Containment Accident Pressure Credit Risk Assessments for Selected Plants

FINAL REPORT
(USER NEED REQUEST NRR-2011-006)

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EXECUTIVE SUMMARY

In 2009 and 2010, the U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Regulatory Research (RES) performed informal probabilistic risk assessments (PRAs) of the use of containment accident pressure (CAP) by two boiling-water reactors (BWRs) with Mark I containments: Browns Ferry and Monticello. The scope of these risk assessments was a Level 1, internal events PRA that produced an estimate for the change in core damage frequency (CDF). The Office of Nuclear Reactor Regulation (NRR) identified the need to have these two PRAs formally documented for knowledge management purposes, as well as the need to perform and formally document additional PRAs for six other plants. The NRC outlined this need in user need request (UNR) NRR-2011-006, "User Need Request: Office of Nuclear Regulatory Research Technical Support for Nuclear Reactor Regulation Risk Assessment of Licensees' Use of Containment Accident Pressure To Provide Adequate Net Positive Suction Head for Emergency Core Cooling System and Containment Heat Removal System Pumps," dated May 4, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML110330244).

The UNR also requested that RES develop generic estimates of the risk associated with the use of CAP credit for both pressurized-water reactors (PWRs) and BWRs with various containment designs, taking into account any differences in containment performance that might be important to the risk results, and documenting these risk assessments. NRR requested that risk assessments of similar scope to the Browns Ferry and Monticello risk assessments be performed for six additional reactor plants; two BWRs with Mark I containments and four PWRs with large dry and subatmospheric containments. Additionally, the risk assessments were expanded beyond the development of estimates of the change in CDF to include development of estimates of the change in large early release frequency (LERF).

Throughout the course of performing the work under this UNR, NRR and RES engaged in multiple discussions on various technical aspects of the work. In particular, a key issue discussed between NRR and RES was if it was necessary to develop values for the containment leakage rates that are sufficient to defeat CAP for PWRs or result in a large early release. Existing industry risk assessments provided values of leakage rates for BWRs with a Mark I containment; however, corresponding containment leakage rate values for PWR containments were not readily available. Determining these values for PWR containments would require significant NRC resources for performing thermal-hydraulic analyses, including the development of thermal-hydraulic models (i.e., GOTHIC input decks), for the PWRs being analyzed. With regard to other BWR containment types other than a Mark I, NRR and RES staff determined that assessments were unnecessary because (1) plants with a Mark III type containment do not require the use of CAP and (2) it was determined that it is relatively unlikely that a plant with a Mark II type containment would require CAP given the relatively low required net positive suction head for the pumps of interest.

NRR and RES agreed that additional thermal-hydraulic analyses for the PWRs were not warranted for the continuation of the investigation into the risk associated with the use of CAP credit and that the final report should only discuss the PRAs for the four plants that had been analyzed and documented to date (i.e., Browns Ferry, Monticello, Beaver Valley, and Fort Calhoun). Since no additional thermal-hydraulic analyses were to be performed, the NRC staff agreed that the risk assessments for the two PWRs—Beaver Valley and Fort Calhoun—would assume the same containment leakage rates as were used for the BWRs for defeating CAP or producing a large early release, despite the fact that these values were likely to be highly conservative for the PWR analyses.

The risk results of these assessments (i.e., the change in CDF and LERF) primarily depend on two parameters: the containment failure rate and containment leakage rate surveillance test interval (STI). The results are linearly dependent on the containment failure rate, but are nonlinear with respect to STI. The results are presented as a function of STI to demonstrate the nonlinear behavior of the results over a range of STIs and not just for certain STIs of interest. Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” provides both prescriptive and performance-based requirements for testing containment leakage of water-cooled power reactors. The prescriptive requirements dictate that, “After the preoperational leakage rate tests, a set of three Type A tests shall be performed, at approximately equal intervals during each 10-year service period.” The performance-based requirements dictate that, “A Type A test must be conducted…at a periodic interval based on the historical performance of the overall containment system as a barrier to fission product releases to reduce the risk from reactor accidents.” Since some licensees have applied for and been granted containment leakage STIs that require one test in a 10-year period and, in some cases, one test in a 15-year period, the results are presented as a continuous function of STIs ranging from zero (i.e., continuous testing) to a 15-year period.

The NRC developed the approach used for the performance of these risk assessments to satisfy NRR’s technical support needs; the NRC has not accepted or endorsed it for any other purpose. Further, without additional thermal-hydraulic analyses, the NRC does not consider the PWR results to be technically defensible for any regulatory decision absent a more realistic technical basis for the containment leakage rate values used in the analysis.

1. INTRODUCTION

This report documents probabilistic risk assessments (PRAs) that have been performed for selected nuclear plants that account for the use of containment accident pressure (CAP) credit during design-basis accident (DBA) conditions. In some accident scenarios, CAP is relied upon to maintain adequate net positive suction head (NPSH) for low-head emergency core cooling system (ECCS) and containment heat removal system (CHRS) pumps during DBA conditions. The purpose of the CAP credit PRA is to characterize the overall impact of the risk associated with the use of CAP credit and, in doing so, develop quantitative estimates of the changes in plant risk associated with the use of CAP credit. This report specifically documents the CAP credit PRAs that have been performed for the reactor plants listed below in Section 1.2.

1.1 Problem Statement

During accident conditions such as a loss-of-coolant accident (LOCA), the reactor coolant system transfers enormous quantities of mass and energy to the containment atmosphere, resulting in pressurization of the containment structure. This increase in the containment pressure above the normal operating containment pressure is known as containment accident pressure. In some accident scenarios, CAP is relied upon to ensure that adequate NPSH is provided to the low-head ECCS and CHRS pumps, so as to prevent cavitation of those pumps and the potential functional failure of the ECCS and CHRS.

The reliance on CAP is effectively a reliance on the ability of the containment to successfully maintain the pressure increase for as long as the additional pressure is needed. If the containment is not intact when CAP is needed (i.e., the containment leakage rate is greater than the operational limits) and the leakage rate is great enough, the low-head ECCS and CHRS pumps may experience cavitation at some point during the course of the accident. Prolonged pump cavitation could significantly damage the pumps and potentially prevent the system from performing its intended safety function, which could result in core damage. Further, in addition to causing core damage, the containment leakage rate could potentially be large enough to result in a large early release of radioactivity to the environment. The result is that the reliance on CAP during DBA conditions creates a potential dependency between the performance of the containment—a fission product barrier—and the ability to prevent core damage.

1.2 Objectives and Scope

The objective of this work is to provide support to the Office of Nuclear Reactor Regulation (NRR) in response to the user need request designated as NRR-2011-006, “User Need Request: Office of Nuclear Regulatory Research Technical Support for Nuclear Reactor Regulation Risk Assessment of Licensees’ Use of Containment Accident Pressure To Provide Adequate Net Positive Suction Head for Emergency Core Cooling System and Containment Heat Removal System Pumps,” dated May 4, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML110330244). The user need request asks for generic estimates of the risk of CAP credit for pressurized-water reactors (PWRs) and boiling-water reactors (BWRs), taking into account any differences in containment performance that might be important to the risk results. Originally, this work included documenting the previous CAP credit risk assessments for Browns Ferry and Monticello and performing CAP credit risk assessments for six additional reactor plants that use CAP credit in their accident analyses, as chosen by NRR and the Office of Nuclear Regulatory Research (RES). However, after multiple technical discussions between NRR and RES, the staff agreed that the work under NRR-2011-006 would only involve two additional reactor plants, those being Fort Calhoun and

Beaver Valley. The complete list of subject plants from the original user need request are listed below and include the reactor type, vendor, and containment type in parenthesis for reference.

Reactor Plants Previously Analyzed

- Browns Ferry (BWR, General Electric-4, Mark I containment)
- Monticello (BWR, General Electric-3, Mark I containment)

Reactor Plants Selected by NRR and RES:

- Beaver Valley (PWR, Westinghouse 3-loop, subatmospheric containment)
- Callaway (PWR, Westinghouse 4-loop, large dry containment)
- Fort Calhoun (PWR, Combustion Engineering, large dry containment)
- Oconee (PWR, Babcock & Wilcox, large dry containment)
- Cooper (BWR, General Electric-4, Mark I containment)
- Peach Bottom (BWR, General Electric-4, Mark I containment)

The scope of these CAP credit PRAs consists only of an internal events analysis for the full-power mode of operation. The quantitative risk metrics of interest in this study are the change in core damage frequency (ΔCDF); and the change in large early release frequency (ΔLERF). These risk metrics are calculated using the Birnbaum importance measure for the loss of containment integrity (LOCI) event.

1.3 Background

Since the early 1970s, the use of CAP to ensure adequate NPSH for low-head ECCS and CHRS pumps during DBA conditions has arisen as a potential safety issue. Over the past several decades, the U.S. Nuclear Regulatory Commission (NRC) staff and the Advisory Committee on Reactor Safeguards (ACRS) have debated if it is appropriate to use CAP (a.k.a., CAP credit) for this purpose, as well as what type of analyses should be used to justify the use of the CAP credit.

Additionally, regulatory guidance on the use of CAP credit is inconsistent. In 1970, the NRC issued Regulatory Guide (RG) 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps" (Ref. 1), which states the following:

> Emergency core cooling and containment heat removal systems should be designed so that adequate net positive suction head (NPSH) is provided to system pumps assuming maximum expected temperatures of pumped fluids and no increase in containment pressure from that present prior to postulated loss of coolant accidents.

More recently, the NRC issued RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (Ref. 2), which contained a qualification of the guidance in RG 1.1 that states the following:

> For certain operating PWRs for which the design cannot be practicably altered, conformance with Regulatory Position 1.3.1.1 may not be possible. In these cases, no additional containment pressure should be included in the determination of available NPSH than is necessary to preclude pump cavitation. Calculation of available containment pressure and sump water temperature as a function of time should underestimate the expected containment pressure and

overestimate the sump water temperature when determining available NPSH for this situation.

RG 1.82 provides a similar qualification for BWRs. Additionally, recent licensee applications for extended power uprates (i.e., Vermont Yankee, Browns Ferry, and Monticello) have included the use of CAP credit in their analysis. Section 3.0 of this report provides a discussion of the regulatory perspectives related to the use of CAP credit.

2. TECHNICAL APPROACH

The general technical approach used to analyze the different containment types in this study consists of the following major steps:

1. Define the scope.

2. Select and modify an appropriate risk model to account for the dependency created between the low-head emergency core cooling system (ECCS) and containment heat removal system (CHRS) pumps and the containment when containment accident pressure (CAP) credit is used.

3. Quantify the modified risk model to determine risk metrics of interest.

Section 2.3.1 discusses plant-specific analyses and results. This report notes differences between the analyses specific to each containment type, as appropriate. The following sections discuss the details of this technical approach.

2.1 Risk Assessment Scope

The risk assessments are defined in terms of the hazards groups considered, the plant operating states analyzed, and the risk metrics used. These risk assessments only consider internal initiating events, which includes analysis of the following events:

- general transients
- loss-of-coolant-accident (small, medium, and large)
- steam line break outside of containment
- loss of service water
- loss of plant control air
- loss of offsite power
- loss of main feedwater
- inadvertent open relief valve
- loss of condenser heat sink
- interfacing system loss-of-coolant-accident

These risk assessments only include analysis of at-power operations. No operator actions were identified as being required for the implementation of CAP credit (e.g., manual tripping of drywell coolers) and, as such, none were included in these assessments. The risk metrics calculated for these PRAs include the Brinbaum importance measure for the loss of containment integrity event, the change in core damage frequency (ΔCDF), and the change in large early release frequency (ΔLERF).

