NRC INSPECTION MANUAL

APOB

INSPECTION MANUAL CHAPTER 0609 APPENDIX H

CONTAINMENT INTEGRITY
SIGNIFICANCE DETERMINATION PROCESS

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0609H-01 INTRODUCTION

Core damage accidents that lead to large, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population have the potential to cause early health effects, e.g. prompt fatalities. The frequency of all accidents of this type is called the large early release frequency (LERF) as described in Regulatory Guide 1.174 (reference 1). Such accidents include unscrubbed releases associated with early containment failure at or shortly after reactor vessel breach, containment bypass events, and loss of containment isolation.

The relationship of LERF thresholds to core damage frequency (CDF) thresholds found in Regulatory Guide 1.174 provides the basis for the risk significant characterizations found in Table 1.1 below. The LERF based approach is one order of magnitude more stringent than the CDF based approach. Therefore, it may be necessary under some circumstances to characterize the risk significance of an inspection finding using the LERF based approach. The purpose of this appendix to provide guidance for assessing the impact of inspection findings in relation to the containment barrier cornerstone of safety. The basis for the guidance presented in this appendix is discussed in IMC 0308, Reactor Oversight Process (ROP) Basis Document.

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

The significance determination process (SDP) assigns a risk characterization to inspection findings based on LERF considerations. This process is designed to interface directly with the SDP for Type A findings, derived from IMC 0609, Appendix A (at power) and Appendix G (shutdown), that are important LERF contributors. In addition, the guidance addresses findings related to structures, systems, and components (SSCs) that do not influence CDF determinations but can impact the containment function (i.e., Type B findings). It is recommended that inspectors, working with senior reactor analysts (SRAs) as needed, evaluate both Type A and Type B findings for at power findings. It is further recommended that SRAs evaluate both Type A and Type B findings for shutdown.

Note: Type A and Type B findings are defined in section 03.02 Definitions.

01.01 Applicability

The guidance in this SDP is designed to provide NRC inspectors, SRAs and NRC management with a simple probabilistic risk framework for use in identifying which findings are potentially risk-significant from a LERF perspective. Appendix H also helps facilitate communication of the basis for significance between the NRC and licensees. In addition, it identifies findings that do not warrant further NRC engagement, due to very low risk significance, given the findings are entered into the licensee's corrective action program.

01.02 Entry Conditions

The entry conditions for the containment integrity SDP described in this document are related to:

- Findings evaluated under IMC 0609 Appendix A (at power) or Appendix G (shutdown) that potentially increase LERF. or
- Degraded conditions affecting containment barrier integrity (that can potentially increase LERF without affecting CDF).

Appendix H provides simplified risk-informed guidance for estimating the increase in LERF associated with inspection findings related to deficient licensee performance during full power (see IMC 0609, Appendix A) and shutdown operations (see IMC 0609, Appendix G).

01.03 Appendix H Outline

The guidance presented in this appendix is based on a number of assumptions and modeling approximations. Section 02 presents the limitations and precautions that must be considered when evaluating inspection findings. Abbreviations and definitions used in this appendix are presented in Section 03. Section 04 is an overview of the approach and the procedure. Section 05 describes consequential steam generator tube ruptures (C-SGTR). Section 06 presents the procedure for analyzing those findings that have an impact on CDF (i.e., Type A findings) and Section 07 presents the procedure for analyzing those findings that only impact the containment function (i.e., Type B findings). Findings related to power operation and findings related to shutdown operations are both addressed.

01.04 Use of SAPHIRE Software to Calculate LERF

Although this manual chapter provides the methods to estimate LERF manually, LERF can now be calculated automatically with Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) software. The LERF assessment factors for Type A LERF findings have now been programmed into all standardized plant analysis risk (SPAR) models for all plants using global linkage rules.

The use of SAPHIRE to calculate LERF for Type A findings is the preferred method since it eliminates the need to manually list sequences and sum them using the worksheets and methods in this manual chapter. SAPHIRE can also provide values for LERF in Type A findings using the SDP analysis tool. It is important to note, however, that even the SAPHIRE results will produce a Phase 2 Initial Risk Significance Approximation and further refinement might be appropriate.

01.05 Use of Licensee Models for LERF

If provided, LERF risk insights from the licensee risk model can be a source of risk information. The SRA should determine if the PRA model in question is capable of adequately evaluating the risk associated with the finding (e.g., licensee PRA may not model C-SGTR or type B findings). Any evaluation using licensee provided information should be done by an SRA during the detailed risk evaluation.

0609H-02 LIMITATIONS AND PRECAUTIONS

Appendix H generates a reasonably conservative, order-of-magnitude assessment of the risk significance of inspection findings. The intent of Appendix H is to provide guidance for NRC inspectors to easily obtain a quick assessment of risk significance. If appropriate, a more detailed assessment may be performed in a SDP Phase 3 evaluation.

The approach in this appendix has numerous assumptions and limitations which include the following:

- This revision incorporates the AP1000 reactor design into Appendix H. Since this is a new reactor design that hasn't been previously assessed by the SDP, if an analyst has a basis for why this procedure is not adequately capturing the risk, they may depart from this procedure and perform a Phase 3 detailed risk evaluation.
- Since this SDP is focused on LERF, i.e., early fatality risk, long-term risk effects such as population dose and latent cancer fatalities are not addressed in this guidance. In addition, long term accident sequences that involve failure of containment heat removal and ultimately progress to containment failure, e.g., loss of containment heat removal sequences in BWRs, are assumed not to contribute to LERF. It is assumed that effective emergency response actions can be taken within the long time frame of these accident sequences.
- For the evaluation of risk significance during shutdown, only the period within eight days of the beginning of the outage is considered. After eight days, it is assumed that the short-lived, volatile isotopes that are principally responsible for early health effects have decayed sufficiently such that the finding would not contribute to LERF. In addition, all core damage sequences are considered as candidate LERF sequences, because there is greater variability regarding when evacuation would begin.
- LERF determinations depend on the containment design, plant specific attributes and features, which have considerable variability.
- It was conservatively assumed for all interfacing system loss-of-coolant-accidents (ISLOCAs) that the path outside containment is not submerged, nor does it benefit from other means of fission product retention (i.e. the release is not scrubbed).
- 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.
- For those findings that impact the containment function (i.e., Type B findings), baseline CDFs for full power were assumed in order to simplify the calculation of the change in risk. The baseline CDFs for full power assumed were10^{-4}/ry for PWRs,10^{-5}/ry for BWRs, and 10^{-6}/ry for AP1000 plants.
- It was assumed, conservatively, that a main steam isolation valve (MSIV) leakage rate in excess of 10,000 scfh in BWRs (reference 2) with Mark I and Mark II containments is significant to LERF.

0609H-03 ABBREVIATIONS AND DEFINITIONS

03.01 Abbreviations

ADS	Automatic Depressurization System (AP1000)
ATWS	Anticipated Transient Without Scram
CAP	Corrective Action Program
CCFP	Conditional Containment Failure Probability
CCW	Component Cooling Water
CD	Core Damage
CDF	Core Damage Frequency
CE	Combustion Engineering
C-SGTR	Consequential Steam Generator Tube Rupture
DF	Decontamination Factor
DHR	Decay Heat Removal
ECCS	Emergency Core Cooling System
IMC	Inspection Manual Chapter
LER	Licensee Event Report
LERF	Large Early Release Frequency
LOIA	Loss of Instrument Air Initiator
LOOP	Loss of Offsite Power
LORHR	Loss of RHR Initiating Event
LOSW	Loss of Service Water Initiator
LTOP	Low Temperature Over Pressure Events
POS	Plant Operating State
PRA	Probabilistic Risk Assessment
PRHR	Passive Residual Heat Removal System (AP1000)
RCS	Reactor Coolant System
RHR	Residual Heat Removal
ROP	Reactor Oversight Process
RPV	Reactor Pressure Vessel
SCFH	Standard Cubic Feet per Hour
SDP	Significance Determination Process
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SPAR	Standardized Plant Analysis Risk
SSC	Structure, System, or Component
TS	Technical Specifications
TW	Time Window
TW-E	Early Time Window, before refueling operation
TW-L	Late Time Window, after refueling operation

03.02 Definitions

LERF: The frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects.

Close-in population: The population living or transiting within one mile of the reactor site boundary. This also includes nonessential plant personnel being evacuated from the site, any temporary population or local workforce, and any population that may be transiting through the area. Per the Commission's Safety Goal Policy if there are no individuals residing within a mile of the plant boundary, an individual for evaluation purposes, should be assumed to reside 1 mile from the site boundary.

Effective Evacuation: A set of actions by the licensee and local authorities that results in reasonable assumption that the close-in population has been evacuated. This does not require verification or certainty that every individual has left the area, only that all reasonable efforts have been completed and the population has had time to leave the area.

Definitions related to shutdown plant conditions can be found in IMC 0609, Appendix G, Shutdown Operations Significance Determination Process.

Appendix H Phases of Significance Determination:

- Phase 1 - Characterization and Initial Screening of Findings: Precise characterization of the finding and an initial screening of very low-significance findings for disposition by the licensee's corrective action program.

- Phase 2 - Initial Risk Significance Approximation and Basis: Initial approximation of the risk significance of the finding and development of the basis for this determination for those findings that are not screened out in Phase 1 screening.

- Phase 3 - Risk Significance Finalization and Justification: Also known as a detailed risk evaluation, this is a review and as-needed refinement of the risk significance estimation results from Phase 2, or development of any risk analysis outside of this guidance, by an NRC risk analyst (any departure from the guidance provided in this document constitutes a Phase 3 analysis and must be performed by an NRC risk analyst or SRA).

0609H-04 OVERVIEW OF THE APPROACH AND PROCEDURE FOR SIGNIFICANCE DETERMINATION

The guidance described in this section provides an assignment of a significance level (color) to inspection findings based on LERF considerations. This guidance considers findings resulting from deficient licensee performance during full power operations as well as shutdown operations. In Section 04.01, two distinct types of inspection findings that can potentially affect LERF are defined. Section 04.02 provides details of the overall approach taken to the assessment of their significance.

04.01 Types of Findings

An inspection finding associated with a licensee performance deficiency during full power or shutdown operations is characterized by its potential impact on SSCs, by an estimate of the duration of this degradation, and by other information needed to assess the impact on accident likelihood or barrier cornerstone. Two types of findings are encountered:

<u>Type A Findings</u>:

Type A findings can influence the likelihood of accidents leading to core damage that are also identified as contributors to LERF. Such a finding will already have been processed using Appendix A of IMC 0609 for findings at full power, or IMC 0609 Appendix G for findings related to shutdown operations to determine their contributions to ΔCDF.

