

U.S. NRC Level 3 Probabilistic Risk Assessment Project

Volume 5a: Reactor, At-Shutdown, Level 1
PRA for Internal Events

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U.S. NRC Level 3 Probabilistic Risk Assessment Project

Volume 5a: Reactor, At-Shutdown, Level 1 PRA for Internal Events

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Prepared by:
J. Wood¹

¹ U.S. Nuclear Regulatory Commission

A. Kuritzky, NRC Level 3 PRA Project Program Manager

ABSTRACT

The U.S. Nuclear Regulatory Commission performed a full-scope site Level 3 probabilistic risk assessment (PRA) project (L3PRA project) for a two-unit pressurized-water reactor reference plant. The scope of the L3PRA project encompasses all major radiological sources on the site (i.e., reactors, spent fuel pools, and dry cask storage), all internal and external hazards, and all modes of plant operation. A full-scope site Level 3 PRA for a nuclear power plant site can provide valuable insights into the importance of various risk contributors by assessing accidents involving one or more reactor cores as well as other site radiological sources. This report, one of a series of reports documenting the models and analyses supporting the L3PRA project, specifically addresses the reactor, at-shutdown, Level 2 PRA model for internal events for a single unit. The analyses documented herein are based information for the reference plant as it was designed and operated as of 2012 and does not reflect the plant as it is currently designed, licensed, operated, or maintained.¹

CAUTION: The L3PRA project was developed to meet the specific objectives outlined in SECY-11-0089 using state of practice methods and data. While the study provides valuable insights and addresses its key objectives, due to limitations in time, resources, and plant information, some technical aspects of the study were subjected to simplifications or were not fully addressed. As such, inclusion of approaches in the L3PRA project documentation should **not** be viewed as an endorsement of these approaches for regulatory purposes.

¹ An overview report, which covers all three PRA levels, has been created for each major element of the L3PRA project scope (e.g., for the combined internal event and internal flood PRAs for a single reactor unit operating at full power). These overview reports include a reevaluation of plant risk based on a set of updated plant equipment and PRA model assumptions (e.g., incorporation of the current reactor coolant pump shutdown seal design at the reference plant and the potential impact of the U.S. nuclear power industry's proposed safety strategy, called Diverse and Flexible Coping Strategies [FLEX], both of which reduce the risk to the public).

FOREWORD

The U.S. Nuclear Regulatory Commission (NRC) performed a full-scope site Level 3 probabilistic risk assessment (PRA) project (L3PRA project) for a two-unit pressurized-water reactor reference plant. The staff undertook this project in response to Commission direction in the staff requirements memorandum dated September 21, 2011 (Agencywide Documents and Management System [ADAMS] Accession No. ML112640419) resulting from SECY-11-0089, “Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities,” dated July 7, 2011 (ML11090A039).

Licensee information used in performing the Level 3 PRA project was voluntarily provided based on a licensed, operating nuclear power plant. The information provided reflects the plant as it was designed and operated as of 2012 and does not reflect the plant as it is currently designed, licensed, operated, or maintained. In addition, the information provided for the reference plant was changed based on additional information, assumptions, practices, methods, and conventions used by the NRC in the development of plant-specific PRA models used in its regulatory decision-making. **As such, the L3PRA project reports will not be the sole basis for any regulatory decisions specific to the reference plant.**

Each set of L3PRA project reports covering the Level 1, 2, and 3 PRAs for a specific site radiological source, plant operating state, and hazard group is accompanied by an overview report. The overview reports summarize the results and insights from all three PRA levels.

To provide results and insights better aligned with the current design and operation of the reference plant, the overview reports also provide the results of a parametric sensitivity analysis based on a set of new plant equipment and PRA model assumptions for all three PRA levels. The sensitivity analysis reflects the current reactor coolant pump shutdown seal design at the reference plant, as well as the potential impact of FLEX strategies,² both of which reduce the risk to the public.

A full-scope site Level 3 PRA for a nuclear power plant site can provide valuable insights into the importance of various risk contributors by assessing accidents involving one or more reactor cores as well as other site radiological sources (i.e., spent fuel in pools and dry storage casks). These insights may be used to further enhance the regulatory framework and decision-making and to help focus limited agency resources on issues most directly related to the agency’s mission to protect public health and safety. More specifically, potential future uses of the L3PRA project can be categorized as follows (a more detailed list is provided in SECY-12-0123, “Update on Staff Plans to Apply the Full-Scope Site Level 3 PRA Project Results to the NRC’s Regulatory Framework,” dated September 13, 2012 [ML12202B170]):

- enhancing the technical basis for the use of risk information (e.g., obtaining updated and enhanced understanding of plant risk as compared to the Commission’s safety goals)
- improving the PRA state of practice (e.g., demonstrating new methods for site risk assessments, which may be particularly advantageous in addressing the risk from

² FLEX refers to the U.S. nuclear power industry’s proposed safety strategy, called Diverse and Flexible Coping Strategies. FLEX is intended to maintain long-term core and spent fuel cooling and containment integrity with installed plant equipment that is protected from natural hazards, as well as backup portable onsite equipment. If necessary, similar equipment can be brought from off site.

advanced reactor designs, a multi-unit accident, or an accident involving spent fuel; and using PRA information to inform emergency planning)

- identifying safety and regulatory improvements (e.g., identifying potential safety improvements that may lead to either regulatory improvements or voluntary implementation by licensees)
- supporting knowledge management (e.g., developing or enhancing in-house PRA technical capabilities)

In addition, the overall L3PRA project model can be exercised to provide insights regarding other issues not explicitly included in the current project scope (e.g., security-related events or the use of accident tolerant fuel). Furthermore, some future advanced light water reactor (ALWR) and advanced non-light water reactor (NLWR) applicants may rely heavily on the results of analyses similar to those used in the L3PRA project to establish their licensing basis and design basis by using the Licensing Modernization Project (LMP) (NEI 18-04, Rev. 1) which was endorsed via Regulatory Guide 1.233 in June 2020. Licensees who use the LMP framework are required to perform Level 3 PRA analyses. Therefore, another potential use of the methodology and insights generated from this study is to inform regulatory, policy, and technical issues pertaining to ALWRs and NLWRs.

The results and perspectives from this report, as well as all other reports prepared in support of the L3PRA project, will be incorporated into a summary report to be published after all technical work for the L3PRA project has been completed.

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EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) performed a full-scope site Level 3 probabilistic risk assessment (PRA) project (L3PRA project) for a two-unit pressurized-water reactor reference plant. The staff undertook this project in response to Commission direction in the staff requirements memorandum dated September 21, 2011 (Agencywide Documents and Management System [ADAMS] Accession No. ML112640419) resulting from SECY-11-0089, “Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities,” dated July 7, 2011 (ML11090A039).

As described in SECY-11-0089, the objectives of the L3PRA project are the following:

- Develop a Level 3 PRA, generally based on current state-of-practice methods, tools, and data, that (1) reflects technical advances since the last NRC-sponsored Level 3 PRAs (ML040140729), which were completed over 30 years ago, and (2) addresses scope considerations that were not previously considered (e.g., low-power and shutdown risk, multi-unit risk, other radiological sources).
- Extract new insights to enhance regulatory decision-making and to help focus limited NRC resources on issues most directly related to the agency’s mission to protect public health and safety.
- Enhance PRA staff capability and expertise and improve documentation practices to make PRA information more accessible, retrievable, and understandable.
- Demonstrate technical feasibility and evaluate the realistic cost of developing new Level 3 PRAs.

This report documents an evaluation of the single-unit operation during low power and shutdown and the Level 1 PRA for internal events for shutdown operating states during a refueling outage that supports the L3PRA project. Licensee information used in performing the L3PRA project was voluntarily provided based on a licensed, operating nuclear power plant. The information provided reflects the plant as it was designed and operated as of 2012 and does not reflect the plant as it is currently designed, licensed, operated, or maintained. (For example, the L3PRA does not reflect the current reactor coolant pump shutdown seal design or the potential impact of FLEX strategies.³) In addition, the information provided for the reference plant was changed based on additional information, assumptions, practices, methods, and conventions used by the NRC in the development of plant-specific PRA models. **As such, this report will not be the sole basis for any regulatory decisions specific to the reference plant.**

As a part of the L3PRA study, the NRC has constructed a Level 1, at-shutdown, PRA model for internal events using the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) software. The shutdown Level 1 PRA model was developed to assess the risk associated with the shutdown plant operating states (POSS) of a refueling outage. Six

³ FLEX refers to the U.S. nuclear power industry’s proposed safety strategy, called Diverse and Flexible Coping Strategies. FLEX is intended to maintain long-term core and spent fuel cooling and containment integrity with installed plant equipment that is protected from natural hazards, as well as backup portable onsite equipment. If necessary, similar equipment can be brought from off site.

categories of initiating events are assessed: loss of offsite power, loss of RHR cooling, loss of inventory, loss of power to a 4,160V AC safety-related bus, overdraining entering mid-loop, and loss of nuclear service cooling water.

The total modeled shutdown CDF during refueling outages is estimated to be 1.3×10^{-5} per year on average. The shutdown CDF is less than the total at-power CDF for internal events of approximately 6.0×10^{-5} per year, as documented in (NRC, 2022). While the shutdown CDF is nominally less than the at-power CDF when considered on an average per year basis, it is important to remember that the shutdown risk is concentrated during a much shorter time period (about 20 days assuming a 30-day refueling outage every 18 months) compared to full power operation (93 percent of a calendar year or about 340 days). When considering the relative durations of shutdown and at-power operation, the core damage probability is more than three times higher during the modeled shutdown refueling outage than at-power operation.

Notable observations from the shutdown CDF results include:

- The highest contributing initiating event category to the shutdown CDF is loss of offsite power. The significant contribution from losses of offsite power is similar to the results from the L3PRA project at-power, internal events PRA (NRC, 2022).
- The next highest contributing initiating event category is overdrain entering mid-loop prior to refueling. Mid-loop operation is generally recognized as a high-risk evolution in qualitative risk assessments. The short time available until RCS boiling occurs upon loss of cooling and the reliance on operator actions to diagnose and respond to the event in a short timeframe are challenges that contribute to the risk significance of overdrain scenarios.
- The loss of inventory with RCS at pressure is also a significant initiating event category. The relatively high initiating event frequency and significant human failure events are important contributors to this scenario.
- A significant portion of the shutdown CDF (80 percent) occurs during POSs when the containment equipment hatch is initially open at the time of the initiating event occurrence. While actions can be taken to close containment (these actions are considered in the Level 2 PRA), the initial containment status is an important attribute to consider in assessing the potential for radiological releases.
- The highest contributing plant operating states in terms of CDF include POS 8E (refueling operations early), POS 8L (refueling operations late), POS 5A (operation with pressurizer solid), and POS 6 (mid-loop operations). POS 8E and POS 8L represent operation with the refueling cavity flooded. While the accident scenarios in these states progress slowly due to the large water inventory available in the refueling cavity, the unavailability of equipment for maintenance and the relatively long POS durations are important factors influencing the risk in these states.

Sources of model uncertainty for the shutdown Level 1 PRA are identified and assessed. Several model uncertainties were identified that can have significant impacts on the CDF results. Sensitivity studies were performed for selected model uncertainties to quantify the impacts on the results. Some of the model uncertainties with the largest potential impacts on CDF include consideration of possible steam generator nozzle dam failures, re-quantifying

significant human error probabilities with upper-bound estimates, and alternative evaluations of loss of offsite power frequencies for shutdown operations.

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¹ Formerly of U.S. Nuclear Regulatory Commission

² U.S. Nuclear Regulatory Commission

³ Innovative Engineering & Safety Solutions, LLC

⁴ Energy Research, Inc.

⁵ Formerly of Sandia National Laboratories

⁶ Formerly of Idaho National Laboratory

⁷ Idaho National Laboratory

⁸ Formerly of Westinghouse Electric Company

ABBREVIATIONS AND ACRONYMS

AC	Alternating current
ADAMS	Agencywide Documents and Management System
AFW	Auxiliary feedwater
ALWR	Advanced light water reactor
AMSAC	ATWS mitigation system actuation circuitry
ANS	American Nuclear Society
AOP	Abnormal operating procedure
ARV	Atmospheric relief valves
ASME	American Society of Mechanical Engineers
ATWS	Anticipated transient without scram
CCDP	Conditional core damage probability
CCP	Centrifugal charging pump
CCW	Component cooling water
CD	Core damage
CDF	Core damage frequency
COPS	Cold over-pressure protection system
CST	Condensate storage tank
CVCS	Chemical and volume control system
DC	Direct current
ECCS	Emergency core cooling system
EDG	Emergency diesel generator
EOP	Emergency operating procedure
EPRI	Electric Power Research Institute
ESF	Engineered safety feature
ESFAS	Engineered safety features actuating system
ET	Event tree
FLEX	Diverse and Flexible Mitigation Capability
FSAR	Final safety analysis report
FV	Fussell-Vesely
GPM	Gallons per minute
HEP	Human error probability
HFE	Human failure event
HPI	High-pressure injection
HRA	Human reliability analysis
HVAC	Heating, ventilation, and air-conditioning
IE	Initiating event
ISLOCA	Interfacing system loss-of-coolant accident
kV	Kilovolt
LCO	Limiting condition of operation

LER	Licensee event report
LMP	Licensing Modernization Project
L3PRA	Level 3 Probabilistic Risk Assessment (project)
LO4160V	Loss of power to 4,160V AC bus
LOCA	Loss-of-coolant accident
LOINV	Loss of inventory
LOOP (or LOSP)	Loss of offsite power
LORHR	Loss of residual heat removal
LPI	Low pressure injection
LPSD	Low-power and shutdown
MDP	Motor-driven pumps
MFW	Main feedwater
MOV	Motor-operated valves
MSIV	Main steam isolation valves
NLWR	Non-light water reactor
NRC	Nuclear Regulatory Commission
NSCW	Nuclear service cooling water
ODML	Overdrain entering mid-loop
PORV	Power-operated relief valve
POS	Plant operating state
PRA	Probabilistic risk assessment
PSV	Pressurizer safety valve
PTL	Pull-to-lock
PTLR	Pressure temperature limits report
PWR	Pressurized-water reactor
RAT	Reserve auxiliary transformer
RCP	Reactor coolant pump
RCS	Reactor coolant system
RHR	Residual heat removal
RHUT	Recycle holdup tank
RIR	Risk Increase Ratio
RPV	Reactor pressure vessel
RRR	Risk Reduction Ratio
RVLIS	Reactor vessel level indicating system
RWST	Refueling water storage tank
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
SAT	Standby auxiliary transformer
SBO	Station blackout
SD	Shutdown
SDC	Shutdown cooling
SDP	Significance Determination Process

SFP	Spent fuel pool
SFPC	Spent fuel pool cooling
SG	Steam generator
SGTR	Steam generator tube rupture
SI	Safety injection
SPAR	Standardized plant analysis risk
SSPS	Solid state protection system
TBV	Turbine bypass valve
TDP	Turbine-driven pump
T-H	Thermal-hydraulic
UAT	Unit auxiliary transformer
V	Volt

1 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) performed a full-scope site Level 3 probabilistic risk assessment (PRA) project (L3PRA project) for a two-unit pressurized-water reactor reference plant. The staff undertook this project in response to Commission direction in the staff requirements memorandum dated September 21, 2011 (Agencywide Documents and Management System [ADAMS] Accession No. ML112640419) resulting from SECY-11-0089, “Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities,” dated July 7, 2011 (ML11090A039).