2.2 Risk Model Selection and Modification

These PRA analyses used the NRC's standardized plant analysis risk (SPAR) models. The SPAR models are standardized, plant-specific risk models that use the event-tree and fault-tree linking methodology. The SPAR models used for each analysis were constructed with and evaluated in the NRC's SAPHIRE PRA code. The staff used version 8.0.7.18 of the SAPHIRE code for the modeling and evaluation of the Fort Calhoun and Beaver Valley SPAR models and version 8.0.7.19S for the Browns Ferry and Monticello SPAR models in these analyses. The process of modifying the SPAR model consists of the following three steps:

1. Define the loss of containment integrity (LOCI) as a basic event.
2. Determine the LOCI event probability.
3. Modify the appropriate parts of the plant SPAR model.

The following sections provide detailed descriptions of each step.

2.2.1 Definition of the Loss of Containment Integrity as a Basic Event

The LOCI event is defined in terms of (1) the containment leakage rate, S_{CDF}, that is sufficient to defeat the CAP needed to maintain adequate net positive suction head (NPSH) for the low-head ECCS and CHRS pumps and, for this study, is assumed to cause core damage, and (2) the containment leakage rate, S_{LERF}, that is large enough to result in a large early release of radiological material. The LOCI event also is defined in terms of three timeframes: pre-initiating event, concurrent with an initiating event, and post-initiating event. As such, the LOCI event probability is composed of five major components: small and large pre-initiating event leaks, on-demand failure of containment isolation, and small and large post-initiating event leaks.

A small leak is defined as a leakage rate, S, that is greater than or equal to S_{CDF} but is less than S_{LERF}; a large leak is defined as a leakage rate, S, that is greater than or equal S_{LERF}; and the on-demand failure of containment isolation is assumed to always result in a leakage rate that is greater than S_{LERF}. When expressed as a logic diagram, the LOCI basic event can be structured as illustrated below.

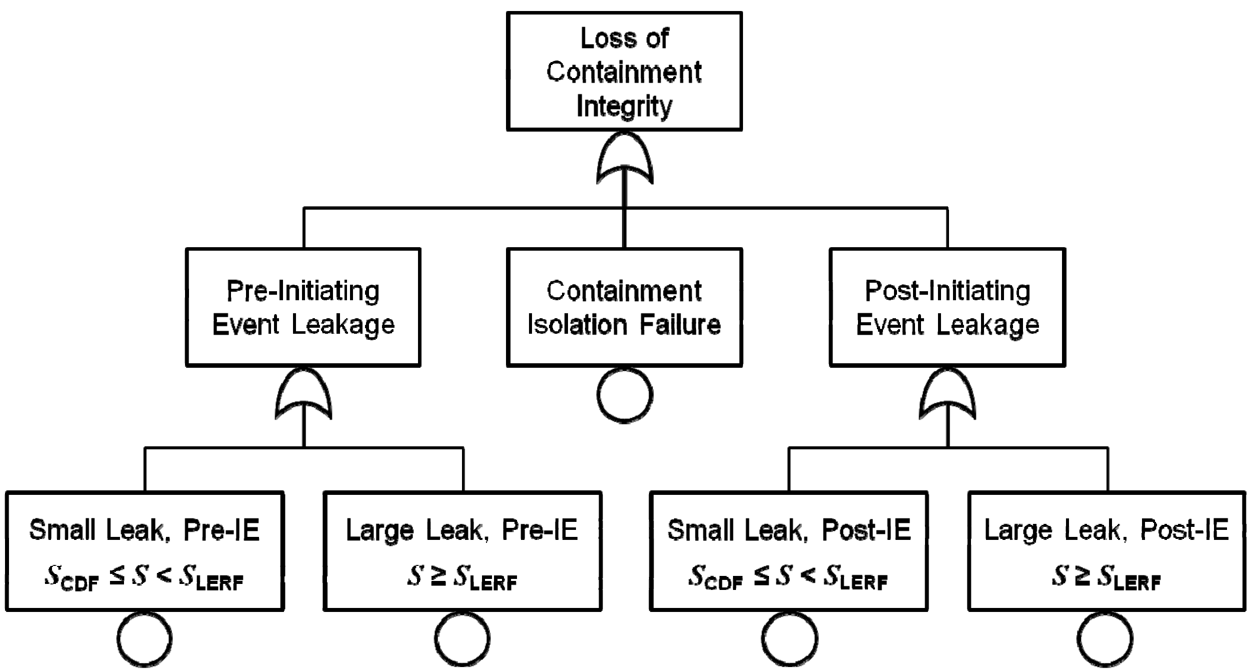

Figure 1. LOCI event logic diagram

The total probability of the LOCI event is expressed as the sum of the probabilities of its five components. Sections 2.2.2 and 2.2.3 discuss the derivation of the probability estimates for the five components and the modeling of the LOCI event in the SPAR model.

For the boiling-water reactors (BWRs) Browns Ferry and Monticello, is the staff assumed that the low-head ECCS and CHRS pumps, which rely on CAP for adequate NPSH, are functionally failed if CAP is defeated. If CAP is defeated, the accident is assumed to result in core damage. Although it is recognized that, in some cases, the degree and duration of cavitation of the low-head pumps may be minimal for a given accident sequence, no consideration is given for the actual pump and fluid conditions following the loss of CAP. Additionally, it is assumed that the ECCS and CHRS pumps require CAP whenever they take suction from the suppression pool.

For the pressurized-water reactors (PWRs) Beaver Valley and Fort Calhoun, the staff assumed that the only time that CAP is important for maintaining adequate NPSH is when the plant is operating in high-pressure recirculation mode. In this mode the low-head ECCS pumps draw suction from the containment sump and provide adequate NPSH for the high-pressure pumps. The staff assumed that the low-head ECCS pumps require CAP whenever they are in the high-pressure recirculation mode of operation.

The containment leakage rates discussed above are described in terms of the unit L_a. The unit L_a is the maximum allowable leakage rate and is defined in Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." As described in Appendix J, the unit L_a is the percentage of weight of the original content of containment air that is lost to the environment in a 24-hour period when the containment is pressurized to the peak internal containment pressure P_a related to the design-basis accident. As an example, if L_a is defined as 0.1 weight-percent of the containment air, a leakage rate of 50 L_a equates to a loss of 5 percent of a containment atmosphere, by weight, in a 24-hour period under design-basis accident conditions. The definition of L_a is plant-specific and may vary from 0.1 to 2.0 weight-percent, depending on the plant. Table 1 shows the various definitions of L_a at a pressure of P_a for each plant analyzed, as taken from their respective technical specifications.

Table 1. Plant-Specific Definitions of L_a

	Plant	L_a Definition (weight-percent)	P_a (psig)
BWRs	Browns Ferry (Unit 1)	2.0	48.5
	Monticello	1.2	42
PWRs	Beaver Valley (Unit 1)	0.1	43.1
	Fort Calhoun	0.1	60

For a given plant, the actual value of S_{CDF} would need to be derived from a plant-specific thermal-hydraulic analysis. For some BWRs with Mark I containments, recent licensee application submittals for extended power uprates (EPUs) provide values of S_{CDF} that were used as a basis for calculating the changes in CDF associated with the use of CAP credit. For example, Vermont Yankee determined S_{CDF} values of 27 L_a, based on the containment analysis approach in 10 CFR Part 50, Appendix K, “ECCS Evaluation Models” (Ref. 3). Similarly, the Browns Ferry EPU application used a leakage rate of 20 L_a for S_{CDF} [4].

For Browns Ferry and Monticello, the staff used a generic leakage rate estimate of 20 L_a for S_{CDF}, based on values used in previous licensee applications, and a generic leakage rate estimate of 35 L_a for S_{LERF}, based on industry guidance (Ref. 5).

Since it is unclear if analyses similar to that performed for the BWR Mark I-type containment has been performed to determine appropriate values of S_{CDF} and S_{LERF} for large dry and subatmospheric containment types, the staff used the same leakage rate estimates for S_{CDF} and S_{LERF} in the Browns Ferry and Monticello analyses as were used for the values of S_{CDF} and S_{LERF} for the Beaver Valley and Fort Calhoun analyses.

2.2.2 Determination of the Loss of Containment Integrity Event Probability

The staff calculated the LOCI basic event probability for each of the three timeframes using different analytical models. The pre-initiator LOCI event probability was calculated using a semi-Markov model; the upon-initiator LOCI event probability was represented by the containment isolation failure probability; and the post-initiator LOCI event was calculated using a Poisson model. These models are discussed in more detail below.

Pre-Initiating Event Timeframe

The pre-initiating event (i.e., pre-initiator) LOCI event describes the likelihood that a containment leak exists prior to the occurrence of an initiating event and that the leak is large enough to defeat CAP credit or to defeat CAP credit and result in a large early release. The pre-initiator LOCI event probability depends on the size of the leak and the surveillance interval of the Type A containment leakage test, which determines the overall integrated leak rate.

The size-dependent containment leakage rate probabilities used in this study were taken from Electric Power Research Institute (EPRI) report TR-1009325, “Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals” (Ref. 6). This report includes containment failure probability estimates that were developed through expert elicitation and account for containment failure caused by construction error or deficiency; human error related to testing, maintenance, or design error; corrosion; and fatigue failures. The staff developed probability estimates for both large and small containment types, whereby a large or small containment was defined as having more than or less than 1 million cubic feet of free volume, respectively. Although separate probability estimates were developed for small containments, these estimates were only slightly lower than those of large containments, and the difference between the estimates is considered to be insignificant (Ref. 6). As such, for the purpose of this study, probability estimates for the large containment type were used to represent the pre-initiator LOCI event probabilities for all containment types.

The staff used a semi-Markov model to determine the dependency of the pre-initiator LOCI event probabilities on multiple parameters, and they performed a separate analyses for each leakage size (i.e., small or large). The semi-Markov process is used to describe a system that only exists in one of any number of discrete states at a given point in time. The time spent by the system in a given state is randomly determined using an arbitrary probability distribution, and the probability that the system transitions to another state only depends on its current state and is independent of the system’s past history. The staff incorporated several parameters into the model, including the containment failure rate for a given leakage rate; the mean time repair of a containment leak; the surveillance test interval (STI); the time allowed by technical specifications to make an at-power repair; the time allowed by technical specifications to transition to shutdown; and the efficiency of the leak test. The semi-Markov model used for this study accounts for the following four system states and is independent of containment type:

- State 1: The containment is intact.
- State 2: The containment has an undetected leak.
- State 3: The leak is detected and a repair is initiated.
- State 4: The repair is not completed in the allotted time, and the reactor is transitioned to shutdown.

For a given leak size (i.e., small or large), the pre-initiator LOCI event probability is represented as the long-run fraction of time that the system spends in States 2, 3, and 4. A semi-Markov state diagram shows the relationships between the system states and their respective transitions, as shown in Figure 2.

