide inventory. Such a large release of aerosols readily plugs filters and renders filtration systems ineffective. (Non-power accidents, e.g., fuel handling accidents, do not release sufficient quantities of radionuclides to constitute a large release and therefore are not expected to influence LERF.) Finally, the release sequences that are of importance are those that bypass the suppression pool, e.g., containment (predominantly drywell) bypass. Therefore, filtration systems are not addressed further in this SDP. Nor are the filters on the control room air vents addressed.

4 BOILING WATER REACTORS WITH MARK II CONTAINMENTS

This chapter presents the technical bases for the risk significance categories recommended for BWRs with Mark II containments in Chapter 2. These containment designs rely on water in the suppression pool to condense steam and scrub fission products released from the reactor coolant system (RCS). Mark I and II containments have a smaller volume and a higher design pressure than Mark III containments and are inerted to prevent combustion of non-condensable gases released during the accident (e.g., hydrogen deflagration or detonation). As shown by the results of the Individual Plant Examination (IPE) program, BWRs generally have low core damage frequencies, on the average an order of magnitude lower than PWRs, because they have multiple ways of providing water to the core following an initiating event. However, the relatively smaller volumes of some BWR containments may result in a higher conditional probability of containment failure given the occurrence of core damage. The accident sequences in Mark II containments that contribute to LERF involve both early containment failure and bypass of the suppression pool. Accident sequences leading to releases that pass through the suppression pool are scrubbed (i.e., most of the fission products are retained in the pool); hence these releases are not large. Therefore, containment failures involving failures of the wetwell airspace alone are not contributors to LERF.

This chapter follows the format of previous chapters, discussing Type A and Type B findings separately.

4.1 Type A Findings

Type A findings are associated with accidents that have undergone the CDF-based SDP but may influence the LERF determination. Each of the accident classes affected by the CDF-based SDP, therefore, has to be evaluated in terms of its influence on LERF. This section describes the technical bases for the "Factors" recommended for Type A findings for BWRs with Mark II containments.

4.1.1 ISLOCA

An important insight from NUREG-1150, the IPE program, and NUREG/CR-5124 (an ISLOCA study) is that these accident sequences were not significant contributors to LERF for any of the BWR containment designs. The frequency of occurrence is low and the release path is tortuous so fission products are held up and scrubbed in the compartments and buildings along the path of release to the environment. Hence ISLOCA is not likely to be a contributor to LERF in BWR containments.

4.1.2 ATWS

Another important insight from the IPE program and numerous published PRAs is that ATWS accident sequences are significant contributors to LERF for BWRs with Mark II containment designs. These accident sequences put more heat into containment than can be removed by the normal containment heat removal systems (CHRSs). The resulting rapid pressure rise may cause the containment to fail before or shortly after core damage. If the suppression pool is bypassed, a large release may ensue. The results of past PRAs and IPEs were reviewed to determine how important this failure mode is to BWRs. For Mark II containments, the conditional probability of early containment failure and suppression pool bypass due to ATWS sequences was estimated. The results of this survey are expressed by the equation:

$$\text{Factor} = C_{PEF} * C_{PPB}$$

where: Factor is the multiplier on the core damage frequency,
C_{PEF} is the conditional probability of early containment failure given ATWS, and
C_{PPB} is the conditional probability of pool bypass given early containment failure.

In this case, based on data from the IPE program, the C_{PEF} is 0.6 and C_{PPB} is 0.7. This results in a Factor of 0.4 for ATWS sequences.

The above Factor indicates that the staff should consider increasing the risk significance categorization of findings processed through the CDF-based SDP for an ATWS in a BWR with a Mark II containment.

4.1.3 Transients

This class of accidents includes a wide range of transient-initiated events, including station blackout (SBO) scenarios. Published PRAs and the IPE results were reviewed to determine if any of the attributes of these accidents might influence LERF determinations.

Mark II containments rely on suppression pools to mitigate the consequences of design basis accidents. Not all releases during severe accidents that are scrubbed by the suppression pool are large, and therefore, not all releases contribute to LERF. Only accident sequences where the containment fails early and the release bypasses the suppression pool are likely to contribute to LERF. Early failure of the containment at or close to the time of vessel breach can occur due to (1) ex-vessel steam explosions (in-pedestal) and (2) potential high-pressure melt ejection sequences (including vessel blow down and direct heating of the containment atmosphere) and results in over pressure failure of containment, usually from drywell head seal failure (bypass) as the result of lifting of the drywell head. (Mark II containment atmospheres, however, are inert during operation, and hydrogen combustion (deflagration and detonation) is not possible and therefore not a contributor to containment failure.) In each case, the release path to the environment is tortuous and significant fission product holdup and scrubbing are expected to occur. No credit for any fission product holdup or scrubbing is credited in this SDP. Extensive studies have been performed on the response of Mark II containments to severe accidents. The results of these studies indicate significant variability due to differences in the configurations of the various Mark II containment designs.

As with Mark I containment designs, the probability of a Mark II containment failure is relatively independent of whether the pedestal or drywell floor is flooded if the RCS is at *high*-pressure at the time of vessel breach. The likelihood of containment failure given a high-pressure vessel breach is lower than for a Mark I containment (i.e., approximately 0.3 with or without water in a Mark II versus 1.0 for a Mark I; refer to NUREG/CR-6595). The containment failure probability, however, is reduced (by a factor of three from 0.3 to 0.1 in NUREG/CR-6595) if there is water on the pedestal or drywell floor at the time of vessel breach for *low*-pressure sequences. (In the 0.3 case the staff should consider the CDF-based SDP for a potential increase because of potential LERF considerations; no additional considerations are necessary in the 0.1 case.) The likelihood of water in-pedestal depends on the configuration of the various Mark II containment designs. Water inside the pedestal area affects the likelihood of ex-vessel steam explosions inside the pedestal area and can be affected by downcomers when they are located directly below the vessel.

The failure pressures for Mark II containments have been predicted to range between 120 and 160 psig (NUREG/CR-2442, "Reliability Analysis of Steel Containments Strength"; Limerick Generating Station Probabilistic Risk Assessment; and Long Island Lighting Company's request for power increase). The most probable over pressure failure location for Mark II plants appears to be the primary containment liner in the wetwell airspace above the surface of the suppression pool. Calculations in support of the Containment Performance Improvement program demonstrated that the effects of high-pressure vessel failure, i.e., high pressure melt ejection (HPME), for short term station blackout events (the most containment-challenging event) would not fail containment (or the pedestal wall) for at least 1 hour and 40 minutes and as much as 8 hours after vessel failure (NUREG/CR-5565).

Certain design features of Mark II containments, especially the reactor pedestal design, play an important role in the response of some plants to severe accidents. La Salle Units 1 and 2 and Columbia Power Station[3] (CPS) have Mark II containments with a recessed in-pedestal region (in effect, a reactor cavity). Limerick Generating Station Units 1 and 2 and Susquehanna Units 1 and 2 have a flat in-pedestal floor at about the same elevation as the ex-pedestal drywell floor. Nine Mile Point Unit 2 has a recessed in-pedestal floor that communicates directly (via in-pedestal downcomers) with the suppression pool. In La Salle and CPS, the core-concrete interaction (CCI) could continue unabated on a dry drywell floor and all of the generated products would be scrubbed in the suppression pool before being vented to the environment. Water from drywell sprays would accumulate on the drywell floor, ex-pedestal, until it overflowed the downcomers and entered the suppression pool. The entrance to the in-pedestal region is above the ex-pedestal downcomer opening height. The in-pedestal drains would remove the water in the pedestal. If CCI were not mitigated before failure of the pedestal floor, the debris would fall into the suppression pool and be quenched and scrubbed.

In the Limerick plant, core debris discharged after vessel breach can spread out over the drywell floor thinly enough not to challenge the integrity of the downcomer. In the Susquehanna plant, the floor slopes up from the pedestal and only the near downcomers may be challenged. The integrity of the downcomer is only challenged when the corium is deep enough to overflow down the inside of the

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3 The Columbia Power Station was formerly known as Washington Nuclear Project Number 2 (WNP-2).

downcomer, heating the inside of the downcomer and CCI attack at the junction of the downcomer and the drywell floor.

In Nine Mile Point plant, there are downcomer pipes in the in-pedestal region and core debris could enter the suppression pool and create a potential for a fuel-coolant interaction (FCI). The water would cool the debris and terminate or greatly reduce CCI. Other Mark II plants have drain lines in the drywell floor that become plugged with core debris and prevent suppression pool bypass. If the drain line containment isolation valve were open or if the valve failed open, e.g., from water hammer, the drain line, if not plugged, could result in a suppression pool bypass at plants where the drain line penetrates containment.

As noted above, there is considerable variability in the response of Mark II containments to core melt accidents. However, these studies indicate that if the RCS is at high pressure, then on average the conditional probability is close to 0.3 that the Mark II containment will fail whether or not the drywell floor is flooded. The conversion Factor for Type A findings is therefore 0.3 for that fraction of the transient accident class that has a high RCS pressure during core meltdown. Therefore, the staff should consider increasing the risk significance categorization of findings related to transients with high RCS pressure processed through the CDF-based SDP.

Data from NUREG/CR-6595 indicates that if there is no water on the drywell floor or the pedestal area, the conditional probability that the containment will fail, leading to a large early release, is approximately 0.3 for transients with low RCS pressure at core meltdown. This number is uncertain; IPE data for a couple of Mark II plants indicates that the conditional containment failure probability may be closer to 0.2. However, as long as this conditional probability is > 0.1, the risk significance of the finding remains essentially the same. Hence if a finding related to a transient with low RCS pressure (where the drywell floor cannot be flooded) is processed through the CDF-based SDP, the staff should consider increasing the risk significance categorization.

These studies also indicate that if the RCS is depressurized and the drywell floor is flooded, then the conditional probability is less than 0.1 that the Mark II containment will fail. The conversion Factor for a Type A finding is therefore <0.1 for that fraction of the transient accident class that has a low RCS pressure and a flooded drywell floor during core meltdown. The risk significance determined by the CDF-based SDP for transients with the RCS depressurized in BWR Mark II containments appears to be appropriate and need not be changed because of LERF considerations provided the drywell floor is flooded.