As described in SECY-11-0089, the objectives of the L3PRA project are the following:

- Develop a Level 3 PRA, generally based on current state-of-practice methods, tools, and data,¹ that (1) reflects technical advances since the last NRC-sponsored Level 3 PRAs (NRC, 1994a), which were completed over 30 years ago, and (2) addresses scope considerations that were not previously considered (e.g., low-power and shutdown risk, multi-unit risk, other radiological sources).
- Extract new insights to enhance regulatory decision-making and to help focus limited NRC resources on issues most directly related to the agency’s mission to protect public health and safety.
- Enhance PRA staff capability and expertise and improve documentation practices to make PRA information more accessible, retrievable, and understandable.
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This report documents an evaluation of the single-unit operation during low power and shutdown and the Level 1 PRA for internal events for shutdown operating states during a refueling outage that supports the L3PRA project.

Licensee information used in performing the L3PRA project was voluntarily provided based on a licensed, operating nuclear power plant. The information provided reflects the plant as it was designed and operated as of 2012 and does not reflect the plant as it is currently designed, licensed, operated, or maintained. (For example, the L3PRA does not reflect the current reactor coolant pump shutdown seal design or the potential impact of FLEX strategies.²) In addition, the information provided for the reference plant was changed based on additional information, assumptions, practices, methods, and conventions used by the NRC in the development of

¹ “State-of-practice” methods, tools, and data refer to those that are routinely used by the NRC and industry or have acceptance in the PRA technical community. While the L3PRA project is intended to be a state-of-practice study, note that there are several technical areas within the project scope that necessitated advancements in the state of practice (e.g., modeling of multi-unit site risk, modeling of spent fuel in pools or casks, and of human reliability analysis for other than internal events and internal fires).

² FLEX refers to the U.S. nuclear power industry’s proposed safety strategy, called Diverse and Flexible Coping Strategies. FLEX is intended to maintain long-term core and spent fuel cooling and containment integrity with installed plant equipment that is protected from natural hazards, as well as backup portable onsite equipment. If necessary, similar equipment can be brought from off site.

plant-specific PRA models. **As such, the L3PRA project reports will not be the sole basis for any regulatory decisions specific to the reference plant.**

Since the L3PRA project involves multiple PRA models, each of these models should be considered a “living PRA” until the entire project is complete. It is anticipated that the models and results of the L3PRA project are likely to evolve over time, as other parts of the project are developed, or as other technical issues are identified. As such, the final models and results of the project (which will be documented in a summary report to be published after all technical work for the L3PRA project has been completed) may differ in some ways from the models and results provided in the current report.

The series of reports for the L3PRA project are organized as follows:

Volume 1: Final summary report

Volume 2: Background, site and plant description, and technical approach

Volume 3: Reactor, at-power, internal event and flood PRA (overview report)

Volume 3a: Level 1 PRA for internal events

Volume 3b: Level 1 PRA for internal floods

Volume 3c: Level 2 PRA for internal events and floods

Volume 3d: Level 3 PRA for internal events and floods

Volume 4: Reactor, at-power, internal fire and external event PRA (overview report)

Volume 4a: Level 1 PRA for internal fires

Volume 4b: Level 1 PRA for seismic events

Volume 4c: Level 1 PRA for high wind events and other hazards evaluation

Volume 4d: Level 2 PRA for internal fires and seismic and wind-related events

Volume 4e: Level 3 PRA for internal fires and seismic and wind-related events

Volume 5: Reactor, at-shutdown, internal event PRA (overview report)

Volume 5a: Level 1 PRA for internal events

Volume 5b: Level 2 PRA for internal events

Volume 5c: Level 3 PRA for internal events

Volume 6: Spent fuel pool all hazards PRA (no overview report)

Volume 6a: Level 1 and Level 2 PRA

Volume 6b: Level 3 PRA

Volume 7: Dry cask storage, all hazards, Level 1, Level 2, and Level 3 PRA

Volume 8: Integrated site risk, all hazards, Level 1, Level 2, and Level 3 PRA

CAUTION: The L3PRA project was developed to meet the specific objectives outlined in SECY-11-0089 using state of practice methods and data. While the study provides valuable insights and addresses its key objectives, due to limitations in time, resources, and plant information, some technical aspects of the study were

subjected to simplifications or were not fully addressed. As such, inclusion of approaches in the L3PRA project documentation should **not** be viewed as an endorsement of these approaches for regulatory purposes.

1.1 **Objectives and Scope**

The objective of this report is to describe the evaluation of lower power and shutdown (LPSD) risk and insights. The results presented here are the LPSD Level 1 PRA internal event results for core damage frequency (CDF). The model and documentation are intended to support a Level 2 PRA for LPSD conditions, but Level 2 PRA results are not presented in this report.

The scope of the model is limited to certain hazard categories, accident types, and plant states. Some areas that are not addressed here may require additional study (some of these are identified and discussed in 12APPENDIX A Candidates for Future Work). Some notes on the model scope:

- The model represents LPSD risk for reference plant unit 1 only. Unit 2 is expected to have similar risk contributions as those that apply to unit 1 due to the symmetry of the two units' designs and operation. This model does not attempt to represent the interactions and dependencies between the units for multi-unit accidents. Multi-unit accidents are addressed as part of the L3PRA project integrated site risk task.
- The risk from the spent fuel pools is not included in this model. The spent fuel pools are addressed in the L3PRA project spent fuel pool PRA.
- Low power plant operating states (POSS)³ are assessed qualitatively and quantitatively through a selection of sensitivity studies, but low power operating states are not included in the quantified model results.
- The model and report address internal events only. Other hazard categories (e.g., internal fire, internal flooding, heavy load drops, and seismic events) are not in the scope of this report.
- Several potential initiating event categories are identified (in Section 3). Each identified initiating event category is assessed qualitatively, and a subset of initiating events are selected for modeling. The initiating event categories that are modeled are:
 - Loss of residual heat removal (RHR) cooling
 - Loss of reactor coolant system (RCS) inventory
 - Over draining when entering mid-loop
 - Loss of electrical power to the class 1E 4,160V alternating current (AC) bus supplying power to the operating RHR train

³ As discussed in Section 2.4, a POS is a standard arrangement of the plant during which the plant conditions are relatively constant, are modeled as constant, and are distinct from other configurations in ways that impact risk.

- Loss of offsite power (LOOP)
- Loss of nuclear service cooling water (NSCW)

Addressing the broad range of outage types and plant conditions that can occur is a significant challenge associated with LPSD PRA. Several different types of outages and shutdown conditions can occur at the plant. This model does not attempt to represent all possible outages and plant configurations or an average across these conditions. The modeled outage is informed by reference plant's experience with past outages. The assumptions about plant configurations and equipment unavailabilities have been defined based on actual plant conditions as documented in reference plant outage reports or observed during a visit to the reference plant site. Some additional notes regarding the modeling of plant conditions:

- This report discusses different types of plant outages, but the model focuses on representing a refueling outage. Other types of outages are not modeled but are addressed qualitatively and CDF contributions are estimated in sensitivity studies.
- The model results are sensitive to the assumptions about equipment unavailability. Testing and maintenance practices and equipment unavailability during outages can vary from plant to plant and from outage to outage. Therefore, care should be taken to interpret the model results in the context of the unavailability assumptions and not to generalize to all plant outage practices.
- Frequencies of initiating events can be influenced by the plant conditions, the types of maintenance activities being performed, the type of outage, and other factors. While these factors can be important, the available data on shutdown initiating events have limited information about what the plant conditions were during those events. The initiating event frequencies were estimated for specific plant conditions when sufficient data and information were available to support the estimates. In some cases, the initiating event frequencies remain constant across different plant states and are not specific to the plant conditions.

Even with these limitations the analysis and results presented in this report provide useful insights on the risks associated with LPSD operation.

1.2 PRA Approach and Report Organization

The approach for developing the shutdown PRA involved several activities to develop an understanding of the refueling outage operations at the reference plant and the broader industry-wide operating experience with shutdown states. These activities included:

- Review outage information provided by the reference plant, including procedures, past outage summary reports, and drafted shutdown PRA documents
- Review of LPSD operating experience (documented in 12APPENDIX D
- A visit to the reference plant site to observe refueling outage operations, observe the plant simulator, and review outage training and procedures.

The above information was reviewed and used as inputs to the PRA development tasks. The plant outage types and POSs to be included in the model were developed. The initiating events to be included in the model were identified. One of the key activities was defining the approach for human reliability analysis (HRA). The SPAR-H method (NRC, 2005a) was selected to be used for L3PRA project, and guidance was developed to apply a consistent approach for assessing performance shaping factors. The above activities and other PRA development tasks are documented in this report. In performing and documenting the analysis the trial use standard for LPSD PRA methodology (ASME/ANS, 2014) was consulted. Each applicable technical element identified in the standard is addressed in this report. The report is structured as follows:

- Section 2 addresses POSs and the types of outages that occur during the reference plant LPSD operations. The frequencies and durations of the modeled POSs are provided here.
- Section 3 presents the initiating event analysis. Initiating events are identified for LPSD operating conditions.
- Section 4 documents the accident sequence analysis. This section describes the event tree models that are used to represent the plant response to initiating events.
- Section 5 describes the success criteria that are applied to modeled end states and to safety functions and systems used for accident mitigation.
- Section 6 describes the fault tree models used to represent the failures of systems used for accident mitigation.
- Section 7 presents the HRA performed for the shutdown PRA.
- Section 8 provides the data analysis. This includes the approaches used for equipment failure rates, testing and maintenance unavailabilities, initiating event frequencies, and other basic events introduced for shutdown modeling.
- Section 9 describes the model quantification and results.
- Section 10 describes the uncertainty analysis.
- Section 11 summarizes conclusions and insights from the LPSD PRA.
- Section 12 contains a list of references.
- The report appendices include a list of candidates for future work (12APPENDIX A), documentation of cut sets and importance measure results (12APPENDIX B), identified model uncertainties (12APPENDIX C), and a review of LPSD operating experience (12APPENDIX D).

2 PLANT OPERATING STATES DEFINITIONS

2.1 Introduction

Although the plant is shutdown, the reactor is subcritical, the primary system depressurized, and the plant is being cooled by the decay heat removal system, accidents during specific plant configurations can lead to fuel damage. Due to the uncertainties of scheduling from outage to outage and the variability of plant configurations that can be encountered in outage schedules, the results from a shutdown PRA can vary considerably, depending on out-of-service equipment and plant operating states. Thus, the characterization of the plant outage type and the plant operating state must be sufficient to understand the accident analysis under consideration, along with the appropriate plant response.

It should be noted that the LPSD PRA standard (ASME/ANS, 2014) has been under development since the late 1990s and represents an ambitious scope of PRA elements. Only recently has the industry piloted a peer review of the draft standard, documented in the 2015 report, *EPRI Low Power and Shutdown Probabilistic Risk Assessment Standard Pilot: Palo Verde Self-Assessment* (EPRI, 2015). This effort identified 21 recommendations to improve the LPSD PRA standard. Among them, the industry self-assessment team found that “the outage procedures provided a more inclusive basis for identifying plant-operating states than did the concept of plant evolutions as provided in the LPSD PRA standard.” Although the plant operating state development described below is primarily based on the concept of plant evolutions as provided in the LPSD PRA standard, reference plant procedures have been utilized to the extent practicable.

2.2 Plant Technical Specification Modes

2.2.1 Mode Definitions

A plant mode corresponds to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 2-1 with fuel in the reactor vessel.

The plant modes are defined in the Standard Technical Specifications for Westinghouse plants, as documented in NUREG-1431 (NRC, 2012a).

2.2.2 High Level Equipment Availability by Mode

Some of the major equipment or functions based on the front-line systems modeled in the at-power event trees of the L3PRA project (NRC, 2022) may be out of service during some plant modes, by virtue of the mode (not considering already inoperable components causing technical specification-directed shutdowns). Such equipment and functions must be identified, at least at the mode level, so that they can be used in definition and selection of plant operating states later.

Table 2-1 Plant Technical Specification Modes

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER ^(a)	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown ^(b)	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown ^(b)	< 0.99	NA	≤ 200
6	Refueling ^(c)	NA	NA	NA
(a) Excluding decay heat. (b) All reactor vessel head closure bolts fully tensioned. (c) One or more reactor vessel head closure bolts less than fully tensioned.				

The need for adequate decay heat removal is essential for reactor safety while the reactor is shutdown. The following operational requirements assist in maintaining the decay heat removal function while the reactor is shutdown.

- In Modes 1, 2, and 3, the RHR is required to have an operable RHR pump and heat exchanger in each train, per Technical Specification limiting condition of operation (LCO) 3.5.2.
- In Mode 4, at least one train of RHR is required to be operable, per Technical Specification LCO 3.4.6 and LCO 3.5.3.
- In Mode 5, the RHR is required to be operable and in operation, per Technical Specification LCO 3.4.7 and LCO 3.4.8.
- In Mode 6, the RHR is required to be operable and in operation, per Technical Specification LCO 3.9.5 and LCO 3.9.6.
- In Mode 6 only, minimum RHR flow of ≥ 3000 gpm is required.
- In Mode 6 only, one train of RHR may be used as an OPERABLE boron injection flow path provided a flow path is established from the OPERABLE refueling water storage tank (RWST) via an RHR pump through the cold legs with water level ≥ 23 feet above the reactor vessel flange. The RHR pump may not be the pump that is being applied to meet LCO 3.9.5.

2.3 Plant Outage Types

The identification of outage types has a significant influence on the plant operating states. The number of shutdowns per year, specific plant operating configurations, and the resulting

conditional system and component availabilities all affect the plant's exposure to potential low power and shutdown initiating events. Therefore, it is important to explicitly define the outage types to be included in the low power and shutdown PRA. The definition of outage types used here follows typical PWR plant design and operating experience gained through a review of existing studies and references (ASME/ANS, 2014), (NRC, 1994a), and (NUMARC, 1991).

