TECHNICAL BASIS

CONTAINMENT INTEGRITY SIGNIFICANCE DETERMINATION PROCESS
(IMC 0609, App H)

FOR TYPE A AND TYPE B FINDINGS

FULL POWER AND SHUTDOWN OPERATIONS

TABLE OF CONTENTS

LIST OF TABLES AND FIGURES

List of Tables

List of Figures

Technical Basis for Type A and Type B Findings - Full Power Operations

Introduction

The screening criteria and technical basis for the Phase 2 assessment guidance for Type A and Type B findings are discussed below, for each of the five reactor and containment design combinations. The screening criteria and the numerical factors are based largely on NUREG-1765, "Basis Document for Large Early Release Frequency (LERF) Significance Determination Process," (Reference 6), to which the reader is referred for more detail.

1.0   Type A Findings

A subset of the findings processed through the core damage frequency (CDF) based significance determination process (SDP) can also affect LERF. Findings of this nature are classified as "Type A" in Figure 4.1. Guidance for the assessment of Type A findings is presented in the form of screening criteria to identify those accident sequences that contribute to LERF. These screening criteria are based on the characteristics of the sequence (see Table 5.1). If a finding is evaluated to affect an accident sequence that contributes to LERF, that sequence is examined further in a Phase 2 assessment.

The containment SDP (Appendix H to IMC 609) is based on the identification of a factor that represents the conditional probability of a large early release for each LERF-contributing sequence. The factor is dependent on the nature of the accident sequence (see Table 5.2). The change in LERF is derived from the changes in the accident sequence frequencies in the following way:

$$\Delta LERF = \sum_i f_i * \Delta CDF_i \qquad\qquad (1)$$

where the index i runs over the accident sequences that contribute to LERF.

Only certain core damage sequences represent a challenge to containment and are different for each reactor/containment type combination. The sequences are summarized below. Core damage sequences that do not pose a threat to containment integrity would not result in a significant increase in LERF and are therefore not included in Equation (1) above.

In general, the information used to address containment performance for Type A findings has been derived from the NUREG-1150 study (Reference 7) and supplemented by studies addressing resolution of selected severe accident issues. The factors for Type A findings given below are based on average values for each reactor/containment type combination. Plant specific values may be used to obtain the risk significance of a finding for individual plants when performing a Phase 3 assessment.

1.1   Accident Sequence Screening Criteria for Type A Findings

The CDF-based SDP will result in a risk categorization for the finding, and an assessment of the types of accident sequences whose frequency would increase as a result of the finding. Initially all inspection findings pass through the screening decision criteria shown in Table 5.1. The finding is assessed for each of the accident categories contributing to the risk significance ranking from the CDF-based SDP for various reactor and containment type combinations. The outcome of the screening process shown in Table 5.1 is whether or not a Phase 2 assessment needs to be performed with respect to LERF.

BWR Accidents Important to LERF

For BWR plants with Mark I and Mark II containments, findings related to interfacing system loss of coolant accidents (ISLOCA), anticipated transient without scram (ATWS), and accidents resulting in high reactor coolant system (RCS) pressure (i.e., transients and small break LOCA) need to be further evaluated in Phase 2. For Mark I plants, core damage 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 2 to Table 5.1. For BWR Mark III plants, findings related to ISLOCA, transients, small break LOCAs, and station blackout (SBO) categories need to be further evaluated in Phase 2.

ATWS sequences are <u>not</u> an important contributor 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 by the pool and a large early release does not occur.

PWR Accidents Important to LERF

For PWR plants with large dry and sub-atmospheric containments, findings related to the accident categories ISLOCA and steam generator tube rupture (SGTR) need to be further evaluated in Phase 2. For the PWR plants with ice condenser containments, findings related to ISLOCA, SGTR, and SBO accident categories need evaluation in Phase 2.

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. However, if the igniters are not available (as they would be in an SBO), NUREG/CR6527 indicates a conditional containment failure probability (CCFP) close to 1.0.

ATWS sequences are <u>not</u> significant contributors to LERF for PWRs. During ATWS in PWRs, the containment pressure increases slowly so it can be considered a late failure

mode. Therefore, the risk significance determined by the CDF based SDP alone for ATWS events in PWRs is appropriate.

High and low pressure core damage sequences (in which the containment is not bypassed) are <u>not</u> 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 on the occurrence of 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.

1.2   LERF Factors for LERF-Significant Sequences

LERF factors for sequence types affecting LERF associated with Type A findings are shown in Table 5.2 and discussed individually below.

<u>ISLOCA</u>

The ISLOCA scenario occurs when isolation valves between the high pressure RCS and a low pressure system fail causing a LOCA outside containment. If core damage occurs, the release path can bypass containment and cause a large release to the environment. For BWRs and PWRs, an ISLOCA is potentially a high consequence accident sequence since the containment is bypassed. Although some fission product holdup and scrubbing would occur along the release path, this depends on the break location and plant-specific features and is not credited in this SDP. Thus, the factor is equal to 1.0 for this accident class.

<u>ATWS</u>

ATWS accident sequences are significant contributors to LERF for BWRs with Mark I and Mark II containment designs. These accident sequences result in an energy input to containment that cannot be adequately removed by the normal containment heat removal systems. This results in a rapid pressure rise that may cause the containment to fail before or shortly after core damage. If the suppression pool is bypassed, a large release may occur. The NRC review of individual plant examinations (IPEs), NUREG-1560, Volume 2, indicated that the significance of ATWS events in the various IPEs depends on plant-specific features such as the ability of pumps to operate with saturated water, and on assumptions about the power level, the period in the fuel cycle when the event occurs, and the effectiveness of operator response.