<u>Type B Findings</u>:

Type B findings are related to a degraded condition that has potentially important implications for the integrity of the containment, without affecting the likelihood of core damage. Table 4.1 shows a list of SSCs (associated with maintaining containment integrity in different containment types). The LERF significance of these SSCs is also addressed in the table.

04.02 LERF Based Significance Determination Process

Figure 4.1 describes the process flow of typical inspection findings. Findings processed through a CDF based SDP will be processed for potential ΔLERF contribution as Type A findings. Findings that only impact the containment function without affected core damage sequences will be processed as Type B findings.

<u>Type A Findings:</u>

For type A findings, the CDF based SDP guidance is used to determine the risk significance based on ΔCDF. If the total ΔCDF for the finding is less than 1E-7 per reactor year, then the finding should be assigned a Green significance level.

If the total ΔCDF ≥ 1E-7 per reactor-year, then a screening is conducted using LERF screening criteria to assess whether any of the core damage sequences affected by the finding are potential LERF contributors. If none of the sequences is a LERF contributor there is no increase in risk and the risk significance based on ΔCDF applies. If one or more of the affected sequences is identified as a LERF contributor, an assessment is performed to estimate ΔLERF and determine the increase in risk significance based on LERF considerations as discussed in detail in Section 05.

Type B Findings:

Type B findings have no impact on the determination of ΔCDF and therefore will not have been processed through the CDF based SDP. These findings, however, are potentially important to ΔLERF contribution and have to be allocated an appropriate risk category based on LERF considerations. As shown in Figure 4.1, an initial screening is conducted to determine if a finding is related to a containment SSC (see Table 4.1) or containment status that has an impact on LERF. If the answer is NO, the finding is Green. If the answer is YES, an assessment of the risk significance is performed using guidance provided in Section 06.

Table 4.1 Containment-Related SSCs Considered for LERF Implications[1]

SSC	LERF Significance
Containment penetration seals: – BWR Mark I and II drywell or PWR containment – BWR Mark III wetwell	Failure of penetration seals that form a barrier between the containment and the environment can be important to LERF
Containment isolation valves in lines: – connecting BWR drywell or PWR containment airspace to environment – connecting RCS to environment or open systems outside containment – connected to closed systems inside/outside containment	Large lines connecting containment airspace to environment (e.g., vent/purge) can contribute to LERF Small lines (< 1–2 inch diameter) and lines connecting to closed systems would not generally contribute to LERF Isolation valves connecting to RCS can contribute to ISLOCA
Main steam isolation valves	Excessive MSIV leakage can contribute to LERF in high pressure accident sequences in BWR Mark I and II plants
BWR drywell/containment sprays	Mark I and II drywell sprays and Mark III containment sprays are important to preventing liner melt-through and mitigating suppression pool bypass
Containment flooding system(s)	Important to preventing liner melt-through in Mark I's
PWR containment sprays and fan coolers	Impact late containment failure and source terms, but not LERF

[1] Some of the listed SSCs could affect the core damage frequency as well as LERF.

Table 4.1 Containment-Related SSCs Considered for LERF Implications[1]

SSC	LERF Significance
Hydrogen control system	
– igniters	Important to LERF in Mark III and ice condenser plants For AP1000, a significant loss of function of hydrogen igniters should be assessed for LERF impacts (e.g., diffusion flames, deflagration-to-detonate transition) until more experience with that containment type is gained
– air return fans and hydrogen mixing systems	Not essential to hydrogen control if igniters are available
Suppression pool (SP) systems	
– components important to SP integrity/scrubbing (e.g., vacuum breakers)	Important to LERF in all BWR plants
– suppression pool cooling	Impacts late containment failure but not LERF
Ice condenser system	
– ice condenser doors and ice bed	Significant flow blockage can be important to LERF
– air return fans	Not important to LERF (similar to containment sprays)
– ice mass air return fans	Deviations in weight of ice not important to LERF
– foreign objects in ice compartment	Not important to LERF (unless CDF is affected)
Filtration systems – Standby Gas Treatment System – control room ventilation	Not important to LERF due to unavailability in dominant sequences (e.g., SBO), plugging from high aerosol loadings in severe accident, and other considerations
Spent fuel assemblies (individual) – fuel handling accidents within pool – fuel handling accidents outside pool	Not important to LERF due to small fission product inventory contained in single fuel bundle. Scrubbing by water in the spent fuel pool further reduces releases.

Table 4.1 Containment-Related SSCs Considered for LERF Implications[1]

SSC	LERF Significance
ADS system (AP1000)	The capability to depressurize the RCS in a high-pressure transient mitigates the consequences of having high RCS pressure during melt progression and vessel rupture. Such accidents have a potential to fail the steam generator tubes or to lead to energetic phenomena at the time of vessel rupture that can challenge containment. Operation of ADS stage 4 provides a vent path for the severe accident hydrogen to the steam generator compartments, bypassing the IRWST, and mitigating the conditions required to produce a diffifusion flame near the containment wall.

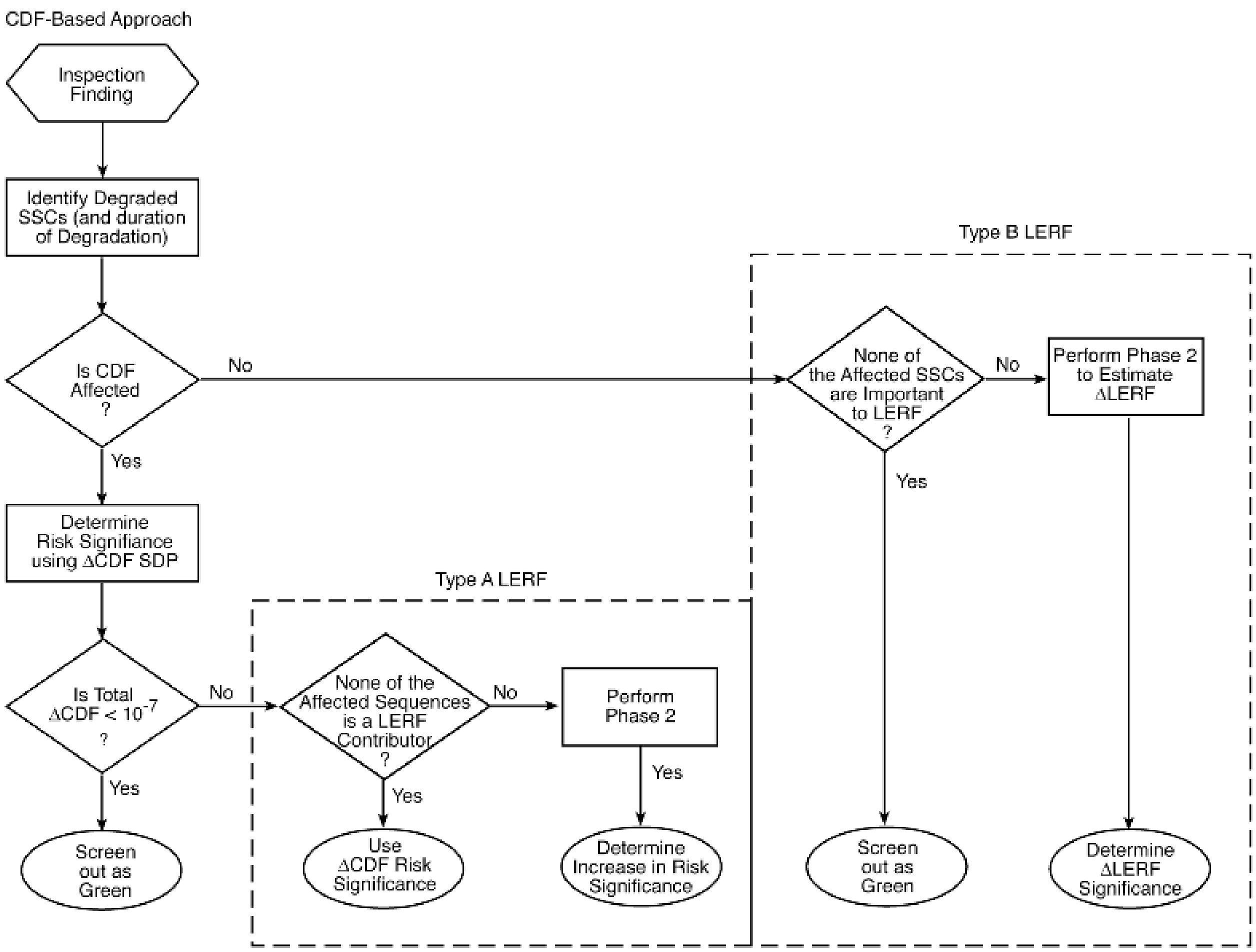

Figure 4.1 LER-based Significance Determination Process

0609H-05 CONSEQUENTIAL STEAM GENERATOR TUBE RUPTURE (C-SGTR)

Consequential Steam Generator Tube Rupture (C-SGTR) is an event in which steam generator tubes leak or fail as a consequence of the high differential pressure or elevated temperatures during accident conditions.

The main accident scenarios of interest for C-SGTR are those that lead to core damage with high reactor pressure, dry steam generator, and low steam generator pressure (High-Dry-Low or HDL) conditions. A typical example of such an accident scenario is a station blackout with loss of auxiliary feedwater. Though other situations can lead to the potential for C-SGTR (e.g., over-pressure from ATWS, a large main steam line break, deliberate action to isolate feed to a faulted steam generator), these other sources are generally understood to be lower contributors to LERF. All of these situations are distinct from SGTR as an initiating event, which should continue to be treated as described elsewhere in this appendix.

NUREG-2195 concluded that the overall contribution of C-SGTR scenarios to containment bypass is about a factor of 10 larger for CE plants than Westinghouse plants[2]. Since C-SGTR is expected to contribute no more than 1-2% additional LERF for a typical Westinghouse plant, it is on par with other sources of LERF for these plants. Conversely, C-SGTR has the potential to be a much more significant contributor to LERF for CE plants, depending on the nature of the finding and its impact on the risk evaluation.