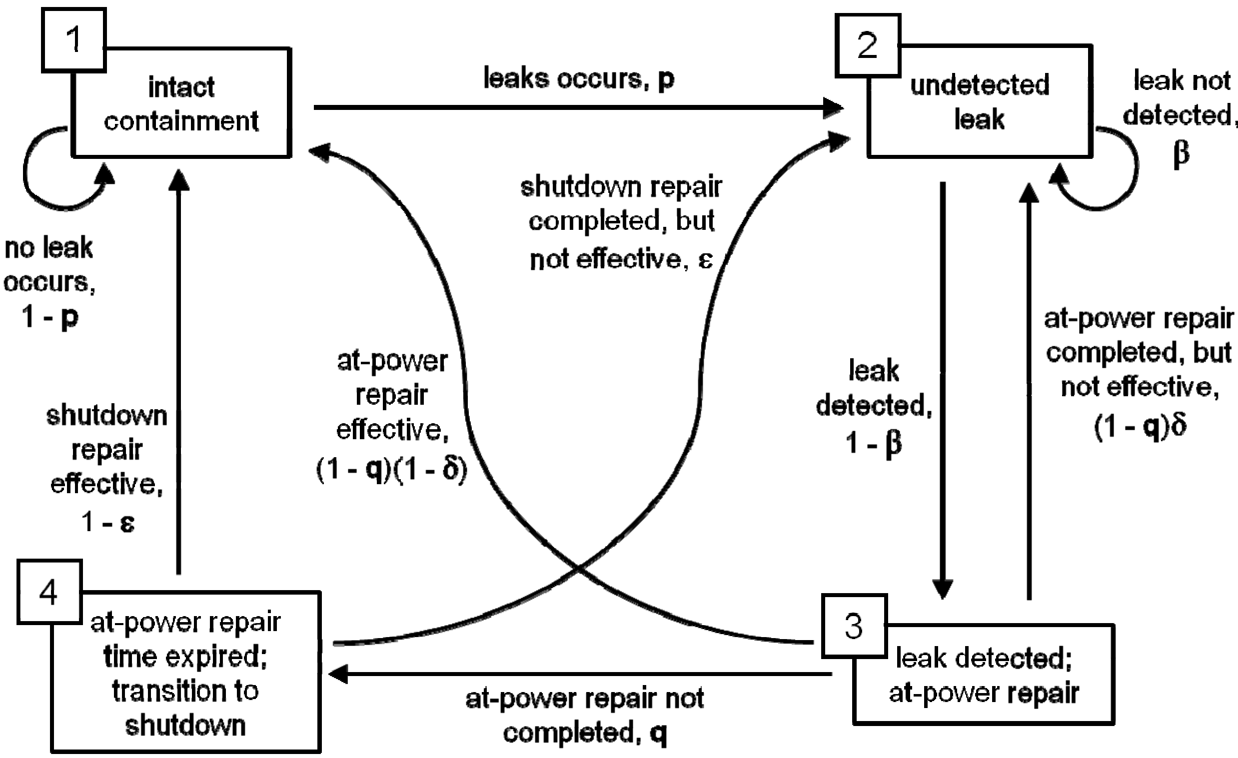

Figure 2. Semi-Markov model state diagram for development of the pre-initiator LOCI probability

Although the staff included several parameters in the semi-Markov analysis, they performed sensitivity studies that indicated that the pre-initiator LOCI probability primarily depends on the containment leakage failure rate and the surveillance test interval of the leak test over a wide range of STIs. As such, the pre-initiator LOCI probability for each leakage rate of interest is discussed primarily as a function of the STI. Appendix A of this report provides a more detailed discussion of the semi-Markov model and includes a derivation of the pre-initiator LOCI probability as a function of the STI.

The staff determined the containment failure rate for a given leakage rate using data from NUREG-0933, “Resolution of Generic Safety Issues” (Ref. 7) and the EPRI report TR-1009325 (Ref. 6). Under Task II.E.4, “Containment Design,” of Section 1 of NUREG-0933, the estimated frequency of an undetected breach in containment integrity is approximately 1.1×10^{-2} per reactor-year, which can be expressed as 1.3×10^{-6} per reactor-hour. This failure rate is assumed to represent a containment leakage rate of at least 1 L_a, based on the definition of an unavailable containment described by the standard technical specifications. Using this information, the appropriate containment failure rate was calculated by multiplying the 1 L_a containment failure rate by a scaling factor. Equation 1 shows this relation relative to the calculation of the containment failure rate for a leakage rate that is sufficient to defeat CAP.

$$\lambda(S_{CDF}) \;=\; \frac{Pr\{S \geq S_{CDF} | leak\}}{Pr\{S \geq 1\ L_a | leak\}} \times \lambda(1\ L_a) \qquad \text{(Equation 1)}$$

The probability expressions in the ratio represent the conditional probabilities that the actual containment leakage rate, S, is greater than or equal to a predetermined value, in this case S_{CDF}, given that a containment leak exists. Similarly, the containment failure rate for a leakage rate that is sufficient to result in a large early release is determined by substituting the numerator of the scaling factor with the conditional probability for S greater than or equal to S_{LERF}. Using probability estimates taken from EPRI report TR-1009325 and the assumed S_{CDF} and S_{LERF} of 20 L_a and 35 L_a, respectively, the actual containment failure rates are determined for small and large leaks, as shown below.

$$Pr\{S \geq 1\ L_a | leak\} \;=\; 2.7 \times 10^{-2} \qquad \text{(Equation 2)}$$

$$Pr\{S \geq 20\ L_a | leak\} \;=\; 1.9 \times 10^{-3} \qquad \text{(Equation 3)}$$

$$Pr\{S \geq 35\ L_a | leak\} \;=\; 9.9 \times 10^{-3} \qquad \text{(Equation 4)}$$

$$\lambda(20\ L_a) = \frac{(1.9 \times 10^{-3})}{(2.7 \times 10^{-2})} \times \left(1.3 \times 10^{-6}/\text{reactor-hr}\right) = 9.2 \times 10^{-8}/\text{reactor-hr} \qquad \text{(Equation 5)}$$

$$\lambda(35\ L_a) = \frac{(9.9 \times 10^{-4})}{(2.7 \times 10^{-2})} \times \left(1.3 \times 10^{-6}/\text{reactor-hr}\right) = 4.7 \times 10^{-8}/\text{reactor-hr} \qquad \text{(Equation 6)}$$

The staff expressed the pre-initiator LOCI probability as a function of STI for the leak test. Table 2 below lists some of the resulting probabilities calculated for this analysis.

Table 2. Calculated pre-initiator LOCI event probability values

STI (hours)	20 $L_a \leq S <$ 35 L_a	$S >$ 35 L_a	*Total*
1	2.6E-06	1.4E-06	4.0E-06
2	2.6E-06	1.4E-06	4.0E-06
4	2.8E-06	1.4E-06	4.2E-06
8	3.0E-06	1.6E-06	4.5E-06
16	3.4E-06	1.8E-06	5.2E-06
24	3.8E-06	2.0E-06	5.8E-06
48	5.1E-06	2.7E-06	7.8E-06
72	6.4E-06	3.4E-06	9.8E-06
168	1.2E-05	6.1E-06	1.8E-05

336	2.1E-05	1.1E-05	3.1E-05
730	4.2E-05	2.2E-05	6.4E-05
1,460	8.1E-05	4.2E-05	1.2E-04
2,190	1.2E-04	6.3E-05	1.8E-04
4,380	2.4E-04	1.2E-04	3.6E-04
8,760	4.7E-04	2.5E-04	7.2E-04
17,520	9.4E-04	4.9E-04	1.4E-03
29,200	1.6E-03	8.2E-04	2.4E-03
87,600	4.7E-03	2.5E-03	7.1E-03
131,400	7.0E-03	3.7E-03	1.1E-02

Concurrent with Initiating Event Timeframe

The LOCI event that is concurrent with an initiating event (i.e., upon-initiator LOCI event) represents a failure of the containment to isolate on-demand. A containment isolation (CI) failure is assumed to result in a direct containment by-pass and, following core damage, a release of radiological material to the environment. The magnitude of containment leakage from this type of event is assumed to always be greater than S_{LERF}.

For BWRs with a Mark I containment type, the staff assumed that no release paths to the environment exist during normal at-power operations. However, a potential release path does exist during the process of inerting and de-inerting the containment atmosphere. Since these processes may take up to 24 hours to complete following startup and prior to shutdown of the reactor, respectively, an accident could occur that requires CAP for maintaining adequate NPSH. Therefore, the inerting and deinerting periods are the only relevant periods of time for which the probability of an upon-initiator LOCI event needs to be determined. Since these release paths only exist twice during a single fuel cycle—assumed here to be 24 months—the total probability of the upon-initiator LOCI event occurring is expressed as the total probability of the on-demand CI failure in that timeframe, 10^{-3}—including independent and common-cause failures—weighted by the fraction of time that the release paths are available over a single fuel cycle, as shown in Equation 7.

$$\begin{aligned} Pr\{\text{leak concurrent with initiator}\} &= (10^{-3}) \times \left[\left(\frac{2\times 24\ hrs}{24\ months}\right) \times \left(\frac{1\ month}{730\ hrs}\right)\right] \\ &= 2.7 \times 10^{-6} \end{aligned} \quad \text{(Equation 7)}$$

The staff considered an additional release path for the Mark I containment that could potentially exist when at least one pair of main steam isolation valves (MSIVs) fails to close (FTC) on-demand following the occurrence of a loss-of-coolant accident (LOCA) that communicates with the containment atmosphere (i.e., small, medium, or large LOCAs). If this occurs, it is assumed that the containment would be depressurized through the LOCA break location and then through stuck-open MSIVs, which is assumed to defeat CAP and create a hole large enough for a large early release. Because the small, medium, or large LOCA initiating event is expressed as a frequency, the combination of the LOCA and the MSIV-FTC must be modeled outside of the LOCI event logic, as shown in Figure 3, and quantified as its own contribution to the ΔCDF, as discussed in Section 2.3.

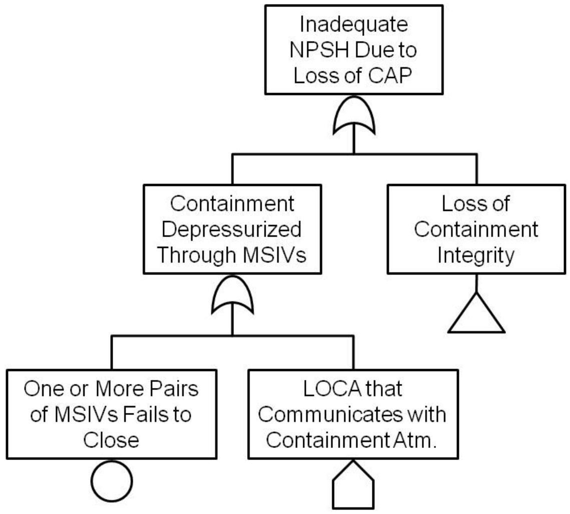

Figure 3. BWR logic modeling for inadequate NPSH caused by loss of CAP

Figure 3 shows the LOCI event as being modeled with a transfer gate for simplicity; however, the LOCI event is representative of the logic structure shown in Figure 1.