4.1.4 LOCAs

LOCAs are a class of accidents initiated by a wide range of break sizes. The RCS response depends on the break size and location. Published PRAs and the IPE results were reviewed to determine if any of the attributes of these accidents might influence LERF determinations. The RCS pressure during core meltdown was found to have the largest influence on LERF determinations. Thus, that fraction of accidents initiated by LOCAs that result in the highest RCS pressure (i.e., small-break LOCAs) needs to be assessed in terms of LERF.

If a finding related to a small break LOCA is processed through the CDF-based SDP, the risk significance should be examined for a potential increase because of LERF considerations. Small-break LOCA findings that result in high RCS pressure at vessel breach should therefore be combined with transients with the RCS at high pressure and treated in the same way (refer to Section 4.1.3).

4.2 Type B Findings

Type B findings are associated with SSCs that do not impact the CDF determination and, therefore, have not undergone the CDF-based SDP. These findings, however, are potentially important to LERF determinations and do have to be allocated an appropriate risk category. This section presents the technical bases for the risk significance categories recommended for Type B findings for BWRs with Mark II containments in Section 2.3.

4.2.1 Containment Penetration Seals, Isolation Valves, and Purge and Vent Lines

An important insight from the IPE program and other published PRAs is that containment leakage and loss of containment isolation accident sequences do not significantly contribute to the LERF for plants with Mark II containments. Mark II containments are operated with inert atmospheres. A gross failure of a penetration seal, isolation valve, or venting (e.g., the vent and purge valves) is identified by the failure to inert containment or the loss of the inert containment atmosphere. The risk significance of the failure depends on its location and size. For failures involving the drywell pressure boundary, fission product releases into the drywell will not receive the benefit of suppression pool scrubbing. This primarily impacts LOCAs and the ex-vessel phase of other severe accidents. A breach in the drywell pressure boundary that results in a leakage to the environment greater than $200 \times L_a$ (see footnote in Section 3.2.1) could constitute a large early release. Drywell sprays, if available, reduce the amount of the release. Data generated in the IPE program and reported in published PRAs suggests that for BWRs with Mark II containments, on average, about 15% of the core damage frequency consists of early containment failure sequences and about 35% of the early containment failure sequences lead to large releases. Hence, on average, about 0.05 of the core damage frequency in BWRs with Mark II containments constitutes LERF. Thus, if a finding implies that a breach in the drywell pressure boundary results in a drywell leakage rate $> 200 \times L_a$, the large release probability of 0.05 conservatively increases essentially to 1.0. The conversion factor for Type B findings is, therefore, approximately (1.0 - 0.05) = 0.95 for findings of this type. This assumption neglects the effect of pool scrubbing for sequences in which the in-vessel release passes through the suppression pool. Any consideration of pool scrubbing and fission product retention eliminates these sequences as significant to LERF. The early decontamination factor for particulates (fission products important to LERF) would be well over 100 (typical early pool decontamination factors are in the range of 10,000 to 100,000). The risk significance can be determined by using the relationship given in Section 2.3 and assuming a total CDF of 10^{-5}/ry for BWRs:

$$\Delta \text{LERF} = 0.95 \times 10^{-5} \times \text{(multiplier based on duration of degraded condition)}$$

Using the multipliers given in Section 2.3 for each of the three (degraded condition) durations, the following three ΔLERFs and the corresponding risk significance categories are obtained:

Duration	ΔLERF	Significance Category
> 30 days	9.5×10^{-6}	yellow
30–3 days	9.5×10^{-7}	white
< 3 days	9.5×10^{-8}	green

If a finding identifies a degraded condition that involves a breach of the drywell pressure boundary that can potentially result in a leakage rate in excess of $200 \times L_a$ and the duration of the degraded condition is determined, then one of the significance categories given above can be assigned to the finding.

Another possibility is that a valve seal may appear to be intact and fully functional, but may contain a flaw that would fail to prevent leakage under full pressure conditions. The presence of the two metal components in such close proximity, held apart only by a seal, limits the size of the opening to a small hole the size of which is related to the actual dimensions of the valve. The seal's physical presence reduces the theoretical size of the opening and thereby limits the amount of material released past the valve. Furthermore, redundant containment isolation valves are required. If the second valve seal is functional, there is no release. A release requires the failure of both seals in a single line. A similar failure of the second seal is unlikely. If a similar failure occurs, the release is substantially reduced (by increased holdup time, plateout, agglomeration and gravitational settling, and impaction, and by reduced flow as the result of pressure reduction across the seals). It has been suggested that such small leakage pathways may actually become plugged by particulates, thereby terminating the release. Therefore, such failures are not addressed in this SDP.

4.2.2 Containment Sprays

Early containment failures in Mark II plants arise from several different failure mechanisms, depending on the plant. As shown in the IPE report (NUREG-1560), these failure mechanisms range from a containment overpressure failure due to loss of containment heat removal to an ex-vessel core debris-induced pedestal drain line isolation valve failure (potentially a small containment bypass). NUREG/CR-5623, "BWR Mark II Ex-Vessel Corium Interaction Analysis," has demonstrated that drain failure is only possible if the oxides exit the bottom of the reactor vessel before the zirconium or steel. This has a very low probability of occurrence and low potential offsite consequences. With no FCI events, there is a very low likelihood that there will be drain line failure. As pointed out above, only those accidents that bypass the suppression pool (i.e., involve failure of the drywell) contribute to LERF.

The containment sprays, if they are available, could impact these accidents by scrubbing the containment atmosphere and, thus, converting a potentially "large release" into a non-large release. In one IPE submittal, under conditions of suppression pool bypass, the use of drywell sprays to scrub the containment atmosphere or wetwell sprays to scrub the wetwell atmosphere is an accident management action. However, according to the results of the IPE program (reported in NUREG-

1560), the emergency operating procedures (EOPs) for the Nine Mile Point Unit 2 indicate that sprays are used only if adequate core cooling is assured.

Containment sprays are effective in removing fission products from the containment atmosphere and are hence capable of mitigating drywell-to-wetwell bypass events. However, containment systems were designed for the thermal conditions associated with the design basis LOCA. The spray pumps are driven by AC power and, thus, are not available for SBO accident sequences. However, licensees have modified their spray systems to permit the operation of the containment sprays from the fire suppression system using the diesel-driven pump. This pump is manually connected to the RHR line that supplies the containment spray header. Calculations have demonstrated that the reduced flow rate and duration of containment sprays using the fire suppression system diesel-driven water pump delay late containment failure by approximately 6 hours for Mark II containments.

For those sequences where sprays are available, spray operation can be beneficial, as mentioned above, for those accidents that challenge containment. In these sequences, timing is important. The sprays can be operated manually or actuated automatically at a predetermined containment pressure or temperature or a high-radiation signal (according to the Boiling Water Reactor Owners Group (BWROG)).

There is thus a great deal of uncertainty in how much spray operability reduces LERF. An approximate estimate of the contribution of sprays can be made based on the results reported in NUREG/CR-6595. The conditional containment failure probability for low-pressure sequences that fail containment early is 0.3 without water on the drywell floor and 0.1 with water on the drywell floor. (The conditional containment failure probability (CCFP) for high-pressure sequences that fail containment early is unaffected by the presence of water.) If we assume, conservatively, that water on the drywell floor can be ascribed to successful spray operation, then based on NUREG/CR-6595 the worth of sprays is (0.3 - 0.1) multiplied by the frequency of low-pressure scenarios. If we also conservatively assume (based on NUREG-1560) that about 30% of the CDF sequences are low-pressure sequences, then the Type B Factor for containment sprays is approximately (0.3 - 0.1) x 0.3 = 0.06.

The risk significance can be determined by using the relationship given in Section 2.3 and assuming a total CDF of 10^{-5}/ry for BWRs:

$$\Delta\text{LERF} = 0.06 \times 10^{-5} \times \text{(multiplier based on duration of degraded condition)}$$

Using the multipliers given in Section 2.3 for each of the three (degraded condition) durations, the following three ΔLERFs and corresponding risk significance categories are obtained:

Duration	ΔLERF	Significance Category
> 30 days	6×10^{-7}	white
30–3 days	6×10^{-8}	green
< 3 days	6×10^{-9}	green

If a finding identifies a degraded condition that could lead to inoperability of the containment sprays and the duration of the degraded condition is known, then one of the significance categories given above can be assigned to the finding.

4.2.3 Suppression Pool Cooling

Heat removal is needed to maintain the suppression pool temperature within the pool's operating limits. Failure of the cooling system could eventually result in containment failure, which could result in inadequate net positive suction head (NPSH) for the reactor heat removal (RHR) system water pumps for some Mark II containments. Failure of the RHR pumps leads to loss of coolant to the core, dryout, meltdown, and, eventually, reactor vessel failure. In this particular sequence, however, core degradation typically takes about 30 hours. Thus, loss of suppression pool cooling is unlikely to influence LERF. Therefore, findings related to loss of suppression pool cooling should be processed through the CDF-based SDP to be assigned a risk significance category.

Another potential concern is plugging of the suction strainers such that pool cooling or reactor vessel injection is lost. The resolution of this concern is documented in NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors," Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss of Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," and Regulatory Guide 1.82, Revision 2, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," published in May 1996. Findings related to potential plugging of the strainers are not anticipated. Findings of this nature are not expected to influence LERF. However, if such a finding is made, it should be processed through the CDF-based SDP in order to be assigned a significance category.

4.2.4 Fan Coolers

Fan coolers are normally operating. They are designed to maintain the containment atmosphere within the plant technical specification operating limits. They are not capable of removing large amounts of heat deposited into containment over short periods such as after a pipe break or vessel failure. The fan coolers have minimal capability to remove fission products and no credit is given for any such removal. Therefore, the unavailability of the fan coolers may result in operational problems but will not significantly affect the timing or quantity of fission products released to the environment as the result of a severe core damage accident. Thus, the fan coolers are not an important factor in LERF determinations.

4.2.5 Containment Flooding Systems

Containment flooding in BWRs with Mark II containments is a long-term strategy to cool the core in-vessel and prevent reactor vessel failure. The containment is flooded to prevent reactor vessel failure or to arrest or mitigate further CCI. Core debris exiting the reactor vessel is not a structural issue for Mark II containments. In all cases, CCI stops and the material cools inside the containment, albeit the drywell or wetwell. Containment flooding adds water to the suppression pool, cools the debris, and potentially retards CCI or prevents CCI from failing the drywell floor. However, because CCI and containment flooding for Mark II containments occur relatively late in an accident, they do not affect LERF. Therefore, findings associated with containment flooding do not influence LERF determinations.