Therefore, findings that could significantly influence the likelihood of having high RCS pressure during core damage or that involve the reliability of feedwater for a CE plant should be evaluated for potential LERF findings from C-SGTR. The RASP Handbook provides the technical basis and a simplified worksheet to estimate LERF resulting from a C-SGTR. Westinghouse plants can also experience C-SGTR but since the potential for it becoming a significant LERF contributor is lower, Appendix H does not require Westinghouse plants to be screened for C-SGTR.

05.01 Evaluation of C-SGTR in AP1000 Reactors

For AP1000 reactors, conditions that may significantly affect the conditional probability of having a consequential (a.k.a., severe accident-induced) steam generator tube rupture should not be screened out. Generally, such conditions would involve an increase in the likelihood of accident sequences associated with the onset of core damage at high pressure, coincident with one or more steam generators having boiled dry. Such instances may include station blackout or transients with failure to depressurize the RCS (e.g., due to ADS and PRHR failures). For accident sequences when core damage occurs with high RCS pressure, a dry SG, and low secondary side pressure, it is likely that full-loop natural circulation conditions will develop, leading to creep damage to both the RCS piping (hot leg and surge line nozzles) and steam generator tubes. The order and timing of failure of these components dictates whether LERF is a concern. These accident sequences could have a greater contribution to LERF, similar to the

[2] Two of the primary factors driving this difference are hot leg diameter and steam generator inlet plenum design. CE plants tend to have larger hot legs that connect to the steam generator closer to the tube sheet, along with flat-bottomed steam generator inlet plenums. Westinghouse plants tend to have smaller hot legs that connect lower in the steam generator inlet plenum, along with rounded-bottom steam generator inlet plenums. These design features tend to dictate the degree of mixing in the inlet plenum under high-dry-low conditions, resulting in a greater challenge to the tubes in the design typical of CE plants.

other containment bypass events that have been screened in (e.g., ISLOCA). AP1000 is not subject to loop seal blockage conditions that can tend to mitigate the threat to SG tubes for other Westinghouse designs, though it is estimated to be less likely to incur such accident sequences to begin with. Additional experience with C-SGTR modeling for AP1000 design is necessary before these findings can be more efficiently screened.

0609H-06 PROCEDURE FOR TYPE A FINDINGS

The CDF-based SDPs (Appendix A and Appendix G to IMC-0609) provide guidance for assessment of the significance of findings that impact CDF. This leads to identification of CDF sequences associated with each finding, evaluation of the increase in frequency of each of the contributing sequences, and determination of the finding significance to ΔCDF based on all contributing sequences collectively.

Evaluation of the impact of the finding on LERF for these sequences is addressed using this appendix. Section 06.01 presents the procedure for Type A findings at full power, and Section 06.02 presents the procedure for Type A findings at shutdown.

06.01 Approach for Assessing Type A Findings at Power

This section provides the step-by-step process (as shown in Figure 6.1) for assessing the risk significance with respect to LERF of Type A findings at full power. As a reminder, SAPHIRE can also be used to calculate LERF of Type A findings, and is the preferred method.

STEP 1 – Finding Characterization

Determine the total ΔCDF of the finding and identify the associated CDF sequences which may be LERF contributors.

Step 2 – Accident Sequence Screening

Generally, only a subset of those sequences contributing to CDF significance of a finding has the potential to impact LERF. A more detailed discussion of these sequences for each containment type is provided in IMC 0308, and briefly summarized below.

<u>BWRs</u>

- For BWR Mark I and Mark II plants, findings related to ISLOCA, ATWS, and accidents with high RCS pressure (i.e., transients and small break LOCA).
- For BWR Mark I plants, accidents that involve a dry drywell floor at vessel breach regardless of whether the RCS is at low or high pressure also need to be evaluated in Phase 2 as indicated in Note 3 to Table 6.1.
- For BWR Mark III plants, findings related to ISLOCA, transients, small break LOCAs, and station blackout (SBO) categories.

<u>PWRs</u>

- For PWR plants with large dry and sub-atmospheric containments, <u>as well as AP1000</u>, findings related to the accident categories ISLOCA and SGTR. Certain accident sequences that lead to core damage with high reactor pressure, dry steam generator,

and low steam generator pressure (High-Dry-Low or HDL) conditions can lead to a consequential steam generator tube rupture (C-SGTR). A typical example of such an accident scenario in an exsting PWR reactor, is a station blackout with a loss of auxiliary feedwater. A typical example of such an accident scenario for an AP-1000 reactor would be a station blackout or transients with failure to depressurize the RCS (e.g., due to ADS and PRHR failures). Though other situations can lead to the potential for C-SGTR (e.g. over-pressure from ATWS, a large main steam line break, (deliberate action to isolate feed to a faulted steam generator), these other sources are generally understood to be lower contibutors to LERF. A C-SGTR is more of a concern for Combustion Engineering (CE) plants. Consult the Risk Assessment for Operational Events (RASP) Manual Volume 5 for more information.

- For the PWR plants with ice condenser containments, findings related to ISLOCA, SGTR, and SBO accident categories.

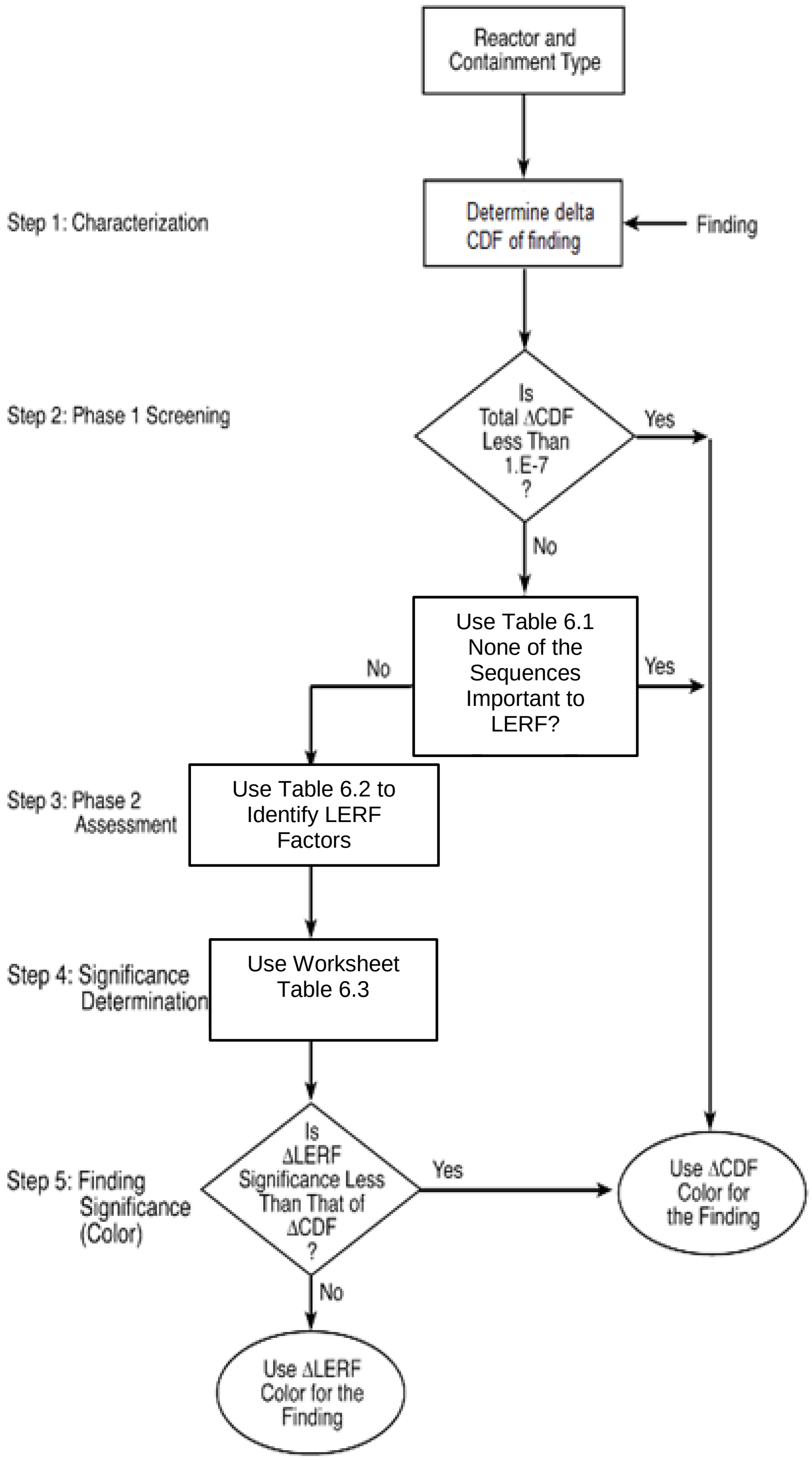

Figure 6.1 Road Map for LERF-based Risk Significance Evaluation for Type A Findings at Power

Accident categories that are screened out in Phase 1 include:

- LOOPs with successful emergency AC power operation (non-SBO events).
- LOOPs with failure of emergency AC power in which power is recovered prior to core damage.

In general, sequences with late core damage (i.e., sequences that proceed to core damage due to loss of containment heat removal) will not contribute to LERF. Other sequences that are screened out are summarized below. When screening out these events for PWR's use caution not to overlook High-Dry-Low sequences which could result in a C-SGTR that would be significant for LERF.

<u>BWRs</u>

- ATWS sequences are not important contributors to LERF for BWRs with Mark III containment. Containment failure from ATWS sequences occurs due to gradual over-pressurization of containment prior to core damage. However, these sequences leave the drywell and suppression pool intact, hence the releases are scrubbed and a large early release does not occur.

<u>PWRs</u>

- ATWS sequences are usually not significant contributors to LERF for PWRs. During a PWR ATWS, containment pressure increases slowly and is therefore a late failure mode. The risk significance determined by the CDF based SDP for ATWS events in PWRs is sufficient. An exception to this would be an ATWS sequence coincident with a loss of feed that could lead to a C-SGTR.
- High and low pressure core damage sequences (in which the containment is not bypassed) are not significant contributors to LERF for PWRs with large dry and sub-atmospheric containments. An important insight from the IPE program and other PRAs is that the conditional probability of early containment failure is less than 0.1 for core damage accident scenarios that leave the RCS at high pressure. If the RCS is depressurized, the probability of early containment failure is less than 0.01.
- In PWRs with ice condenser containments, severe accident studies indicate that the most significant factor is the availability of hydrogen igniters and the ice condenser to mitigate severe accidents. If the igniters are available (i.e., non-SBO accidents), the conditional early containment failure probability is less than 0.1 even during accidents that leave the RCS at high pressure.