For Mark I and II containments, the Factor which represents conditional probability of early containment failure and suppression pool bypass due to ATWS sequences is estimated by the equation below:

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

where: Factor is the multiplier on the CDF
              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.

Based on data from the IPE program, averaged across Mark I plants, C_{PEF} = 0.6 and C_{PPB} = 0.5. This results in a Factor of 0.3 for ATWS sequences. The corresponding average values for Mark II containments are C_{PEF} = 0.6 and C_{PPB} = 0.7 resulting in a Factor of 0.4 .

Steam Generator Tube Ruptures (SGTR)

SGTRs are initiating events that can lead, after further failures, to core damage. If core melt occurs, and the secondary side is open, the release path can bypass containment. It was conservatively assumed that all SGTR core damage sequences would release sufficiently large amounts of radioactive material to the atmosphere outside containment that, for SDP purposes, would be treated as sequences important to LERF. As the containment is bypassed for SGTR sequences, the conversion factor for Type A findings is 1.0 for this accident class.

Mark I Plants: Scenarios with the RCS at High Pressure and Drywell Floor Flooded/Dry

Information provided in NUREG/CR-6595 (Reference 9) for Mark I plants indicates that if the RCS is at high pressure at vessel breach and the drywell floor is dry, then the conditional probability of containment failure is 1.0. If the RCS is at high pressure at vessel breach and the drywell floor is flooded, then the average conditional probability of containment failure in Mark I plants is 0.6. NUREG/CR-5423 (Reference 10) documented an analysis to evaluate the probability of containment liner melt through in Mark I plants for sequences where the vessel fails at low pressure. This analysis considered both oxidic and metallic pours with and without water on the drywell floor. If there is no water on the drywell floor, the conditional containment failure probability of the containment from corium attack was estimated to be 1.0 for oxidic melts and 0.63 for metallic melts. Given the uncertainty in core melt progression and molten debris composition, for sequences involving drywell floor at vessel breach, a factor of 1.0 is conservatively assumed in the SDP, regardless of RCS pressure. The same study indicates that if the reactor coolant system pressure is low and the drywell floor is flooded then the conditional containment failure probability and factor is less than 0.1.

Mark II Plants: Scenarios with the RCS at High Pressure

Based on data provided in NUREG/CR-6595 (Reference 9) for Mark II plants, the likelihood of containment failure given a vessel breach is approximately 0.3 if the RCS is at high pressure. This probability is relatively independent of whether the pedestal or drywell floor is flooded or not for the high pressure scenarios.

Mark III Plants: Scenarios with RCS at High Pressure and SBO Sequences with RCS at High or Low Pressure

Mark III plants have a drywell and suppression pool that is totally enclosed within a containment structure. Thus, both the containment and the drywell have to fail or be bypassed for a large release to occur. The NUREG-1150 (Reference 7) study indicates that the likelihood that an accident sequence contribute to LERF depends on two factors: (1) whether the accident is a SBO sequence (hydrogen igniters not operating and containment sprays unavailable) and (2) the pressure in the RCS at the time of vessel breach. In a SBO sequence, the conditional probability of a large early release is approximately 0.2 (due to hydrogen combustion) regardless of whether the RCS is at high or low pressure. If the RCS is at high pressure, the conditional probability of failure of both containment and the drywell is also about 0.2 (due to overpressure failure). Thus, for findings related to any SBO or transient with high RCS pressure the Factor would be 0.2.

Ice Condenser Plants: Station Blackout Scenarios

Like BWR Mark III plants, ice condenser containments rely on AC-powered glow plug igniters to control hydrogen released during core damage accidents. A recent detailed study of severe accident phenomena in ice condenser plants NUREG/CR-6427 (Reference 11) indicates that the most significant challenge to containment integrity arises from SBO sequences in which the igniters are not available. The average CCFP for SBO sequences from hydrogen combustion across five ice condenser plants is about 0.6 (range from 0.22 to 0.97). For a screening SDP, this Factor is conservatively assessed a value of 1.0.

2.0   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. Type B findings are fundamentally different from Type A findings in that they are not processed through the CDF based SDP and have to be allocated significance categories based only on LERF considerations. Table 1A-1 provides a list of SSCs associated with maintaining containment integrity in different containment types that were reviewed to determine if ΔLERF would be affected if the SSCs were found to be degraded.

As the containment function may be compromised for a Type B finding it can potentially affect either all core damage accidents or a subset of those accidents. 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}/reactor year for PWRs and 10^{-5}/ reactor-year for BWRs. Assuming a baseline CDF is a limitation that has to be recognized in the light of the relatively wide ranges associated with Plant-Specific CDF estimates.

The risk significance categories are obtained using the following relationship:

$$\Delta LERF = F_B \times F_R \times CDF \times \text{(multiplier for the duration of degraded condition)}$$

where:

F_B = increase in the conditional large early release probability resulting from the finding, and

F_R = fraction of the CDF for which the containment failure mode affected by the finding impacts LERF.

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

<u>Duration Multiplier</u>
>30 days 1.0
30-3 days 0.1
< 3 days 0.01

For Type B findings, the factor F_B is the difference between the conditional probability of large early release assuming complete failure of the SSC and the original conditional probability of large early release assumed in the baseline risk estimate. F_B is multiplied by the fraction F_R of CDF that pertains to the phenomena that impact (or are impacted by) the failure of the SSC under consideration.