4.2.6 Suppression Pool Bypass

Suppression pool bypass is a failure of a component or structure that would limit the ability of the suppression pool to perform its intended safety function. An example of such a finding is a large drywell vent line open. The bypass has to be shown to prevent most of the flow from passing through the suppression pool, e.g., drywell leakage is not suppression pool bypass (in the context of this report) because most of the flow would (or could) pass through the suppression pool, preventing a large release.

Information from the IPE program for Mark II plants and published PRAs indicates that, on average, about 15% of core damage accident sequences progress to early containment failure and about 35% of the early containment failure sequences consist of large releases. Hence, on average, 0.05 of the core damage frequency in BWR Mark II containments contributes to LERF. If a finding implies suppression pool bypass, then the large release probability 0.05 increases to about 0.15. In addition, if the suppression pool is bypassed, accidents that fail the containment late in the accident sequence could potentially fail it early due to a lack of heat removal by the pool, though only if the bypass is caused by failures of the vacuum breakers in the wetwell airspace. However, even if these potential failure modes are taken into account, the ΔLERF significance category given below remains the same.

The conversion Factor for Type B findings is therefore 0.15 - 0.05 = 0.1 for suppression pool bypass findings and potentially higher if some late failures occur early. The risk significance can be determined by using the relationship given in Section 2.3 and assuming a total CDF of 10^{-5}/ry for BWRs:

$$\Delta\text{LERF} = 0.1 \times 10^{-5} \times \text{(multiplier based on duration of degraded condition)}$$

Using the multipliers given in Section 2.3 for each of the three (degraded condition) durations, the following three ΔLERFs and corresponding risk significance categories are obtained:

Duration	ΔLERF	Significance Category
> 30 days	10^{-6}	Yellow
30–3 days	10^{-7}	white
< 3 days	10^{-8}	green

If a finding identifies a degraded condition that could lead to suppression pool bypass and the duration of the degraded condition is known, then one of the significance categories given above can be assigned to the finding.

4.2.7 Main Steam Isolation Valve Leakage

Accidents that involve excessive leakage of the main steam isolation valves (MSIVs) can lead to releases that bypass the containment. Although excessive MSIV leakage has not been demonstrated to be a risk-significant accident in terms of contribution to LERF in risk assessments of BWRs with Mark II containments, it remains a potential bypass mechanism and should be considered in the SDP. Core melt accidents involving excessive MSIV leakage with the reactor coolant system at high pressure may be somewhat similar to induced steam generator tube ruptures in PWRs. Excessive leakage (potentially leading to early health effects) is defined as greater than 10,000 scfh passed through both the inboard and its associated outboard MSIV (PRAB-02-01) or the inability to quantify the leakage rate.

Consequently, if a finding reveals excessive MSIV leakage (as defined above), its significance category can be determined by assuming a bypass probability of 1.0 for all high-pressure core melt accidents and from the duration of the degraded condition. The conversion Factor for Type B findings, then, is the difference between the conditional probability of 1.0 and the original conditional probability of a large early release (i.e., 0.3 if the drywell is flooded). Assuming that the ratio of high-pressure to low-pressure sequences is about 70:30 based on IPE data, the conversion Factor for Type B findings is therefore approximately 0.7(1.0 - 0.3) = 0.49 for findings of this type. The risk significance can be determined by using the relationship given in Section 2.3 and assuming a total CDF of 10^{-5}/ry for BWRs:

$$\Delta\text{LERF} = 0.49 \times 10^{-5} \times \text{(multiplier based on duration of degraded condition)}$$

Using the multipliers given in Section 2.3 for each of the three (degraded condition) durations, the following three ΔLERFs and corresponding risk significance categories are obtained:

Duration	ΔLERF	Significance Category
> 30 days	4.9×10^{-6}	yellow
30–3 days	4.9×10^{-7}	white
< 3 days	4.9×10^{-8}	green

If a finding identifies a degraded condition that could lead to excessive MSIV leakage and the duration of the degraded condition is known, then one of the significance categories given above can be assigned to the finding.

4.2.8 Filtration Systems

Filtration systems, e.g., the standby gas treatment system (SGTS), remove particulates and, in some cases, condition the air. These systems are used extensively outside containment and require AC power to operate. Thus these systems are not important for station blackout sequences. Accidents sequences which are important for LERF considerations release a substantial portion of the reactor core radionuclide inventory. A large release of aerosols readily plugs filters and renders filtration systems ineffective. (Non-power accidents, i.e., fuel handling accidents, do not release sufficient quantities of radionuclides to result in an estimated prompt (or early) fatality and therefore do not influence LERF.) Finally, the release sequences that are of importance are those that bypass the suppression pool, e.g., containment (predominantly drywell) bypass. Therefore, filtration systems are not addressed further in this SDP, nor are control room air vents addressed.

5 BOILING WATER REACTORS WITH MARK III CONTAINMENTS

This chapter presents the technical bases for the risk significance categories recommended for BWRs with Mark III containments in Chapter 2. These containment designs rely on water in the suppression pool to condense steam and to scrub fission products released from the reactor coolant system (RCS). Mark III containments are similar in volume to ice condenser plants and have igniter systems installed to promote hydrogen combustion at low concentration levels, significantly below the detonation levels. As shown by the results of the Individual Plant Examination (IPE) program, BWRs have generally low core damage frequencies, on the average an order of magnitude lower than PWRs, due to the multiple ways of providing water to the core following an initiating event. The accident sequences in Mark III containments that contribute to LERF involve both early containment failure and bypass of the suppression pool. Accident sequences leading to releases that pass through the suppression pool are scrubbed, i.e., most of the fission products are retained in the pool: hence these releases are not large. Thus containment failures involving failures of the wetwell airspace alone will not be contributors to LERF.

This chapter follows the same format as previous chapters by discussing Type A and Type B findings separately.

5.1 Type A Findings

Type A findings are associated with accidents that have been assessed using the CDF-based SDP but which may influence the determination of LERF. Each of the accident classes impacted by the CDF-based SDP therefore has to be evaluated in terms its influence on LERF. This section describes the technical bases for the "Factors" recommended in Section 2.2 for Type A findings in BWRs with Mark III containments.

5.1.1 ISLOCA

An important insight from NUREG-1150, the IPE program, and NUREG/CR-5124 (an ISLOCA study) is that these accident sequences are not significant contributors to LERF for any of the BWR containment designs. The frequency of occurrence is low and the release path is tortuous, so significant fission product holdup and scrubbing in the compartments and buildings along the path of release to the environment are expected. Hence ISLOCA is not likely to be a contributor to LERF in BWR containments.

5.1.2 ATWS

Another important insight from the IPE program and numerous published PRAs is that ATWS accident sequences are not significant contributors to LERF for BWRs with Mark III containment designs. Early containment failure can occur only from ATWS sequences that overpressurize the containment wall before vessel breach due to excessive safety relief valve (SRV) discharge to the suppression pool. However, these sequences leave the drywell and suppression pool intact: hence the releases are scrubbed by the pool and a large early release does not occur. Therefore, LERF implications of ATWS sequences are not addressed by the LERF SDP.

5.1.3 Transients

This class of accidents includes a wide range of transient-initiated events, including station blackout (SBO) scenarios. Published PRAs and the IPE results were reviewed to determine if any of the attributes of these accidents might influence LERF determinations.

Mark III containments have a double-layer containment, with the drywell and suppression pool forming one layer and the outer containment structure (wetwell) the second layer. BWR Mark III containments also rely on the suppression pool to condense steam and scrub fission products released from the RCS during a severe accident. Releases that are scrubbed do not contribute to LERF (e.g., sequences in which the discharge is through the safety or relief valves and sequences with the drywell intact). Thus, the major contributors to LERF are accidents that fail the containment and the drywell or directly bypass the suppression pool.

Mark III containments have a significantly greater free volume than Mark I and Mark II plants but are not inert during operation. This containment design therefore has to depend on glow plug hydrogen igniters to control pressure loads resulting from hydrogen combustion events. If the igniters are not operating due to lack of AC power (the dominant sequence being a station blackout) or operator failure to manually actuate them, there is a possibility of an energetic hydrogen combustion (deflagration or detonation) event at the time of vessel failure (or at other times if the operators fail to follow procedures and the igniters are actuated when a significant amount of hydrogen has accumulated). These energetic combustion events were reported in NUREG/CR-1150 and the supporting documentation for Grand Gulf (NUREG/CR-4551, "Evaluation of Severe Accident Risks: Grand Gulf, Unit 1," Volume 6) to result in early containment failure with a relatively high conditional probability (~ 0.5). However, as noted above a large release requires failure of the drywell in addition to containment failure. Drywell failure can occur (1) directly as a result of loads associated with vessel breach or from hydrogen combustion, or (2) indirectly as a result of structural failure of the pedestal.

The only significant event that was found in NUREG/CR-4551, Volume 6, to cause drywell failure before vessel breach was hydrogen combustion in the wetwell. However, at the time of vessel breach loads from direct containment heating, ex-vessel steam explosions, hydrogen combustion, and reactor pressure vessel blow down contribute to the probability of drywell failure.