<u>Step 2.1</u>

If the total ΔCDF (i.e., sum of all sequences) is <1E-7 per year, the LERF significance is Green and further LERF-related evaluation is not needed. Otherwise, proceed to Step 2.2.

Step 2.2

Compare the attributes of all core damage sequences with a ΔCDF of ≥1E-8 per year with those in Table 6.1 to identify those sequences which have the potential to affect LERF. Individual sequence results that are <1E-8 are not significant and are not evaluated further. However, those LERF sequences that are ≥1E-8 (sequence result of 8 or less) are evaluated for the

overall LERF contribution. If none of the sequences impacts LERF, the risk significance obtained from the ΔCDF assessment is used for the significance of the finding and no further LERF-related evaluation is necessary. If ΔCDF sequences are identified as having the potential to affect LERF[3], proceed to Step 3.

Step 3 – Phase 2 Assessment

For sequences needing Phase 2 analysis, risk significance determination is performed using the following two substeps:

Step 3.1 – LERF Factor Determination

Identify the LERF factor associated with each of the sequences remaining after screening using Table 6.2. Document these sequences and their associated LERF factors as discussed in the next substep.

Step 3.2 – ΔLERF Significance Evaluation

Document details of LERF significance assessment using the LERF worksheet (Table 6.3). List each sequence assessed in Phase 2 in column 1 together with its CDF score (in column 2).

Document the sequence attributes that make it a potential LERF contributor (e.g. high RCS pressure, drywell floor status for BWRs, etc.) in column 3.

Document the LERF factor (see Step 3.1) in column 4.

Document the LERF score in column 5. The LERF score is calculated by multiplying the ΔCDF score (column 2) by the LERF factor (column 4). For example, if a sequence has a ΔCDF score of 7 (i.e., 1E-7) and the associated LERF factors is 0.4, the LERF score is 4×10^{-8}.

Step 4 – LERF Significance

Sum the scores for all of the LERF contributing sequences associated with the finding and enter the total ΔLERF score in the space below Column 5. Use the numerical result to determine the ΔLERF significance (color), using Table 1.1.

Step 5 – Finding Significance

Compare the CDF significance (color) with that for the LERF significance for the same finding. The higher (color) is the preliminary risk significance of the finding.

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[3]No extra credit should be given for Severe Accident Management operator recovery actions (e.g., actions to depressurize the RCS or to flood Mark I drywell) unless recovery is explicitly modeled in the CDF sequence. Defer such recovery credit to Phase 3 assessment if needed.

Table 6.1 Phase 1 Screening-Type A Findings at Full Power

Reactor Type	Containment Type	Attributes of Accident Sequence Related to Finding: ISLOCA	SGTR	ATWS	SBO (Note 1)	High RCS Pressure (Note 2)	All Others
BWR	Mark I	Perform Phase 2	Not Applicable	Perform Phase 2	Perform Phase 2	Perform Phase 2	Note 3
BWR	Mark II	Perform Phase 2	Not Applicable	Perform Phase 2	Perform Phase 2	Perform Phase 2	Screen Out (Note 4)
BWR	Mark III	Perform Phase 2	Not Applicable	Screen Out (Note 4)	Perform Phase 2	Perform Phase 2	creen Out (Note 4)
PWR	Large Dry and Sub-Atmospheric	Perform Phase 2	Perform Phase 2	Screen Out (Note 4)	Screen Out (Note 4)	Screen Out (Note 4)	Screen Out (Note 4)
PWR	Combustion Engineering Plants	Perform Phase 2	Perform Phase 2	(Note 5)	(Note 5)	(Note 5)	(Note 5)
PWR	Ice Condenser	Perform Phase 2	Perform Phase 2	Screen Out (Note 4)	Perform Phase 2	Screen Out (Note 4)	Screen Out (Note 4)
PWR	AP1000	Perform Phase 2	Perform Phase 2	Screen Out (Note 5)	Screen Out (Note 5)	Screen Out (Note 5)	Screen Out (Note 5)

Note 1: SBO is defined as a LOOP sequence with loss of emergency AC and failure to recover AC power.

Note 2: High pressure is defined as greater than 250psi at the time of reactor vessel breach. Transients and small break LOCAs (smaller than about 2-inch equivalent break size in BWRs and 0.75 - 1 inch in PWRs) will usually result in pressures in the RCS greater than 250psi at the time of reactor vessel melt-through in the absence of manual depressurization.

Consider a Sequence to be low pressure in case of:

- Large or intermediate LOCA
- Sequences that include successful depressurization (DEP)
- Availability of low pressure injection (LPI) is questioned on sequence branch

Consider a sequence to be high pressure in case of:

- The sequence includes failure of depressurization (DEP)
- None of the low pressure considerations identified above apply

Note 3: A phase 2 assessment should be performed for any sequences that are expected to proceed to reactor vessel breach into a dry reactor cavity. Therefore, all other transients with successful RCS depressurization should be assessed. Sequences involving LOCAs in the drywell or drywell spray operation are excluded because they result in a flooded drywell floor. LOCAs involving stuck open relief valve sequences do not result in flooded drywell.

Note 4: Screen out means that the accident sequence related to the finding is not significant to LERF and is Green.

Note 5: CE plants should be screened for C-SGTR. Refer to the RASP Manual Volume 5 for more information. AP1000 should be screened for C-SGTR only if conditions described in section 05.01 are met.

Table 6.2 Phase 2 Assessment Factors -Type A Findings at Power

Reactor Type	Containment Type	Attributes of Accident Sequence Related to Finding					
		ISLOCA	SGTR	ATWS	SBO (Note 1)	High RCS Pressure (Note 2)	Low RCS Pressure (Note 2)
BWR	Mark I	1.0	Not Applicable	0.3	(Note 3)	0.6 If drywell is Flooded	<0.1 If drywell is Flooded
						1.0 If drywell is Dry	1.0 If drywell is Dry
BWR	Mark II	1.0	Not Applicable	0.4	(Note 4)	0.3	Screen Out in Phase 1
BWR	Mark III	1.0	Not Applicable	Screen Out in Phase 1	0.2	0.2	Screen Out in Phase 1
PWR	Large Dry and Sub-Atmospheric	1.0	1.0	Screen Out in Phase 1	Screen Out in Phase 1	Screen Out in Phase 1	Screen Out in Phase 1
PWR	AP1000	1.0	1.0	Screen Out in Phase 1	Screen Out in Phase 1	Screen Out in Phase 1	Screen Out in Phase 1
PWR	Ice Condenser	1.0	1.0	Screen Out in Phase 1	1.0	Screen Out in Phase 1	Screen Out in Phase 1

Note 1: SBO is defined as a LOOP sequence with loss of emergency AC and failure to recover AC power.

Note 2: High pressure is defined as greater than 250psi at the time of reactor vessel breach. Transients and small break LOCAs (smaller than about 2-inch equivalent break size in BWRs and 0.75–1 inch in PWRs) will usually result in pressures in the RCS greater than 250psi at the time of reactor vessel melt- through in the absence of manual depressurization.

Note 3: If the RCS is at high pressure during the SBO then the Factors for the high pressure column apply. If the RCS is at low pressure during the SBO, the factors for the low pressure column apply.

Note 4: If the RCS is at high pressure during the SBO then the Factor is 0.3. If the RCS is at low pressure during the SBO, the finding can be screened out.

Table 6.3 Manual Worksheet for ΔLERF

(1) Sequences	(2) ΔCDF Score (X)	(3) Sequence Attributes	(4) LERF Factor (Table 6.2 for power, Table 6.4 for shutdown) (F)	(5) ΔLERF Score F * ($1x10^{-X}$)
Total ΔLERF Score				

06.02 Approach for Assessing Type A Findings During Shutdown

This section provides a step-by-step process (shown in Figure 6.2) for assessing the risk significance with respect to LERF of Type A findings applicable to shutdown operation.

STEP 1 – Finding Characterization

Step 1.1

Review the assessment performed using IMC 0609, Appendix G, to identify the sequences affected by the finding, and the POSs and time windows (TWs) applicable to the finding.

Step 1.2

Determine the status of containment when the finding occurred for each POS and TW:

For PWRs and BWR Mark IIIs, the status of containment is either open or intact.

For BWRs Mark I and IIs, the status of containment is either intact, de-inerted, or open.

STEP 2 – Accident Sequence Screening

Step 2.1

For each shutdown core damage scenario identified in Step 1, determine if the following conditions were met:

- The finding occurred while the plant was in POS 1E or POS 2E.
- The finding occurred within the first eight days of the outage.

Step 2.2

If both conditions in Step 2.1 were met, go to Step 3. Otherwise, the LERF significance is Green and further evaluation for LERF implications is not needed.

STEP 3 – Phase 2 Assessment

For sequences needing Phase 2 analysis, risk significance determination is performed using the following two substeps:

Step 3.1

Determine the LERF factor for each core damage scenario affected by the finding for the appropriate containment status using Table 6.4.

Step 3.2

Document details of LERF significance assessment for the finding being evaluated using the LERF worksheet (Table 6.3). List each sequence assessed in Phase 2 in column 1 together with its CDF score (in column 2). Since all core damage sequences are potential LERF contributors, column 3 may be left blank. Document the LERF factor (see Step 3.1) in column 4. Document the LERF score in column 5. The LERF score is calculated by multiplying the ΔCDF score (column 2) by the LERF factor (column 4). For example, if a sequence has a ΔCDF score of 7 and the associated LERF factor is 0.2, the LERF score is 2×10^{-8}.

STEP 4 – LERF Significance

Sum the scores for all of the LERF contributing sequences associated with the finding being evaluated and enter the total ΔLERF score in the space below column 5 of the completed Table 6.3. Use the numerical result to determine the ΔLERF significance (color), using Table 1.1.

STEP 5 – Finding Significance

Compare the CDF significance (color) with that for the LERF significance for the same finding. The higher (color) is the preliminary risk significance of the finding.