The identified outage types include those outages that are expected to be routinely entered as part of normal plant operations. Extended outages due to significant event occurrences, major equipment issues, or regulatory actions are excluded from the scope of the analysis. For example, after experiencing a core damage event at a reactor unit, other units may be ordered to shutdown to assess their own vulnerability to the event. Such outages would require an analysis of the unique conditions leading to the extended shutdown.

2.3.1 Plant Outage Type Definitions

For the Level 3 PRA low power and shutdown model, the plant outage types are defined as follows:

POT-1: Non-Drained Maintenance without the Use of RHR

POT-1 corresponds to forced outage maintenance in hot standby (i.e., reactor coolant temperature above 350°F). The reactor is subcritical (control rods fully inserted), the boron concentration is above the required concentration for hot shutdown, the RCS is maintained hot, the reactor pressure vessel head remains closed, and decay heat is removed by the use of feedwater and steam generators dumping steam either to the condensers via turbine bypass valves or through the atmospheric dump valves. This plant state will be reached if short outage times are expected, such as an inadvertent reactor trip, or if there are maintenance activities to be performed that do not require RCS cooldown.

POT-2: Non-Drained Maintenance with the Use of RHR

POT-2 corresponds to maintenance while in hot shutdown or cold shutdown without the need to drain the RCS. The reactor is subcritical (control rods fully inserted), the boron concentration is above the required concentration for cold shutdown, the RCS is cooled to below 350°F, the reactor pressure vessel head remains closed, and decay heat is removed by use of the residual heat removal system aligned in the shutdown cooling mode. This plant state will be reached if longer outage times are expected, if there is a loss of the main heat sink, or if there are maintenance activities to be performed that require cooldown of the RCS.

POT-3: Drained Maintenance with the Use of RHR

POT-3 corresponds to reduced inventory operation with RCS level just below the flange and with fuel in the reactor pressure vessel (RPV). This state is entered if maintenance requires low RCS level without unloading the core. The reactor coolant system is drained to near the reactor vessel head flange level with the reactor vessel head in place. The RCS level may be further lowered to at or near the mid-loop level for certain types of maintenance activities. Decay heat is removed by the residual heat removal system aligned in the shutdown cooling mode.

POT-4: Refueling

POT-4 corresponds to a refueling outage. The first activities are similar to those for POT-3 up to the point of draining to below the flange. Starting with RCS level below the RPV flange, the reactor basin/fuel transfer canal is filled for transfer of the fuel assemblies to the spent fuel pool (SFP). During refueling outages, many maintenance activities and surveillance tests are performed. Decay heat is removed by the RHR system and/or SFP cooling (SFPC) system, depending on the location of the spent fuel assemblies. Refueling outages are expected to be the longest duration and include the most extensive level of maintenance and testing activities in comparison with the outage types. Refueling outages also will typically involve large numbers of contract workers or temporary employees at the site to perform specialized activities. Advanced planning, scheduling and work controls are also expected to be more extensive than other outage types.

Other outages not covered by these four types, such as for repair or replacement of components like steam generators or main steam lines, are not planned to occur during the designed life of the plant. The risk of such events is best evaluated when such a repair or replacement is to be performed. At that juncture, it is likely that these four plant outage types can still represent the outages planned for such activities after appropriately adjusting the POS durations, especially if the core is completely offloaded to the SFP prior to beginning such unique maintenance.

2.3.2 Plant Outage Frequencies and Durations

Characterizing the frequency and duration of each outage type is important for understanding the exposure to different plant operating configurations. The plant-specific operating experience for the reference plant was examined. The results of that examination formed the basis of the POT frequency and duration estimates. A number of different information sources were used to identify and characterize the types of outages that have occurred at the reference plant. These information sources include:

- NRC Reactor Operating Experience Database (NROD) maintained by the Idaho National Laboratory (INL, 2013)
- Draft reports on development of a shutdown PRA model for the reference plant, including a plant trip log
- NRC Licensee Event Reports
- NRC Inspection Reports
- NRC Power Reactor Status Reports

The analyzed data covers all outages that occurred from January 1990 through December 2014. The year 1990 was selected as the starting point for the data period to exclude the early operating history of both units at the reference plant site. There is limited operating history prior to 1990, and the plant unavailability is expected to be elevated during the early part of the plant's operating history and not representative of later operating experience. The POT frequencies and average durations for the years 1990 through 2014 are shown in Table 2-2. The data in Table 2-2 cover both Unit 1 and Unit 2 operating experience. The period spans

25 calendar years for each unit, and a total of 50 calendar years are used to estimate the outage frequency at the combined site. The hours reported in the table identify the outage times measured from the reactor trip breakers opening until closing the generator output breaker during startup. The outage durations are taken from the reference plant trip log. The trip log covers all trip events up to October 2006 for Unit 1 and through May 2007 for Unit 2. For later outages, the durations were estimated from the NROD data. The POT frequency and duration estimates only include events that resulted in a reactor trip or manual shutdown. Power reductions, taking the main generator offline, or other activities that did not involve shutting down the reactor are excluded from this analysis.

Outage procedures and practices have evolved since 1990. Discussions with the reference plant staff indicate that refueling outage practices changed around 2009 in an effort to reduce the durations of refueling outages. Since this time the reference plant has routinely entered a mid-loop condition prior to defueling the reactor (i.e., hot mid-loop) and average outage durations have decreased. Table 2-2 includes the POT frequencies and average durations for the years 2009 through 2014. The data period covers 12 calendar years (six years for each unit). The data from 2009 to 2014 are most applicable to this project. Older data back to 1990 are included in the table to show the past experience with POT-2 and POT-3 outages. There has been limited experience with these outage types in recent years.

Table 2-2 Plant Outage Frequencies and Average Durations

		POT-1	POT-2	POT-3	POT-4
		No-drain w/o RHR	No-Drain w/RHR	Drained	Refueling
Years 1990 - 2014	# of Shutdowns	64	6	15 ⁴	34
	Ave. Hours per event (Rx Trip to Gen Breaker Closure)	53.1	270.7	213.4	904.9
	Frequency/Calendar Hours	1.46E-04	1.37E-05	3.42E-05	7.76E-05
	Frequency/Calendar Years	1.3	0.12	0.30	0.68
Years 2009 - 2014	# of Shutdowns	13	0	1	8
	Ave. Hours per event (Rx Trip to Gen Breaker Closure)	60.9	0	192	684
	Frequency/Calendar Hours	1.24E-04	0.00E+00	9.51E-06	7.61E-05
	Frequency/Calendar Year	1.08	0.00	0.08	0.67

⁴ There are seemingly a relatively large number of drained outages performed by the reference plant during this period. The number and types of outages were self-identified by the reference plant. Most of the outages involved RCS leakage with repairs to valves or weld locations. Such outages are not uncommon at other plants. However, a complete accounting of industry-wide outages is not available, and it is unclear if the number of drained outages experienced at the reference plant is an outlier from industry experience.

2.4 Plant Operating State Development

The use of plant operating states (POSs) to categorize the many different states of the plant during an outage provides for discrete sets of boundary conditions affecting the LPSD PRA.

The identification of POSs is a two-step process. First, the plant outage types are defined as previously discussed, and then the unique plant operating states within each modeled outage type are identified and characterized. The second step is accomplished by examining the various configurations that may be encountered during a unit outage, identifying the reactor and plant conditions associated with these low power and outage system configurations, and then defining a set of unique POSs (including transitions) that includes the entire range of LPSD evolutions. The resulting POSs form the set of initial conditions for the model event sequences.

A POS is a standard arrangement of the plant during which the plant conditions are relatively constant, are modeled as constant, and are distinct from other configurations in ways that impact risk. POSs are a basic modeling device used for a phased-mission risk assessment that discretizes the plant conditions for specific phases of a LPSD evolution. Examples of such plant conditions include: core decay heat level, primary water level, primary temperature, primary vent status, containment status, and decay heat removal mechanisms. Examples of risk impacts that are dependent on POS definition include the selection of initiating events, initiating event frequencies, definition of accident sequences, success criteria, and accident sequence quantification (ASME/ANS, 2014).

The plant conditions considered for use in characterizing POSs included:

- Reactor criticality or shutdown margin
- Decay heat level
- Technical Specification mode
- RCS temperature
- RCS pressure
- RCS water level
- RCS vent status (open or closed)
- Status of RCS loops (e.g., isolated by nozzle dam or isolation valves)
- Location of the fuel (reactor vessel versus SFP)
- Availability of safety and support systems (e.g., mechanism for removing decay heat)
- System alignments
- Status of the containment

The initial list of POSs considered is shown in Table 2-3. This list is based on reference (ASME/ANS, 2014) and information obtained from the reference plant.

Table 2-3 Candidate Plant Operating States

POS		TS	RCS			POS applicable when transitioning to outage type			
No.	Description	TS Mode	Power	T _{avg} (°F)	Boundary (Vent Status)	Refueling (POT-4)	Maintenance w/o Drain, w/o RHR (POT-1)	Maintenance w/o Drain (POT-2)	Maintenance w/Drain (POT-3)
0	Full power operation	1	100%	Normal Operations Temp (NOT) at 100%	RPV Head Intact	X	X	X	X
1	Low power and reactor shutdown	1,2	>0%	< NOT at 100%	RPV Head Intact	X	X	X	X
2	Cooldown with steam generators to 350°F	3	0%	350<T _{avg} <NOT at 100%	RPV Head Intact	X	X	X	X
3	Cooldown with residual heat removal system to 200°F	4	0%	200<T _{avg} <350	RPV Head Intact	X		X	X
4	Cooldown to ambient temperature with residual heat removal system only	5	0%	T _{avg} <200	RPV Head Intact	X		X	X
5	Draining the reactor coolant system to mid-loop	5,6	0%	T _{avg} <100	RCS Vented*, RPV Head Intact	X			X
6	Mid-loop operation	5,6	0%	T _{avg} <100	RCS Vented*, RPV Head Intact, PORV may be open	X			X
7	Filling refueling cavity for refueling operation	6	0%	T _{avg} <100	RCS Vented*, RPV Head de-tensioned	X			
8	Refueling operation (OLD CORE)	6	0%	T _{avg} <100	RCS Vented*, RPV Head removed	X			
	DEFUELED					X			
8	Refueling operation (NEW CORE)	6	0%	T _{avg} <100	RCS Vented*, RPV Head removed	X			
9	Draining the reactor coolant system to mid-loop after refueling operation	6	0%	T _{avg} <100	RCS Vented*, RPV Head removed	X			
10	Mid-loop operation after refueling	5,6	0%	T _{avg} <100	RCS Vented*, RPV Head tensioned	X			
11	Refill reactor coolant system	5,6	0%	T _{avg} <100	RPV Head Intact	X			X
12	Reactor coolant system heatup/draw bubble in pressurizer	5	0%	200<T _{avg}	RPV Head Intact	X		X	X
13	Reactor coolant system heatup to 350°F	4	0%	200<T _{avg} <350	RPV Head Intact	X		X	X
14	Startup with steam generators (AFW) to Hot Standby	3	0%	350<T _{avg} <NOT at 100%	RPV Head Intact	X		X	X

Table 2-3 Candidate Plant Operating States (cont.)

POS		TS	RCS			POS applicable when transitioning to outage type			
No	Description	TS Mode	Power	Tavg (°F)	Boundary (Vent Status)	Refueling (POT-4)	Maintenance w/o Drain, w/o RHR (POT-1)	Maintenance w/o Drain (POT-2)	Maintenance w/Drain (POT-3)
15	Reactor startup and low power operation	1,2	>0%	< NOT at 100%	RPV Head Intact	X	X	X	X
*RCS may be vented by removal of pressurizer manways or safety valves. Purge valves are not considered vent paths.									

2.5 Plant Operating State Selection

The candidate list of POSs in Table 2-3 was reviewed against reference plant-specific information to develop the projects final list of POSs, which is shown in Table 2-4 below. A description of each of the POSs is provided following the table.

2.5.1 POS 0: Full Power Operation

This POS is modeled by the full-power PRA. The plant is assumed at 100% power operating with main feedwater (MFW) used for cooling and the turbine rolling. The plant is in technical specification Mode 1.

Power descent begins at the start of a planned outage. During descent to 50% power, no other significant changes occur. Therefore, this period of the outage is grouped with POS 0 and evaluated using the models for full power operation. Only those plant evolutions eventually resulting in plant shutdowns are included in the low power shutdown evaluation.

2.5.2 POS 1: Low Power and Reactor Shutdown

The next period of outage covers an approximately 6-hour period as the power is reduced from 50% power to 20% power. During this period, one MFW pump is removed from service. The second MFW pump continues to provide cooling in manual control and the turbine continues to roll. The auxiliary feedwater (AFW) motor-driven pumps (MDPs) are aligned to mini-flow per reference plant system procedures, but do not actually inject to the steam generators. Non-1E 13.8kV and 4.16kV buses are also transferred from the unit auxiliary transformers (UATs) to the reserve auxiliary transformers (RATs) during this period. The plant is still in technical specification Mode 1. The conditions for POS 1 are very similar to that of POS 0. The only significant distinction is the reduction to just one running MFW pump.

A summary of the key equipment status and activities for POS 1 is provided in Table 2-5.

Table 2-4 Final Plant Operating States

POS		TS	RCS			POS applicable when transitioning to outage type				Scope of Guiding Procedures
No.	Description	TS Mode	Power	T _{avg} (°F)	Boundary (Vent Status)	Refueling (POT-4)	Maintenance w/o Drain, w/o RHR (POT-1)	Maintenance w/o Drain (POT-2)	Maintenance w/Drain (POT-3)	
0	Full power operation	1	50% to 100%	Normal Operations Temp (NOT) at 100%	RPV Head Intact	X	X	X	X	Power operation
1	Low power and reactor shutdown	1,2	<50% to 20%	T _{avg} < NOT at 100%	RPV Head Intact	X	X	X	X	Power operation, AFW system
2	Cooldown with steam generators to 350°F	3	Reactor tripped to 0%	350 < T _{avg} < NOT at 100%	RPV Head Intact	X	X	X	X	Power operation, Cooldown to cold shutdown
3	Cooldown with residual heat removal system to 200°F	4	0%	200 < T _{avg} < 350	RPV Head Intact	X		X	X	Cooldown to cold shutdown
4	Cooldown to ambient temperature with residual heat removal system only	5	0%	175 < T _{avg} < 200	RPV Head Intact	X		X	X	Cooldown to cold shutdown
5A	Pressurizer water solid for degassing	5	0%	T _{avg} = 175	RPV Head Intact	X			X	Cooldown to cold shutdown
5B	Draining the reactor coolant system to reduced inventory	5	0%	~110 < T _{avg} < 175	RCS Vented ¹ , RPV Head Vent opened	X			X	Cooldown to cold shutdown
6	Mid-loop operation ²	5,6	0%	71 < T _{avg} < 130	RCS Vented ¹ , RPV Head in place, de-tensioning begins	X			X	Refueling operations, Mid-loop operations
7	Filling refueling cavity for refueling operation	6	0%	71 < T _{avg} < 130	RCS Vented ¹ , RPV Head de-tensioned and removed	X				Refueling operations, RHR system
8E	Refueling operation (OLD CORE)	6	0%	71 < T _{avg} < 130	RCS Vented ¹ , RPV Head removed	X				Refueling operations

Table 2-4 Final Plant Operating States (cont.)