Structural failure of the pedestal can occur as a result of the loads accompanying vessel breach or from core-concrete interactions. The loads at vessel breach and the potential for core-concrete interactions depend on whether or not there is water in the pedestal prior to vessel breach. Analyses have shown that water will not overflow the weir wall. However, the latest revision of the Boiling Water Reactor Owners Group (BWROG) Emergency Procedures and Severe Accident Guidelines places greater emphasis on flooding the containment and the in-pedestal cavity. As a result, the in-pedestal cavity is more likely to be flooded prior to vessel failure. The top of the core debris (which is relocated core materials) with 100% of the corium (which is core debris plus all other materials that have left the reactor vessel such as steel) inside the in-pedestal cavity is 3.75 feet below the drywell floor elevation. The cavity walls are 23.5 feet thick with about 6 feet of concrete remaining below the debris (which is corium plus all CCI products and by products). Any ex-vessel steam explosion is unlikely to damage either the pedestal walls or the floor. The energy would be directed upward where it would impact the approximately 366 control rod drive tubes, instrument tubes, and the structural support steel beams before approaching the reactor vessel. Estimates of the dynamic effects on a BWR4 with a Mark I containment of steam explosions (both in-vessel and ex-vessel) indicate that there is inadequate energy to loft the reactor vessel or launch the reactor vessel head into containment (an alpha mode containment failure). It is expected that the results would be similar for steam explosions and significantly less for an alpha mode containment failure for BWR6s in Mark III containments. The potential for an ex-vessel steam explosion, while uncertain, is related to the pours of material from the reactor vessel into the water (the composition of the pours, the timing of the pours, and the rate the material enters the water). Based on experiments attempting to generate steam explosions, analytical predictions, and expert judgment, it was determined in NUREG/CR-4551, Volume 6, that ex-vessel steam explosion are unlikely to significantly affect the integrity of either the containment or the drywell. Accordingly, loads from high-pressure vessel breach and hydrogen combustion were determined to be the leading causes of containment and drywell failure.

The Grand Gulf results (NUREG/CR-4551, Volume 6) are summarized in the table below. This table indicates that accident sequences that contribute to LERF (which require failure of the drywell in addition to containment failure) are sensitive to the type of accident (i.e., SBO vs. non-SBO) and the pressure in the reactor pressure vessel at the time of vessel breach (i.e., transient vs. large break LOCA).

Table 5.1 Mark III Conditional Containment Failure Probabilities

	Station Blackout, SBO (Igniters and Sprays Unavailable)		Non-SBO (Igniters and Sprays Available)	
RCS Pressure at Vessel Breach	Containment Fails	Containment and Drywell Fail	Containment Fails	Containment and Drywell Fail
High	~ 0.5	~ 0.2	~ 0.5	~ 0.2
Low	~ 0.5	~ 0.2	~ 0.01 - 0.02	~ 0.01

As shown in the table, if the RCS is at high pressure the likelihood of containment failure is relatively independent of whether the igniters are operating. In addition, the likelihood of simultaneous failure of the drywell is also independent of igniter operation if the RCS is at high pressure. The important difference between SBO and non-SBO for high-RCS-pressure sequences relates to the availability of the sprays (not available for SBO, available for non-SBO sequences). Although spray availability does not impact the likelihood of containment and drywell failure as shown in the above table, it does scrub the containment atmosphere and thus reduces the quantity of radionuclides released (i.e., potentially making the release no longer large).

As the above table indicates, if the RCS is depressurized at vessel breach the likelihood of containment failure is dependent on whether the igniters are operating. If the igniters are not available, the conditional probability of containment failure is approximately 0.5 even with the RCS at low pressure. The likelihood of simultaneous failure of the drywell is also about 0.2 at the time of vessel breach. Thus all SBO sequences have a conditional probability of 0.2 of a large release, regardless of the pressure in the RCS.

The potential for containment failure at the time of vessel breach when the RCS is at low pressure and the igniters are operating is not directly assessed in NUREG/CR-4551, Volume 6. However, the conditions prior to vessel breach should be applicable to this situation because the RCS is depressurized and none of the issues associated with high pressure melt ejection (HPME) would occur. The results prior to vessel breach indicate a conditional probability of containment failure in the range of 0.01 to 0.02 if the igniters are operating.

In summary, the conditional probability for transient sequences with the RCS at high pressure and for all SBO sequences is close to 0.2 so that the Mark III containment fails at the same time that the suppression pool is bypassed. The conversion Factor for Type A findings is therefore 0.2 for all SBO sequences and for that fraction of the transient accident class that has a high RCS pressure during core meltdown. Thus if a finding related to any SBO or transient with high RCS pressure is processed through the CDF-based SDP, the risk significance should be evaluated for a potential increase because of LERF considerations.

However, if the RCS is depressurized and the igniters are operating, then the conditional probability is less than 0.1 that the Mark III containment will fail. The conversion Factor for a Type A finding is therefore <0.1 for that fraction of the transient accident class with a low RCS pressure during core meltdown and without a station blackout. The risk significance determined by the CDF-based SDP for transients with the RCS depressurized in BWR Mark III containments appears to be appropriate and need not be changed because of LERF considerations. The importance of the hydrogen igniters themselves is discussed separately in Section 5.2.3 as a Type B finding.

5.1.4 LOCAs

This class of accidents includes events initiated by a wide range of break sizes, which result in significantly different RCS responses. Published PRAs and the IPE results were reviewed to determine if any of the attributes of these accidents might influence LERF determinations. The RCS pressure during core meltdown was found to have the largest influence on LERF determinations. Thus, that fraction of accidents initiated by LOCAs that result in the highest RCS pressure (i.e. small-break LOCAs) needs to be assessed in terms of LERF considerations.

If a finding related to a small-break LOCA is processed through the CDF-based SDP, the risk significance should be examined for a potential increase because of LERF considerations. Small-break LOCA findings that would result in high RCS pressure at vessel breach should therefore be combined with transients with the RCS at high pressure and treated in exactly the same way (refer to Section 5.1.3).

5.2 Type B Findings

Type B of findings are associated with SSCs that do not impact the CDF determination and therefore have not been assessed using the CDF-based SDP. These findings, however, are potentially important to LERF determinations and do have to be allocated an appropriate risk category. This section presents the technical bases for the risk significance categories recommended for Type B findings in BWRs in Section 2.3.

5.2.1 Containment Penetration Seals, Isolation Valves, and Vent and Purge Lines

Mark III containments are not inert so containment leakage or loss of containment isolation may not be detected. In this context containment leakage and loss of containment isolation mean the containment (wetwell) atmosphere can communicate with the environment. Failure of these containment penetration seals, isolation valves, and vent and purge lines would not, of itself, result in suppression pool bypass.

The leakage may become important from a LERF perspective if the leakage rate is greater than 500 times the design basis leakage (L_a) (refer to the footnote in Section 3.2.1), but this assumes that the release is unscrubbed, which is not the case for Mark III containments. The effect of suppression pool bypass has to be taken into account when considering leakage from containments of this type. Suppression pool decontamination factors (expressed as the ratio of the amount of material released into the pool to the amount of material released from the pool) have been shown to vary as a function of time from a factor of 10,000 or more to something less than 100. A conservative factor of 10 has historically been used to represent a pool over the *entire* accident period. In terms of "early" this is *extremely* conservative. To determine the containment leakage criterion of importance (provided there is no significant pool bypass), the decontamination factor is multiplied by the containment leakage rate ($10 \times 500 \times L_a$) for a LERF-important leakage rate of $5000 \times L_a$. However, if there is enough drywell-to-wetwell bypass flow to make pool scrubbing ineffective, then $500 \times L_a$ is an appropriate criterion.

IPE information for Mark III plants indicates that, on average, about 25% of core damage frequency comprises early containment failure sequences and about 10% of the early containment failure sequences consist of large releases. Hence, on average, about 0.025 of the core damage frequency in Mark III BWRs contributes to LERF. The relatively low LERF value for these plants reflects the fact that containment failure has to be coupled with loss of drywell integrity for a large release to occur. The conversion Factor for Type B findings, therefore, is the difference between assuming complete loss of containment integrity coupled with loss of drywell integrity (i.e., 0.1) and the original early failure probability (0.025) (i.e., (0.1 - 0.025) = 0.0975). The risk significance can be determined by using the relationship given in Section 2.3 and assuming a total CDF of 10^{-5}/ry for BWRs:

$$\Delta \text{LERF} = 0.0975 \times 10^{-5} \times \text{(multiplier based on duration of degraded condition)}$$

Using the multipliers given in Section 2.3 for each of the three (degraded condition) durations, the following three ΔLERFs and corresponding risk significance categories are obtained:

Duration	ΔLERF	Significance Category
> 30 days	9.8×10^{-7}	white
30–3 days	9.8×10^{-8}	green
< 3 days	9.8×10^{-9}	green

If a finding identifies a degraded SSC that could lead to a containment leakage rate in excess of 500 times L_a and the duration of the degraded condition is known, then one of the significance categories given above can be assigned to the finding.

5.2.2 Containment Sprays

As shown in Section 5.1.3 above, a large early release in Mark III containments requires both early containment failure and loss of drywell integrity, which are mainly caused by hydrogen combustion events. SBO accidents, in which the hydrogen igniters are unavailable, and high-pressure scenarios (with the igniters operating) are the dominant contributors to these failure modes.

The pressure pulse from hydrogen combustion is too rapid for the sprays to have a significant impact on averting containment failure. In principle, the sprays could make an impact by scrubbing the containment atmosphere and thus converting a potentially "large release" into a non-large release.

However, the effectiveness of the sprays would be extremely uncertain under these conditions. The spray pumps are driven by AC power, and thus are not available for SBO accident sequences. However, BWR licensees have modified their spray systems to permit the containment sprays to operate with the diesel-drive fire suppression system water pump. This pump is manually connected to the RHR line that supplies the containment spray header. When AC power is available, spray operation can be beneficial for accidents that fail containment. In these cases, timing is an important issue. The sprays can be operated manually or actuated automatically at a predetermined containment pressure or temperature or a high-radiation signal. Containment spray systems can also be useful for mitigating severe accidents. For example, they could be used to provide fission product scrubbing and containment cooling as core materials leave a failed reactor vessel.

IPE information (refer to Section 5.2.1) indicates that, on average, about 0.025 of the core damage frequency in Mark III BWRs contributes to LERF. Spray operation, if available and effective, could potentially mitigate the fraction of the CDF that contributes to LERF. The risk significance of a finding related to spray operation therefore conservatively approaches a conditional probability of 0.025 for a large early release, assuming that the sprays are effective for mitigating the LERF contributors.

The conversion Factor for a Type B finding is therefore 0.025 for a finding related to spray operation. The risk significance can be determined by using the relationship given in Section 2.3 and assuming a total CDF of 10^{-5}/ry for BWRs:

$$\Delta LERF = 0.025 \times 10^{-5} \times \text{(multiplier based on duration of degraded condition)}$$

Using the multipliers given in Section 2.3 for each of the three (degraded condition) durations, the following three ΔLERFs and corresponding risk significance categories are obtained:

Duration	ΔLERF	Significance Category
> 30 days	3×10^{-7}	white
30–3 days	3×10^{-8}	green
< 3 days	3×10^{-9}	green

If a finding identifies a degraded condition that could impact spray operation and the duration of the degraded condition is known, then one of the significance categories given above can be assigned to the finding.