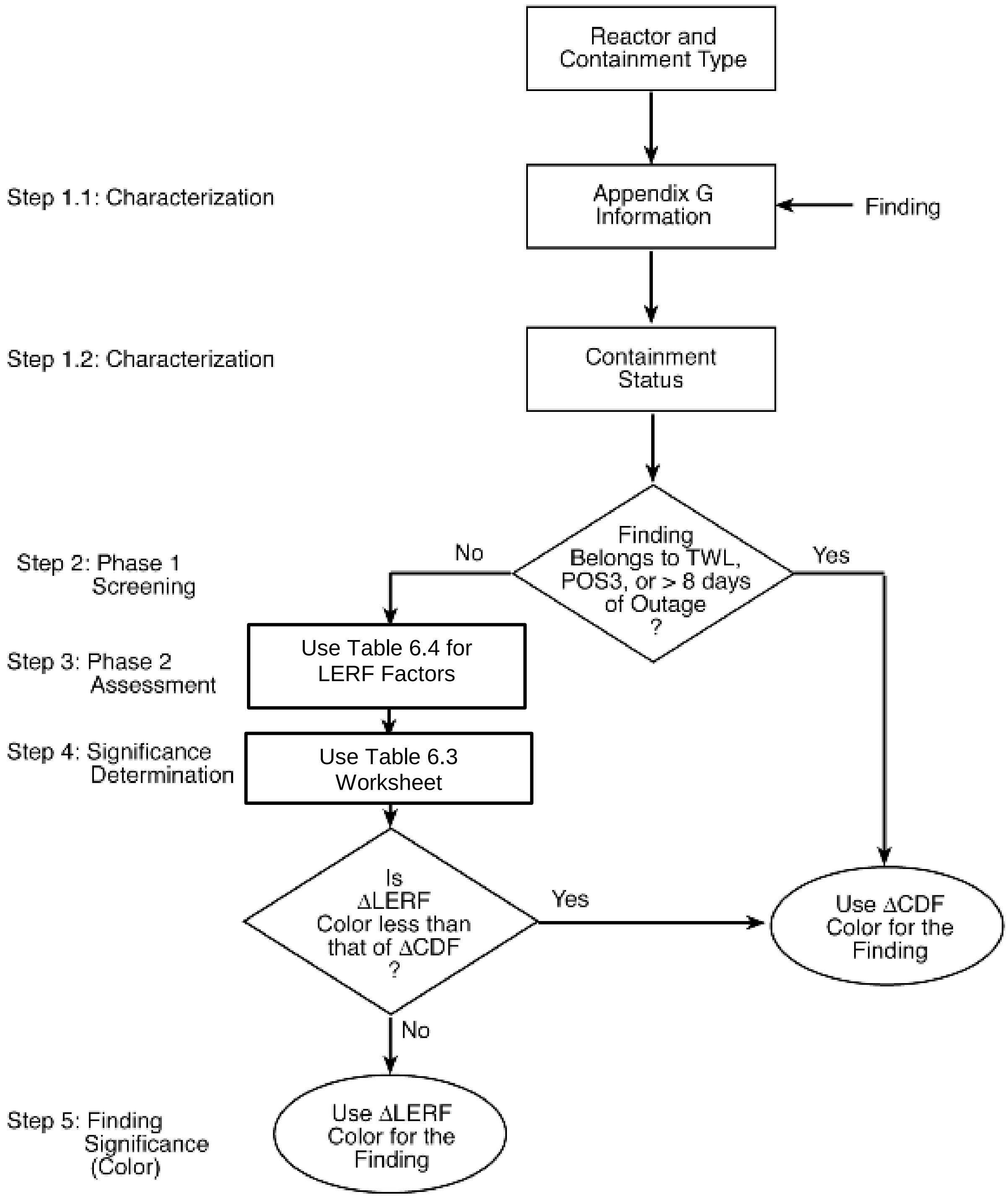

Figure 6.2 Road Map for LERF-based Risk Significance Evaluation for Type A Findings at Shutdown

Table 6.4 Phase 2 Assessment Factors -Type A Findings at Shutdown

Reactor/ Containment Type	Containment Status (Note 1)	Accident Sequence Related to Finding	
		Finding occurs: (1) in POS 1E or POS 2E within first 8 days of outage.	All Others
BWR Mark I and II	De-inerted	1.0	Screened Out
BWR Mark III	Intact	0.2 – if igniters are not available (Note 2)	Screened Out
		Screen Out – if igniters are available (Note 3)	
PWR Large Dry and Sub-Atmospheric	Intact	Screen Out (Note 3)	Screened Out
AP1000	Intact	Screen Out (Note 3)	Screened Out
Ice Condenser	Intact	1.0 – if igniters are not available (Note 2)	Screened Out
		Screen Out – if igniters are available (Note 2 and Note 3)	
All	Open	1.0	Screened Out

Note 1: An intact containment is one in which, the licensee intends to: (1) close all containment penetrations with a single barrier or can be closed in time to control the release of radioactive material, and (2) maintain the containment differential pressure capability necessary to stay intact following a severe accident at shutdown. When the RCS is open, an intact containment means that containment can be re-closed prior to boiling the RCS inventory. If the licensee does not intend to maintain an intact containment, then containment is open.

A de-inerted containment is one in which limits on the primary containment oxygen concentration as defined in TS are no longer met.

Note 2: There are no TS for igniters to be operable during shutdown. However, it is possible that igniters could be recovered by operator action, in which case the finding could be screened out (i.e. not significant to LERF)

Note 3: To screen out the finding, the analyst must verify that the licensee's plans for containment closure consider: 1) time to boiling given a loss of RCS inventory which shortens time to boiling compared to a loss of the operating train of RHR. (NOTE: selecting time to boiling based on RCS level at the bottom of the hot leg should always meet the recommendation) and (2) a potential loss of offsite power and a loss of all vital AC power.

0609H-07 PROCEDURE FOR TYPE B FINDINGS

Type B findings have no direct impact on the likelihood of core damage but have potentially important implications for containment integrity. This section provides the procedure for evaluation of LERF significance of Type B findings. Similar to the Type A findings approach, a step wise process (Figure 7.1) is used, which leads to a conservative estimate of LERF significance. Section 07.01 presents the procedure for findings at full power, and Section 07.02 for findings at shutdown.

07.01 Approach for Assessing Type B Findings at Power

STEP 1 – Finding Characterization

Characterize the finding in terms of its relationship to the containment barrier function. Collect information needed for significance determination: SSCs affected and the nature of the degradation; the duration (i.e., >30days, 30-3 days, and < 3 days) of the degraded condition; information such as the magnitude of the leakage or number and location of inoperable hydrogen igniters. The type of information required can be inferred from Table 7.2 below.

STEP 2 – Screening of Finding

Determine if the finding is associated with an SSC(s) important to LERF, using Table 7.1. If the finding is screened out then no further assessment is needed and the finding is Green. Otherwise, proceed to Step 3 below. Note that a detailed description of finding to be assessed in Step 3 is included in Table 7.2.

STEP 3 – Phase 2 Assessment

Use Table 7.2 to provide a significance assignment to a Type B finding. For inspection findings involving leakage rates (e.g., MSIV leakage, containment leakage), if the as-found leakage rate is less than the values listed in Table 7.2, the finding is Green.

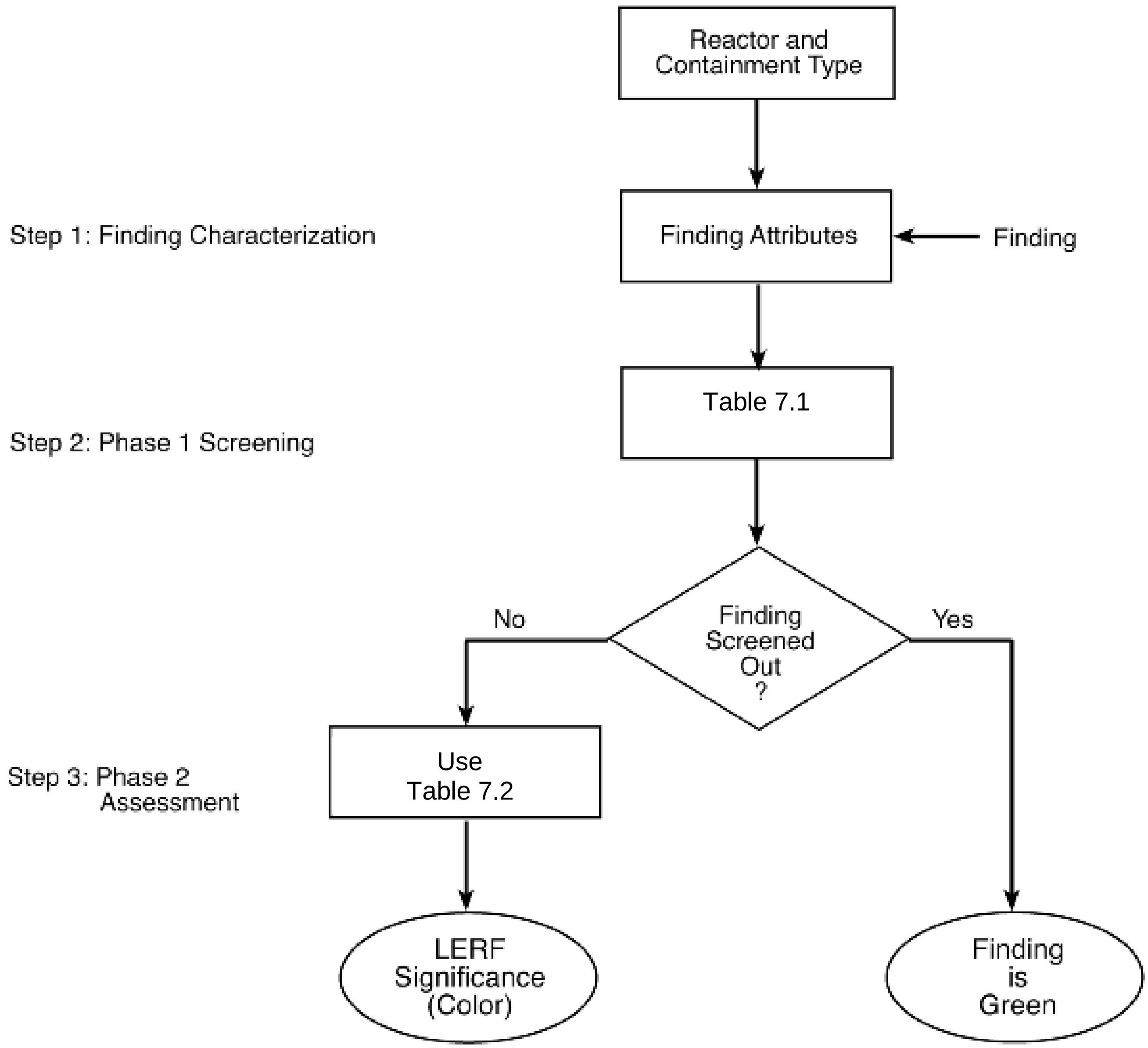

Figure 7.1 Road Map for LERF-based Risk Significance for Evaluation Type B Findings at Power