For the PWRs, it is assumed that no potential release paths exist during operations when CAP would be needed. The Fort Calhoun Final Safety Analysis Report indicates that all of the penetrations directly connecting the inside containment atmosphere to the outside atmosphere have at least one locked-closed valve located outside the containment (Ref. 8). As such, it is assumed that for such a potential release path, the release path would only be available if there was a failure to close the outside containment valve as a result of human error combined with a CI failure. The staff used a probability of $5\text{x}10^{-3}$ for the on-demand CI failure for the PWRs, which is taken from the description of the accident progression event tree in NUREG/CR-4551, "Evaluation of Severe Accident Risks: Zion, Unit 1," Vol. 7, Revision 1, Part 2A (Ref. 9). This probability is weighted by the human error probability of the failure to secure the outside containment valve as shown in Equation 8.

$$
\begin{aligned}
Pr\{\text{CI and FTC valve (human error)}\} &= (5 \times 10^{-3}) \times (3 \times 10^{-3}) \\
&= 1.5 \times 10^{-5}
\end{aligned}
\qquad \text{(Equation 8)}
$$

Post-Initiating Event

Following the occurrence of an initiating event, the containment is initially assumed to be intact. The probability of a post-initiating event (i.e., post-initiator) LOCI describes the likelihood of a containment failure at some time after the initiating event occurs, but before the CAP is no longer required (i.e., the CAP credit mission time). The CAP credit mission time is the only parameter in the SPAR model that relates to the duration of CAP credit following the occurrence of an initiating event. The staff determined the probability of occurrence of a post-initiator LOCI using a Poisson model as shown in Equation 9.

$$
Pr\{\text{post-initiator leak}\} = 1 - e^{-\lambda T} \approx \lambda T \qquad \text{(Equation 9)}
$$

The staff assumed a mission time of 72 hours for these risk assessments, and assumed λ is the same containment unavailability frequency used for the semi-Markov model to determine the pre-initiating event LOCI probability. Equations 10 through 13 show the calculated probabilities

$$\lambda(20\ L_a) \ = \ 9.2 \times 10^{-8}/\text{reactor-hr} \qquad \text{(Equation 10)}$$

$$\lambda(35\ L_a) \ = \ 4.7 \times 10^{-8}/\text{reactor-hr} \qquad \text{(Equation 11)}$$

$$Pr\{\text{post-initiator leak (CDF)}\} \ = \ (9.2 \times 10^{-8}) \times (72\ hrs) = 6.6 \times 10^{-6} \qquad \text{(Equation 12)}$$

$$Pr\{\text{post-initiator leak (LERF)}\} \ = \ (4.7 \times 10^{-8}) \times (72\ hrs) = 3.4 \times 10^{-6} \qquad \text{(Equation 13)}$$

These post-initiator LOCI event probabilities are assumed to be the same for BWRs and PWRs.

Total LOCI Event Probability

As discussed in Section 2.2.1, the total LOCI event probability is expressed as the sum of the individual probability contributions from the pre-initiator, upon-initiator, and post-initiator timeframes. The total LOCI event probability is the sum of the pre-initiator LOCI probability values and the constant upon-initiator and post-initiator LOCI probability values for all STIs. As an example, Figure 3 shows the plots of the individual contributions to and the total LOCI event probability for the BWRs with Mark I containments. The PWR total LOCI event probability exhibits the same behavior as shown in Figure 3. Noteable STIs are indicated.

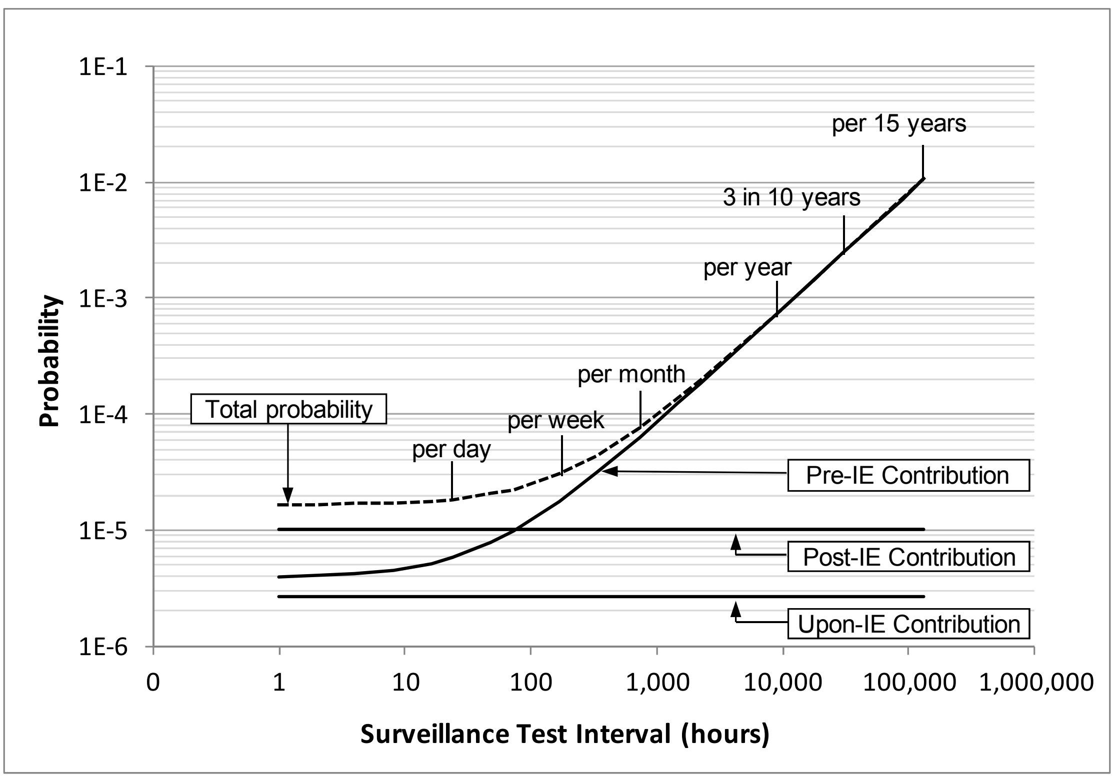

Figure 4. LOCI event probability vs. surveillance test interval for BWR Mark I containments

Additionally, for the BWRs, the contribution of the MSIV-FTC concurrent with a LOCA that communicates with the containment atmosphere is discussed further in Section 2.3. The LOCI event probability determined for each plant is used to directly calculate the ∆CDF and ∆LERF results, as discussed in Section 2.3. The values of the total LOCI event probability that were evaluated for each type of containment are shown below in Table 3.

Table 3. Total LOCI event probabilities

STI (hours)	Total LOCI Event Probability	
	BWR, Mark I	PWR, Large Dry and Subatmospheric
1	1.4E-05	5.0E-03
2	1.4E-05	5.0E-03
4	1.5E-05	5.0E-03
8	1.5E-05	5.0E-03
16	1.5E-05	5.0E-03
24	1.6E-05	5.0E-03
48	1.8E-05	5.0E-03
72	1.9E-05	5.0E-03
168	2.6E-05	5.0E-03
336	3.7E-05	5.0E-03
730	6.4E-05	5.1E-03
1,460	1.1E-04	5.1E-03
2,190	1.6E-04	5.2E-03
4,380	3.1E-04	5.4E-03
8,760	6.1E-04	5.7E-03
17520	1.2E-03	6.4E-03
29200	2.0E-03	7.3E-03
87,600	6.0E-03	1.2E-02
131,400	8.9E-03	1.5E-02

Although the total LOCI event probability curve shown in Figure 4 has the same shape for BWRs and PWRs, realistically, the total LOCI event probability for BWRs with Mark I containments is not dictated by this curve for large STIs. Standard Technical Specification 3.6.3.2 for General Electric (GE) BWR/4 reactors (Ref. 10) requires that the oxygen concentration in the containment atmosphere be monitored on a weekly basis to ensure that the concentration is less than 4.0 volume-percent when the reactor is operating at full power. If oxygen concentration is not restored below 4.0 volume-percent within 24 hours, the plant has 8 hours to reduce power to 15 percent of rated thermal power.

As discussed in the bases for the technical specification, this oxygen concentration is maintained by inerting the containment atmosphere with nitrogen gas. The staff assumed that if a containment leak exists that is sufficient to defeat CAP, the technical specification for the primary containment oxygen concentration would fail to be met since the nitrogen makeup system would be depleted well in advance of the scheduled resupply, or the resupply frequency would need to substantially increase beyond that expected for normal operating conditions.

Although there is no technical specification for the rate of nitrogen usage, it is assumed that the increase in the need for nitrogen makeup to meet the oxygen concentration technical specification would prompt an investigation into the cause of the increase.

The result is that, realistically, the total LOCI event probability for the BWRs analyzed has an effective upper limit, as dictated by this effective STI. The total LOCI event probability for the BWRs is assumed to be constant for STIs that are greater than the effective STI. Conversely, PWRs do not have this type of monitoring capability and, as such, the total LOCI event probability for PWRs is assumed to be dictated by the pre-initator probability contribution for large STIs.

2.2.3 Modification of the Appropriate Parts of the Plant SPAR Model

For each plant, the staff modified the associated SPAR model to include the LOCI event in the appropriate fault trees. The LOCI event was added to fault trees based primarily on the assumptions that (1) for a BWR with a Mark I containment, CAP was required whenever the low-head ECCS or CHRS pumps are aligned to draw water from the suppression pool, and (2) for a PWR with a subatmospheric or large dry containment, CAP was required when the ECCS and CHRS pumps are aligned to draw water from the containment sump during relevant recirculation modes of operation. Figure 3 in Section 2.2.2 shows the BWR logic modeling used for the LOCI event and the MSIVs event concurrent with small, medium, or large LOCAs. Section 2.3.1 discusses the plant-specific fault tree modifications.

2.3 Risk Metric Quantification and Plant-Specific Results

The risk metrics of interest for this study are ΔCDF and ΔLERF. These risk metrics are calculated using the Birnbaum importance for the LOCI and MSIV-FTC events. The Birnbaum importances for these events depend on the placement of the LOCI and MSIV-FTC events in the logic model and the frequencies of other initiating events and basic events in the model. However, the Birnbaum importances are independent of the probabilities used to generate the cut sets for the analysis. As such, the LOCI event probability was artificially set to 10^{-2} to ensure that a sufficient number of cut sets containing the LOCI event survived truncation when the model was solved and could be used to provide an accurate determination of the Birnbaum importance for the event. The staff used the probability of 10^{-4} for the MSIV-FTC event, which is a previously established value from the SPAR models.

For the BWR analyses, the ΔCDF risk measure is calculated by taking the sum of (a) the product of the LOCI event probability and the Birnbaum importance measure for the LOCI event and (b) the product of the MSIV-FTC event probability and Birnbaum importance measure for the MSIV-FTC event, as shown in Equations 14a. For the PWR analyses, the ΔCDF risk measure is calculated in the same way, but it does not include the contribution from the MSIV-FTC event, as shown in Equations 14b.

$$\Delta\mathrm{CDF}_{\mathrm{BWR}} = I_B\{LOCI\} \cdot Pr\{LOCI\} + I_B\{MSIV\text{-}FTC\} \cdot Pr\{MSIV\text{-}FTC\} \qquad \text{(Equation 14a)}$$

$$\Delta\mathrm{CDF}_{\mathrm{PWR}} = I_B\{LOCI\} \cdot Pr\{LOCI\} \qquad \text{(Equation 14b)}$$

Equations 15a and 15b show Equations 14a and 14b as a function of the individual probability contributions.

$$\Delta \text{CDF}_{\text{BWR}} = I_B\{LOCI\} \cdot [Pr\{B_S\} + Pr\{B_L\} + Pr\{A_s\} + Pr\{A_L\} + Pr\{CI\}] + I_B\{MSIV\text{-}FTC\} \cdot Pr\{MSIV\text{-}FTC\ event\} \quad \text{(Equation 15a)}$$

$$\Delta \text{CDF}_{\text{PWR}} = I_B\{LOCI\} \cdot [Pr\{B_S\} + Pr\{B_L\} + Pr\{A_s\} + Pr\{A_L\} + Pr\{CI\}] \quad \text{(Equation 15b)}$$

where

$I_B\{LOCI\}$ = LOCI event Birnbaum importance

$Pr\{B_S\}$ = Pre-initiator LOCI event probability, small leak

$Pr\{B_L\}$ = Pre-initiator LOCI event probability, large leak

$Pr\{A_S\}$ = Post-initiator LOCI event probability, small leak

$Pr\{A_L\}$ = Post-initiator LOCI event probability, large leak

$Pr\{CI\}$ = Containment isolation failure probability

$I_B\{MSIV\text{-}FTC\}$ = MSIV-FTC Birnbaum importance

$Pr\{MSIV\text{-}FTC\ event\}$ = MSIV-FTC event probabilty

The ΔLERF risk measure was similarly calculated for the BWR and PWRs as shown in Equations 16a and 16b.