POS		TS	RCS			POS applicable when transitioning to outage type				Scope of Guiding Procedures
No.	Description	TS Mode	Power	T _{avg} (°F)	Boundary (Vent Status)	Refueling (POT-4)	Maintenance w/o Drain, w/o RHR (POT-1)	Maintenance w/o Drain (POT-2)	Maintenance w/Drain (POT-3)	
DF	DEFUELED	n/a	0%	$71 < T_{avg} < 130$	RCS Vented ¹ , RPV Head removed	X				Refueling operations
8L	Refueling operation (NEW CORE)	6	0%	$71 < T_{avg} < 130$	RCS Vented ¹ , RPV Head removed	X				Refueling operations
9	Draining the reactor coolant system after refueling operation	5, 6	0%	$71 < T_{avg} < 130$	RCS Vented ¹ , RPV Head installed and tensioned	X				Post refueling operations
10	Mid-loop operation after refueling ²	5	0%	$71 < T_{avg} < 130$	RCS Vented ¹ , RPV Head Intact	X				Mid-loop operations
11	Refill reactor coolant system	5	0%	$T_{avg} \sim 125$	RPV Head Intact	X			X	RCS refill
12	Reactor coolant system heatup/draw bubble in pressurizer	5	0%	$\sim 125 < T_{avg} < 175$	RPV Head Intact	X		X	X	RCS refill
13	Reactor coolant system heatup to 350 °F	4	0%	$175 < T_{avg} < 350$	RPV Head Intact	X		X	X	Heatup to hot shutdown, RCS refill
14	Startup with steam generators (AFW) to Hot Standby	3	0%	$350 < T_{avg} < 557$	RPV Head Intact	X		X	X	Heatup to hot shutdown
15A	Reactor startup and low power operation	1,2	0% to <5%	< NOT at 100%	RPV Head Intact	X	X	X	X	Reactor startup
15B	Reactor startup and low power operation	1,2	5% to <50%	< NOT at 100%	RPV Head Intact	X	X	X	X	Power operation

¹ RCS may be vented by removal of pressurizer manways or safety valves. Purge valves are not considered vent paths.

² POSs 6 and 10 contain the duration independent initiating event for overdraining the RCS.

Table 2-5 POS 1 Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
Electrical Power	Transfer offsite power to RATs
Steam Generator Cooling	One MFW pump is removed from service. AFW available if demanded.

2.5.3 POS 2: Cooldown with Steam Generators to 350 °F

This POS begins with the trip of the turbine and subsequent reactor trip from 20% power. The plant is then transitioned directly to Mode 3. The remaining operating MFW pump is tripped about an hour later. All but one condensate pump is also stopped. Steam generator (SG) cooling is then provided by the two AFW MDPs. The AFW actuation signal on MFW pump trip is subsequently defeated. Reactor coolant pumps (RCPs) 2 and 3 are removed from service before beginning RCS cooldown to RHR entry conditions. During the cooldown, the power operated relief valve (PORV) block valves are verified open and the cold over pressure protection system (COPS) is enabled. The safety injection (SI) accumulators are also blocked. The SI signals on low pressurizer pressure and main steam line pressure are blocked. The SI pumps are also put in pull-to-lock (PTL). These equipment status changes are assumed to apply for the entire period of this POS. Note that many of these equipment changes might not be performed in POS 2 for a POT1 outage, since the plant will not be entering RHR cooling conditions.

At the end of this POS a transition to Mode 4 (<350 F) occurs. Just prior to this, the AFW actuation signal on low-low level is defeated and the AFW MDP cross-connect between the A and B trains is opened to permit one AFW MDP to feed all four steam generators.

A summary of the key equipment status and activities for POS 2 is provided in Table 2-6.

Table 2-6 POS 2 Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
Steam Generator Cooling	MFW pump is secured. RCPs 2 and 3 are secured. AFW MDPs provide steam generator level.
Safety Injection	Accumulators are blocked. Some SI actuation signals inhibited. SI pumps placed in PTL.
Cold Overpressure Protection System	COPS is enabled with PORVs, at setpoints specified in the plant's pressure temperature limits report (PTLR).

2.5.4 POS 3: Cooldown with RHR System to 200°F

The residual heat removal (RHR) cooling will be started when RCS pressure is less than 365 psig and RCS temperature drops below 340 °F. It is assumed that the first train of RHR is started with RCS temperature still at 340 °F. One AFW MDP is secured, and the other AFW MDP is cooling all four steam generators. The running AFW MDP and one RHR train (either the A or B train) are then operated in parallel. The personnel airlocks are bypassed, but not yet

opened. This Mode 4 POS only lasts about 3 hours. RHR letdown is placed in service while normal charging continues.

At the end of POS 3, the second RHR pump is started and containment entry authorized according to reference plant procedures. POS 3 is over when the second RHR pump is started (i.e., just before entering Mode 5, when RCS temperature drops below 200 °F).

A summary of the key equipment status and activities for POS 3 is provided in Table 2-7.

Table 2-7 POS 3 Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
Steam Generator Cooling	One AFW MDP feeding all four steam generators.
RHR Cooling	RHR train B placed in service for shutdown cooling. RHR train A hot leg suction isolation valves closed. Train cross-connect valves closed.
Cold Overpressure Protection System	COPS is enabled with PORVs at setpoints as specified in PTLR and with RHR train B suction relief valve.

2.5.5 POS 4: Cooldown to Ambient Temperature with RHR System Only

POS 4 continues the RCS cooldown to 180°F using two RHR loops. The plant is in Mode 5. All AFW pumps and the remaining condensate pump are stopped, and the containment spray pumps are disabled. The equipment hatch is opened early in this POS and containment purge is placed in service. Also, the main steam isolation valves (MSIVs) are closed, and condenser vacuum is broken. Therefore, the condenser steam dumps are no longer operable. The solid state protection system (SSPS) is placed in test. The containment coolers are shutdown. Maintenance and testing on the unprotected⁵ train of 120V AC vital instrument power and DC power can occur after entering Mode 5 and prior to entering mid-loop. This includes maintenance on the safety features sequencer system. Maintenance on the sequencer makes automatic actuation of the related train emergency diesel generator unavailable. This condition is exited prior to entering mid-loop. Maintenance and testing on the opposite train 120V AC vital instrument power and DC power is performed during POS 8L, after swapping the protected train.

RCS level remains in the pressurizer throughout this approximately 6-hour POS. At the end, RCS pressure is still at 350 psig and two RCPs are running.

A summary of the key equipment status and activities for POS 4 is provided in Table 2-8.

⁵ As used here, “unprotected” means the train can be removed from service for test or maintenance; “protected” means administrative controls are in place to prevent the train from being removed for test or maintenance.

Table 2-8 POS 4 Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
AC Power	Testing and maintenance on train A 120V AC power and distribution panel is unavailable. Primary activity involves A sequencer testing and maintenance. EDG A automatic actuation is unavailable, but action to restore EDGs is credited if both EDGs are failed or unavailable.
DC Power	Testing and maintenance on train A 125V DC power and DC bus is unavailable. Activities can include battery testing/maintenance and battery charger maintenance.
Steam Generator Cooling	AFW is stopped. Main condenser is unavailable. Steam generator level is held at 60-70% of normal range. Steam generators available for heat removal through atmospheric relief valves, if needed.
RHR Cooling	Both RHR trains placed in service for shutdown cooling.
Safety Injection	SSPS placed in test mode. Operator action required to establish RCS injection.
Cold Overpressure Protection System	COPS is enabled with PORVs at setpoints as specified in PTLR and with both RHR suction relief valves.
Containment Status	Containment equipment hatch and personnel airlock are open.
Containment Sump	Protective covers for sump screens are installed, causing sump to be unavailable during all of Mode 5 and Mode 6 operation.

2.5.6 POS 5A: Pressurizer Water Solid for Degassing

Two RHR pumps are running throughout this mode 5 POS. This POS is a portion of POS 5 as defined in Table 2-3. It is separated from the remainder of the Table 2-3 POS 5 definition because of the water solid condition with the RCS closed and the potential for a rapid pressure transient causing pressure relief valves to lift. POS 5A begins when the operators take the pressurizer solid while at 180°F by adjusting charging and RHR letdown to slowly raise pressurizer level within a 2-hour period. The centrifugal charging pumps (CCPs) are placed in PTL and the pressurizer backup heaters are de-energized during this POS.

When RCS hydrogen concentration is determined to be acceptable, then pressurizer cooldown is begun while still water solid. Pressurizer spray flow is maximized for this purpose. RCS cooldown proceeds to less than 110°F. Both of the remaining RCPs that are operating are tripped before RCS pressure is reduced below 250 psig. POS 5A ends when the last RCP is tripped. Also, preparing steam generators for wet layup begins in POS 5A.

A summary of the key equipment status and activities for POS 5A is provided in Table 2-9.

Table 2-9 POS 5A Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
AC Power	Testing and maintenance on train A 120V AC power and distribution panel is unavailable. Primary activity involves A sequencer testing and maintenance. EDG A automatic actuation is unavailable, but action to restore EDGs is credited if both EDGs are failed or unavailable.
DC Power	Testing and maintenance on train A 125V DC power and DC bus is unavailable. Activities can include battery testing/maintenance and battery charger maintenance.
Steam Generator Cooling	AFW is stopped. Main condenser is unavailable. Steam generator level is held at 60-70% of normal range. Steam generators available for heat removal through atmospheric relief valves, if needed.
RHR Cooling	Both RHR trains placed in service for shutdown cooling.
Safety Injection	Centrifugal charging pumps are placed in PTL to avoid inadvertent injection.
Cold Overpressure Protection System	COPS is enabled with PORVs at setpoints as specified in PTLR and with both RHR suction relief valves.
Containment Status	Containment equipment hatch and personnel airlock are open.

2.5.7 POS 5B: Draining the RCS to below Reactor Vessel Head Flange

This Mode 5 POS begins when the last RCP is tripped. Both RHR trains remain operating throughout. The RCS is subsequently fully depressurized. RCP seal return is isolated before RCS pressure is less than 150 psig. The centrifugal charging pump control switches are in PTL position, and the pumps are available for operators to start, if required. RCS loop drains on each RCS loop are opened at 100 psig to drain RCS level to 80% in the pressurizer. RHR letdown is then used to depressurize the RCS to atmospheric pressure. The pressurizer head vent is then opened to help the subsequent drain down. The three pressurizer safety valves are removed with pressurizer level still at 80% (in rare cases, the pressurizer manway will be removed for venting, but for this project, it is always assumed to be the three PSVs). Therefore, this POS is assumed to be vented throughout. RHR letdown is used to drain the RCS to just below the reactor flange for vessel disassembly. The reactor vessel head vent is also opened once level is below top of the reactor vessel head. The RCS drainage path is via RHR train B letdown to the recycle holdup tank (RHUT). Additional level indications are made available during this POS. About two-thirds of the POS duration is spent with RCS level below the reactor vessel head, with the remainder of the duration involving the drain down from water solid conditions.

During POS 5B, all four steam generators are filled to 95% on the wide range and held there for a minimum of six hours. Steam generator cooling is not available because the RCS is fully vented.

POS 5B differs from the Table 2-3 definition of POS 5 in that the water solid condition has been separated out as POS 5A. Another key difference is that RCS draining in POS 5B stops just below flange level, rather than proceeding to mid-loop.

A summary of the key equipment status and activities for POS 5B is provided in Table 2-10.

Table 2-10 POS 5B Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
AC Power	Testing and maintenance on train A 120V AC power is completed. EDG A automatic actuation is restored as part of standard contingency plan prior to draining below 15% pressurizer level.
DC Power	Testing and maintenance on train A 125V DC power is completed.
Steam Generator Cooling	RCS is depressurized and vented. Steam generators are no longer available for heat removal.
RHR Cooling	Both RHR trains placed in service for shutdown cooling.
Safety Injection	Centrifugal charging pumps are restored to standby condition.
Cold Overpressure Protection System	Three pressurizer safety valves are removed. COPS is no longer required.
Containment Status	Containment equipment hatch and personnel airlock are open.

2.5.8 POS 6 – Early Mid-loop Operation

Though not employed during earlier refueling outages, the current typical refueling outage does include an early mid-loop period. Interviews with the reference plant staff indicated that around year 2009 the plant underwent an effort to improve consistency from outage to outage and reduce the overall time spent with the reactors shutdown. Since that time the plant has routinely performed an early mid-loop evolution during refueling outages. Mid-loop operation allows installation of SG nozzle dams. The nozzle dams (i.e., 22 to 24-inch pipe plugs) provide a watertight seal between the RPV and the bottom of the SGs on the primary side. Once installed, service air with nitrogen backup is used to keep them in place. Leak and low pressure alarms are provided for problem detection. The nozzle dams remain in service until POS 10, where they are removed just prior to establishing RCS vacuum.

Once the nozzle dams are installed maintenance activities on the reactor coolant pumps and steam generators (e.g., tube inspections) can be performed with higher RCS water levels.

The RCS level would first be held just below the reactor vessel head flange in order to begin the head removal. Head removal preparation is begun, but not completed, before draining to mid-loop per reference plant procedures. By technical specification definition, Mode 5 ends and Mode 6 begins with the de-tensioning of the first reactor vessel stud.

POS 6 definition includes draining the RCS from just below the reactor vessel head to mid-loop in order to install SG nozzle dams. The top of the RHR suction nozzle where it connects to the hot leg is approximately five inches below the centerline of the hot leg. To prevent RHR pump cavitation, water level is maintained slightly above these elevations during mid-loop operation, at one foot above the hot leg centerline. At the beginning of the POS, RHR heat removal is provided by both RHR trains. Prior to draining the RCS more than 2 feet below the reactor head vessel flange, the preferred RHR configuration is one RHR loop operating and one RHR pump in standby and aligned for shutdown cooling. This configuration prevents possible damage to both pumps if cavitation were to occur.