5.2.3 Hydrogen Igniters

This section deals with findings related to how well the igniter system is operating (i.e., are all the glow plugs functioning?). Findings that cause the complete loss of the system (e.g., due to loss of AC power) are dealt with as Type A findings in Section 5.1.3.

If a finding implies that a portion of the glow plug igniter system is inoperable, the only impact on the probability of early containment failure will be for non-SBO sequences in which the RCS is depressurized. All SBO sequences and sequences with the RCS at high pressure have a conditional probability of early containment failure and simultaneous failure of the drywell close to 0.2. As discussed in Section 5.1.3, the probability of early containment failure from hydrogen combustion events for non-SBO sequences with the RCS depressurized is close to 0.01. This is because the igniters are operating and burn the hydrogen at low concentrations as it released from the top of the suppression pool with no containment-challenging pressure spike.[4] If some of the igniters were not operating, the local concentration of hydrogen would increase until it was ignited, either by a working igniter elsewhere or random ignition (e.g., static discharge). At the extreme, if none of the

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4 This has been demonstrated by industry experiments (quarter-scale Hydrogen Control Owners Group proprietary tests that showed a pressure rise of only about 3 psig in the wetwell. The tests were verified by NRC independent calculations which were peer-reviewed (NUREG/CR-5571, Figure 5.5.1)). It should be noted that three NRC teams (two separate reviews for NUREG-1150 and one for the Containment Performance Improvement program) have visited the Grand Gulf site and none of them could identify any possible ignition source (other than the igniters) inside containment.

igniters were operating, the probability of early containment failure from non-SBO depressurized sequences would approach 0.2 from hydrogen detonation or energetic deflagration.

Another consideration is the locations of the non-operating igniters. For example, if failure of multiple igniters results in loss of igniter coverage in two adjacent regions along the release path of hydrogen, the concentration of hydrogen would become higher before ignition than if the failed igniters were uniformly distributed throughout containment, or located in a stairwell. Since no studies have been performed on the sensitivity of containment failure to the number of inoperative igniters and their placement, both in relation to each other and to the expected path of hydrogen, it is assumed that having all the igniters in two adjacent regions inoperative at any time would significantly increase the probability of early containment failure from hydrogen combustion events.

The conversion Factor for Type B findings is therefore 0.2 - 0.01 = 0.19 for findings that imply loss of igniter system effectiveness. The risk significance can be determined using the relationship given in Section 2.3 and by assuming a contribution to CDF for those accidents where the igniters are normally effective in preventing containment and drywell failure, namely non-SBO sequences with the RCS at low pressure at the time of vessel breach (i.e., LOCAs and transients with the RCS depressurized). For the four BWR6 plants that are housed in Mark III containments, the IPE report (NUREG-1560) reveals that the fraction of core damage frequency arising from SBO is about 50% (ranging from 15% to 90%). The IPE database on the plant damage states for these plants was searched to determine the fraction of plant damage states (PDSs) that have low RCS pressure. The average across the four plants for PDSs with this attribute is approximately 40%. Assuming that this fraction applies to both SBO and non-SBO sequences, the contribution to CDF from non-SBO sequences with RCS at low pressure is about 0.5 x 0.4 = 0.2. Hence, if a total CDF of 10^{-5}/ry is assumed, the contribution to LERF from non-SBO sequences with the RCS depressurized is about 2×10^{-6}/ry, so the risk significance of the loss of igniter system effectiveness is:

$$\Delta\text{LERF} = 0.19 \times 2 \times 10^{-6} \times \text{(multiplier based on duration of degraded condition)}$$

Using the multipliers given in Section 2.3 for each of the three (degraded condition) durations, the following three ΔLERFs and corresponding risk significance categories are obtained:

Duration	ΔLERF	Significance Category
> 30 days	4×10^{-7}	white
30–3 days	4×10^{-8}	green
< 3 days	4×10^{-9}	green

If a finding identifies that more than 10% of the hydrogen igniters would be unable to perform their intended function and the duration of the degrade condition is known, then one of the significance categories given above can be assigned to the finding. Further analyses would be needed to assign a more realistic risk significance to any of the findings that exceed the failure criterion.

5.2.4 Suppression Pool Cooling

Heat removal is needed to maintain the suppression pool temperature within its operating limits. Failure of the cooling system could eventually result in containment over pressure. In this particular sequence, however, core degradation typically takes about 30 hours. Thus loss of suppression pool cooling is unlikely to influence LERF. Therefore findings related to loss of suppression pool cooling should be processed through the CDF-based SDP in order to assign a risk significance category.

Another potential concern is plugging of the suction strainers such that pool cooling or reactor vessel injection is lost. Resolution of this problem is documented in NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors," Generic Letters 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," and 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss of Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," and Regulatory Guide 1.82, Revision 2, published in May 1996. Therefore findings related to potential plugging of the strainers are not anticipated. Findings of this nature are not expected to influence LERF. However, if a finding does occur it should be processed through the CDF-based SDP in order to assign a significance category, and to determine if a further assessment for LERF consideration is needed.

5.2.5 Fan Coolers

Fan coolers are normally operating. They are designed to maintain the containment atmosphere within the plant technical specification operating limits. They are not capable of removing large amounts of heat deposited into containment over short periods from a pipe break or vessel failure. The fan coolers have minimal capability to remove fission products and no credit is given for any such removal. Therefore, the unavailability of the fan coolers may result in operational problems but will not significantly affect the timing or quantity of fission products released to the environment as the result of a severe core damage accident. Thus, the fan coolers are not an important factor in LERF determinations.

5.2.6 Containment Flooding Systems

Some plants have an upper pool dump capability. The pool dump provides a supplemental water supply for the suppression pool and causes it to overflow the weir wall into the drywell to provide partial flooding of the drywell. However, preventing or reducing the amount of water that could be dumped into the suppression pool is unlikely to influence the LERF determination. A finding related to upper pool dump does not therefore warrant a LERF-based risk category.

5.2.7 Suppression Pool Bypass

Suppression pool bypass is a failure of some component or structure that would limit the ability of the suppression pool to perform its intended safety function. An example of such a finding is a large drywell vent line open. The bypass has to be shown to prevent most of the flow from passing through the suppression pool, e.g., drywell leakage is not suppression pool bypass (in the context of this report) because most of the flow would (or could) pass through the suppression pool, preventing a large release.

As stated above in Section 5.2.1, IPE information for Mark III plants indicates that, on average, about 25% of core damage frequency consists of early containment failure sequences and about 10% of the early containment failure sequences consist of large releases. Hence, on average, about 0.025 of the core damage frequency in Mark III BWRs contributes to LERF. If a finding on an SSC impacts suppression pool integrity and implies suppression pool bypass, then the large release probability 0.025 increases to about 0.25, because all of the early failure sequences are potentially unscrubbed.

The conversion Factor for Type B findings is therefore 0.25 - 0.025 = 0.23 for suppression pool bypass findings. The risk significance can be determined by using the relationship given in Section 2.3 and assuming a total CDF of 10^{-5}/ry for BWRs:

$$\Delta\text{LERF} = 0.23 \times 10^{-5} \times (\text{multiplier based on duration of degraded condition})$$

Using the multipliers given in Section 2.3 for each of the three (degraded condition) durations, the following three ΔLERFs and corresponding risk significance categories are obtained:

Duration	ΔLERF	Significance Category
> 30 days	2×10^{-6}	yellow
30–3 days	2×10^{-7}	white
< 3 days	2×10^{-8}	green

If a finding identifies a degraded condition that could lead to suppression pool bypass and the duration of the degraded condition is known, then one of the significance categories given above can be assigned to the finding.

5.2.8 Main Steam Isolation Valve Leakage

MSIV leakage is only applicable to BWRs with Mark I and II containments. BWRs with Mark III containments have a safety-grade main steam shutoff valve (MSSV) outside of the outboard MSIV. The MSSV is a relatively slow-closing, low-leakage valve. Thus, any leakage past the MSIVs in a main steam line of a Mark III plant is stopped at the MSSV.

5.2.9 Filtration Systems

Filtration systems, e.g., the standby gas treatment system (SGTS), remove particulates and, in some cases, condition the air. These systems are used extensively outside containment and require AC power to operate. Thus these systems are not important for any accident scenario which results in the loss of AC power, i.e., station blackout sequences. Accident sequences which are important for LERF considerations are those which release a substantial portion of the reactor core radionuclide inventory. Such large releases of aerosols readily plug filters and render filtration systems ineffective. (Non-power accidents, i.e., fuel handling accidents, do not release sufficient quantities of radionuclides to result in an estimated prompt (or early) fatality and therefore are not expected to influence LERF.) Finally, the release sequences that are of importance are those that bypass the suppression pool, e.g., containment bypass. Therefore, filtration systems are not addressed further in this SDP. Nor are filters on the control room air vents addressed.

6 PRESSURIZED WATER REACTORS WITH LARGE-VOLUME AND SUBATMOSPHERIC CONTAINMENTS

This chapter presents the technical bases for the risk significance categories recommended for PWRs with large-volume and subatmospheric containments in Chapter 2. The large dry containments rely on large internal volumes and high design pressure capability to mitigate the consequences of design basis and core damage accidents. Subatmospheric containments are operated at a lower pressure than the (outside) atmospheric pressure to reduce the long-term containment pressure due to accidents and to reduce fission product releases to the environment. This design feature reduces the likelihood of pre-existing leaks and containment isolation failures.

This chapter follows the same format as previous chapters by discussing Type A and Type B findings separately.

6.1 Type A Findings

Type A findings are associated with accidents that have been assessed using the CDF-based SDP but which may influence the LERF determination. Each of the accident classes impacted by the CDF-based SDP therefore has to be evaluated in terms of its influence on LERF. This section describes the technical bases for the "Factors" recommended for Type A findings in PWRs with large-volume and subatmospheric containments.