$$\Delta \text{LERF}_{\text{BWR}} = I_B\{LOCI\} \cdot [Pr\{B_L\} + Pr\{A_L\} + Pr\{CI\}] + [I_B\{LOCI\} \cdot CCFP] \times [Pr\{B_S\} + Pr\{A_s\}] + I_B\{MSIV\text{-}FTC\} \cdot Pr\{MSIV\text{-}FTC\ event\} \quad \text{(Equation 16a)}$$

$$\Delta \text{LERF}_{\text{PWR}} = I_B\{LOCI\} \cdot [Pr\{B_L\} + Pr\{A_L\} + Pr\{CI\}] + [I_B\{LOCI\} \cdot CCFP] \times [Pr\{B_S\} + Pr\{A_s\}] \quad \text{(Equation 16b)}$$

The first term of the sums in Equations 16a and 16b represents the total contribution to ΔLERF from large containment leaks (i.e., $S \geq S_{\text{LERF}}$). The second term of the sums in Equations 16a and 16b represents the total contribution of ΔLERF from small containment leaks (i.e., $S_{\text{CDF}} \leq S < S_{\text{LERF}}$). The second terms are weighted by the product of the LOCI event Birnbaum importance and the conditional containment failure probability ($CCFP$). The weighting of the second term by the $CCFP$ accounts for the fact that a given core damage accident sequence that involves a small containment leak also may result in a large early release. Although not shown, the first term in Equations 16a and 16b also is weighted by the $CCFP$, but only shows $I_B\{LOCI\}$ since, by definition, the $CCFP$ is equal to one for the large leakage events. The third term of the sum in Equation 16a represents the risk contribution from the MSIV-FTC event. The $CCFP$ used in this equation is calculated by taking the ratio of the known baseline LERF and CDF values for each plant as shown in Equation 17.

$$CCFP = \frac{LERF_{base}}{CDF_{base}} \quad \text{(Equation 17)}$$

Plant-specific baseline LERF and CDF values for each were obtained from licensee PRA models.

2.3.1 Plant-Specific Risk Results

This section presents the results of the plant-specific CAP credit PRA analyses. A summary description of each analysis is provided that includes a brief description of the reactor plant, the version of SAPHIRE and the plant SPAR model used, and a description of the modifications that were made to the SPAR model. Additionally, the values of the key parameters are presented, followed by the plots of the total ΔCDF and ΔLERF and the individual contributions from the different timeframes as a function of STI. As discussed in Section 2.2.1, both the BWR and PWR analyses assumed that a leakage rate of 20 L_a was sufficient to defeat CAP and a leakage rate of 35 L_a was sufficient to defeat CAP and result in a large early release.

Browns Ferry Unit 1

The reactor at Browns Ferry Nuclear Plant, Unit 1 is a GE Type 4 BWR with a Mark I containment. The staff used SAPHIRE version 8.0.7.19S to modify version 8.22 of the Browns Ferry Unit 1 SPAR model (April 20, 2012). The SPAR model was modified by adding the logic structure shown in Figure 3 of Section 2.2.2 to the SPAR model to account for the failure of low pressure systems and functions and for a loss of containment integrity, either through a loss of containment integrity or failure of one or more MSIV pairs, that takes a suction from the suppression pool, including the containment spray system (CSS), low-pressure coolant injection (LCI), low-pressure core spray (LCS), and suppression pool cooling (SPC). The staff modified the following fault trees: CSS, CS1, LCI, LCS, LCS01, SPC, and SP1.

The Birnbaum importance of the LOCI and MSIV-FTC events in the Browns Ferry Unit 1 SPAR model are 7.7×10^{-4} and 1.1×10^{-3}, respectively. The baseline CDF and LERF values for Browns Ferry Unit 1 are 4.9×10^{-6} and 8.8×10^{-7}, respectively, which results in an overall $CCFP$ value of 0.18. The ΔCDF and ΔLERF as a function of STI are shown below in Figures 5 and 6, respectively.

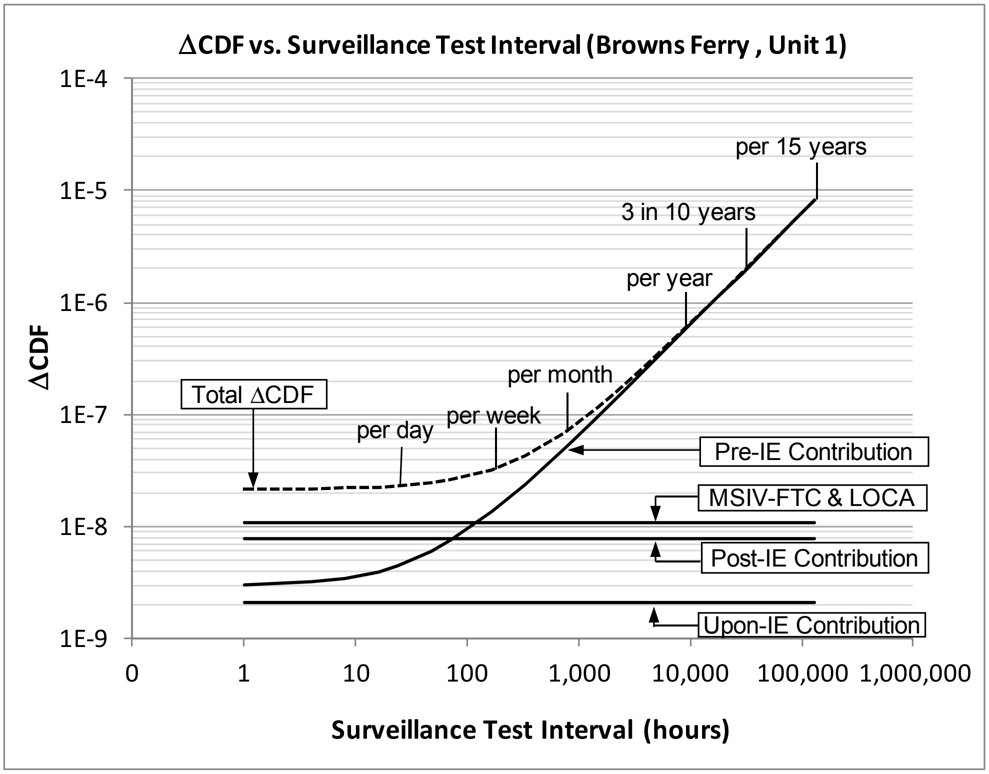

Figure 5. ΔCDF vs. surveillance test interval for Browns Ferry Unit 1, Mark I

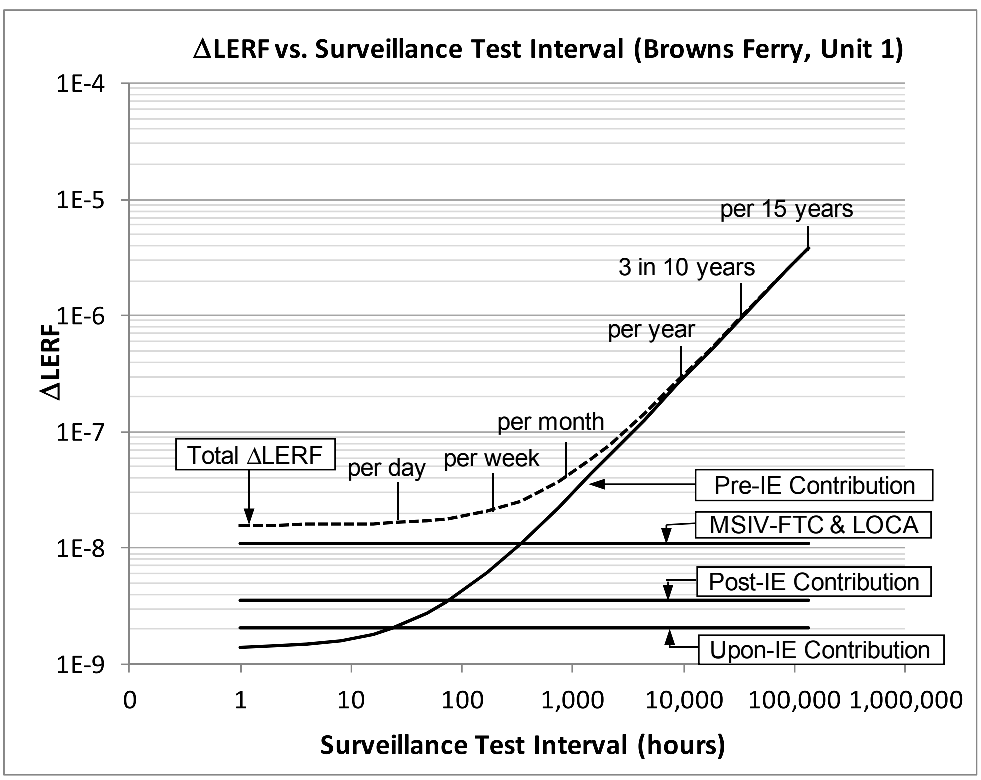

Figure 6. ΔLERF vs. surveillance test interval for Browns Ferry Unit 1, Mark I

Monticello

The reactor at Monticello Nuclear Generating Plant is a GE Type 3 BWR with a Mark I containment. The staff used SAPHIRE version 8.0.7.17 to modify version 8.20 of the Monticello SPAR model (May 2012). The SPAR model was modified by adding the logic structure shown in Figure 3 of Section 2.2.2 to account for the failure of low pressure systems and functions for a loss of containment integrity (either through a loss of containment integrity or failure of one or more MSIV pairs), that take a suction from the suppression pool, including the CSS, LCI, LCS, and SPC. The staff modified the following fault trees: CSS, CS01, LCI, LCS, SPC, and SP1.

The Birnbaum importance of the LOCI and MSIV-FTC events in the Monticello SPAR model are 1.4×10^{-3} and 1.1×10^{-4}, respectively. The baseline CDF and LERF values for Monticello are 7.4×10^{-7} and 1.2×10^{-7}, respectively, which result in an overall $CCFP$ value of 0.16. The ΔCDF and ΔLERF as a function of STI are shown below in Figures 7 and 8, respectively.