Table 7.1 Phase 1 Screening-Type B Findings at Power

Reactor Type	Containment Type	SSC Affected by Finding					
		Containment Penetration Seals, Isolation Valves, Vent and Purge Systems	Ice Condenser Flow Blockage	Suppression Pool Integrity	MSIV Leakage	Drywell / Containment Sprays	Igniters
BWR	Mark I	Perform Phase 2	Not Applicable	Perform Phase 2	Perform Phase 2	Perform Phase 2	Not Applicable
BWR	Mark II	Perform Phase 2	Not Applicable	Perform Phase 2	Perform Phase 2	Perform Phase 2	Not Applicable
BWR	Mark III	Perform Phase 2	Not Applicable	Perform Phase 2	Not Applicable[1]	Perform Phase 2	Perform Phase 2
PWR	Large Dry and Sub-Atmospheric	Perform Phase 2	Not Applicable	Not Applicable	No Applicable	Not Applicable	Not Applicable
PWR	AP1000	Perform Phase 2	Not Applicable	Not Applicable	Not Applicable	Not Applicable[2]	Not Applicable
PWR	Ice Condenser	Perform Phase 2	Perform Phase 2	Not Applicable	Not Applicable	Perform Phase 2	Perform Phase 2

Note 1: Some BWR Mark lll containments may have a safety-grade low-leakage Main Steam Shutoff Valve (MSSV) outside of the out- board MSIV. (This may have been abandoned in some plants) Reference (2)

Note 2: AP1000 is being treated akin to a large dry containment in the absence of operating experience, however, a particular performance deficiency relating to local hydrogen effects (e.g., the potential for a diffusion flame near the containment wall) could warrant further investigation.

Table 7.2 Phase 2 Risk Significance -Type B Findings at Power

Reactor Type	Containment Type	Finding	Risk Significance >30 days	Risk Significance 30–3 days	Risk Significance <3 days
BWR	Mark I and Mark II	Leakage from drywell to environment >100 % containment volume/day through containment penetration seals, isolation valves or vent and purge systems	Yellow	White	Green
		Failure of systems/components critical to suppression pool integrity/scrubbing (vacuum breakers or other bypass mechanisms)	Yellow	White	Green
		Main steam isolation valve leakage >10,000 scfh through the best-sealing valve in any steam line (see Reference 2)	Yellow	White	Green
	Mark I	Drywell sprays unavailable	Yellow	White	Green
	Mark II	Drywell sprays unavailable	White	Green	Green
BWR	Mark III (NOTE 1)	Leakage from wetwell to environment >1,000 % containment volume/day through containment penetration seals, isolation valves or vent and purge systems	White	Green	Green
		Failure of systems/components critical to suppression pool integrity/scrubbing (vacuum breakers or other bypass mechanisms)	Yellow	White	Green
		Failure of multiple igniters such that coverage is lost in two adjacent compartments	White	Green	Green
		Containment sprays unavailable	White	Green	Green
PWR	Large Dry and Sub-Atmospheric	Leakage from containment to environment >100 % containment volume/day through containment penetration seals, isolation valves or vent and purge systems	Red	Yellow	White
PWR	AP1000	Leakage from containment to environment >100 % containment volume/day through containment penetration seals, isolation valves or vent and purge systems	White	Green	Green
		Significant loss of function to hydrogen igniters	Note 2		
PWR	Ice Condenser (NOTE 1)	Leakage from containment to environment >100 % containment volume/day through containment penetration seals, isolation valves or vent and purge systems	Red	Yellow	White
		Blockage of more than 15% of the flow passage into or through the ice bed	Red	Yellow	White
		Failure of multiple igniters such that coverage is lost in two adjacent compartments	Red	Yellow	White

Note 1: For BWR Mark III containments and PWR ice condenser plants, the term compartments is used interchangeably with the term regions, or zones and relates to the likelihood that hydrogen concentrations could rise to levels that could challenge containment. For a particular finding, the intent is to determine if the igniter system would remain effective at controlling concentrations in this regard, and "two adjacent compartments" is used as a rule-of-thumb. If it is not clear whether the igniter system would remain effective, the inspector should refer to the text in section 5.2.3 (Mark III containments) or section 7.2.4 (ice condenser containments) of NUREG-1765, and to consult the design-basis analysis associated with the igniter system. If it is still unclear, the inspector should contact the regional SRA and headquarters staff knowledgeable in this area.

Note 2: For AP1000, a significant loss of function to the hydrogen igniters should be assessed for LERF impacts.

07.02 Approach for Assessing Type B Findings at Shutdown

STEP 1 – Finding Characterization

Figure 7.2 shows the process flow for this approach. Characterize the finding in terms of its relationship to the containment barrier function. Collect information needed for significance determination, specifically the SSCs affected and the nature of the degradation, the duration of the degraded condition if less than the complete outage and if the condition had existed before shutdown (during power operation), or could exist upon change of plant/containment status (e.g. return to power) and information such as the magnitude of the leakage or the number and location of the inoperable hydrogen igniters. The type of information required can be inferred from Table 7.4 below. In addition, identify each POS(s) and time windows with which the finding is associated.

STEP 2 – Accident Sequence Screening

STEP 2.1 – Screen on the Basis of POS and Time Window

If the finding occurs (1) in POS 1 or POS 2 AND (2) in TW-E, AND (3) within eight days of the start of the outage, THEN, go to Step 2.2. Otherwise, screen the finding as Green.

STEP 2.2 – Screen on the Basis of the Impact of the Finding

Determine if the finding is associated with an SSC(s) important to LERF using Table 7.3. Consideration of items A through D (as applicable) facilitates the use of Table 7.3.

A. Did the finding involve the licensee failing to maintain the capability to close containment (maintain an intact containment) when the licensee planned to maintain an intact containment consistent with NRC expectations (GL 88-17) and Industry expectations (NUMARC 91-06)? This question applies to PWR and BWR Mark III licensees only. If yes, Go to Table 7.3, containment status is intact. If no, continue with Step B.

B. Did the finding involve hydrogen igniters in a BWR Mark III or a PWR ice condenser containment and the licensee maintained an intact containment?

If yes, Go to Table 7.3, containment status is intact. If no, continue with Step C.

C. Did the finding occur when the containment was de-inerted for a Mark I or Mark II containment?

If yes, go to Table 7.3, containment status is de-inerted. If no, continue with Step D.

D. Did the licensee intend to have an open containment without the capability to reclose containment?

If yes, Go to Table 7.3, containment status is open.

NOTE: If a PWR licensee is not maintaining an intact containment during POS 1E and POS 2E, this may be a significant finding under the Maintenance Rule. Check with an SRA for further guidance.

If no, Screen out the finding.

If the finding is screened out, it is assigned Green significance, and no further assessment is needed. Otherwise, proceed to Step 3 below.

STEP 3 – Phase 2 Assessment

Determine if shutdown mitigation capability is minimal or in-depth or closely resembles an in-depth or minimal capability. Use Tables 7.5 and 7.6 for BWRs, or Tables 7.7 and 7.8 for PWRs, to help make this determination.

NOTE: For PWRs, if mitigation capability does not match with the tables, choose between in-depth or minimal capability based on: (1) availability of SGs and (2) availability of ECCS pumps and charging pumps

Use Table 7.4 to determine color of finding.

NOTE: Should the duration of a Type B finding exist for less than eight hours, then the color finding is reduced by one order of magnitude.

NOTE: Findings that may have existed before shutdown (during power operation) or that could impact LERF upon change of plant/containment status (e.g. return to power) should be assessed. In case the finding is judged to impact power operation, Section 07.01 guidance should be used in the assessment.

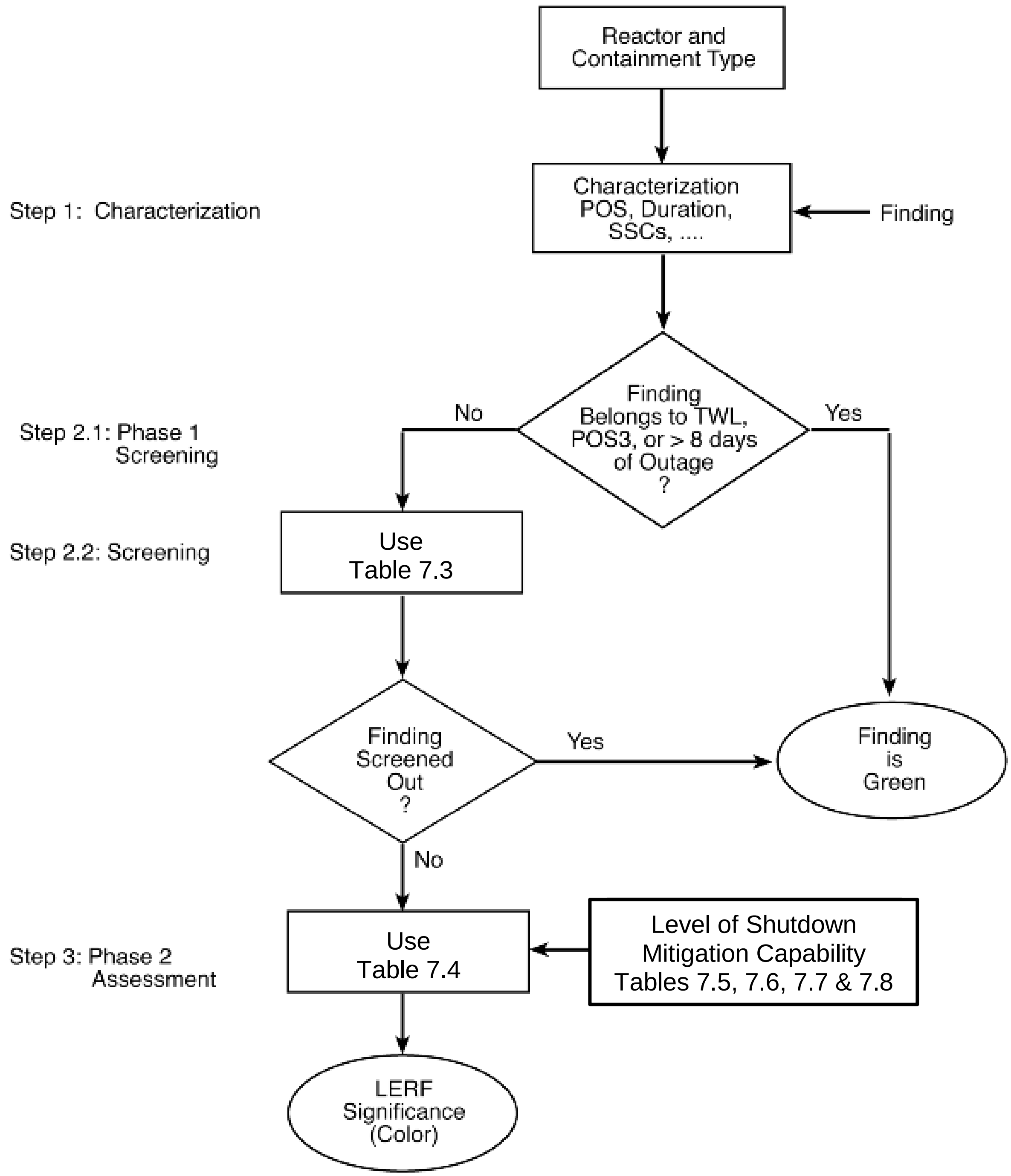

Figure 7.2 Road Map for LERF-based Risk Significance Evaluation for Type B Findings at Shutdown