For example, suppose a finding in a BWR with a Mark III containment or a PWR with an ice condenser containment indicates that a significant number of hydrogen igniters are not operable. Potentially, such a finding implies that the containment is vulnerable to failure from a hydrogen deflagration or detonation after a core damage accident. However, the igniters require AC power to operate and would therefore not be available anyway in a SBO accident. Hence, the risk significance of the finding pertains only to the non-SBO portion of the total CDF. Generic values for these parameters are provided in the phase 2 guidance contained herein. In a phase 3 evaluation, the plant-specific CDF for the major accident classes, as well as a more precise estimate of the multiplier (if the actual duration can be established), should be taken into consideration in making a final significance determination.

If F_B is 1.0 and duration of the degraded condition is >30 days, the implication is that ΔLERF is equivalent to either the total CDF or the fraction F_R of CDF that is relevant to the containment function that is compromised. If the total CDF is involved, then, under these circumstances (refer to Table 1.1), the risk significance based on LERF is red for both PWRs and BWRs. As the product of F_B and F_R decreases, the influence of the containment SSC on the determination of LERF also decreases correspondingly. When this product reaches 0.1, the implication is that the significance category becomes one order of magnitude lower, i.e. the LERF significance is equivalent to the CDF significance.

The bases for the significance categories for each of the five reactor/containment combinations are provided in summary below. 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 (Reference 8). Containment related findings that are not

addressed in this LERF SDP are not expected to be risk significant with respect to LERF.

The characteristics of the severe accidents that contribute to LERF and those SSCs important to maintain containment integrity are plant specific and depend upon the containment design (i.e., large containment volume versus the various pressure suppression designs). The screening criteria are therefore presented below in the context of five containment designs: BWRs with Mark I, II, or III containments, PWRs with large dry and sub-atmospheric containments, and PWRs with ice condenser containments.

2.1   Phase 1 Screening Decision Criteria for Type B Findings

Table 1A-1 lists the containment SSCs that were reviewed to determine the impact on ΔLERF if they were found to be in a degraded. Table 6.1 provides the Phase 1 screening criteria for Type B findings for all reactor and containment type combinations. Findings involving containment SSCs that do not appear in Table 6.1 may be considered to be screened out in Phase 1. Main steam isolation valve (MSIV) leakage as a containment SSC failure mode is screened out for BWR Mark III plants since these containments have a safety-grade main steam shut-off valve (MSSV) that is a relatively slow closing low leakage valve. Thus, leakage past the MSIV in a Mark III plant would likely be stopped by the MSSV.

2.2   LERF Factors for LERF-Significant SSCs

Each Type B finding that was not screened out in Phase 1 requires a Phase 2 assessment. The risk significance of the various findings, i.e., color assignment, are summarized in Table 6.2 in terms of the duration that the condition was estimated to exist. Table 1A.2 provides details of the values of F_B and F_R for different reactor and containment type combinations that were used to generate the risk significance color assignments. The bases for assigning these values are discussed below.

The values of F_B and F_R given in the tables below are average values for each reactor/containment type combination. Plant specific values of these parameters may be substituted to obtain the risk significance of a finding for individual plants when performing a Phase 3 assessment.

Reactor and Containment Type	Finding	F_B	F_R	Risk Significance		
				> 30 days	30 - 3 days	< 3 days
BWR Mark I	Leakage from drywell to environment > 100% volume/day through containment penetration seals, isolation valves or vent and purge systems	0.9	1.0	Yellow	White	Green
	Failure of systems/components critical to suppression pool integrity or scrubbing (vacuum breakers or other bypass mechanisms)	0.2	1.0	Yellow	White	Green
	Main steam isolation valve leakage >10,000 scfh through the best sealing valve in any steam line	0.2	0.5	Yellow	White	Green
	Drywell sprays unavailable	0.7	1.0	Yellow	White	Green
BWR Mark II	Leakage from drywell to environment > 100% volume/day through containment penetration seals, isolation valves or vent and purge systems	0.95	1.0	Yellow	White	Green
	Failure of systems/components critical to suppression pool integrity or scrubbing (vacuum breakers or other bypass mechanisms)	0.1	1.0	Yellow	White	Green
	Main steam isolation valve leakage >10,000 scfh through the best sealing valve in any steam line	0.7	0.7	Yellow	White	Green
	Drywell sprays unavailable	0.2	0.3	White	Green	Green
BWR Mark III	Leakage from wetwell to environment >1000% containment volume/day through containment penetration seals, isolation valves or vent and purge systems	0.975	1.0	Yellow	White	Green
	Failure of systems/components critical to suppression pool integrity or scrubbing (vacuum breakers or other bypass mechanisms)	0.23	0.1	Yellow	Green	Yellow
	Failure of multiple igniters such that coverage is lost in two adjacent compartments	0.19	0.2	White	Green	Yellow
	Containment sprays unavailable	0.025	1.0	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	0.9	1.0	Red	Yellow	White
PWR Ice Condenser	Leakage from containment to environment > 100% containment volume/day through containment penetration seals, isolation valves or vent and purge systems	0.9	1.0	Red	Yellow	White
	Blockage of more than 15% of the flow passage into or through the ice bed	0.9	1.0	Red	Yellow	White
	Failure of multiple igniters such that coverage is lost in two adjacent compartments	0.9	0.9	Red	Yellow	White

Table 1A.2 Phase 2 Risk Significance -Type B Findings at Full Power

Containment Penetration Seals, Isolation Valves, Vent and Purge Systems

The risk significance of a finding relative to a loss of containment penetration seals, isolation valves or vent and purge systems is assessed in terms of the leakage rate from containment (drywell to environment for BWR Mark I and II, wetwell to environment for BWR Mark III, and containment airspace to environment for PWR plants). Several studies, including NUREG/CR-4330, NUREG-1493, and NUREG/CR-6418, 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% containment volume/day appears to constitute an approximate threshold beyond which the release may become significant to LERF. In Mark III plants, however, the impact of suppression pool decontamination factors (DFs) has to be taken into account when considering the leakage from these containments. Conservatively, a DF of 10 has historically been used to represent a pool over the entire accident period. In terms of an "early release" this factor is extremely conservative. Including this DF to determine the containment leakage criterion of importance to LERF would imply a wetwell to environment leak rate of about 1000% containment volume/day.