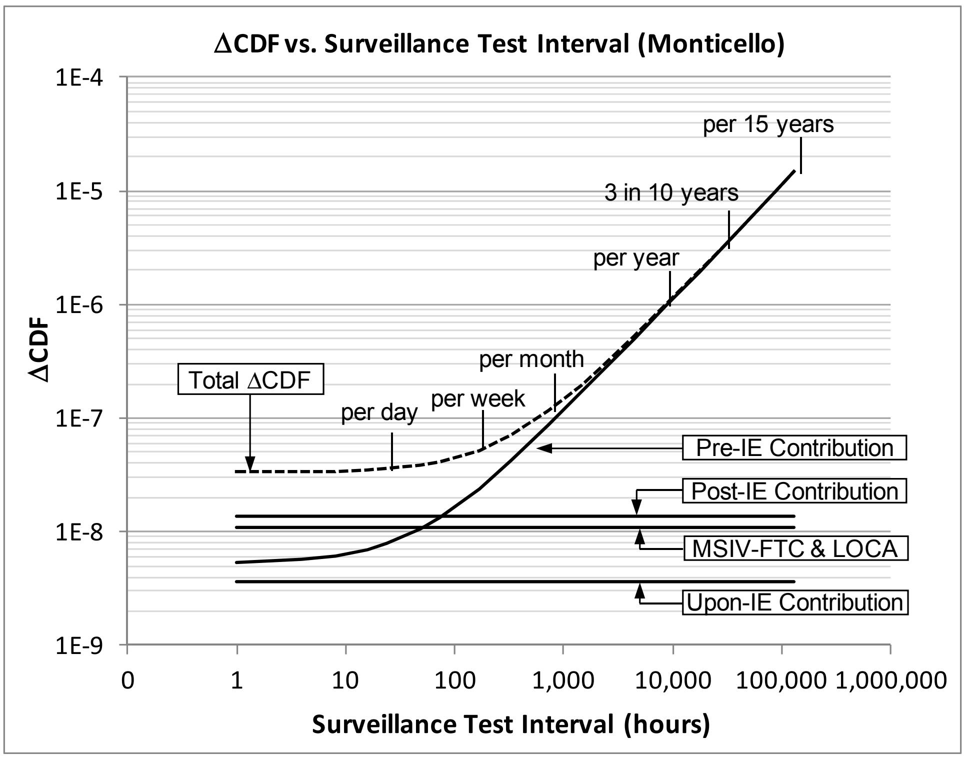

Figure 7. ΔCDF vs. surveillance test interval for Monticello, Mark I

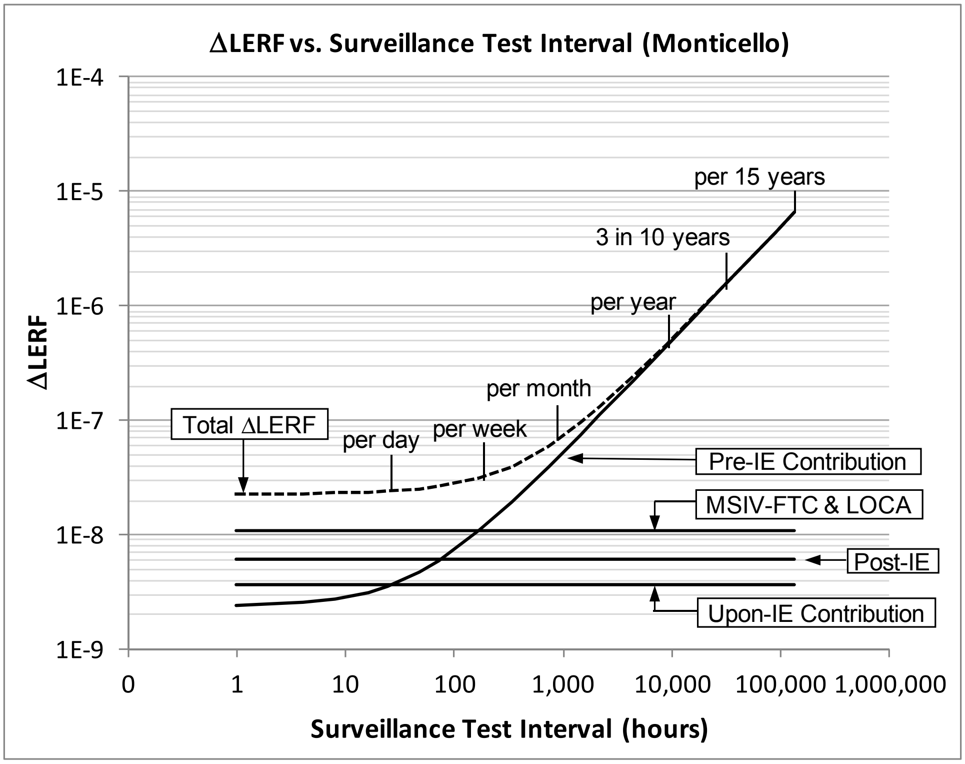

Figure 8. ΔLERF vs. surveillance test interval for Monticello, Mark I

Beaver Valley Unit 1

The reactor at Beaver Valley Power Station, Unit 1, is a Westinghouse three-loop PWR with a subatmospheric containment. The staff used SAPHIRE version 8.0.7.17 to modify version 8.17 of the Beaver Valley Unit 1 SPAR model, version 8.17 (January 2011). The SPAR model was modified to account for the loss of low-pressure recirculation (LPR) and/or high-pressure recirculation (HPR) caused by a loss of containment integrity.[1] The staff added the basic event to the LPR train fault trees (LPR-TRAINA and LPR-TRAINB). No logic changes were made to the recirculation spray system (RSS) fault tree. The purposes of the RSS system are to cool the containment sump water and to maintain containment pressure and temperature. The RSS system has two sets of pumps that take water from the containment sump, send it through a heat exchanger, and then spray the water into the containment.

The Birnbaum importance of the LPR-LOCI event in the Beaver Valley Unit 1 SPAR model is 3.0×10^{-4}. The baseline CDF and LERF values for Beaver Valley Unit 1 are 6.2×10^{-6} and 4.0×10^{-8}, respectively, which result in an overall *CCFP* value of 0.0065. The ΔCDF and ΔLERF as a function of STI are shown below in Figures 9 and 10, respectively.

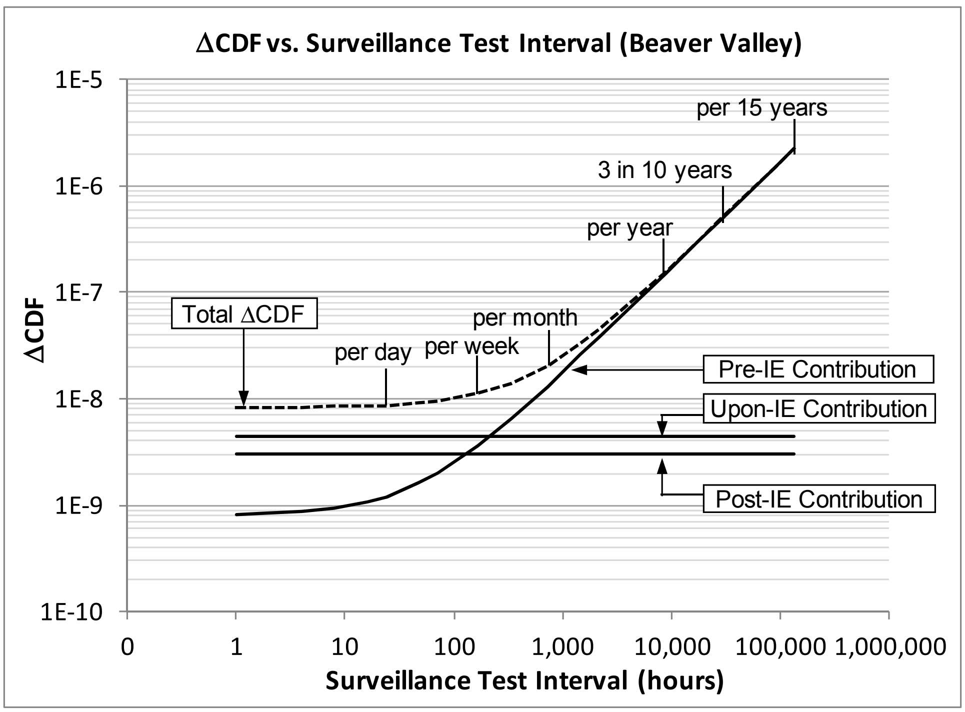

Figure 9. ΔCDF vs. surveillance test interval for Beaver Valley, subatmospheric

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[1] HPR at Beaver Valley, Unit 1, requires the LPR pumps (i.e., residual heat removal pumps) to provide adequate NPSH to the high pressure injection pumps during HPR.

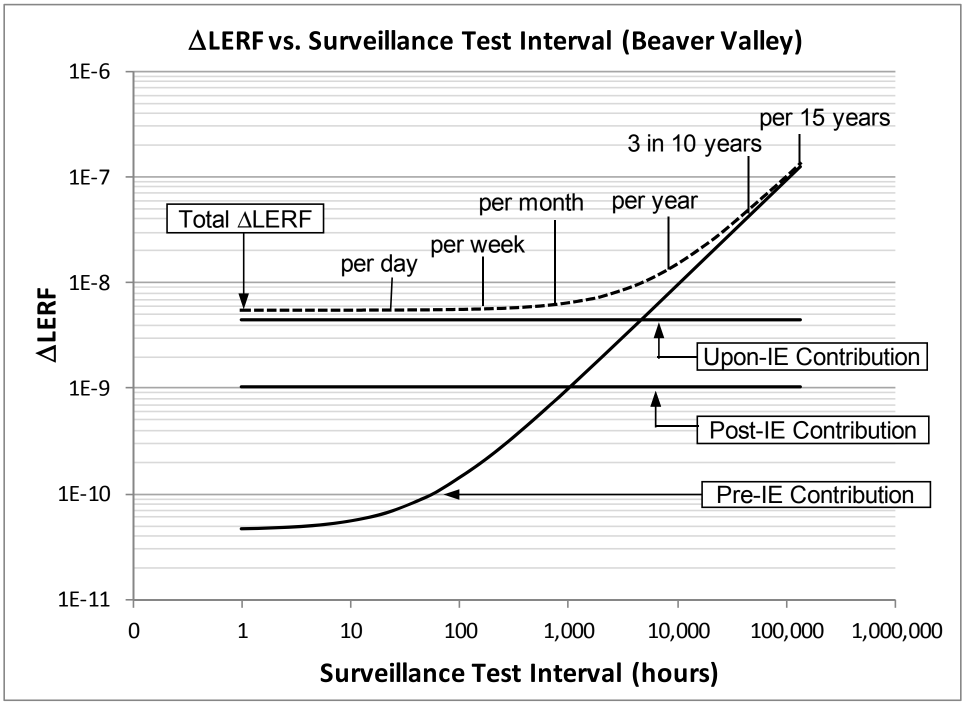

Figure 10. ΔLERF vs. surveillance test interval for Beaver Valley, subatmospheric

Fort Calhoun

The reactor at Fort Calhoun Station is a Combustion Engineering PWR with a large dry containment. The staff used SAPHIRE version 8.0.7.18 to modify version 8.15 of the Fort Calhoun SPAR model (January 2011). The staff modified the SPAR model by adding the basic event LOCI-FAILED to the SPAR model to account for the failure of LPR, HPR, and low-pressure, hot-leg recirculation (LPR-HL). The LPR-TRNA-F (*LPR Train A Fails*), LPR-TRNB-F (*LPR Train B Fails*), HPR-TRNA-F (*HPR Train A Fails*), HPR-TRNB-F (*HPR Train B Fails*), LPR-HL (*Low-Pressure, Hot Leg Recirculation Fails*) fault trees. No logic changes were made to the containment spray recirculation (CSR) system fault tree. The CSR system takes suction from the containment sump and delivers the water through the shutdown cooling heat exchangers to cool it down before discharging it into containment. In addition, containment air coolers (CACs) are modeled as a backup to the CSR system for sump water cooling.

The Birnbaum importance of the LOCI event in the Fort Calhoun SPAR model is 8.07×10^{-4}. The baseline CDF and LERF values for Fort Calhoun are 1.1×10^{-5} and 1.2×10^{-6}, respectively, which results in an overall $CCFP$ value of 0.11. The ΔCDF and ΔLERF as a function of STI are shown below in Figures 11 and 12, respectively.