6.1.1 ISLOCA

This accident scenario occurs when isolation valves between the high-pressure RCS and a low-pressure secondary system fail, causing a loss-of-coolant accident (LOCA) outside of containment. If adequate coolant makeup is eventually lost, core damage can occur and, since the release path bypasses containment, a large fraction of the radionuclides can be released to the environment. An important insight from the IPE program and the ISLOCA study is that these accident sequences are significant contributors to LERF for all of the PWR containment designs.

Since the containment is bypassed for ISLOCA sequences, the conversion Factor for Type A findings is 1.0 for this accident class. Therefore, if a finding related to this accident class is processed through the CDF-based SDP, the risk significance (i.e., color assignment) should be increased by one order of magnitude.

6.1.2 ATWS

Another important insight from the IPE program and numerous published PRAs is that ATWS accident sequences are not significant contributors to LERF for PWRs. This is for two reasons. First, the frequency of ATWS scenarios is very low. Secondly, for the few plants that performed an analysis of an ATWS sequence, the containment pressure increased slowly enough to be a late containment failure mode. Therefore, the risk significance determined by the CDF-based SDP for ATWS events in PWRs is appropriate and need not be changed because of LERF considerations.

6.1.3 SGTR

This section addresses SGTRs as initiating events that can lead, after further failures or refueling water storage tank (RWST) depletion, to core damage. If core melt occurs, and the secondary side is open, a release path can exist that bypasses containment. With an SGTR as an initiating event, the impact on CDF is addressed within the context of the CDF-based SDP. It should be noted that this sequence is different from other Type A findings in that it is related to the integrity of only one component, the steam generator tube. Other Type A findings (e.g., ISLOCA findings) relate to one component out of many different components and paths.

Since the containment is bypassed for SGTR sequences, the conversion Factor for Type A findings is 1.0 for this accident class. Therefore, if a finding related to this accident class is processed through the CDF-based SDP, the risk significance (i.e. color assignment) should be increased by one order of magnitude.

The assessment of SGTR as a large early release is conservative. The SGTR fraction of CDF includes contribution from accident sequences where the relief valves do close and where core damage occurs well after emergency evacuation has been completed and thus no prompt fatality is expected. Furthermore, preliminary calculations indicate that at least some SGTR sequences with direct release to the environment do not result in any prompt fatalities. The potential decontamination factor associated with steam generators and piping to the atmospheric release valves is currently being evaluated. It will further reduce the possibility of a SGTR resulting in any prompt fatality.

6.1.4 Transients

This class of accidents includes a wide range of transient-initiated events, including station blackout (SBO) scenarios. Published PRAs and the IPE results were reviewed to determine if any of the attributes of these accidents might influence LERF determinations.

As noted above, PWRs with large dry and subatmospheric containment designs rely on large internal volumes and plant design features to mitigate the consequences of design basis accidents (DBAs). Subatmospheric containments are maintained below atmospheric pressure during normal plant operation and rely on sprays to return the containment to subatmospheric conditions during accidents, reducing the potential release of radioactive fission products to the environment. The containment size and design pressure for subatmospheric designs are comparable to those for large dry containments. As there is significant margin between the design pressures and the ultimate capacities of these containment structures, both of these designs are extremely robust when subjected to the harsh conditions associated with core damage accidents.

An important insight from the IPE program and other published PRAs is that core damage accidents in which the RCS remains at high pressure during core meltdown and up to the time of reactor pressure vessel failure result in the most severe challenges to the structural integrity of these two containment designs. However, the probability of early containment failure resulting from the various challenges has been determined to be low for these containments. In general, the probability was reported to be less than 0.1 conditional on the occurrence of core damage accident scenarios at high pressure. If the RCS is depressurized during core meltdown, the probability of early containment failure is expected to be less than 0.01.

For example, the staff has reviewed all large dry and subatmospheric PWR designs and found that the conditional containment failure probability given a high-pressure melt ejection was less than 0.01 (NUREG-CR-6338, NUREG/CR-6475). Thus, the staff has concluded that direct containment heating from high-pressure melt ejection is resolved for plants with large dry and subatmospheric containments and, therefore, not important to risk. Many workshops have been held where experts have evaluated the potential for in-vessel steam explosions and their potential effects. The conclusion of these workshops is that in-vessel steam explosions are highly unlikely and given a steam explosion, there is insufficient energy to launch the reactor head or the vessel as a rocket and damage containment (NUREG-1524). Ex-vessel fuel-coolant interactions that could lead to a steam explosion are extremely dependent on plant design (geometry), pool condition (size, shape, temperature, and pressure), and corium condition (temperature, pour rate, and pour composition) and are highly uncertain and unpredictable. The only consideration that could affect the potential for energetic fuel-coolant interactions is whether water can be put below the reactor vessel.

The published data on the likelihood of early containment failure for containments of this design indicates that the conditional failure probability is less than 0.1 whether or not the RCS is at high pressure. The conversion Factor for a Type A finding is therefore <0.1 for all accidents initiated by transient events. The risk significance determined by the CDF-based SDP for all transients in PWRs with this type of containment design appears to be appropriate and need not be changed because of LERF considerations.

6.1.5 LOCAs

This class of accidents includes events initiated by a wide range of break sizes, resulting in significantly different RCS responses. Published PRAs and the IPE results were reviewed to determine if any of the attributes of these accidents might influence LERF determinations. Nothing was found in the published data that would suggest that accidents initiated by LOCAs should be treated differently than transient events. The risk significance determined by the CDF-based SDP for all LOCAs in PWRs is therefore appropriate and need not be changed because of LERF considerations.

6.2 Type B Findings

Type B findings are associated with SSCs that do not impact the CDF determination and, therefore, the findings have not been assessed using the CDF-based SDP. These findings, however, are potentially important to LERF determinations and do have to be allocated an appropriate risk category. This section presents the technical bases for the risk significance categories recommended for Type B findings in PWRs in Section 2.3.

6.2.1 Penetration Seals, Isolation Valves, and Purge and Vent Lines

An important insight from the IPE program and other published PRAs is that containment leakage or loss of containment isolation accident sequences do not significantly contribute to the LERF for plants with subatmospheric containments. As the atmosphere in these containments has to be maintained below atmospheric pressure during operation, such leaks or isolation failures are readily identified by excessive operation of the vacuum pumps or difficulty in maintaining a subatmospheric pressure inside containment.

PWR large dry containments, however, do not have a controlled environment and so containment leakage or loss of containment isolation may not be detected. The leakage, from a LERF prospective, may become important if the leakage rate to the environment is greater than 10 times the design basis leakage (L_a) (see footnote to Section 3.2.1). As this failure mode opens a direct path outside of containment, the conditional probability of a large early release approaches 1.0 for leakage or isolation failures. Since the average conditional probability of early failure, including bypass, is about 0.1 in these containments, the conversion Factor for Type B findings is therefore (1.0 - 0.1) = 0.9 for leakage or isolation failures. The risk significance can be determined by using the relationship given in Section 2.3 and assuming a total CDF of 10^{-4}/ry for PWRs:

$$\Delta\text{LERF} = 0.9 \times 10^{-4} \times \text{(multiplier based on duration of degraded condition)}$$

Using the multipliers given in Section 2.3 for each of the three (degraded condition) durations, the following three ΔLERFs and corresponding risk significance categories are obtained:

Duration	ΔLERF	Significance Category
> 30 days	9.0×10^{-5}	red
30–3 days	9.0×10^{-6}	yellow
< 3 days	9.0×10^{-7}	white

If a finding identifies a degraded SSC that could lead to a high containment leakage rate in excess of 1000 times L_a and the duration of the degraded condition is known, then one of the significance categories given above can be assigned to the finding.

6.2.2 Containment Sprays

Although containment sprays are effective in removing fission products from the containment atmosphere, they were designed for the thermal conditions associated with the design basis LOCA. The sprays, therefore, are not effective in preventing early containment failures associated with core melt accident sequences. The spray pumps are operated by AC power, and thus are not available for SBO accident sequences. When AC power is available, spray operation can be beneficial for those accidents that fail containment by scrubbing fission products from the atmosphere. In these cases, timing is an important issue. The sprays must be operated when the fission products are in the atmosphere to be scrubbed. The sprays can be operated manually or actuated automatically at a predetermined containment pressure and temperature and a high-radiation signal (Westinghouse Owners Group (WOG)). The pressure or temperature setting may not activate the sprays at the time of the release of radioactivity to the environment (WOG). If the sprays are available during high radiation inside containment, they reduce the atmospheric source term available to be released to the environment.

As noted above in Section 6.1.4, the likelihood of early containment failure is < 0.1, so the risk significance determined by the CDF-based approach is applicable to these containment designs. Under these circumstances the operability of the spray system has no impact on the SDP.

6.2.3 Fan Coolers

The severe accident management (SAM) guidelines (WOG) instruct the operators to start the fan coolers, if they are available, when the containment atmosphere reaches a pressure, temperature, or high-radiation limit. The fan coolers were designed for slowly increasing temperature conditions, such as heat loss from the reactor coolant system. In some cases, the fan coolers are safety grade, i.e., designed for design basis accident heat losses to the atmosphere. Some of the fan coolers have filters which may collect particulates, including fission products. However, under severe accident conditions, these filters may quickly become clogged and prevent or greatly reduce fan cooler flow. Because the flow rates through the fan coolers are relative low, potential fission product removal by the fan coolers is not expected to significantly affect the potential atmospheric source term available for release to the environment.

LERF is also not influenced by the fan coolers because of the nature of the dominant challenges to containment integrity that can occur during a severe core damage accident scenario. These challenges are characterized by the very short dynamic duration of hydrogen deflagration and detonation events. The challenge to containment integrity from these short-duration events cannot be effectively mitigated by the active heat removal systems (i.e., fan coolers) because of the relatively long response time of these systems. In addition, the likelihood of a large release is low enough in plants with these containments that the operability or availability of fan coolers will have no appreciable impact on LERF.

Consequently, LERF is not significantly affected by the operability of these systems.

7 PRESSURIZED WATER REACTORS IN ICE CONDENSER CONTAINMENTS

This chapter presents the technical bases for the risk significance categories recommended for PWRs with ice condenser containments in Chapter 2. Ice condenser containments have a "pressure suppression" design and rely on ice in the containment to condense steam released from the RCS during an accident. Ice condenser containments have smaller volumes and lower design pressures than large dry and subatmospheric designs.