Design basis leakage from containment is determined by regulatory requirements to assure that the containment leakage will be below the design basis (or maximum allowable) leak rate L_a set by the 10CFR Part 100 dose limits that are incorporated in the plant technical specifications. While individual plants have been licensed with various different values of L_a, typical values of L_a are 0.1% containment volume/day for PWRs and 0.5% containment volume/day for BWRs. Thus a LERF significant leakage rate from containment of 100% containment volume/day would correspond to about 1000 L_a for PWRs and 200 L_a for BWR Mark I and Mark II plants. In Mark III plants, the LERF significant leakage rate of 1000% containment volume/day would correspond to 2000 L_a.

The comparison between % containment volume/day and L_a is shown in Table 1A.3 for various reactor/containment type combinations. The 100 volume percent per day leakage rate is approximately equivalent to a hole size in containment of 2.5-3.0 inches in diameter for PWRs with large dry containments, 2 inches for PWRs with ice condenser containments, 1 inch for BWRs with Mark I or II containments, and 2.5 inches for BWRs with Mark III containments.

Table 1A.3 Containment Leakage Rate Significant to LERF		
Reactor/Containment Type	LERF-Significant Containment Leakage Rate	
	% Containment Volume/Day	Design Basis Leakage Rate, L_a
BWR Mark I and II	100	200 L_a
BWR Mark III	1000	2000 L_a
PWR - All	100	1000 L_a

Data generated in the IPE program and in published PRAs suggests that, on average, about 0.1 of the CDF in BWR Mark I plants and 0.05 of the CDF in Mark II plants consists of LERF. Thus, if a finding were to imply the existence of a breach in the drywell pressure boundary that would result in a LERF significant drywell leakage rate shown in Table 1A.3 above, the large release probability would increase to essentially 1.0. The factor F_B is thus (1.0 - 0.1 = 0.9) for Mark I plants and (1.0 - 0.05 = 0.95) for Mark II plants. In Mark III plants, IPE information indicates that, on average, about 25% of CDF consists of early containment failure sequences and about 10% of early containment failure sequences consist of large releases. Hence, on average, about 0.025 of the CDF consists of LERF. The relatively low baseline value of LERF for Mark III plants reflects the fact that containment failure has to be coupled with loss of drywell integrity (i.e., a factor of 0.1) for a large release to occur. If a finding reveals that the containment leakage would exceed the LERF-significant leakage rate (1000% per day for a Mark III, which includes a factor of 10 credit for suppression pool scrubbing), the conditional probability of a large early release (given core damage) would approach 1.0 for all core damage events. The factor F_B for a LERF significant containment leak in Mark III plants is thus (1.0 - 0.025 = 0.975). As the above results are based on the total assumed CDF for BWRs, the value of F_R is 1.0 for findings of this type since essentially all core damage events would lead to a large early release given a large containment leak.

For PWR plants, the conditional probability of early containment failure and bypass given core damage is approximately 0.1. A finding that implies the existence of a breach in containment that would result in a LERF significant containment leak rate shown in Table 1A.3 above would open a direct path to the environment, and hence the factor F_B is (1.0 - 0.1 = 0.9) for PWR plants. As the above are based on the total assumed CDF for PWRs the value of F_R is 1.0 for findings of this type.

Systems Important to Suppression Pool Integrity - BWR Only

These findings pertain to failures of those systems that are important for maintaining suppression pool integrity and preventing suppression pool bypass in BWR plants, vacuum breakers in the wetwell airspace or other bypass mechanisms. The values for F_B and F_R in Table 1A.2 were obtained by reference to previous PRAs and in particular the results of the IPE program, as documented in NUREG-1560 (Reference 8). These values are described in the basis document for LERF SDP (NUREG-1765). The approach adopted for determining the risk results was to determine the average

conditional probability of early containment failure (given core damage) if the suppression pool was bypassed. The following conditional early containment failure probabilities are provided in NUREG-1765: 0.3 for Mark I containments, 0.15 for Mark II containments and 0.25 for Mark III containments. The baseline average conditional probability of large early release given core damage was then determined for each containment, type. The baseline conditional early containment failure probabilities reported in NUREG-1765 are: 0.1 for Mark I containments, 0.05 for Mark II containments and 0.025 for Mark III containments. The finding is assumed to result in complete suppression pool bypass and therefore the difference between the above conditional probabilities (e.g., 0.3 - 0.1 = 0.2 for BWR Mark I plants, 0.15 - 0.05 = 0.1 for Mark II plants, and 0.25 - 0.025 = 0.23 for Mark III plants) for each containment type represents F_B (i.e., the increase in conditional probability of the finding). As the above results are based on the total assumed CDF for BWRs, the value of F_R is 1.0 for findings of this type.

Main Steam Isolation Valves - BWR Mark I and II Only

Excessive leakage from the MSIVs during a core melt accident in BWR Mark I plants can lead to a release path that bypasses containment and can impact LERF. Core melt accidents involving excessive MSIV leakage in Mark I and Mark II plants with the RCS at high pressure are similar to induced SGTRs in PWR plants. A finding in a BWR Mark I plant that indicates excessive leakage from the MSIV in excess of 10,000 scfh (standard cubic feet per hour) or an inability to quantify the leakage rate would lead to the risk significance category indicated in Table 1A.2.