Table 7.3 Phase 1 Screening-Type B Findings at Shutdown

Reactor/ Containment Type	Containment Status (Note 1)	SSC Affected by Finding			
		Containment Penetration Seals, Isolation Valves, Vent and Purge Systems	Suppression Pool Integrity	Drywell/ Containment Sprays	Igniters
BWR Mark I and II	De-inerted	No Type B Findings Important to ΔLERF (Note 2)			
BWR Mark III	Intact	Perform Phase 2	Perform Phase 2	Screen Out (Not important to LERF)	Perform Phase 2
PWR Large Dry and Sub-Atmospheric	Intact	Perform Phase 2	Not Applicable	Screen Out (Not important to LERF)	Not Applicable
PWR AP1000	Intact	Perform Phase 2 contact HQ for further assistance	Not Applicable	Not Applicable	(note 4)
PWR Ice Condenser	Intact	Perform Phase 2	Not Applicable	Screen Out (Not important to LERF)	Perform Phase 2
All	Open	No Type B Findings Important to ΔLERF (Note 3)			

Note 1: An intact containment is one in which, the licensee intends to: (1) close all containment penetrations with a single barrier or can be closed in time to control the release of radioactive material, and (2) maintain the containment differential pressure capability necessary to stay intact following a severe accident at shutdown. When the RCS is open, an intact containment means that containment can be reclosed prior to RCS boiling. A Type B performance deficiency results when a licensee intends to have an intact containment but cannot maintain that capability due to a performance deficiency. For Mark III containments, the definition of intact containment applies to primary containment.

If the licensee does not intend to maintain an intact containment, then containment is open. If a PWR licensee is not maintaining an intact containment during POS 1E and POS 2E, then this observation could be risk significant under the Maintenance Rule and should be reported to an SRA.

A de-inerted containment is one in which limits on the primary containment oxygen concentration as defined in TS are no longer maintained.

Note 2: Type B findings would be unimportant to ΔLERF because containment would be de-inerted and expected to fail due to hydrogen combustion, regardless of Type B finding. However, findings that may have existed before shutdown or that could impact LERF upon change of plant/containment status (e.g. return to power) should be assessed.

Note 3: Type B findings would be unimportant to Δ LERF because containment is already open and cannot be re-closed. However, findings that may have existed before shutdown or that could impact LERF upon change of plant/containment status (e.g. return to power) should be assessed. If a PWR licensee is not maintaining an intact containment during POS 1E and POS 2E, then this observation could be risk significant under the Maintenance Rule and should be reported to an SRA.

Note 4: For AP1000, a significant loss of function should be assessed for LERF impacts.

Table 7.4 Phase 2 Risk Significance -Type B Findings at Shutdown
(For POS 1/TW-E and POS 2/TW-E in which the finding occurs during the first eight days of the outage)

Reactor/ Containment Type	Containment Status (NOTE 1)	Finding	Risk Significance (NOTE 2)	
			Minimal Capability	In-depth Capability
BWR Mark I, II	De-inerted	Screened Out in Phase 1	N/A	N/A
BWR Mark III	Intact	Leakage from containment to environment > 1000% containment volume/day through containment penetration seals, isolation valves or vent and purge systems with suppression pool integrity (NOTE 3)	POS 1E -Yellow POS 2E - Yellow	POS 1E- White POS 2E - Green
BWR Mark III	Intact	Loss of suppression pool integrity (NOTE 4)	POS 1E -Yellow POS 2E - Yellow	POS 1E- White POS 2E - Green
BWR Mark III (NOTE 5)	Intact	Failure of multiple igniters such that coverage is lost in two adjacent compartments given that primary containment is intact	POS 1E - White POS 2E - White	POS 2E- Green POS 2E - Green
PWR Large Dry and Sub-Atmospheric	Intact	Leakage from containment to environment > 100 % containment volume/day through containment penetration seals, isolation valves or vent and purge systems	POS 1E -Yellow POS 2E - Red	POS 1E - White POS 2E - White
PWR Ice Condenser (NOTE 5)	Intact	Leakage from containment to environment >100 % containment volume/day through containment penetration seals, isolation valves or vent and purge systems	POS 1E - Yellow POS 2E - Red	POS 1E - White POS 2E - White
		Failure of multiple igniters such that coverage is lost in two adjacent compartments	POS 1E - Yellow POS 2E - Red	POS 1E - White POS 2E - White
All	Open	Screened Out in Phase 1	Green	Green

Note 1: An intact containment- is one in which the licensee intends to: (1) close all containment penetrations with a single barrier or can be closed in time to control the release of radioactive material, and (2) maintain the containment differential pressure capability necessary to stay intact following a severe accident at shutdown. When the RCS is open, an intact containment means that containment can be re-closed prior to RCS boiling. A type B performance deficiency results when a licensee intends to have an intact containment but cannot maintain that capability due to a performance deficiency. For Mark III containments, the definition of intact applies to primary containment.

If the licensee does not intend to maintain an intact containment, then containment is open. If a PWR licensee is not maintaining an intact containment during POS 1E and POS 2E, then this observation could be risk significant under the Maintenance Rule and should be reported to a SRA.

A de-inerted containment is one in which limits on the primary containment oxygen concentration as defined in Technical Specifications are no longer maintained.

Note 2: The results assume that each shutdown scenario results in a LERF if the licensee fails to maintain an intact containment or the containment fails due to loss of hydrogen control in Ice Condenser and Mark III containments. In phase 3 analysis, if the staff concludes that failures involving long term cooling can be eliminated from LERF because the licensee would have evacuated given successful short-term cooling, then the color of the finding would be reduced.

When using this table, there are no duration factors associated with findings at shutdown. The generic shutdown CDFs include the frequency and duration that POS 1 and POS 2 are entered into per calendar year for both PWRs and BWRs. For BWRs, POS 1 is assumed to last four days; POS 2 is assumed to last two days. For PWRs, POS 1 is assumed to last two days; POS 2 is assumed to last six days. Should the duration of a type B finding exist for less than eight hours, then the color finding is reduced by one order of magnitude.

Note 3: As discussed in Regulatory Guide 1.174, releases that pass through the pool would be scrubbed and would not contribute to LERF. Rather than crediting the pool with completely eliminating LERF, a decontamination factor (DF)

of 10 is assigned to pool scrubbing in the SDP. This DF results in the LERF-significant leak rate increasing from 100% containment volume per day to 1000% containment volume per day

Note 4: With the suppression pool unavailable, fission products will not be scrubbed and steam generated by decay heat is assumed to lead to gradual over-pressurization of containment and the need to vent prior to effective evacuation. Thus, the finding could be LERF significant even if leak rate is less than 100% containment volume per day.

Note 5: For BWR Mark III containments and PWR ice condenser plants, the term compartments is used interchangeably with the term regions, or zones, and relates to the likelihood that hydrogen concentrations could rise to levels that could challenge containment. For a particular finding, the intent is to determine if the igniter system would remain effective at controlling concentrations in this regard, and “two adjacent compartments” is used as a rule-of-thumb. If it is not clear whether the igniter system would remain effective, the inspector should refer to the text in section 5.2.3 (Mark III containments) or section 7.2.4 (ice condenser containments) of NUREG-1765, and to consult the design-basis analysis associated with the igniter system. If it is still unclear, the inspector should contact the regional SRA and headquarters staff knowledgeable in this area.