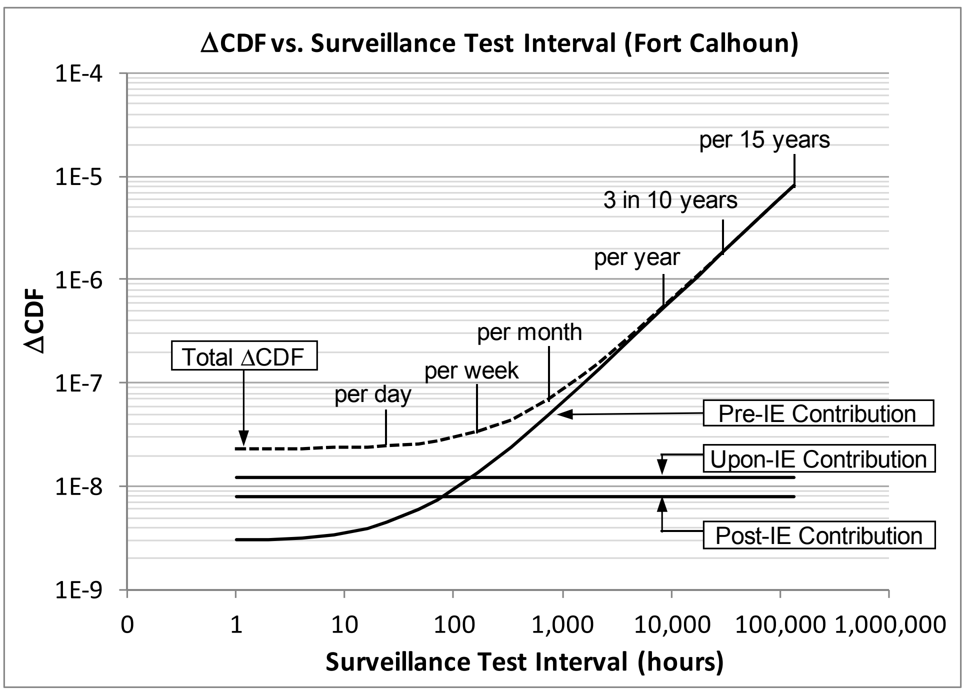

Figure 11. ΔCDF vs. surveillance test interval for Fort Calhoun, large dry

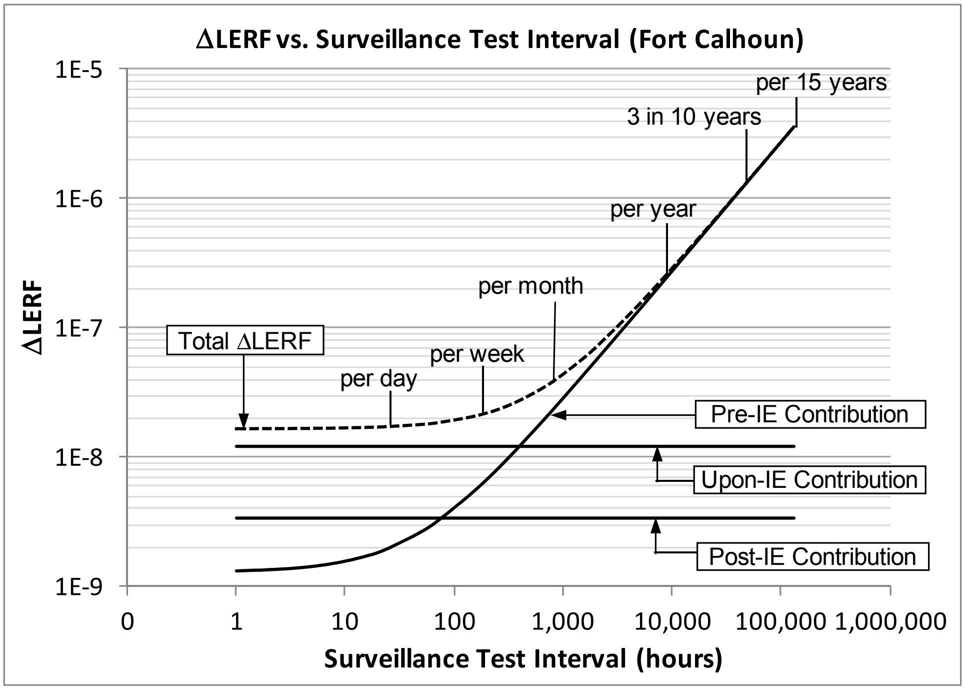

Figure 12. ΔLERF vs. surveillance test interval for Fort Calhoun, large dry

3. REGULATORY PERSPECTIVES

The previous sections have estimated the change in core-damage frequency (CDF) and large early release frequency (LERF) caused by crediting containment accident pressure (CAP) to provide adequate net positive suction head (NPSH) to the low-head emergency core cooling system (ECCS) and containment heat removal system (CHRS) pumps as a function of the containment integrity surveillance test interval (STI). To place these estimates into perspective when making regulatory decisions, it is useful to consider the circumstances under which licensees may request CAP credits and staff guidance that is relevant to these circumstances.

Historically, most licensee requests for CAP credit are part of broader nonrisk-informed license amendment requests. In fact, the motivation for preparing this report originally stemmed from concerns expressed by the staff and the Advisory Committee for Reactor Safeguards during the review of requests for extended power uprates (EPUs). However, the NRC cannot rule out the possibility that licensees may request CAP credits as part of nonrisk-informed or risk-informed license amendment requests. The approach taken in this report to estimate the change in CDF and LERF due to a CAP credit is independent of the reason (EPU or other) for the license amendment request or the type (nonrisk-informed or risk-informed) of license amendment request.

For risk-informed license amendment requests, licensees are expected to follow the guidance in Regulatory Guide (RG) 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis” (Ref. 11), and the staff will use Section 19.2 of the Standard Review Plan (SRP) (Ref. 12) to guide its review. RG 1.174 provides five key principles of risk-informed decisionmaking. Relevant to the change in CDF and LERF, the fourth key principle states that, “When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission’s Safety Goal Policy Statement.” RG 1.174 provides numerical risk acceptance guidelines (change in CDF vs. baseline CDF and change in LERF vs. baseline LERF) for judging if the fourth key principle has been met. With respect to risk-informed license amendment requests, the following apply:

- Licensees will provide plant-specific risk estimates, which may be compared to the estimates provided in this report.
- The change in CDF and LERF used in the RG 1.174 numerical risk acceptance guidelines must reflect all aspects of the license amendment request, not merely the CAP credit.
- The change in CDF and LERF should reflect all internal and external hazards (e.g., earthquakes and internal fires). RG 1.174 provides guidance for assessing the results of less than full-scope risk evaluations.
- Uncertainties in the risk evaluation should be considered when reaching a regulatory decision.

For nonrisk-informed license amendment requests, licensees and the staff should consider the guidance provided in Section 19.2, Appendix D, of the SRP, which addresses the use of risk information in review of nonrisk-informed license amendment requests. As stated in SRP Section 19.2, Appendix D:

> When a license amendment request complies with the regulations and other license requirements, there is a presumption by the Commission of adequate protection of public health and safety (Maine Yankee, ALAB-161, 6 AEC 1003 (1973)). However, circumstances may arise in which new information reveals an unforeseen hazard or a substantially greater potential for a known hazard to occur, such as identification of an issue that substantially increases risk. In such situations, the NRC has the statutory authority to require licensee action above and beyond existing regulations to maintain the level of protection necessary to avoid undue risk to public health and safety…. Under unusual circumstances that could introduce significant and unanticipated risks, the NRC staff reviewers would assume the burden of demonstrating that the presumption of adequate protection is not supported by the bases for the existing staff positions despite the fact that currently specified regulatory requirements are met.

Additionally, for nonrisk-informed license amendment requests, the staff should assess the requested changes and the need for and the effectiveness of any compensatory measures that might be warranted because of risk considerations by evaluating the changes relative to the safety principles and integrated decisionmaking process defined in RG 1.174. With respect to the estimates of the change in CDF and LERF provided in this report that result from the use of a CAP credit, it should be noted that:

- The numerical risk acceptance guidelines may serve as a point of reference for gauging risk impact but are not legally binding requirements and do not constitute a definition of adequate protection.

- The applicability of the estimates provided in this report to a specific nonrisk-informed license amendment request must be established.

- Uncertainties in the analyses must be considered in any finding that adequate protection is achieved.

Staff reviewers also may consider the guidance in the Office of Nuclear Reactor Regulation Office Instruction LIC-504, "Integrated Risk-Informed Decision Making Process for Emergent Issues." Specifically, Section 4.1.1 provides numerical risk criteria for determining if an emergent issue requires an immediate plant shutdown. It should be noted that the Commission has not provided a definition of adequate protection in terms of risk metrics. Nevertheless, it is reasonable to conclude that a nonrisk-informed license amendment request that includes a CAP credit would be unacceptable if the numerical risk criteria provided in Section 4.1.1 of LIC-504 are not met.

4. REFERENCES

(1) U. S. Nuclear Regulatory Commission, “Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps,” Safety Guide 1.1, November 1970 (Agencywide Document Access and Management System (ADAMS) Accession No. ML003739925).

(2) U. S. Nuclear Regulatory Commission, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” Regulatory Guide 1.82, March 2012 (ADAMS Accession No. ML052440060).

(3) Entergy letter to the U.S. Nuclear Regulatory Commission, “Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263–Supplement No. 38, Extended Power Uprate–Containment Overpressure Credit,” BVY 05-098, October 21, 2005 (ADAMS Accession No. ML053000281).

(4) ERIN Engineering and Research, Inc., “BFN EPU Containment Overpressure (COP) Credit Risk Assessment,” June 2006 (ADAMS Accession No. ML062090482).

(5) Nuclear Energy Institute (NEI), “Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Leakage Rate Test Surveillance Intervals,” Developed for NEI by the Electric Power Research Institute (EPRI) and Data Systems and Solutions, November 2001 (ADAMS Accession No. ML012990239).

(6) Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, EPRI, Palo Alto, CA, 1009325 R1, 2005 (ADAMS Accession No. ML053550424).

(7) U. S. Nuclear Regulatory Commission, “Resolution of Generic Safety Issues,” NUREG-0933, December 2011 (ADAMS Accession No. ML11353A382).

(8) Fort Calhoun Updated Safety Analysis Report (ADAMS Accession No. ML003717756).

(9) U.S. Nuclear Regulatory Commission, “Evaluation of Severe Accident Risks: Zion, Unit 1,” NUREG/CR-4551, Volume 7, Revision 1, Part 2A, March 1993 (ADAMS Accession No. ML070530496).

(10) U. S. Nuclear Regulatory Commission, “Standard Technical Specifications, General Electric, BWR/4 Plants,” NUREG-1433, Volume 1, Revision 4.0, April 2012 (ADAMS Accession No. ML12104A192).

(11) U. S. Nuclear Regulatory Commission, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Regulatory Guide 1.174, May 2011 (ADAMS Accession No. ML100910006).

(12) U. S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800.