The values for F_B and F_R in Table 1A.2 were again obtained by reference to previous PRAs and in particular the results of the IPE program (as documented in NUREG-1560). In order to determine F_R it was assumed that MSIV leakage is only risk significant for those accident sequences in which the RCS is at high pressure during core melt. The IPE results for BWR Mark I and Mark II plants were reviewed to determine the fraction of the average CDF that consists of high RCS pressure sequences for the two containment types of interest. These fractions were then equated to F_R. Based on the IPE results, F_R = 0.5 for Mark I plants and F_R = 0.7 for Mark II plants.

The values for F_B were determined by obtaining the average conditional probability of early failure and large release for high RCS pressure sequences for each of the two containment types (i.e., 1.0 for Mark I containments if the drywell floor is dry, 0.6 for Mark I containments if the drywell floor is flooded and 0.3 for Mark II containments independent of whether the drywell floor is dry or flooded). The finding is assumed to result in a bypass probability of 1.0 for all high pressure sequences and therefore the difference between the above conditional probabilities for each containment type represents F_B (i.e., the increase in conditional probability of the finding). For Mark II containments, F_B = 1.0 - 0.3 = 0.7. However, for Mark I containments, F_B depends on whether or not the drywell floor is flooded. If it is assumed that the drywell floor is flooded 50% of the time, based on IPE results, the change in conditional probability can be determined to be as follows:
F_B = 0.5 x (1.0-1.0) + 0.5 x (1.0-0.6) = 0.2.

Drywell/Containment Sprays - BWR only

Using drywell sprays to flood the drywell floor during core melt accidents in Mark I plants is an important strategy for reducing the likelihood of LERF. In Mark II plants, containment sprays can impact accidents that bypass the suppression pool by scrubbing the containment atmosphere thus converting a potentially large release into a scrubbed release. In Mark III plants, containment sprays can be used to provide some fission product scrubbing.

As noted above, using the drywell sprays to flood the drywell floor during core meltdown accidents in Mark I containments is an important strategy for preventing liner melt-through and hence lowering the likelihood of LERF. In addition to the status of the drywell floor (flooded or dry), the pressure in the reactor vessel also has a significant impact on the likelihood of LERF. Results 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 were to imply that the drywell floor could not be flooded (using any available means such as residual heat removal (RHR) pumps or diesel-driven pumps) then the large release probability of 0.6 would approach 1.0 for high pressure scenarios and would lie in the range of <0.1 to 1.0 for low pressure scenarios. F_B is therefore approximately equal to {0.5 x (1.0 - 0.6) + 0.5 x (1.0 - <0.1)} = 0.7 for findings of this type. This assumption assumes that all liner melt-through failures result in LERF and also neglects the effect of pool scrubbing for those sequences in which the in-vessel release passes through the suppression pool. As the above results are based on the total assumed CDF for BWRs the value of F_R is 1.0 for findings of this type.

For those sequences where sprays are available, spray operation can be beneficial, as mentioned above, for those accidents that challenge Mark II containments. In these cases, timing is an important issue and there is also a great deal of uncertainty in assessing the significance of spray operability to reducing LERF. An approximate estimate of the contribution of sprays can be made based on the results reported in NUREG/CR-6595. The CCFP for low pressure sequences that fail containment early is 0.3 without water on the drywell floor and 0.1 or less with water on the drywell floor. (The CCFP for high pressure sequences that fail containment early is unaffected by the presence of water). Assuming conservatively, that water on the drywell floor can be ascribed to successful spray operation, then based on the results reported in NUREG/CR-6595, F_B = 0.3 - 0.1 = 0.2 for accident sequences in Mark II plants in which the RCS is at low pressure. If we further assume, again based on the IPE results, that about 30% of the CDF sequences are low pressure sequences, then F_R = 0.3 in Mark II plants.

IPE information indicates that, on average, about 0.025 of the CDF 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 approaches conservatively a conditional probability of 0.025 for a large early release assuming that the sprays are effective for mitigating the LERF contributors. Therefore F_B = 0.025 in Mark III plants for a finding related to spray operation. As these results are based on the assumed CDF for BWRs the value of F_R is 1.0 for findings of this type.

Hydrogen Igniters - BWR Mark III Plants

These findings pertain to the operability of the glow plug hydrogen igniter system (i.e., are all the glow plugs functioning) in BWR Mark III plants under conditions where AC power is available. The loss off all igniter coverage in two or more adjacent compartments is considered potentially risk significant for hydrogen combustion events.

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 (high pressure and low pressure) and non-SBO 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. 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. If a portion 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.

In the extreme, if none of the igniters were operating, the probability of early containment failure from non-SBO depressurized sequences would approach 0.2 from hydrogen detonation or energetic deflagration. Therefore F_B = 0.2 - 0.01 = 0.19 for findings that imply loss of igniter system effectiveness.

F_R can be determined by assuming a contribution to the 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). Based on IPE results, the fraction of core damage involving SBO is about 0.5, and the fraction of core damage events involving low RCS pressure is about 0.4. Assuming that this latter 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. Therefore, F_R = 0.2 for findings of this type in Mark III plants.

Hydrogen Igniters - PWR Ice Condenser Plants

These findings pertain to the operability of the glow plug hydrogen igniter system (i.e., are all the glow plugs functioning) in PWR ice condenser plants under conditions where AC power is available. The loss off all igniter coverage in two or more adjacent compartments is considered potentially risk significant for hydrogen combustion events.

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). The probability of early containment failure from non-SBO sequences from hydrogen combustion events is zero because the igniters are operating and burning the hydrogen at low concentrations as it enters the upper compartment of the containment.

If a portion of the igniters were not operating, the local concentration of hydrogen would increase until it was ignited, either by a working igniter elsewhere or by random ignition(e.g., static discharge).