This chapter follows the same format as previous chapters by discussing Type A and Type B findings separately.

7.1 Type A Findings

Type A findings are associated with accidents that have been assessed using the CDF-based SDP but which may influence the LERF determination. Each of the accident classes impacted by the CDF-based SDP therefore has to be evaluated in terms its influence on LERF. This section describes the technical bases for the "Factors" recommended for Type A findings in PWRs with ice condenser containments.

7.1.1 ISLOCA

This accident scenario occurs when isolation valves between the high-pressure RCS and a low-pressure secondary system fail, causing a loss-of-coolant accident (LOCA) outside of containment. If adequate coolant makeup is eventually lost, core damage can occur and, since the release path bypasses containment, a large fraction of the radionuclides can be released to the environment. An important insight from the IPE program and the ISLOCA study is that these accident sequences are significant contributors to LERF for all of the PWR containment designs.

As the containment is bypassed for ISLOCA sequences, the conversion Factor for Type A findings is 1.0 for this accident class. Therefore, if a finding related to this accident class is processed through the CDF-based SDP, the risk significance (i.e. color assignment) should be increased by one order of magnitude.

7.1.2 ATWS

Another important insight from the IPE program and numerous published PRAs is that ATWS accident sequences are not significant contributors to LERF for PWRs. This is for two reasons. First, the frequency of ATWS scenarios was very low. Secondly, for the few plants that performed an analysis of an ATWS sequence, the containment pressure increased slowly enough to be a late containment failure mode. Therefore, the risk significance determined by the CDF-based SDP for ATWS events in PWRs is appropriate and need not be changed because of LERF considerations.

7.1.3 SGTR

SGTRs are initiating events that can lead, after further failures or RWST depletion, to core damage. If core melt occurs, and the secondary side is open, a release path can exist that bypasses containment. With an SGTR as an initiating event, the impact on CDF is addressed within the context of the CDF-based SDP. It should be noted that this sequence is different from other Type A findings in that it is related to the integrity of only one component, the steam generator tube. Other Type A findings (e.g., ISLOCA findings) relate to one component out of many different components and paths.

Since the containment is bypassed for SGTR sequences, the conversion Factor for Type A findings is 1.0 for this accident class. Therefore, if a finding related to this accident class is processed through the CDF-based SDP, the risk significance (i.e., color assignment) should be increased by one order of magnitude.

The assessment of SGTR as a large early release is conservative. The SGTR fraction of CDF includes contribution from accident sequences where the relief valves do close and where core damage occurs well after emergency evacuation has been completed and thus no prompt fatality is expected. Furthermore, preliminary calculations indicate that at least some SGTR sequences with direct release to the environment do not result in any prompt fatalities. The potential decontamination factor associated with steam generators and piping to the atmospheric release valves is currently being evaluated. It would further reduce the possibility of a SGTR resulting in any prompt fatality.

7.1.4 Transients

This class of accidents includes a wide range of transient-initiated events, including station blackout (SBO) scenarios. Published PRAs and the IPE results were reviewed to determine if any of the attributes of these accidents might influence LERF determinations.

These containment designs are pressure suppression designs as they rely on ice in the upper compartment of the containment to condense steam and mitigate the consequences of design basis accidents (DBAs). The internal volume and design pressure of containment are determined by assuming that the steam released from a large-break LOCA in the RCS is condensed in the ice chest. Like BWR Mark III containments, ice condensers rely on glow plug igniters to control hydrogen released during postulated accidents in which the core is damaged. The igniter systems are designed to burn hydrogen at low concentrations and thus reduce the potential for large deflagrations or detonations that could challenge containment integrity.

Phenomena which can challenge the structural integrity of ice condenser containments include hydrogen combustion during high- and low-pressure sequences, in-vessel steam explosions, and rapid steam generation caused, for example, by core debris contacting water in the reactor cavity. The IPE program results show that for ice condenser containments the most severe challenges to containment structural integrity arise from core damage accidents in which the RCS remains at high pressure during core meltdown and reactor pressure vessel failure. Important containment failure mechanisms associated with high-pressure melt ejection include hydrogen combustion associated with direct containment heating (NUREG/CR-6427) and impingement of molten corium on the containment wall.

Table 7.1 Conditional Probability of Containment Over Pressure Failure in Ice Condenser Containment Plants

Plant	Direct Containment Heating		Non-Direct Containment Heating			
	SBO	Non-SBO	SBO		Non-SBO	
			H_2 Burn	Steam Spike	H_2 Burn	Steam Spike
Catawba	1.00	0.0	0.29	0.0	0.0	0.0
D.C. Cook	0.82	0.0	0.93	0.0	0.0	0.08*
McGuire	0.98	0.0	0.55	0.0	0.0	0.0
Sequoyah	0.99	0.0	0.97	0.0	0.0	0.0
Watts Bar	0.99	0.0	0.22	0.0	0.0	0.0

* Spray inoperable after 36 minutes due to recirculation failure.

Ref: NUREG/CR-6427, "Assessment of the DCH Issue for Plants With Ice Condenser Containments."

Sandia National Laboratories (SNL) recently carried out a detailed study (NUREG/CR-6427) of severe accident phenomena in ice condenser containment plants. The study focused on the direct containment heating (DCH) issue. The study considered all the significant early containment failure mode issues examined in NUREG-1150, including (1) DCH over pressure failures, (2) thermal failures of the containment liner resulting from accumulation of the dispersed debris against the containment liner following high-pressure melt ejection, (3) non-DCH hydrogen combustion over pressure failures in scenarios where core damage is arrested in-vessel or when the reactor pressure vessel fails at low pressure, and (4) non-DCH steam spike over pressure failures when the vessel lower head fails at low (<200 psi) RCS pressures. An evaluation of the containment event trees used in the plant IPEs showed that all ice condenser plants, except McGuire, have an early containment failure probability in the range of 0.35% to 5.8%. The early containment failure probability for McGuire was found to be 13.9%, which is dominated by a higher SBO probability.

In assessing the containment response to severe accidents, the study used the CONTAIN code which had been extensively validated for predictions of containment response to steam sources and non-DCH-related hydrogen combustion deflagration. This validation includes International Standard Problem (ISP) 16 for the Heissdampf Reactor steam source tests and the Nevada Test Site large-scale hydrogen combustion tests (NUREG/CR-6533). The DCH modeling has been extensively assessed with the NRC-sponsored DCH tests at SNL and elsewhere (NUREG/CR-4896, NUREG/CR-5586, LA-12866, 1995 American Nuclear Society Meeting). The results of the SNL study are shown above in Table 7.1.

The results of the CONTAIN calculations indicate that the ice condenser containment integrity is not challenged except for SBO (no igniters available) accident sequences that are associated with high hydrogen concentrations. This conclusion holds broadly, as shown in Table 7.1, for both DCH and non-DCH events regardless of whether the reactor pressure vessel is at low or high pressure.

For non-DCH events, the average conditional containment failure probability (CCFP) for SBO sequences is about 0.6 (range of 0.2 to 0.97, depending on plant) from hydrogen combustion events and zero from steam spikes. For DCH events, the corresponding range of CCFP from SBO sequences is from 0.82 to 1.00. Since this SDP is intended for screening purposes only, separately considering the CCFP of non-DCH SBO sequences (0.6) and of DCH SBO sequences (1.0) is not considered necessary.

Based on the above data, ice condenser containments are predicted to fail early with a conditional probability close to unity during SBO transients. Therefore, as the containment fails early for these sequences, the conversion Factor for Type A findings is 1.0 for this accident class. Accordingly, in evaluating the LERF implications of a Type A finding that impacts the frequency of SBO sequences, the risk significance of the finding assigned under the LERF-based SDP should be one order of magnitude higher, i.e., assigned one color more severe, than the ranking (and color) obtained from the CDF-based SDP.

For non-SBO events, in both DCH and non-DCH accident scenarios, the conditional containment failure probability is zero for hydrogen combustion and less than 0.1 (0.0 to 0.08) for steam spikes (NUREG/CR-6427). This is because with the ice available, the ice condenser is able to mitigate any of the accident scenarios. The published data on the likelihood of early containment failure for containments of this design indicates that the conditional failure probability is less than 0.1 if the igniters are operating. The conversion Factor for a Type A finding is therefore < 0.1 for all accidents initiated by transient events, except SBO. The risk significance determined by the CDF-based SDP for transients in PWRs with this type of containment design appears to be appropriate and need not be changed because of LERF considerations.

7.1.5 LOCAs

This class of accidents includes events initiated by a wide range of break sizes, resulting in significantly different RCS responses. Published PRAs and the IPE results were reviewed to determine if any of the attributes of these accidents might influence LERF determinations. Based on the discussion provided above in Section 7.1.4 (refer to Table 7.1), the most important factor affecting containment survivability, and hence LERF, is the availability of igniters. The RCS pressure during core meltdown was found not to influence the likelihood of containment failure provided the igniters were operating. The conversion Factor for a Type A finding is therefore < 0.1 for all accidents initiated by LOCA events. The risk significance determined by the CDF-based SDP for LOCAs in PWRs with this type of containment design appears to be appropriate and need not be changed because of LERF considerations.

7.2 Type B Findings

Type B findings are associated with SSCs that do not impact the CDF determination and therefore have not been assessed using the CDF-based SDP. These findings, however, are potentially important to LERF determinations and do have to be allocated an appropriate risk category. In this section the technical bases for the risk significance categories recommended for Type B findings in PWRs in Section 2.3 are presented.

7.2.1 Penetration Seals, Isolation Valves, and Purge and Vent Lines

PWR ice condenser containments do not have a controlled environment and so containment leakage or loss of containment isolation may not be detected. The leakage, from a LERF perspective, may become important if the leakage rate to the environment is greater than 10 times the design basis leakage (L_a) (see footnote in Section 3.2.1). As this failure mode opens a direct path outside of containment, the conditional probability of a large early release is therefore 1.0 for leakage or isolation failures. Since the conditional probability of early failure and bypass on average is approximately 0.1 for this containment, the conversion Factor for Type B findings is about (1.0 - 0.1) = 0.9 for leakage and isolation failures. The risk significance can be determined by using the relationship given in Section 2.3 and assuming a total CDF of 10^{-4}/ry for PWRs:

$$\Delta\text{LERF} = 0.9 \times 10^{-4} \times \text{(multiplier based on duration of degraded condition)}$$

Using the multipliers given in Section 2.3 for each of the three (degraded condition) durations, the following three ΔLERFs and corresponding risk significance categories are obtained:

Duration	ΔLERF	Significance Category
> 30 days	9.0×10^{-5}	red
30–3 days	9.0×10^{-6}	yellow
< 3 days	9.0×10^{-7}	white

If a finding identifies a degraded SSC that could lead to a high containment leakage rate in excess of 1000 times L_a and the duration of the degraded condition is known, then one of the significance categories given above can be assigned to the finding.