APPENDIX A

DERIVATION OF THE PRE-INITIATOR LOCI PROBABILITY

To determine the effects of the surveillance test interval (STI) on the pre-initiator loss of containment integrity (LOCI) event, the staff developed a system model based on the implementation of a hypothetical technical specification patterned after the boiling-water reactor (BWR)/4 technical specification 3.6.3.2, “Primary Containment Oxygen Concentration.” As shown below in Figure A-1, the hypothetical technical specification accounts for the containment leakage rate, the containment leakage STI, the allowable short-term repair time, and the maximum allowed time to transition to shutdown if the short-term repair is unsuccessful.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Containment leakage at or above [20] L_a	Reduce containment leakage below [20] L_a	T_{ST} [24h]
Required action and Associated completion time not met	Shutdown plant	T_{SD} [8h]

SURVEILLANCE	FREQUENCY
Verify containment leakage less than [20] L_a	T_I [7 days]

Figure A-1. Hypothetical technical specification for the development of pre-initiator LOCI probability

The staff developed the pre-initiator LOCI event probability using a semi-Markov model, as informed by the hypothetical technical specification, to account for the risk contributions from various operational states of the reactor plant. As discussed in Section 2.2.2, the semi-Markov process is used to describe a system that only exists in one of any number of discrete states at a given point in time. The time spent by the system in a given state is randomly determined using an arbitrary probability distribution, and the probability that the system transitions to another state only depends on its current state and is independent of the system’s past history. The model used for this study accounts for the following four system states and is independent of containment type:

- State 1: The containment is intact.
- State 2: The containment has an undetected leak.
- State 3: The leak is detected, a repair is initiated.
- State 4: The repair is not completed in the allotted time and the reactor is transitioned to shutdown.

The semi-Markov state diagram shows the relationships between the system states and the transitions from one state to another, as shown in Figure A-2.

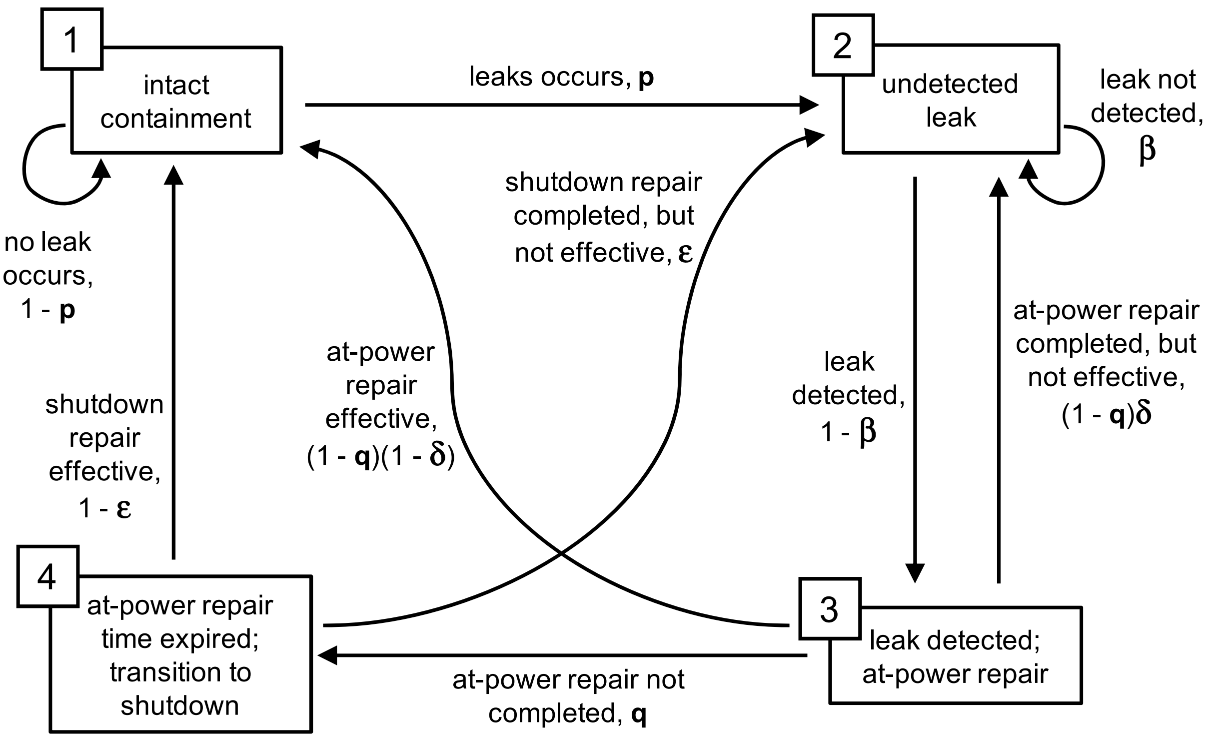

Figure A-2. Semi-Markov model state diagram for the pre-initiator LOCI probability estimate

In this figure, the parameter p represents the likelihood that a containment will develop a pre-existing leak and the parameter is a function of the STI, T_I, and the containment failure rate, λ, for a given leakage rate of interest. The parameters β, δ, and ε represent the probability that the leakage test produces a false-negative result (i.e., the leakage test falsely indicates that the leakage rate is below the value in the technical specification) for states 2, 3, and 4, respectively. The parameter q represents the probability that the short-term at-power repair is not completed in the time interval, T_{ST}, thereby requiring the reactor be shutdown within the time period T_{SD}. The following relations determine variables p and q :

$$p = 1 - \exp\,(-\lambda T_I) \qquad \text{(Equation A-1)}$$

$$q = \exp\,(-\frac{T_{ST}}{\tau}) \qquad \text{(Equation A-2)}$$

The parameter τ represents the mean time to repair the leakage.

For this model, the likelihood that the system exists in a given state is determined by combining the individual likelihoods of making the transition into that state from other states. Using this relationship, together with the diagram in Figure A-2, the following expressions are derived for the probability, expressed here as π, that the system is in one of the four states:

$$\pi_1 = (1-p)\pi_1 \;+\; (1-q)(1-\delta)\pi_3 \;+\; (1-\varepsilon)\pi_4 \qquad \text{(Equation A-3)}$$

$$\pi_2 = p\pi_1 \;+\; \beta\pi_2 \;+ (1-q)\delta\pi_3 + \varepsilon\pi_4 \qquad \text{(Equation A-4)}$$

$$\pi_3 = (1-\beta)\pi_2 \qquad \text{(Equation A-5)}$$

$$\pi_4 = q\pi_3 \quad \text{(Equation A-6)}$$

The steady-state solutions of π_1, π_2, π_3, and π_4 are determined by solving the set of linear equations for each state, which yields the following relations that are expressed only in terms of the defined model parameters,

$$\pi_1 = \frac{1-\beta}{pD}[1-(1-q)\delta - \varepsilon q] \quad \text{(Equation A-7)}$$

$$\pi_2 = \frac{1}{D} \quad \text{(Equation A-8)}$$

$$\pi_3 = \frac{(1-\beta)}{D} \quad \text{(Equation A-9)}$$

$$\pi_4 = \frac{(1-\beta)q}{D} \quad \text{(Equation A-10)}$$

where,

$$D = \frac{(1-\beta)}{p}[1-(1-p)\delta - \varepsilon q] + (2-\beta) + (1-\beta)q \quad \text{(Equation A-11)}$$

The average times that the system spends in a given state are expressed as the following:

$$M_1 = T_I\left(1-\frac{p}{2}\right) \quad \text{(Equation A-12)}$$

$$M_2 = \frac{T_I}{2}(1+\beta) \quad \text{(Equation A-13)}$$

$$M_3 = qT_{ST} + (1-q)\left[\frac{\tau-(\tau+T_{ST})exp\left(\frac{-T_{ST}}{\tau}\right)}{1-\exp\left(\frac{-T_{ST}}{\tau}\right)}\right] \quad \text{(Equation A-14)}$$

$$M_4 = T_{SD} \quad \text{(Equation A-15)}$$

For a given leak size, the total probability that the containment has a pre-existing leak at a time when CAP might be needed during an accident (i.e., the pre-initiator LOCI event probability) is expressed as the long-run fraction of time that the system spends in states 2, 3, and 4, as expressed below in Equation A-16:

$$P_{234} = \frac{M_2+(1-\beta)M_3+(1-\beta)qM_4}{\frac{1-\beta}{p}[1-(1-p)\delta-\varepsilon q]M_1+M_2+(1-\beta)M_3+(1-\beta)qM_4} \quad \text{(Equation A-16)}$$

Although the staff included several parameters in the semi-Markov analysis, sensitivity studies were performed over a wide range of parameter values that indicated that the pre-initiator LOCI probability primarily depends on the containment leakage failure rate and the STI of the leak test, as demonstrated by the sensitivity study results in Figure A-3. The parameters T_{SD} and the ratio of T_{ST}/τ were varied between what were considered to be reasonable extreme minimum and maximum values of the parameters. The two sets of plots produced for different values of λ demonstrate strong dependence on this variable. Similar sensitivity analyses performed for the parameters β, δ, and ε demonstrated little dependence on these parameters between a range of 0.1 and 0.0 (i.e., a perfect test).

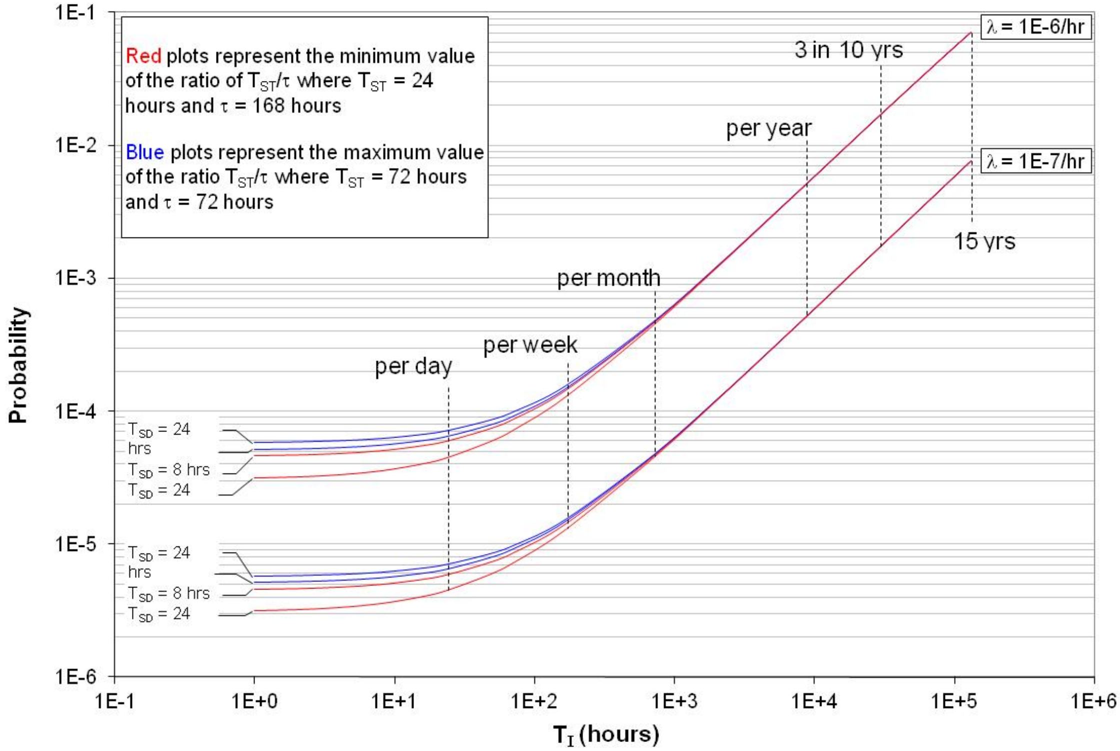

Figure A-3. Sensitivity study results for the semi-Markov model parameters

As such, the pre-initiator LOCI event probability for a given leakage rate is considered to be a function of the STI, with all other parameters held constant. The assumed values of the fixed parameters are as follows,

$$\beta = 0.05$$

$$\delta = 0.05$$

$$\varepsilon = 0.05$$

$$T_{ST} = 24\ hours$$

$$T_{SD} = 8\ hours$$

$$\tau = 72\ hours$$