In 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. Since the average conditional probability of early failure is about 0.1 in ice condenser containments, F_B = 1.0 - 0.1 = 0.9.

However, the unavailability of igniters is a risk contributor only in non-SBO accident sequences, hence we should consider only 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 one percent to 21 percent with an average of approximately 10 percent. Therefore F_R = 0.9 for findings of this type in PWR ice condenser plants.

Ice Condenser Integrity - PWR Ice Condenser Plants Only

Findings related to the integrity of the ice condenser involve failure of some of the ice chest doors to open or a gross build-up of ice or frost that results in a substantial blockage (15 percent or more of the area) of the flow path between ice baskets. There is also a possibility of a significant loss of ice during or prior to an accident. 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, F_B = 1.0 - 0.1 = 0.9 for loss of ice condenser integrity. As these results are based on the assumed total CDF for PWRs, the value of F_R is 1.0 for findings of this type.

Technical Basis for Type A and Type B Findings - Shutdown Operation

Definitions

Similar to information used for Appendix G to IMC 609, the guidance for assessing containment findings for plant shutdown divides an outage into Time Windows (TWs) and Plant Operating States (POSs). Shutdown LERF deficiencies are analyzed according to what TW and POS the finding occurred in. For each TW and POS, the risk of that plant configuration is assumed to stay constant.

Plant Operational States

For PWRs, there are three POSs

POS 1 - This POS starts when the RHR system is placed in service. The RCS is closed such that a steam generator could be used for decay heat removal, if the secondary side of a steam generator is filled. The RCS may have a bubble in the pressurizer. This POS ends when the RCS is vented such that the steam generators cannot sustain core heat removal. This POS typically includes Mode 4 (hot shutdown) and portions of Mode 5 (cold shutdown).

POS 2 - This POS starts when the RCS is vented such that: (1) the steam generators cannot sustain core heat removal and (2) a sufficient vent path exists for feed and bleed. This POS includes portions of Mode 5 (cold shutdown) and Mode 6 (refueling). Reduced inventory operations and midloop operations with a vented RCS are subsets of this POS. Note: performance deficiencies occurring during a vacuum refill of the RCS require use of the POS 1 event trees.

POS 3 - This POS represents the shutdown condition when the refueling cavity water level is at or above the minimum level required for movement of irradiated fuel assemblies within containment as defined by Technical Specifications. This POS occurs during Mode 6.

For BWRs, there are three POSs

POS 1 - This POS starts when the RHR system is put into service. The vessel head is on. This POS covers part of Hot Shutdown (Mode 3) and Cold Shutdown (Mode 4) of the TS Modes.

POS 2 - This POS represents the shutdown condition when the vessel head is removed and reactor pressure vessel water level is less than the minimum level required for movement of irradiated fuel assemblies within the reactor pressure vessel as defined by Technical Specifications. This POS occurs during Mode 5.

POS 3 - This POS represents the shutdown condition when the reactor pressure vessel water level is equal or greater than the minimum level required for movement of irradiated fuel assemblies within the reactor pressure vessel as defined by Technical Specifications. This POS occurs during Mode 5.

Time Windows

For both BWRs and PWRs, there are two 2 Time Windows

Early Time Window (TW-E)- This time widow represents the time before POS 3 is entered. The decay heat is relatively high. The reactor is either in POS 1 or 2.

Late Time Window (TW-L)- This time window represents the time after POS group 3. The decay heat is relatively low. The reactor is either in POS 1, 2, or 3.