7.2.2 Containment Sprays

Although containment sprays are effective in removing fission products from the containment atmosphere, they were designed for the thermal conditions associated with the design basis LOCA. The spray pumps are operated by AC power, and thus are not available for the risk-significant SBO accident sequences. When AC power is available, spray operation can be beneficial for those accidents that fail containment by scrubbing fission products from the atmosphere. In these cases, timing is an important issue. The sprays must be operated when the fission products are in the atmosphere to be scrubbed. The sprays can be operated manually or actuated automatically at a predetermined containment pressure and temperature and at a high-radiation level (WOG). The pressure or temperature setting may not activate the sprays at the time of the release of radioactivity to the environment. If the sprays are available during high radiation inside containment, they reduce the atmospheric source term available to be released to the environment.

As shown above in Section 7.1.4, the dominant early containment failure phenomenon and contributor to LERF in ice condenser containments is hydrogen combustion during SBO accidents when the igniters are inoperable. Since the sprays are not available during SBO accidents, LERF is not significantly affected by the operability of these systems.

7.2.3 Fan Coolers

LERF is not influenced by the fan coolers because of the nature of the dominant challenges to containment integrity that can occur during a severe core damage accident scenario. These challenges are characterized by the very short dynamic duration of hydrogen deflagration or detonation events. The challenge to containment integrity from these short-duration events cannot be effectively mitigated by the active heat removal systems (i.e., fan coolers) because of the relatively long response time of these systems. Consequently, LERF is not significantly affected by the operability of these systems.

7.2.4 Hydrogen Igniters

This section deals with findings related to the functioning of the glow plug igniter system when power is available. Findings that relate to unavailability of the igniter system in a SBO event are dealt with as Type A findings in Section 7.1.4.

If a finding implies that a portion of the glow plug igniter system is inoperable, the only impact on the probability of early containment failure will be for non-SBO sequences. (All SBO sequences have a conditional probability of early containment failure close to unity.) As discussed in Section 7.1.4, the probability of early containment failure from hydrogen combustion events during non-SBO sequences is zero. This is because the igniters are operating and burning the hydrogen at low concentrations as it enters the upper compartment of the containment. If some hydrogen ignited, either by a working igniter elsewhere or by random ignition (e.g., static discharge), the hydrogen would burn and prevent the concentration from increasing to the detonable limit. At the extreme, if none of the igniters were operating, the probability of early containment failure from non-SBO sequences would approach 1.0 from hydrogen detonation or energetic deflagration.

Another consideration is the locations of the non-operating igniters. For example, if failure of multiple igniters results in a loss of all igniter coverage in two adjacent regions in which ignition might have been initiated, the concentration of hydrogen in those regions would become higher before ignition. The significance of this depends on the location. Failed igniters located in the region where the release into containment occurs, i.e., the lower compartment, are not the most critical since this area is likely to be steam inerted. Igniters in the area where most combustion is initiated, i.e., the upper plenum, are more important as failure of these igniters could lead to an increased hydrogen concentration in the upper compartment before the first burn is initiated. In contrast, if the same number of failed igniters were uniformly distributed throughout containment without loss of igniter coverage in any one compartment, the igniter system would continue to operate as designed. Since the design of the ignition system is intended to provide at least one operable igniter in each major compartment within the containment, it is assumed that failure of all igniters in two or more adjacent compartments would significantly increase the probability of early containment failure from hydrogen combustion events.

Since the average conditional probability of early failure is about 0.1 in ice condenser containments, the conditional probability with respect to LERF is (1.0 - 0.1) = 0.9 for a Type B finding related to igniter non-operability. However, the unavailability of igniters is a risk contributor only in non-SBO accident sequences: hence the conditional probability with respect to LERF of 0.9 should be multiplied by the non-SBO fraction of core damage frequency. Based on the IPE database, the SBO frequency as a fraction of CDF at the ice condenser plants ranges from 1% to 21% with an average of approximately 10%. The conditional probability with respect to LERF is then 0.9 x 0.9 = 0.8 for a Type B finding related to igniter non-operability. The risk significance can be determined by using the relationship given in Section 2.3 and assuming a total CDF of 10^{-4}/ry for PWRs:

$$\Delta\text{LERF} = 0.8 \times 10^{-4} \times (\text{multiplier based on duration of degraded condition})$$

Using the multipliers given in Section 2.3 for each of the three (degraded condition) durations, the following three ΔLERFs and corresponding risk significance categories are obtained:

Duration	ΔLERF	Significance Category
> 30 days	8.0×10^{-5}	red
30–3 days	8.0×10^{-6}	yellow
< 3 days	8.0×10^{-7}	white

If a finding identifies that the hydrogen igniters in any two adjacent compartments would be unable to perform their intended function and the duration of the degraded condition is known, then one of the significance categories given above can be assigned to the finding. Further analyses would be needed to assign a more realistic risk significance to any of the findings that exceed the failure criteria.

7.2.5 Ice Condenser Integrity

The ice condenser consists, in part, of baskets of flaked ice. The ice condenses steam released into the lower compartment during an accident. If the ice is depleted or the pathway through the ice bed obstructed, the containment may be over pressurized. There are two possible ways of compromising the integrity/performance of the ice condenser. The first is a failure of some of the ice chest doors to open or a gross build-up of ice or frost that results in a significant blockage of the flow path between the ice baskets. The second is a substantial loss of ice prior to the occurrence of an initiating event. Each possibility is discussed below.

Ice Chest Doors

Within the lower compartment of an ice condenser containment, there are multiple doors that are designed to open on a small pressure differential to admit steam into the ice chest. If these doors do not open, certain accident sequences, e.g., large LOCAs, could over pressurize containment and result in containment failure. Also, if a number of adjacent doors do not open, flow in the ice beds will not be well distributed, and portions of the ice will not be available for condensing steam and removing released fission products. In the past, there have been findings that these doors have been unable to open due to corrosion and blockage by storage containers. Regarding the obstruction of flow due to ice or frost build up or door blockage, Westinghouse has provided an assessment (Westinghouse, 1998) that an approximately 15% reduction in flow area can occur and the ice condenser is still considered functional. The 15% limit is based on a short-term sub-compartment pressure analysis using the TMD (transient mass distribution) code which assumes that flow area blockage will not exceed this value in any of the flow sections of the TMD model. The number of doors that may be blocked to provide an equivalent flow area restriction ≤ 15% varies depending on where these doors are located. For the lower inlet doors (two doors per bay x 24 bays) up to severn doors may be blocked, but not more than one door in any contiguous group of seven doors, to keep within the 15% limit. For the intermediate deck doors (8 doors/bay x 24 bays), the Westinghouse analysis indicates that up to 48 doors may be blocked, but not more than 3 doors within any contiguous group of 11 doors. For the upper deck doors/blankets, consisting of 2 blankets per bay x 24 bays, up to 24 blankets may be blocked and may be in adjacent bays. The lower inlet doors are limiting as their flow area is comparable to the flow area through the ice bed. The flow area through the intermediate and upper deck doors is 3 times the flow area through the ice bed so their failure to open is much less limiting.

Based on the Westinghouse assessment referred to above, the position is taken that if more than 15% of the area of the flow passage through the ice bed is blocked either due to frost build up or because the doors are unable to perform their function, the integrity of the ice condenser is lost. For risk importance, "unable to perform their function" is taken to mean that the doors to the ice chests (especially the lower doors) are not able to open at a differential pressure of twice the plant technical specification maximum differential pressure. Said differently, if the lower doors do not open at a differential pressure of up to twice the plant technical specification differential pressure limit, then they are unable to perform their intended function and need to be evaluated. This is based on the low differential pressure specified in the plant technical specifications and any real event should generate more than twice the plant technical specification differential pressure on the lower doors.

If the integrity of the ice condenser is lost, then it is assumed that containment integrity is also lost. Since the average conditional probability of early failure and bypass is about 0.1 in ice condensers based on IPE data, the conversion Factor for Type B findings is therefore (1.0 - 0.1) = 0.9 for loss of ice condenser integrity. The risk significance can be determined by using the relationship given in Section 2.3 and assuming a total CDF of 10^{-4}/ry for PWRs:

$$\Delta LERF = 0.9 \times 10^{-4} \times \text{(multiplier based on duration of degraded condition)}$$

Using the multipliers given in Section 2.3 for each of the three (degraded condition) durations, the following three ΔLERFs and corresponding risk significance categories are obtained:

Duration	ΔLERF	Significance Category
> 30 days	9.0×10^{-5}	red
30–3 days	9.0×10^{-6}	yellow
< 3 days	9.0×10^{-7}	white

If a finding identifies a degraded condition that could lead to loss of ice condenser integrity and the duration of the degraded condition is known, then one of the significance categories given above can be assigned to the finding. Further analyses would be needed to assign a more realistic risk significance to any of the findings that exceed the failure criteria.

Debris in the Ice Compartment

An issue of concern is that debris at the bottom of the ice compartment, i.e., in the sumps, could damage and fail the sump pumps. This issue impacts the CDF; it is addressed in the Level 1 significance determination process and is not dealt with here.

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A significance determination process (SDP) is proposed that assigns risk charac terization to inspection findings based on large early release frequency (LERF) considerations. This process is designed to int erface directly with the SDP that is based on findings related to those structures, systems, and components (SSCs) that can i nfluence the core damage frequency (CDF). The proposed LERF-based SDP will capture findings for those SSCs that can influ ence CDF determinations but which can also influence LERF. In addition, the proposed LERF-based SDP approach will ad dress findings related to SSCs that do not influence CDF determinations but which can impact the containment function.

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