The containment airlocks are closed, and the equipment hatch is closed using four bolts.

Other key differences with late mid-loop conditions while at vacuum, described later for POS 10, are:

- RCS temperatures are slightly higher,
- decay heat levels are higher,
- the SI pumps are available if needed, though their discharge motor-operated valves (MOVs) are isolated, and
- the RPV head is removed.

POS 6 ends when the operators begin raising RCS level from mid-loop back up to just below the reactor vessel flange. This POS is expected to take 14 hours.

A similar POS must also be considered for POT-3 type maintenance outages requiring drain down.

This POS contains the time (duration) independent initiator for overdraining the RCS leading to a loss of shutdown cooling.

A summary of the key equipment status and activities for POS 6 is provided in Table 2-11.

Table 2-11 POS 6 Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
RHR Cooling	One RHR train in service for shutdown cooling. One RHR train in standby and aligned for shutdown cooling.
Safety Injection	Centrifugal charging pumps are in standby. SI pumps are available, but discharge MOVs are isolated.
Containment Status	Containment equipment hatch and personnel airlock are closed.

2.5.9 POS 7: Filling Reactor Cavity for Refueling Operation

POS 7 begins with RCS level increasing from mid-loop in POS 6. RCS level is raised to just below the reactor flange. Level increases are terminated to allow reactor head disassembly and removal.

The steam generators are also drained and subsequently refilled to 65% using wide range level indication. Since the RCS is fully vented, the unavailability of the steam generators during this period has no affect.

The first reactor vessel flange stud was de-tensioned in POS 6; de-tensioning is completed during this POS, according to reference plant procedures. The equipment hatch is closed (using four bolts) and the containment purge system is operating during reactor vessel head lift. Both RHR trains are operating throughout this POS. Both centrifugal charging pumps are placed in PTL. The transfer tube gate valve is open at the end of this POS.

The sump drain line valve is closed. The flanges are installed on the 12-inch refueling cavity drain line, and the reactor cavity is sealed in preparation for filling. The reactor vessel head is removed and filling of the refueling cavity begins.

Normal charging is used to fill the RCS until overflowing the reactor vessel flange. The SFPC system is used to fill the lower refueling cavity. Next, the non-cooling train of RHR is used to gravity fill from the RWST. The transfer tube gate valve is opened prior to the start of filling the lower reactor cavity. Both the refueling cavity and the transfer canal volumes are filled together.

The normal charging system is placed in standby. RHR letdown is also out of service. One of the two RHR trains is operating in shutdown cooling during this period, with the other in standby with its hot leg suction valve closed. The standby RHR train is used to gravity fill the refueling cavity and is assumed available for shutdown cooling if needed. Testing may require that each RHR train be shut down and run in turn. The two SI pumps are available with the discharge MOVs returned to the open position. One of the two centrifugal charging pumps may be in maintenance, but the other is available in PTL. During RCS filling, the SI accumulators are vented to the containment. Prior to this time, the accumulators were blocked but not depressurized.

As reactor cavity filling reaches the full target height, it is halted, and the upper internals are removed. During the removal, the equipment hatch is closed (with four bolts) and the containment purge system is shutdown. Once the upper internals are removed and filling is completed, the equipment hatch is again opened, and containment purge restored. The refueling canal transfer tube is open except when removing the upper internals.

This POS, which is estimated to last 45 hours, ends when refueling cavity level reaches its full height, and POS 8E, Refueling Operation, begins with the start of fuel offload.

A summary of the key equipment status and activities for POS 7 is provided in Table 2-12.

Table 2-12 POS 7 Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
RHR Cooling	RHR train B in service for shutdown cooling. RHR train A in standby with its hot leg suction valve closed.
Safety Injection	One of two centrifugal charging pumps and SI pumps are in PTL and available. One centrifugal charging pump maintenance may begin during refueling cavity flooding.
Containment Status	Containment equipment hatch and personnel airlock are closed during lifting of reactor vessel head.

2.5.10 POS 8E: Refueling Operation (Early or Old Core)

POS 8E (early or old core) begins at the start of fuel offload, with the refueling cavity flooded completely to a level of approximately 24 feet above the reactor vessel head flange. The entire core is offloaded to the SFP. During this period of approximately 68 hours, one RHR loop is running with RHR letdown isolated. The second RHR loop may be unavailable due to maintenance with its hot leg suction valve closed. Both SI pumps are available, but only one centrifugal charging pump is available, and the other may be unavailable due to maintenance. The equipment hatch is open, but closeable in a short time. The mini-purge system is operating.

One out of the two Unit 1 SFPC trains is running. The other may be, and often is, in maintenance. Both SFPC trains from Unit 2 must be operable.

This POS is similar to POS 8 (old Core) as defined in Table 2-3. A summary of the key equipment status and activities for POS 8E is provided in Table 2-13.

Table 2-13 POS 8E Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
AC Power	Testing and maintenance on train A AC power. Train B and associated train B ESF equipment are marked protected. Activities can include maintenance on train A RAT, EDG A, and safety-related systems. Maintenance on train A cooling water systems (e.g., nuclear service cooling water [NSCW] and component cooling water [CCW]) coincides with AC power system maintenance. Alternate AC power source is available.
RHR Cooling	RHR train B in service for shutdown cooling.
Safety Injection	One of two centrifugal charging pumps and one of two SI pumps are available for injection. ESFAS train A testing may occur during this POS.
Containment Status	Containment equipment hatch and personnel airlock are open.

2.5.11 POS DF: Defueled

This POS applies when the fuel has been offloaded to the SFP (i.e., outside containment). The fuel transfer gate is closed. With the fuel offloaded, the RCS is drained to mid-loop level, and the RHR trains are placed in standby. During this POS, the risks of potential radiological releases are addressed by the L3PRA SFP PRA. This POS is not modeled in the L3PRA Unit 1 reactor LPSD PRA.

Towards the end of this POS, the RCS and refueling cavity are refilled. Even with the reactor cavity flooded, there is still level available in the RWST. Note that the protected RHR trains are typically swapped in the middle of this POS. Test and vacuum fill equipment are installed and one of the RHR trains is then started in the shutdown cooling mode. The fuel transfer tube gate valve is then opened in preparation for fuel reload.

The POS definitions described in Table 2-3 do not apply to this configuration. Of interest for this project is cooling of the SFP for the offloaded core (as mentioned above, this is addressed in the SFP PRA). Therefore, a new POS identifier was created. One out of two Unit 1 SFPC trains is running during this POS. The other may be in maintenance or unavailable due to maintenance on its support systems. Both SFPC trains from Unit 2 must be operable.

2.5.12 POS 8L: Refueling Operation (late or new core)

POS 8L (late or new core) begins at the start of fuel reload with the refueling cavity flooded. The plant is in Mode 6 when the first fuel assembly is loaded. The entire core is loaded into the reactor from the SFP. Decay heat levels are substantially reduced in this POS compared to earlier POSs for two reasons. One reason for this reduction in decay heat is the time spent during the defueled POS. A second reason is the replacement of a fraction of the core with new fuel. It is assumed that 89 out of 193 fuel assemblies are replaced with new fuel.

During POS 8L, which lasts approximately 52 hours, either RHR train is running with normal charging and RHR letdown operating. The second RHR loop may be unavailable due to maintenance, but its hot leg suction valve is open. Both SI pumps are available but only one centrifugal charging pump is available, with the other possibly in maintenance. The equipment hatch is open, but closeable in a short time. The mini-purge system is operating. At the end of this POS, the spent fuel transfer tube is closed and then both RHR trains are protected (i.e., not out for maintenance).

One out of two SFPC trains are running, the other may be, and often is, in maintenance. Both SFPC trains from Unit 2 must be operable throughout this POS.

This POS is similar to POS 8 (new core) as defined in Table 2-4. A summary of the key equipment status and activities for POS 8L is provided in Table 2-14.

Table 2-14 POS 8L Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
AC Power	Testing and maintenance on train B AC power. Swap protected train to train A. Maintenance on train B cooling water systems is also performed at this time.
RHR Cooling	RHR train A in service for shutdown cooling.
Safety Injection	One of two centrifugal charging pumps and one of two SI pumps are available for injection. ESFAS B train testing occurs during this POS. ECCS flow testing on centrifugal charging pump B.
Containment Status	Containment equipment hatch and personnel airlock are open.

2.5.13 POS 9: Draining the RCS to Elevation Below Reactor Vessel Head Flange

This POS covers the activities during draining of the RCS to near the reactor vessel head flange level following the refueling operation. The RCS is first drained to reinstall the reactor internals. Then drainage continues to below the reactor vessel flange to restore the reactor vessel head. Most of the POS duration is with level at 2 feet below the reactor vessel head flange.

During this POS, either train of RHR is operating in shutdown cooling mode with the opposite train of RHR available and used for draining the RCS to the RWST until level is just above the reactor vessel head flange. RHR letdown to the RWST is used to complete the draining to 2 feet below the reactor vessel head flange. The reactor vessel head is then installed and tensioned. After the draining, each RHR train is tested in turn, which involves temporarily shutting down the tested train.

At the end of this POS, the reactor vessel head has been installed, either train of RHR is operating in shutdown cooling mode, and the opposite train is aligned for vacuum refill. Both RHR hot leg suction valves are opened. Normal charging and RHR letdown are in service. RCP seal injection is restored. The SI accumulators are refilled and pressurized, but not yet unblocked. Two of the three removed safety relief valves are reinstalled during this POS. The reactor vessel head vent and pressurizer head vent are aligned and opened. Both centrifugal charging pumps are available. Both SI pumps are isolated, but at least one pump is placed in a state ready to be restored by manual action from the control room. The AFW cross-connect valves are closed. The lower cavity drain covers are removed and the refueling cavity drain valve is installed. The equipment hatch is open, but closeable within a short period of time.

On the secondary side, the circulating water pumps are started and condensate aligned on long cycle. Vacuum is also established in the main condenser.

POS 9 for this project is defined to cover only the activities while RCS level is at or greater than 2 feet below the reactor vessel head flange. This is different from the POS description in Table 2-3. Instead, for this project, the draining to mid-loop and activities while at mid-loop are covered in the next POS.

A summary of the key equipment status and activities for POS 9 is provided in Table 2-15.

Table 2-15 POS 9 Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
RHR Cooling	RHR train A in service for shutdown cooling. RHR train B in standby and available.
Safety Injection	Two of two centrifugal charging pumps and one of two SI pumps are available for injection.
Cold Overpressure Protection System	After reactor vessel head is installed, COPS is restored to operable condition.
Containment Status	Containment equipment hatch and personnel airlock are open.

2.5.14 POS 10: Mid-Loop Operation after Refueling

POS 10 covers the activities when RCS level is less than 2 feet below the reactor vessel head flange. POS 10 begins when draining to mid-loop using RHR letdown. The RCS is drained to a mid-loop elevation to remove the nozzle dams and then install the manhole covers. Most of the POS duration is due to the time to install the manhole covers and pull the nozzle dams. After the nozzle dams are removed and the SG manways installed, the last pressurizer safety valve removed in POS 5B is reinstalled, closing the RCS. Vacuum is established after further draining to the hot leg centerline to clear the pressurizer surge line. The RCS is refilled to near the reactor vessel head using gravity fill via either RHR train. Both the reactor head vent and pressurizer head vent are aligned to the skid-mounted vacuum pump. Normal charging and RHR letdown are aligned. Both RHR hot leg suction valves are opened. COPS remain operable. Both SI pumps are isolated while the steam generator nozzle dams are still in place. Once the nozzle dams are removed and manways are in place, one SI pump is made available. Both centrifugal charging pumps are available throughout this POS.

The containment equipment hatch may remain open during POS 10 if certain conditions are established, as outlined in the reference plant's mid-loop operating procedures. The containment opening is continuously manned by plant personnel dedicated to the equipment hatch for quick closure within 25 minutes. Also, the equipment hatch opening is always kept clear of obstructions that would delay closure. However, a review of the reference plant's past outage experience shows a consistent practice of closing the containment equipment hatch during both hot and cold mid-loop operations (i.e., during both POS 6 and POS 10). The containment equipment hatch and personnel airlock are, therefore, assumed closed during POS 10.

Although there is a period of draindown and then refill at the end, most of this POS is expected to be at mid-loop conditions.

This POS contains the time (duration) independent initiator for overdraining the RCS leading to a loss of shutdown cooling.

A summary of the key equipment status and activities for POS 10 is provided in Table 2-16.

Table 2-16 POS 10 Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
RHR Cooling	RHR train B in service for shutdown cooling. RHR train A in standby and available.
Safety Injection	Two of two centrifugal charging pumps are available for injection. Both SI pump discharge MOVs are isolated.
Cold Overpressure Protection System	Pressurizer safety valves are reinstalled. COPS is operable.
Containment Status	Containment equipment hatch and personnel airlock are closed.

2.5.15 POS 11: Refill Reactor Coolant System (after Refueling)

POS 11 begins when the refill process proceeds above 2 feet below the reactor vessel head flange. RHR B train is running in the shutdown cooling mode. The RCS is vented (via the reactor head vent isolation valve and pressurizer vent valves) and temperatures are less than 125°F. Gravity feed (at ~900 gpm) is proceeding via the idle A train of RHR. Train A of RHR is in PTL. Both hot leg suction valves are open. The reactor head vent isolation valve is closed, and the reactor head vacuum hose is drained. Refill continues until 70% level in the pressurizer is reached. Normal charging and RHR letdown are operating throughout. The SI pumps are both put in PTL, and both centrifugal charging pumps are available. The equipment hatch is open but can be closed within a short time period. The plant is in Mode 5 throughout POS 11.

At the end of this approximately 20-hour POS, the RCS skid-mounted vacuum pump is isolated. The pressurizer head vent valves are closed.

Vacuum refill is used for refueling outages at the reference plant. This process avoids the need for RCP sweeps to remove dissolved gases when in water solid conditions. Note that for POT-3 (non-refueling drained outages), the activities performed during POS 11 are different. For a portion of the POS duration, the pressurizer is filled to water solid conditions and vented to remove dissolved gases. One RCP is running intermittently during this time.

A summary of the key equipment status and activities for POS 11 is provided in Table 2-17.

Table 2-17 POS 11 Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
RHR Cooling	RHR train B in service for shutdown cooling. RHR train A used to gravity fill RCS. Train cross-connect valves closed.
Safety Injection	Two of two centrifugal charging pumps and two of two SI pumps are available for injection.
Containment Status	Containment equipment hatch and personnel airlock are open.