Figures 1B.1 and 1B.2 show a representative BWR and PWR outage divided into TWs and POSs.
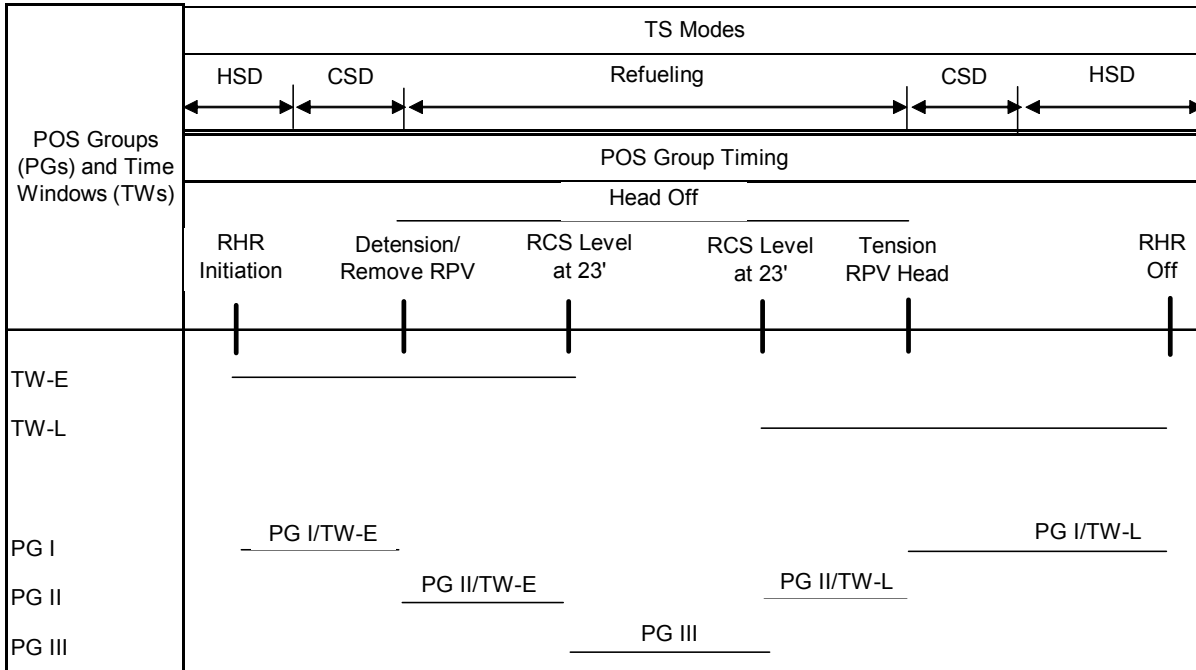
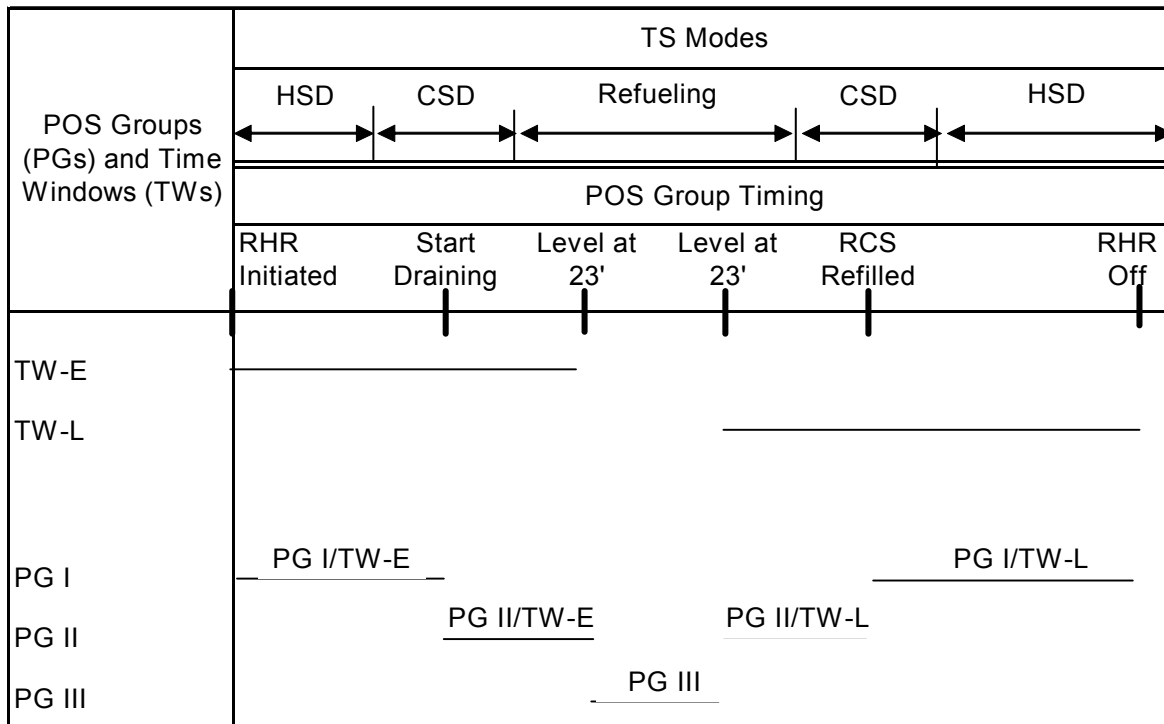**Figure 1B.1   POS and Time Windows for BWRs at Shutdown**

Figure 1B.2   POS and Time Windows for PWRs at Shutdown

Containment Status

Intact Containment

An intact containment is one in which, the licensee intends to: (1) close all containment penetrations with a single barrier, or has procedures to enable the penetrations to 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 the licensee intends to close the containment prior to boiling the RCS inventory. If the licensee does not intend to maintain an intact containment, then containment is open. To be considered an intact containment, the containment leak rate to the environment would be less than the values provided in Table 1A.3.

De-inerted Containment

A de-inerted containment is one in which limits on the primary containment oxygen concentration as defined in Technical Specifications for BWR Mark I and Mark II containments are no longer maintained.

Open Containment

An open containment is one in which the licensee does not intend to maintain an intact containment.

Type A Findings at Shutdown - Phase 1 Screening

For both PWRs and BWRs, In TW-L, it is assumed that LERF cannot occur due to decay of the short-lived isotopes that are principally responsible for early health effects (mainly I and Te). Thus, no LERF significance determination issue is applicable for TW-L. However, LERF can potentially occur in TW-E for both PWRs and BWRs.

For both PWRs and BWRs, a finding that is associated with a core damage scenario is considered a potential LERF scenario during the first eight days of shutdown. The reason for this is twofold: (1) the failures of containment function of most concern happen relatively close to the onset of core damage, and (2) difficulty of making a case that evacuation would have been initiated early enough to prevent a large release. For the findings that are screened out, the risk significance category is determined by the CDF based SDP and need not be changed due to LERF considerations.

Type A Findings at Shutdown - Phase 2 Assessment

As stated above, each core damage scenario occurring during the first eight days of shutdown is considered for LERF at shutdown. The Factor for Type A findings related to these accidents are shown in Table 5.3 for various reactor/containment type combinations. The factors are identified according to the status of containment.

For each core damage scenario that (1) involves an open containment and (2) the finding occurs during the first eight days of the outage, the risk significance category of the finding determined by the CDF based SDP should be increased by one order of magnitude.

For each core damage scenario that (1) involves an intact containment and (2) the finding occurs during the first eight day s of the outage, the Factor and risk significance category of the finding is different for the various reactor and containment types as shown in Table 5.4 of Appendix H.

For BWR Mark I and II plants, the Factor is 1.0 as these plants are vulnerable to hydrogen combustion events since containment is de-inerted at shutdown. Therefore, for each LERF core damage scenario, the risk significance category of the finding determined by the CDF based SDP should be increased by one order of magnitude.