Table 7.5 BWRs With Minimal Shutdown Mitigation Capability

Total Annualized CDF Head on: 3E-6 (per calendar year) Total Annualized CDF Head off: 9E-7 (per calendar year)	
<u>Item</u>	<u>Value</u>
RHR pumps	2 (shared with ECCS)
Other heat removal pumps	0
ECCS pumps (in standby)	2 (Shared with RHR)
SRVs for Power Operated Relief Mode	2
CCW pumps/trains	1 train with 2 pumps
SW pumps/trains	1 train with 2 pumps
Containment Spray pumps	0
Fire Water	No
SW Injection into RCS	No
Path to Suppression Pool	Yes
Suppression Pool	Yes
Other Water sources	No
Other means of removing heat	None
Offsite power sources	2
EDGs	1
Other onsite power sources	0
Level instruments	Yes
Vessel Temperature Instruments	No
Level 3 RHR Isolation	Sometimes Not Used

Table 7.6 BWRs With In-depth Shutdown Mitigation Capability

Total Annualized CDF RCS Head on: 2E-7 (per calendar year) Total Annualized CDF RCS Head off: 4E-8 (per calendar year)	
Item	Value
RHR pumps	2 (Shared with ECCS
Other heat removal pumps	0
ECCS pumps	2 (shared with RHR pumps)
SRVs (in Power Operated Relief mode)	2
CCW pumps/trains	1 train with pumps
SW pumps/trains	1 train with pumps
Containment Spray Pumps	0
Fire Water	Yes
SW Injection into the RCS	Yes
Path to the Suppression Pools	Yes
Suppression Pool	Yes
Other water sources	No
Other means of removing heat	None
Offsite power sources	2
EDGs	2
Other onsite power sources	0
Level instruments	Yes
Vessel temperature Instruments	Yes
Level 3 RHR isolation	Always

Table 7.7 PWRs With Minimal Shutdown Mitigation Capability

Total Annualized CDF RCS open: 3E-5 (per calendar year) Total Annualized CDF RCS closed: 3E-6 (per calendar year)	
Item	Value
RHR pumps	2
Other heat removal pumps	0
ECCS pumps (in standby)	1
RCS vents and pressure control	Yes
CCW pumps/trains	2 trains
SW pumps/trains	2 trains
Containment Spray pumps (as back up to the RHR pumps)	0
Gravity Feed	Yes
Accumulators	0
Steam Generators	Yes
Containment sumps	Yes, but not fully reliable
Other borated water sources	0
Other means of removing heat	0
Offsite power sources	2
EDGs	1
Other onsite power sources	0
Level instruments	2 some of time
Vessel temperature Instruments	2 some of time

Table 7.8 PWRs With In-depth Shutdown Mitigation Capability

Total Annualized CDF RCS open: 1E-7 (per calendar year) Total Annualized CDF RCS closed: 8E-7 (per calendar year)	
Item	Value
RHR pumps	2
Other heat removal pumps	0
Charging Pumps	1
ECCS pumps (in standby)	1
RCS vents and pressure control	Yes
CCW pumps/trains	2 trains
SW pumps/trains	2 trains
Containment Spray pumps	2 as piggy back to the RHR pumps
Gravity Feed	Yes
Accumulators	0
Steam Generators	Yes
Containment sumps	Yes, enhanced reliability
other borated water sources	0
other means of removing heat	0
Offsite power sources	2
EDGs	2
other onsite power sources	0
Level instruments	2 at all times
Vessel temperature Instruments	2 at all times

0609H-08 REFERENCES

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8. NUREG/CR-5432 "The Probability of Liner Failure in Mark-I Containment" August, 1991.

9. NUREG/CR-6427 "Assessment of the DCH Issue for Plants with Ice Condenser" April, 2000.

10. NUREG/CR-4330 "Review of Light Water Reactor Regulatory Requirements" June, 1986.

11. NUREG/CR-1493 "Performance- Based Containment Leak-Test Program" September, 1995.

12. 51 FR 28044 "Safety Goals for the Operations of Nuclear Power Plants; Policy Statement; Republication" August 1986.

13. NUREG-2195 "Consequential SGTR Analysis for Westinghouse and Combustion Engineering Plants with Thermally Treated Alloy 600 and 690 Steam Generator Tubes, Final Report" May 2018.

14. Westinghouse AP1000 Design Control Document Rev. 19 – Tier 2 Chapter 19 – Probabilistic Risk Assessment – Sections 19.59 PRA Results and Insights. ML11171A411. June 2011.

15. Vogtle, Units 3 and 4, Updated Final Safety Analysis Report, Revision 2, Chapter 19, Probabilistic Risk Assesment. ML13206A152. June 2013.

Attachment 1 Guidance for Assessing the Timing of Protective Actions in Detailed Risk Evaluations

SDP LERF treatment typically relies on a general, functional definition of LERF, rather than a more detailed accounting of accident progressing timing against protective action timing. While this more general treatment is usually sufficient for an SDP, experience has shown that it is infrequently necessary to evaluate the timing of protective actions relative to radiological release on an accident sequence basis as part of a detailed risk evaluation. In these situations, the guidance below may be helpful, and should be considered in tandem with other considerations specific to the SDP in question. This level of detail may not be warranted, particularly if the available information on core damage and containment failure timings is not well-characterized.

1. Early declaration, when warranted, may be credited on a probability basis. Example: EALs provide for SRO judgement in some circumstances. 50% probability that SRO declares event early given the plant damage state.

2. It should be assumed that emergency action level monitoring and protective action recommendations are made in a timely manner (e.g. declaration made within 15 minutes of relevant plant conditions and protective action recommendations made 15 minutes thereafter).

3. For external hazards well beyond the design basis (e.g. seismic bins in the upper end of the seismic hazard), some impact on response capabilities is possible but also beyond the state-of-the-practice to model. If these types of events are particularly important to a particular risk evaluation, a sensitivity study could be used to address this aspect.

4. Evacuation time estimate (ETE) studies have been performed, and are periodically updated for all sites. These studies are appropriate sources of information for use in SDP assessments. It is understood that these studies are developed for other purposes, but they often represent the “best available information” with respect to evacuation timing, which meets the intent of SDP. They typically provide estimates of the time between the start of evacuation to the time the last of the individuals have cleared the 10-mile boundary for a range of conditions and assumptions.

5. In using ETE studies as part of the best available information for LERF determination in a detailed risk evaluation:

    a. ETE studies typically present timing for evacuations of emergency response planning areas (ERPAs). ERPAs are typically defined by compass sectors and distances of 2, 5, and 10 miles. For LERF, it is the 2 mile population that is most relevant.

    b. ETE studies typically present timings for the time to evacuate 90% and 100% of the population in particular ERPAs and combinations of ERPAs. The analyst should use the time estimates associated with evacuating 90% of the population, as this represents a reasonable tradeoff between the inclusiveness of the 100% value, versus the fact that the timings represent the time to reach the 10 mile boundary (whereas dose levels will likely drop below those of concern for LERF prior to that distance).

c. ETE studies present times for different scenarios (e.g., day time, night time, winter storms, and roadway impacts). When considering the spectrum of applicable scenarios If the LERF determination is not sensitive to the range of time estimates of relevance, use the most inclusive time.

d. ETE studies also present timings for different evacuation assumptions (i.e., keyhole evacuation, 5-mile 360 degree evacuations, and 10-mile 360 degree evacuations). A judgement should be made as to which of these is most applicable.

6. There are a number of aspects of the above assumptions that are uncertain, and there are aspects of the protective action implementation that are outside of the licensee's control. For this reason, it is appropriate to perform a sensitivity study that shows how ΔLERF would differ if a more optimistic or pessimistic set of assumptions are employed.

Attachment 2: Revision History for IMC 0609, Appendix H

Commitment Tracking Number	Accession Number Issue Date Change Notice	Description of Change	Description of Training Required and Completion Date	Comment Resolution and Closed Feedback Form Accession Number (Pre-Decisional, Non-Public Information)
n/a	04/21/2000 CN 00-007	Intial issuance		N/A
n/a	ML041340009 05/06/04 CN 04-010	Periodic update		N/A
N/A	ML18243A521 02/25/19 CN 19-008	Changes included: • Removed all references to Phase 2 obosolete system notebooks • Added SRA to the list of people that can perform a Phase 3 analysis. • Added a reminder that Saphire can now be used to perform LERF calculations. • Removed the step to multiply the LERF score by a factor of 3.3 from table 5.3 (now table 6.3) since it is not correct to do so. • Revised figure 5.1 (now table 6.1) so it no longer mentions SDP Phase 2 notebooks. • Provided new definitions for “close in population” and “effective evacuation”. • Created a new Attachment 1 which is a Guidance for Assessing the Timing of Protective Actions in Detailed Risk Evaluations. • Addressed feedback form 0609H-2225. Added a new section (section 0609H-05) for C-SGTR and revised table 6.1 to add a row for Combustion Engineering Plants. As a result of adding section 0609H-05, subsequent sections and table numbers were changed.	N/A	ML18247A003 0609H-2225 ML18352A703

Commitment Tracking Number	Accession Number Issue Date Change Notice	Description of Change	Description of Training Required and Completion Date	Comment Resolution and Closed Feedback Form Accession Number (Pre-Decisional, Non-Public Information)
		Table 7.1 has been revised to add a new column for ice condenser flow blockage. Added a note to tables 7.2 and 7.4 to provide more information on failure of multiple igniters for Ice Condenser and BWR Mark III plants. Added a new section 01.05 regarding the use of licensee provided LERF information. Revised definition for Shutdown Operation to better align with IMC 0609, Appendix G.		
	ML20078L336 03/23/20 CN 20-017	IMC 0609, Appendix H has been modified to assess AP1000 reactor design. 0609H2 – Limitations and Precautions: • Alerted analysts that this is the first introduction of AP1000 into Appendix H, and if they have a basis for why this procedure is not adequately capturing the risk, they may depart from this procedure and perform a Phase 3 detailed risk evaluation. • Revised the line item about ISLOCAs that the path outside containment is assumed to be not submerged, and nor does it benefit from other means of fission product retention. This sentence was revised based on insights from SOARCA that substantial ISLOCA fission product retention could result from means other than break submergence. • A CDF value was assigned for AP1000 of 1E-6/ry. This value will be re-visited as operating and PRA modeling experience is gained with the AP1000 design. • Shutdown definitions have been removed from IMC 0609, Appendix H, they are more appropriately located in IMC 0609, App G, Shutdown Operations Significant Determination Process. • Reference #14 has been added to the references.	N/A	ML19352E278

Commitment Tracking Number	Accession Number Issue Date Change Notice	Description of Change	Description of Training Required and Completion Date	Comment Resolution and Closed Feedback Form Accession Number (Pre-Decisional, Non-Public Information)
		• Section 5.01 has been added regarding C-SGTR in AP1000 reactors, and section 6.01 has been revised for C-SGTR. • Tables 6.1, 6.2, 6.4, 7.1, 7.2, 7.3 have been revised to accommodate AP1000 reactor design. • A couple of abbreviations for AP1000 have been added to section 3.01. • Table 4.1 has been modified to include information about AP1000 hydrogen igniters and the ADS system. The information about stage 4 ADS came from the Vogtle Units 3 and 4 FSAR table 19.59-18. • Table 6.4 note 1 – the requirement to notify NRR/SPSB for open containments per SRM 97-168 was removed because the requirement to do so has been replaced by a change to 10 CFR 50.65(a)(4) (64 FR 38557) and the issuance of RG 1.160 that imposed a requirement to manage risk maintenance activities and clarified that the requirement applied during shutdown states. • The wording was changed on note 2 of table 7.4 since the SDP is the NRC staff process not the licensee. • Table 7.1 for PWR Ice Condensers was changed to Perform a Phase 2 for issues with Igniters or Drywell / Containment Sprays. This was mistakenly changed to Not Applicable in the 2019 revision to Appendix H, and is now being corrected.		