2.5.16 POS 12: RCS Heatup/Draw Bubble in Pressurizer (not Solid for Refueling)

This POS begins by initiating RCS heatup from 125 °F to 175 °F. The pressurizer heaters are used to raise the RCS pressure from vacuum to 300 psig, and RHR heat exchanger cooling is throttled to raise average RCS temperature to 175°F. Train B of RHR is running with the other train in standby, but available. Both RHR hot leg suction valves are open. The RCPs remain off for this POS.

This heatup transition is estimated to take 50 hours. During this period, SSPS trains A and B are restored from test. At the conclusion of this POS, the equipment hatch is fully bolted closed and containment purge is shutdown. Containment spray pumps are made operable. The plant is in Mode 5 throughout this POS.

A summary of the key equipment status and activities for POS 12 is provided in Table 2-18.

Table 2-18 POS 4 Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
RHR Cooling	RHR train B in service for shutdown cooling. RHR train A in standby.
Safety Injection	Two of two centrifugal charging pumps and two of two SI pumps are available for injection.
Containment Status	Containment equipment hatch and personnel airlock are open.

2.5.17 POS 13: Continue RCS Heatup to 350 °F

In POS 13, the plant heatup continues to 350 °F, transitioning to Mode 4 above 200 °F. The start of POS 13 is taken to be when the first RCP is started. The other three RCPs are also started in this POS. Either train of RHR is aligned for shutdown cooling and the opposite train is in standby. The heatup is estimated to take 20 hours. Actions to establish containment closure are performed during this POS.

This POS ends when the second RHR train is shut down and isolated. Before isolating RHR, the AFW MDPs are started on mini-flow and the atmospheric relief valves (ARVs) are used to establish RCS temperature control. Alternatively, the main steam lines and condenser steam dumps may be used for temperature control, if the steps to return the main condenser to service have been completed. However, for this project, it is assumed the ARVs are in use for temperature control. The reactor trip breakers are closed.

This POS is similar to POS 13 as defined in Table 2-3. Since the MSIVs are assumed to be closed, only the ARVs are available, if needed, for steam generator cooling.

A summary of the key equipment status and activities for POS 13 is provided in Table 2-19.

Table 2-19 POS 13 Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
Steam Generator Cooling	AFW MDPs are started.
RHR Cooling	RHR train B in service for shutdown cooling. RHR train A in standby.
Safety Injection	Two of two centrifugal charging pumps and two of two SI pumps are available for injection.
Containment Status	Containment equipment hatch is closed and fully bolted. Personnel airlock is closed.
Reactor Protection System	Reactor trip breakers closed.

2.5.18 POS 14: Startup with Steam Generators (on AFW) to Hot Standby

POS 14 begins when the RHR trains are shut down and isolated from the RCS. The AFW system is aligned for steam generator cooling. Transition to plant Mode 3 occurs at greater than 350 °F. At 900 psig, the SI accumulators are aligned for service. Heatup continues to hot standby conditions (i.e., 2,235 psig and 557 °F). This POS is estimated to last 40 hours, ending at the time the operators initiate withdrawal of control rod banks, transitioning the plant to Mode 2.

At the beginning of this POS, COPS is disabled. PORVs are restored to their nominal lift setpoints. Also, at the end of this POS, normal pressurizer spray and heaters are aligned for automatic operation. Both SI pumps and centrifugal charging pumps are aligned for automatic operation and are operable.

This POS is similar to POS 14 as defined in Table 2-3. A summary of the key equipment status and activities for POS 14 is provided in Table 2-20.

Table 2-20 POS 14 Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
Steam Generator Cooling	AFW MDPs providing heat removal through ARVs.
RHR Cooling	RHR trains isolated.
Safety Injection	Both SI pumps and centrifugal charging pumps are aligned for automatic operation and are operable. SI accumulators restored.
Cold Overpressure Protection System	COPS is not enabled. PORVs set to their normal operating setpoints.

2.5.19 POS 15A: Reactor Startup and Low-Power Operation (AFW Running)

POS 15A represents reactor startup from 0% to approximately 5% power following refueling. Cooling is provided by the two AFW MDPs. This POS is initiated when the operators begin the withdrawal of the control rod banks. During this POS, the MSIVs are opened in preparation for transfer to MFW. Steam dumps to the main condenser then become operable. A second condensate pump is started during this POS. The signal to actuate AFW on trip of both MFW pumps is restored. A MFW pump is placed in service and cooling is transferred to MFW, one steam generator at a time. Alignment of one MFW pump signifies the end of POS 15A.

This POS represents only a portion of POS 15 as defined in Table 2-3. Initial startup with cooling provided by AFW distinguishes this period from later times when one MFW pump is operating.

A summary of the key equipment status and activities for POS 15A is provided in Table 2-21.

Table 2-21 POS 15A Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
Steam Generator Cooling	AFW MDPs providing heat removal through ARVs or steam dumps to main condenser. One MFW pump placed in service at end of POS.

2.5.20 POS 15B: Reactor Startup and Low-Power Operation (One MFW Pump Running)

POS 15B begins when the operators transfer from AFW to one MFW pump and condensate for steam generator cooling via the main feedwater regulator valves. Temperature control is provided by the steam dumps. The AFW pumps are stopped.

Reactor power is increased during this POS from 5% to approximately 40%. At 20% power, early on in this POS, the main turbine is rolled, and the main generator output breakers closed. At approximately 40% power, AMSAC⁶ unblocks. The non-1E 13.8kV and 4.16kV buses are swapped back to the UATs from the RATs at about 42% power. These transitions could impact plant operations and may contribute to increased likelihood of a transient initiating event. The estimated duration of this POS is 36 hours. This POS could have been subdivided to distinguish periods before and after turbine roll; however, it was decided to group these periods together and evaluate POS 15B as having the turbine rolling the entire period. The judgment was made that the initiator frequencies are higher while the turbine is rolling and so this grouping is conservative and reduces the number of POSs to be analyzed.

A summary of the key equipment status and activities for POS 15B is provided in Table 2-22.

⁶ ATWS mitigation system actuation circuitry

Table 2-22 POS 15B Key Equipment Status and Activities

Impacted System / Function	Equipment Status and Activities
Steam Generator Cooling	One MFW pump in operation. Main turbine is placed in operation.

2.5.21 POS 0: Low-Power to Full-Power Operation

The rise to full power follows the activities defined by POS 15B. The second MFW pump is started at approximately 50% reactor power. It is assumed that the rise from 50% to full power can be approximated by the conditions for full power. Once the second MFW pump is started, the plant configuration is the same as for full power. The estimated duration for the transition to full power is 80 hours.

2.6 Plant Operating State Duration

Evaluating the duration of POSs is an important aspect of low power and shutdown risk assessment. The POS duration will define how long the plant is exposed to potential accident scenarios in the given plant condition. The durations of POSs will vary for each outage due to a number of different factors, such as, the critical maintenance and testing activities that are scheduled during the outage, and emergent issues or complications that arise during the outage. This outage-to-outage variation is reflected in the reference plant's refueling outage operating experience. In the past ten years, the total duration of refueling outages at the reference plant has ranged from 24 to 49 days.

The outage-to-outage variation introduces an uncertainty in the POS durations. This uncertainty is not explicitly accounted for in the PRA results. An actual outage may be different than the outage modeled in this project.

The POS durations that are used for modeling purposes are shown in Table 2-23. The following assumptions are made regarding the representative POS durations used for the LPSD PRA:

- The POS durations are selected by NRC analysts and based on actual operating experience for refueling outages. The durations are selected to be similar to actual durations documented in the reference plant's refueling outage reports, but numerical averages of the durations are not used. One reason for not using numerical averages is that complete and precise POS durations are not available for all POSs during every refueling outage. In some cases, approximate durations of POSs were inferred based on the documented completion of critical path activities, and these activities do not necessarily align with the POS definitions. A second reason is that some POS durations may be outliers from the average duration, due to conditions specific to that outage. Therefore, an informed selection of POS durations was used to define the type of outage that is modeled. While it is acknowledged that POS durations can vary significantly from outage to outage, the selected durations are not intended to account for the full range of this variation.
- The total duration of a modeled refueling outage (POT-4) is 30 days. This duration is similar to the actual refueling outage durations experienced at the reference plant

between 2009 and 2014. During this time period, the average refueling outage duration was approximately 29 days.

- The total duration of POT-1 is assumed to be three days. Recent operating experience from the reference plant (2009-2014) shows the typical duration for this type of outage ranges from two to three days.
- The conditions leading to a POT-1 outage can vary. A POT-1 outage could be entered from an automatic reactor trip from full power, or an emergent or planned POT-1 outage could involve a controlled power decrease and shutdown. For modeling purposes, a reactor trip at 100% power for POT-1 outages is assumed. Thus, POS 1 is not applicable to POT-1 outages.
- The total durations of POT-2 and POT-3 are both assumed to be 10 days.

Table 2-23 Plant Operating State Durations

POS		TS	RCS			POS Duration (hours) ¹			
						Duration A	Duration B	Duration C	Duration D
No.	Description	TS Mode	Power	T _{avg} (°F)	Boundary (Vent Status)				
1	Low power and reactor shutdown	1,2	50 to 20 %	T _{avg} <NOT	RPV Head Intact	6	N/A	6	6
2	Cooldown with steam generators to 350°F	3	20 to 0%	350<T _{avg} <NOT	RPV Head Intact	5	16	5	5
3	Cooldown with residual heat removal system to 200°F	4	0%	200<T _{avg} <350	RPV Head Intact	3		3	3
4	Cooldown to ambient temperature with residual heat removal system only	5	0%	180<T _{avg} <200	RPV Head Intact	4		68	4
5A	Pressurizer water solid for degassing	5	0%	T _{avg} =180	RPV Head Intact	30			20
5B	Draining the reactor coolant system to reduced inventory	5,6	0%	180<T _{avg} <~125	RCS Vented, RPV Head Intact	40			20
6	Mid-loop operation	5,6	0%	T _{avg} ~125	RCS Vented, RPV Head Intact, PORV may be open	14			20
7	Filling refueling cavity for refueling operation	6	0%	T _{avg} ~125	RCS Vented, RPV Head de-tensioned	45			
8E	Refueling operation (OLD CORE)	6	0%	T _{avg} ~125	RCS Vented, RPV Head removed	68			

Table 2-23 Plant Operating State Durations (cont.)

POS		TS	RCS			POS Duration (hours) ¹			
						Duration A	Duration B	Duration C	Duration D
No.	Description	TS Mode	Power	T _{avg} (°F)	Boundary (Vent Status)				
DF	DEFUELED	n/a	0%	T _{avg} ~125	RCS Vented, RPV Head removed	120			
8L	Refueling operation (NEW CORE)	6	0%	T _{avg} ~125	RCS Vented, RPV Head removed	52			
9	Draining the reactor coolant system after refueling operation	6	0%	T _{avg} ~125	RCS Vented, RPV Head removed	120			
10	Mid-loop operation after refueling	5,6	0%	T _{avg} ~125	RCS Vented, RPV Head tensioned	25			
11	Refill reactor coolant system	5,6	0%	T _{avg} ~125	RPV Head Intact	24			20
12	Reactor coolant system heatup/draw bubble in pressurizer	5	0%	~125<T _{avg} <175	RPV Head Intact	54		48	48
13	Reactor coolant system heatup to 350°F	4	0%	175<T _{avg} <350	RPV Head Intact	20		20	20
14	Startup with steam generators (AFW) to Hot Standby	3	0%	350<T _{avg} <557	RPV Head Intact	40		40	40
15A	Reactor startup and low power operation	2	0=Power<5%	< NOT	RPV Head Intact	20	20	20	20

Table 2-23 Plant Operating State Durations (cont.)

POS		TS	RCS			POS Duration (hours) ¹			
						Duration A	Duration B	Duration C	Duration D
No.	Description	TS Mode	Power	Tavg (°F)	Boundary (Vent Status)				
15B	Reactor startup and low power operation	1,2	5<Power<40%	< NOT	RPV Head Intact	36	36	36	20
Note 1: Duration A: Refueling (POT-4) Duration B: Maintenance w/o Drain, w/o RHR (POT-1) Duration C: Maintenance w/o Drain (POT-2) Duration D: Maintenance w/Drain (POT-3)									

3 INITIATING EVENT ANALYSIS

3.1 Introduction

The objectives of the initiating event analysis are to identify and quantify events that could lead to core damage. The initiating events identification considers the plant-specific features and conditions for each plant operating state during low power and shutdown operations.

3.2 Initiating Event Identification

A systematic approach was developed for identifying initiating events that challenge plant operation during low power and shutdown conditions. The initiating event identification approach involves defining the key safety functions during shutdown POSs and identifying challenges to those safety functions. The initiating event identification for shutdown POSs is discussed in Section 3.2.1. The initiating events during low power and hot standby are discussed in Section 3.2.2. For initiating events during low power and hot standby, the full power initiating event analysis is taken as a starting point, and differences between full power and low power/hot standby operations are identified. In both the low power/hot standby and shutdown approaches, other relevant studies and generic lists of initiating events were reviewed as part of the identification approach.

3.2.1 Shutdown Initiating Event Identification

Shutdown initiating events are identified for the POSs occurring during Modes 4, 5, and 6. The applicable POSs include POS 3 through POS 13, as defined in Section 2.5 of this report.

The approach for identification of shutdown initiating events uses generic and plant-specific information sources to find potential challenges to the plant. The areas that are reviewed include the key safety functions during shutdown operations, plant procedures, monitored safety parameters, plant systems, plant-specific operating experience, interviews with plant staff, and generic lists of initiating events. The potential initiating events identified from these information sources are grouped into a combined set of initiating event categories. The systematic initiating event review process is summarized in Table 3-1.

3.2.1.1 *Challenges to Key Safety Functions*

The key safety functions for LPSD PRA are identified in the ASME/ANS LPSD PRA standard (ASME/ANS, 2014). Other important resources in defining the key safety functions for shutdown POSs are information and results from past low power and shutdown PRA study performed for the nuclear industry. This current study defines four challenges to the key safety functions for Modes 4, 5, and 6, which correspond to the safety functions identified by the ANS/ASME standard. However, the ANS/ASME standard includes a fifth key safety function for containment integrity. The containment integrity safety function is not relevant for identification of initiating events because the containment is routinely opened during shutdown operations, and the loss of containment integrity does not constitute an initiating event. Therefore, four key safety function challenges are considered for identifying shutdown initiating events and are defined below:

Loss of Decay Heat Removal – During shutdown POSs, operation of the RHR system is the primary means of decay heat removal. Loss of heat removal capability can occur due to RHR system failures (e.g., RHR pump failure or loss of flow) or losses of RHR support systems (e.g., loss of cooling to RHR heat exchanger or loss of AC power).