BWR with Mark III containments are also vulnerable to hydrogen combustion events especially since there are no requirements for the igniter system to be available at shutdown. If the igniters are not available, the LERF Factor is 0.2 and the finding becomes a candidate for increase of risk significance category. The factor of 0.2 is derived from the discussion in Section 5.2.3 of NUREG-1765. If the igniters can be recovered by operator action and are made available, the finding can be screened out.

For PWR ice condenser plants, with an intact containment, there are no technical specifications for the hydrogen igniter system to be available once shutdown is entered. If the igniters are not available, it is assumed that the containment is vulnerable to hydrogen combustion events and the LERF Factor is 1.0 for the risk significant core damage accidents. In this case, the CDF based risk significance category should be increased by one order of magnitude. However, if the igniters can be recovered by operator action and are made available, the finding can be screened out.

Type B Findings at Shutdown - Phase 1

Phase 1 screening of Type B findings at shutdown is shown in Table 6.3. The most important feature at shutdown relative to containment SSCs for all reactor/containment type combinations is the status of the containment.

For PWRs and BWRs, the analyst needs to know if the containment is intact or open. If the licensee does not intend to maintain an intact containment, then containment is open. A Type B finding results when a licensee intends to have an intact containment, but cannot maintain an intact containment due to a performance deficiency.

For BWR Mark 1 and Mark II containments, it is assumed that, the containment is de-inerted. Once POS-1 has been entered, and that a core damage accident will lead to loss of containment function with a conditional containment failure probability of 1.0 (due to hydrogen combustion events). Hence containment SSCs do not play a mitigating role in Mark I and II plants at shutdown once the containment is de-inerted, and there are consequently no Type B findings important to ΔLERF in these plants at shutdown.

For BWR Mark III containments and PWR ice condenser containment, the analyst needs to know the status of the hydrogen igniters.

Type B Findings at Shutdown - Phase 2

Phase 2 screening of Type B findings adopts generic baseline CDFs at shutdown during POS 1E and POS 2E. These generic shutdown CDFs were developed to support a quantitative regulatory analysis for the proposed Shutdown Rule described in SECY 97-168. These generic CDFs assumed a 35-day refueling outage each 18 months of operation. These generic shutdown CDFs capture cold shutdown and refueling modes until the refueling cavity is flooded.   For PWRs, the shutdown CDFs cover eight days of cold shutdown and refueling operation until the cavity is flooded . Of these eight days, the first two days are with the RCS closed (POS 1); the last six days are with the RCS "vented" (POS 2). For the BWRs, the analyses covered shutdown operation with the reactor vessel head on for four days (POS 1) and two days with the reactor vessel head off (POS 2).

For both PWRs and BWRs, two voluntary action cases were performed using different interpretations of NUMARC 91-06 and GL 88-17.   (NUMARC 91-06 provides guidance on improving outage management and GL 88-17 provides recommendations concerning the ability of a licensee to mitigate a potential loss of DHR during reduced inventory operations at PWRs.) The higher CDF voluntary case represents a minimum implementation of both guidance documents. The lower CDF voluntary case represents an in-depth implementation of both guidance documents.

Based on staff review of the voluntary action cases, the staff reported in SECY 97-168, that: (1) the existing level of safety at shutdown is largely dependent upon measure that are not traceable to specific underlying regulations, and that could, therefore, be withdrawn by licensee without prior staff approval, and (2) little reduction of risk is achieved by the rule for the licensee who has adopted effective voluntary practices that reduced the for shutdown operation.

A licensee's LERF frequency is strongly dependent on the robustness of a licensee's shutdown mitigation capability. A licensee's shutdown mitigation capability is the set of systems and SSCs that can mitigate a loss or interruption of the RHR function (a shutdown initiating event). Therefore, the analyst must determine whether a licensee has an in-depth shutdown mitigation capability or a minimal shutdown mitigation capability. Tables 6.5, 6.6, 6.7 and 6.8 of Appendix H show what set of systems and SSCs were assumed to be available for a PWR and a BWR with an in-depth mitigation capability and with a minimal shutdown mitigation capability.

To obtain the results in Table 6.4 of Appendix H, the high and low voluntary action case shutdown CDFs from the Regulatory Analysis for SECY 97-168 was used. (See Table B3). For Phase 2 analysis, each shutdown core damage scenario was assumed to lead to a LERF scenarios if: (1) the licensee planned but failed to maintain an intact containment or (2) the containment fails due to loss of hydrogen control in Ice Condenser and Mark III containments (assuming the igniters are not available). To be

considered an intact containment, the containment leak rate to the environment would be less than the values provided in Table 1A.3.

Table B3: POSs Baseline Annualized CDF vs Plant Mitigation Capability

Reactor Type	Shutdown Mitigation Capability	Annualized CDF POS 1E (Head On for four days)	Annualized CDF POS 2E (Head off for two days)
BWR	Minimal	4E-6	1E-6
BWR	In-depth	2E-7	4E-8
PWR	Minimal	3E-6	3E-5
PWR	In-depth	1E-7	8E-7

The risk results in Table 6.4 were obtained by using the following relationship.

ΔLERF = Shutdown CDF (given in Table B3) x Factors given for Type B findings in Table 5.2

NOTE: In Phase 3 analysis, if a licensee can show 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 Table 6.4, there are no duration factors associated with findings at shutdown. As stated earlier, the generic shutdown CDFs include the frequency and duration that POS 1 and POS2 are entered into per calender 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. However, should the duration of a Type B finding exist for less than eight hours, then the color finding should be reduced by one order of magnitude as part of the Phase 2 assessment.

END