Loss of Inventory Control – Loss of inventory events include those events that result in inadvertent diversion or removal of RCS water volume out of the RCS. Loss of inventory events can result from pipe or pressure vessel failures. Maintenance and operational activities can also result in human-induced loss of inventory events.

Loss of Pressure Control – Loss of pressure control events result in pressurization of the RCS and challenge the overpressure relief protection systems. There is a risk of reactor vessel failure given unmitigated overpressure at low temperatures. Loss of pressure control is not applicable for POSs where a large RCS vent has been established.

Loss of Reactivity Control – The potential for reactivity events is limited because control and shutdown rods are fully inserted, and reactor trip breakers are opened during the majority of shutdown operation. Still, reactivity control events can occur due to boron dilution, fuel loading errors, or uncontrolled rod movements.

3.2.1.2 Review of Procedures

The reference plant procedures relevant to shutdown POSs (i.e., shutdown procedures, AOPs, and EOPs) were reviewed to understand how actions being performed during the shutdown POSs could challenge the key safety functions. The types of procedures and the potential challenges that were identified are shown in Table 3-2. In reviewing plant procedures, two additional potential challenges to safe plant operation were identified: heavy load drops and fuel handling events. Although these challenges do not directly impact the key safety functions, they are included for consideration as potential initiating events.

3.2.1.3 Systematic Safety Parameter Review

The reference plant generated a draft initiating event report as part of a shutdown PRA project. Although the shutdown PRA project was not completed, the draft documentation provides useful input to the identification of shutdown initiating events. The approach taken by the reference plant includes a systematic safety parameter review. The approach involves a review of the monitored safety parameters to identify ways in which they can be challenged. The safety parameters and the potential causes that could challenge those parameters are shown in Table 3-3. Table 3-3 also maps the safety parameters identified by the reference plant to the key safety functions

Table 3-1 Shutdown Initiating Event Identification Process

Area Reviewed	Process for Identifying Initiating Events	Information Sources
Key safety functions	Identify potential challenges to key safety functions.	• NUMARC 91-06 (NUMARC, 1991) • Other LPSD PRA studies
Plant procedures	Review procedures for cautions, warnings, and abnormal conditions.	• Reference plant shutdown procedures • Reference plant abnormal operating procedures (AOPs) • Reference plant emergency operating procedures (EOPs)
Monitored safety parameters	Review monitored plant parameters that can indicate challenges to safe operation.	Reference plant and contractors performed a systematic safety parameter search in preparing draft reports for an LPSD PRA.
Plant systems	Review plant systems whose failures can result in an initiating event.	• Reference plant PRA system notebooks • Reference plant outage risk monitoring procedure
Operating experience	Review shutdown initiating events and precursors to initiating events that have occurred at reference plant.	• Licensee Event Reports • Reference plant at-power, internal events PRA report • NUREG-1410 (NRC, 1990)
Interviews with plant staff	Discuss initiating event training scenarios and experience with events that have occurred.	Reference plant site visit to observe a refueling outage
Generic lists of initiating events	Review lists of initiating events that are modeled in other studies.	• Operating experience review of past studies, Licensee Event Reports, and Significance Determination Process findings (documented in 12APPENDIX D) • NUREG/CR-5750 (NRC, 1999) • Reference plant at-power, internal events PRA report • EPRI report 1021176 (EPRI, 2010) • EPRI report 1003113 (EPRI, 2001) • NUREG/CR-6144 (NRC, 1994a)

Table 3-2 Review of Reference Plant Procedures and Potential Challenges

Procedure Scope and Conditions	Relevant Procedure Discussion / Action(s)	Potential Challenge(s) to Shutdown Key Safety Functions
Shutdown Operations Procedures		
Post refueling operations (Mode 6 to Mode 5)	Procedure provides actions for installing reactor vessel head and upper internals and lowering refueling cavity level. Requirements and cautions are provided for RHR operation during different plant states.	Loss of decay heat removal, loss of inventory control
Heatup to hot shutdown (Mode 5 to Mode 4)	Procedure provides instructions for RCS heatup and pressurization, re-establishing COPS, starting RCPs, establishing SG cooling, and placing RHR in standby.	Loss of decay heat removal, loss of inventory control, loss of pressure control
Heatup to normal operation temperature and pressure (Mode 4 to Mode 3)	Procedure provides instructions for RCS heatup and pressurization to normal operating pressure and temperature.	Loss of decay heat removal, loss of inventory control, loss of pressure control
Reactor startup (Mode 3 to Mode 2)	Not applicable to shutdown states.	None
Power operation (Mode 1)	Not applicable to shutdown states.	None
Shutdown to hot standby (Mode 2 to Mode 3)	Not applicable to shutdown states.	None
Cooldown to cold shutdown	Procedure concerns activities related to plant cooldown with steam generators and placing RHR into service. Evolutions involving changes to RCS level and venting are also guided by this procedure (e.g., pressurizer solid operations).	Loss of decay heat removal, loss of inventory control, loss of pressure control
Refueling operations (entry into Mode 6)	Procedure provides actions for removing reactor vessel head and upper internals, raising refueling cavity level, and core alterations. Requirements and cautions are provided for RHR operation during different plant states.	Loss of decay heat removal, loss of inventory control, fuel handling event
Post refueling operations (Mode 6 to Mode 5)	Procedure provides actions for installing reactor vessel head and upper internals and lowering refueling cavity level. Requirements and cautions are provided for RHR operation during different plant states.	Loss of decay heat removal, loss of inventory control

Table 3-2 Review of Reference Plant Procedures and Potential Challenges (cont.)

Procedure Scope and Conditions	Relevant Procedure Discussion / Action(s)	Potential Challenge(s) to Shutdown Key Safety Functions
Shutdown Operations Procedures		
Heatup to hot shutdown (Mode 5 to Mode 4)	Procedure provides instructions for RCS heatup and pressurization, re-establishing COPS, starting RCPs, establishing SG cooling, and placing RHR in standby.	Loss of decay heat removal, loss of inventory control, loss of pressure control
Heatup to normal operation temperature and pressure (Mode 4 to Mode 3)	Procedure provides instructions for RCS heatup and pressurization to normal operating pressure and temperature.	Loss of decay heat removal, loss of inventory control, loss of pressure control
Reactor startup (Mode 3 to Mode 2)	Not applicable to shutdown states.	None
Power operation (Mode 1)	Not applicable to shutdown states.	None
Shutdown to hot standby (Mode 2 to Mode 3)	Not applicable to shutdown states.	None
Cooldown to cold shutdown	Procedure concerns activities related to plant cooldown with steam generators and placing RHR into service. Evolutions involving changes to RCS level and venting are also guided by this procedure (e.g., pressurizer solid operations).	Loss of decay heat removal, loss of inventory control, loss of pressure control
Refueling operations (entry into Mode 6)	Procedure provides actions for removing reactor vessel head and upper internals, raising refueling cavity level, and core alterations. Requirements and cautions are provided for RHR operation during different plant states.	Loss of decay heat removal, loss of inventory control, fuel handling event
Mid-loop operations	Procedure covers draining to reduced inventory and reduced inventory operations. Precautions alert operators to monitor RHR performance and RCS level.	Loss of decay heat removal, loss of inventory control
RCS vacuum refill	Procedure provides actions for reducing RHR cooling rate, installing skid-mounted vacuum pump, establishing RCS vacuum, and filling from RWST through idle RHR train.	Loss of decay heat removal, loss of inventory control, loss of pressure control

Table 3-2 Review of Reference Plant Procedures and Potential Challenges (cont.)

Procedure Scope and Conditions	Relevant Procedure Discussion / Action(s)	Potential Challenge(s) to Shutdown Key Safety Functions
Abnormal Operating Procedures		
Pressurizer spray, safety, or relief valve malfunction	Procedure provides instructions for failed open pressurizer spray valve, PORV, or safety valve, including steps to confirm these valves are closed.	Loss of inventory control, loss of pressure control
RCS leakage	Procedure specifies actions in response to RCS leakage in shutdown modes.	Loss of inventory control
Fuel handling event	Procedure describes actions taken during potential damage of an irradiated fuel assembly.	Fuel handling event
Loss of residual heat removal	Procedure specifies actions to respond to loss of RHR capability during shutdown.	Loss of decay heat removal
Loss of component cooling water	Procedure addresses loss of one CCW train with either RHR or SFPC system in operation.	Loss of decay heat removal
Loss of nuclear service cooling water system	Procedure addresses loss or degraded operation of Nuclear Service Cooling Water. Procedure steps verify CCW and RHR pumps (if required) are running in unaffected NSCW train.	Loss of decay heat removal
Loss of instrument air system	Procedure specifies responses to a loss of instrument air pressure (an attachment specifically addresses loss of instrument air in Mode 4, 5, or 6).	Loss of inventory control, loss of pressure control (if pressurizer solid condition)
Loss of class 1E electrical systems	Procedure addresses loss of 4,160V AC and 480V AC class 1E electrical system (a procedure section specifically addresses loss of power to AC emergency buses in Modes 5 and 6).	Loss of decay heat removal
Loss of 120V AC instrument power	Many of the procedure actions are not applicable during shutdown states. Response actions are applicable during plant states where RCPs are operating and/or steam generator cooling is in use.	Loss of decay heat removal (may result in loss of instrumentation and control of important equipment, e.g., AFW TDP)

Table 3-2 Review of Reference Plant Procedures and Potential Challenges (cont.)

Procedure Scope and Conditions	Relevant Procedure Discussion / Action(s)	Potential Challenge(s) to Shutdown Key Safety Functions
Loss of class 1E 125V DC power	Procedure should be performed concurrent with referenced EOPs, and these are not applicable to shutdown states. However, failure of 125V DC Power could result in loss of instrumentation and control of key equipment that is in use during shutdown states.	Loss of inventory control, loss of decay heat removal (may result in loss of instrumentation and control of important equipment, e.g., RHR, NSCW, and CCPs)
Emergency Operating Procedures		
Reactor trip or safety injection response	Not applicable to shutdown states.	None
Reactor trip response	Not applicable to shutdown states.	None
Natural circulation cooldown	Not applicable to shutdown states.	None
Natural circulation cooldown with void in vessel with level indication	Not applicable to shutdown states.	None
Natural circulation cooldown with void in vessel without level indication	Not applicable to shutdown states.	None
Re-diagnosis	The procedure allows operators to confirm appropriate post-accident recovery procedure. The procedure is applicable in Mode 4, but the referenced accidents and procedures are not relevant to shutdown states.	None
Loss of reactor or secondary coolant	Not applicable to shutdown states.	None
Safety injection termination	Not applicable to shutdown states.	None
Post-LOCA cooldown and depressurization	Not applicable to shutdown states.	None
Cold leg recirculation	Not applicable to shutdown states.	None
Hot leg recirculation	Not applicable to shutdown states.	None
Faulted steam generator isolation	The procedure is applicable while steam generators are in use for heat removal during Mode 4 POSS.	Loss of decay heat removal
Steam generator tube rupture (SGTR)	Not applicable to shutdown states.	None
Post-SGTR cooldown using backfill	Not applicable to shutdown states.	None
Post-SGTR cooldown using steam dump	Not applicable to shutdown states.	None
Loss of all AC power	Procedure responds to a loss of all AC power in Modes 1, 2, 3, and 4.	Loss of decay heat removal

Table 3-2 Review of Reference Plant Procedures and Potential Challenges (cont.)

Procedure Scope and Conditions	Relevant Procedure Discussion / Action(s)	Potential Challenge(s) to Shutdown Key Safety Functions
Loss of all AC power recovery without SI required	Procedure responds to a loss of all AC power in Modes 1, 2, 3, and 4.	Loss of decay heat removal
Loss of all AC power recovery with SI required	Procedure responds to a loss of all AC power in Modes 1, 2, 3, and 4. Normal shutdown conditions will have SI signal blocked and discharge MOVs isolated. This procedure is not relevant for shutdown.	Loss of decay heat removal
Loss of emergency coolant recirculation	Not applicable to shutdown states.	None
LOCA outside containment	Not applicable to shutdown states.	None
Recirculation sump blockage	Procedure responds to a recirculation sump blockage condition. The condition could be entered in responding to an initiating event, but the blockage by itself does not challenge the shutdown key safety functions.	None
Uncontrolled depressurization of all steam generators	Procedure is applicable in Modes 1, 2, 3, and 4, and includes actions to control RCS pressure and cooldown rate and monitor RCS subcooling.	Loss of inventory control, loss of pressure control
SGTR with loss of reactor coolant with subcooled recovery	Not applicable to shutdown states.	None
SGTR with loss of reactor coolant with saturated recovery	Not applicable to shutdown states.	None
SGTR without pressurizer pressure control	Not applicable to shutdown states.	None
Critical safety function status trees	Procedure provides guidance for monitoring safety functions related to subcriticality, core cooling, RCS integrity, and inventory. Applicable to Modes 1, 2, 3, and 4.	Loss of decay heat removal, loss of inventory control, loss of pressure control, loss of reactivity control
Response to nuclear power generation transient	Applicable to Modes 1, 2, 3, and 4. Procedure includes actions to check for possible sources of positive reactivity and eliminate them.	Loss of reactivity control
Response to loss of core shutdown	Applicable to Modes 1, 2, 3, and 4. Procedure includes actions to check and restore an adequate shutdown condition.	Loss of reactivity control

Table 3-2 Review of Reference Plant Procedures and Potential Challenges (cont.)

Procedure Scope and Conditions	Relevant Procedure Discussion / Action(s)	Potential Challenge(s) to Shutdown Key Safety Functions
Response to inadequate core cooling	Not applicable to shutdown states.	None
Response to degraded core cooling	Not applicable to shutdown states.	None
Response to saturated core cooling	Applicable to Modes 1, 2, 3, and 4. Procedure provides actions to establish injection and restore subcooled core cooling.	Loss of decay heat removal
Response to loss of secondary heat sink	Applicable to Modes 1, 2, 3, and 4. Procedure provides actions to restore feed flow to steam generators or establish alternate heat removal.	Loss of decay heat removal
Response to steam generator overpressure	Applicable to Modes 1, 2, 3, and 4. Procedure provides actions for an overpressure condition affecting any steam generator that can impact RCS cool down.	Loss of decay heat removal
Response to steam generator high level	Applicable to Modes 1, 2, 3, and 4. Procedure provides actions to prevent overfilling a steam generator when secondary cooling is being used.	Loss of decay heat removal
Response to loss of normal steam release capabilities	Applicable to Modes 1, 2, 3, and 4. Procedure provides actions to respond to failure of ARVs and condenser dump valves.	Loss of decay heat removal
Response to steam generator low level	Applicable to Modes 1, 2, 3, and 4. Procedure provides actions to respond to a steam generator low level condition.	Loss of decay heat removal
Response to imminent pressurized thermal shock condition	Procedure is applicable in Modes 1, 2, 3, and 4, and includes actions to control RCS pressure and cooldown rate.	Loss of inventory control, loss of pressure control
Response to anticipated pressurized thermal shock condition	Procedure is applicable in Modes 1, 2, 3, and 4, and includes actions to control RCS pressure and cooldown rate.	Loss of inventory control, loss of pressure control
Response to high containment pressure	Procedure is applicable in Modes 1, 2, 3, and 4. Procedure concerns containment integrity, but challenges to containment integrity are not considered shutdown initiating events.	None

Table 3-2 Review of Reference Plant Procedures and Potential Challenges (cont.)

Procedure Scope and Conditions	Relevant Procedure Discussion / Action(s)	Potential Challenge(s) to Shutdown Key Safety Functions
Response to containment flooding	Procedure is applicable in Modes 1, 2, 3, and 4. Procedure provides actions to respond to unexpected containment flooding and may be relevant to RCS loss of inventory scenarios.	Loss of inventory control
Response to high containment radiation level	Procedure is applicable in Modes 1, 2, 3, and 4. Procedure concerns high containment radiation levels.	None
Response to high pressurizer level	Procedure is applicable in Modes 1, 2, 3, and 4. Procedure provides actions to respond to high pressurizer level.	Loss of inventory control, loss of pressure control
Response to low pressurizer level	Procedure is applicable in Modes 1, 2, 3, and 4. Procedure provides actions to respond to low pressurizer level.	Loss of inventory control
Response to voids in reactor vessel	Procedure is applicable in Modes 1, 2, 3, and 4. Procedure provides actions to respond to a void in the reactor vessel.	Loss of inventory control, loss of pressure control

Table 3-3 Review of Reference Plant Safety Parameters

Safety Parameter	Potential Cause	Applicable POSs	Comments	Potential Challenge(s) to Shutdown Key Safety Functions
Low primary flow -RCP trip -RHR trip -Fuel pool cooling trip	- pump failure - loss of seal cooling - loss of bearing cooling - loss of motor cooling - loss of power	RCP trip: Modes 1,2,3,4, and parts of 5 when RCS >100 psig (POSs 1–5A and 11B–15B) RHR trip: RHR operating for Modes 4,5, and 6 (hot shutdown, cold shutdown, mid-loop, and core load/unload). SFPC trip: Fuel pool operations and core load/ unload	Treatment similar to full power for RCP trip. Since only one RHR pump normally operating, add loss of RHR pump as a potential initiating event. Add loss of SFPC pumps as a potential initiating event.	Loss of decay heat removal
High neutron flux	- RCP pump restart in an idle loop - rod withdrawal - rod ejection - boron dilution	All POSs	k/k period limit in intermediate flux channel included in transients with PCS available; others evaluated as reactivity addition accidents.	Loss of reactivity control
High primary pressure	- no single failure; need multiple failures such as pressurizer heaters stuck on and spray failed; or centrifugal charging pump (CCP) inadvertent actuation	All POSs where the RPV head is on and not vented (POSs 1–3 and 7A–15B)	Add cold overpressurization as a potential initiating event.	Loss of inventory control, loss of pressure control
Low primary pressure	- LOCA - main steam line break - SGTR	LOCAs: All POSs Steam line break: Steaming POSs (POSs 1–3 and 13–15B, Modes 1, 2, 3, and part of 4 during startup) SGTR: POSs where the RCS is pressurized >350 psig (Modes 1, 2, and 3)	For applicable POSs, treatment similar to full power. In Modes 5 and 6, add LOCAs as potential initiating events.	Loss of inventory control

Table 3-3 Review of Reference Plant Safety Parameters (cont.)

Safety Parameter	Potential Cause	Applicable POSs	Comments	Potential Challenge(s) to Shutdown Key Safety Functions
High containment pressure	-LOCA -loss of containment cooling/ ventilation -main steam line break in containment	LOCAS: Modes 2–4 Cont cooling: Modes 1, 2 and 3 with RCS >350F and containment isolated Steam line break: Steaming POSs (POSs 1–3 and 13–15B, Modes 1, 2, 3, and part of 4 during startup)	Treatment similar to full power.	None
Low pressurizer level and low pressure	- LOCA	Modes 1 and 2	LOCAS already modeled for all POSs	None
Low fuel pool level	- LOCA in fuel pool	Fuel pool operations	Model as potential initiating event for spent fuel pool	Not applicable for reactor PRA
High pressurizer level	- Chemical and volume control system or other injection source control failure	All POSs with RCS closed or not vented (POSs 1–3 and 13–15B)	Treatment similar to full power when RCS closed; otherwise, screen out.	Loss of inventory control, loss of pressure control
Reactor flux/coolant temperature imbalance	- none unique for power operation or shutdown	Modes 1 and 2	Treatment similar to full power.	None
Low reactor inlet temperature/ overcooling	- main steam line break - open steam generator relief valve - feedwater control failure	Steam line break or open SG relief valve: Steaming POSs (POSs 1–3 and 13–15B, Modes 1, 2, 3, and part of 4 during startup) FW control failure: Only POS 15B in automatic	Treatment similar to full power.	None
Inadequate subcooling/ pressurizer failure	- Pressurizer control failure - LOCA - open pressurizer relief valve	When level in pressurizer controlled: POSs 1–4 and 12–15B	Treatment similar to full power.	Loss of inventory control, loss of pressure control

Table 3-3 Review of Reference Plant Safety Parameters (cont.)

Safety Parameter	Potential Cause	Applicable POSs	Comments	Potential Challenge(s) to Shutdown Key Safety Functions
Low or high steam generator level/ feedwater failures	- main feedwater line break - feedwater control failure - feedwater regulating valve failure -steam generator water level instrumentation failure - MSIV closure	MFW problems: Steaming POSs (POSs 1–3 and 13–15B, Modes 1, 2, 3, and part of 4 during startup) MSIV closure: POSs 1 and 13–15B (Mode 1 and startup from Mode 4)	Treatment similar to full power.	Loss of decay heat removal
High main steam line pressure	- MSIV closure - Turbine trip without bypass - loss of condenser vacuum	MSIV closure: POSs 1 and 13–15B (Mode 1 and startup from Mode 4) Turbine trip: POSs 1 and 15B Loss of condenser vacuum: POSs 1 and 15A–15B	Treatment similar to full power.	Loss of decay heat removal
Low feedwater flow	- Condensate pump failure	Steaming POSs (POSs 1–3 and 13–15B, Modes 1, 2, 3, and part of 4 during startup)	Treatment similar to full power.	Loss of decay heat removal
SGTR activity	- SGTR - fuel element failure	SGTR: POSs where the RCS is pressurized >350 psig (Modes 1, 2, and 3) Fuel element failure: POS 8	SGTR: Treatment similar to full power. Fuel element failure: Add fuel handling accidents as a potential initiating event.	None
Manual trip	- manual trip	POSs 1 and 15A–15B (Modes 1 and 2)	Treatment similar to full power.	None

3.2.1.4 *Initiating Events Due to System Failures*

The reference plant systems were reviewed to assess the possibility of an initiating event occurring due to a failure of the system. The plant systems that are considered for shutdown initiating events are identified in the reference plant's outage risk monitoring procedure. The procedure lists the key safety systems and components that support the key safety functions during outages. The listed systems and components are:

- RCS loops
- RHR
- SI
- CVCS charging
- NSCW
- CCW
- SFPC
- Reserve auxiliary transformers (RATs)
- Standby auxiliary transformer (SAT)
- 1E electrical buses
- Diesel generators
- Containment
- Source range nuclear instrumentation
- RWST
- Fuel handling building (FHB) post-accident (PA) filtration

The assessment of each of these systems is shown in Table 3-4. For each system, the impact of a system failure is described, and it is indicated whether system failure would result in an initiating event. The reference plant's PRA system notebooks for the systems modeled in the at-power PRA and the relevant plant procedures were consulted to determine the impact of system failures.

Table 3-4 Review of Reference Plant Key Safety Systems for Outages

Key Safety System	Impact of System Failure	Causes Shutdown Initiating Event?	Plant Impact References
RCS loops	Each RCS loop includes a steam generator; a reactor coolant pump; and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. RCS loops, in combination with RHR loops, are required to be operable during shutdown POSs. Failure of RCS loops would result in loss of decay heat removal, if RHR loops are unavailable.	Yes	Reference plant bases for Technical Specifications
RHR	The RHR system provides the decay heat removal function during most of the shutdown POSs. RHR is the only system available to provide sustained decay heat removal during Mode 6 and parts of Mode 5 (after the RCS has been vented and RCS loops isolated).	Yes	Loss of residual heat removal procedure
Safety injection	The safety injection system provides an alternate means of core cooling if normal decay heat removal is lost.	No – SI signals and pumps inhibited during POS 2 and re-established during POS 14. Inadvertent SI is not considered during shutdown POSs.	Loss of residual heat removal procedure, PRA system notebook for ECCS
CVCS charging	CVCS provides normal charging for controlling RCS inventory and provides an ECCS function. Two emergency centrifugal charging pumps can provide alternate core cooling if normal decay heat removal is lost, but loss of this function is not an initiating event. Failure of normal charging during reduced inventory operations can impact RHR operation, if RCS level cannot be maintained.	Yes – for inventory control during mid-loop	Mid-loop operations procedure, PRA system notebook for ECCS

Table 3-4 Review of Reference Plant Key Safety Systems for Outages (cont.)

Key Safety System	Impact of System Failure	Causes Shutdown Initiating Event?	Plant Impact References
NSCW	RHR operation for decay heat removal depends on NSCW as the ultimate heat sink for the CCW heat exchangers. NSCW also provides cooling to the RHR pump motor coolers. Failure of NSCW would result in failure of RHR to provide decay heat removal.	Yes	Loss of service water system procedure, PRA system notebook for NSCW
CCW	RHR operation for decay heat removal depends on CCW to remove heat through the RHR heat exchangers. CCW also provides cooling to the RHR pump seal coolers. Failure of CCW would result in failure of RHR to provide decay heat removal.	Yes	Loss of CCW procedure, PRA system notebook for CCW
SFPC	SFPC is not applicable to the reactor PRA.	No	
Reserve auxiliary transformers	After the main turbine and generator are tripped, offsite power for all plant loads is normally taken from the RATs. Failure of the RATs would result in a loss of offsite power and would impact all frontline and support systems that are used to provide decay heat removal.	Yes	Loss of class 1E electrical systems procedure, PRA system notebook for electric power system
Standby auxiliary transformer	The SAT is used to supply power from the alternative AC power substation. The SAT is only used for RAT maintenance, surveillance, or other applicable corrective actions when a RAT is unavailable. Failure of the SAT would not impact offsite power supply for normal electric power distribution alignments.	No	Loss of class 1E electrical systems procedure, PRA system notebook for electric power system
1E electrical buses	The 1E electric buses supply power to systems used for decay heat removal, including RHR, AFW motor driven pumps, and their support systems.	Yes	Loss of class 1E electrical systems procedure, PRA system notebook for electric power system
Diesel generators	Diesel generators provide onsite power to class 1E electric power systems in the event of a loss of preferred power sources. Failure of diesel generators would not impact the plant if the preferred power sources are available.	No	Loss of class 1E electrical systems procedure, PRA system notebook for electric power system

Table 3-4 Review of Reference Plant Key Safety Systems for Outages (cont.)

Key Safety System	Impact of System Failure	Causes Shutdown Initiating Event?	Plant Impact References
Containment	The containment integrity safety function is not relevant for identification of initiating events, because the containment is routinely opened during shutdown operations, and the loss of containment integrity does not constitute an initiating event.	No	
Source range NIs	Source range nuclear instruments serve important functions in monitoring and identifying issues with reactivity control and fuel integrity. Failure of nuclear instrumentation could invoke actions to suspend reactivity changes or core alterations but would not result in an initiating event.	No	Nuclear instrumentation malfunction procedure
RWST	The RWST provides a source of borated water to fill the refueling cavity for refueling operations and for RCS injection if needed in response to an initiating event. Failure of the RWST could limit the plant's accident mitigation capabilities and prevent filling of the refueling cavity but would not result in an initiating event.	No	PRA system notebook for ECCS
FHB PA filtration	The FHB PA filtration system mitigates releases of radioactive material after a fuel handling event or other accidents.	No	

3.2.1.5 Plant-Specific Operating Experience and Staff Interviews

The reference plant's operating experience was reviewed to identify initiating events (or precursors to initiating events) that have occurred during shutdown. Several different information sources were reviewed to identify these events. Most events were identified by searching the licensee event report (LER) database or from the initiating event identification section of the reference plant's at-power, internal events PRA. The complete list of reviewed operating experience sources is given below.

- LER search for the reference plant with keywords: "RHR," "residual heat removal," "shutdown," "offsite power," "cooling," "overdrain," "drain"
- Reference plant at-power, internal events PRA documentation
- Historical record of significance determination process (SDP) evaluations
- Idaho National Laboratory (INL) database of shutdown initiating events.

The initiating event review was supplemented by discussions with reference plant staff on their experiences and training for shutdown initiating events. A member of the reference plant training staff and former operator described the training provided to operating crews prior to entering a refueling outage. Simulator-based training is provided on two shutdown initiating event scenarios: loss of RHR during Mode 4 and loss of RHR during mid-loop. The operator response to a loss of offsite power event during shutdown was also discussed. Also, the reference plant uses a qualitative risk assessment procedure that is performed during each shift during an outage. The qualitative risk assessments assist in identifying plant configurations that can be associated with elevated risk.

3.2.1.6 List of Shutdown Initiating Events

The information resulting from the review activities discussed in the previous sections was compiled to develop the complete set of shutdown initiating events for the LPSD PRA. A summary of the shutdown initiating event categories is presented in Table 3-5.

Table 3-5 Shutdown Initiating Event Categories

Initiating Event Category	Basis for Selecting IE Category	Reference(s)
Loss of RHR cooling	Loss of RHR will result in loss of the decay heat removal safety function during shutdown POSs. EPRI has identified loss of RHR events that have occurred in industry.	• EPRI report 1021176 (EPRI, 2010) • Loss of residual heat removal procedure • Key safety system in outage risk monitoring procedure • Reference plant operating experience review
Loss of CCW	CCW is a support system providing heat removal to the RHR heat exchangers. Loss of CCW will result in loss of the decay heat removal function.	• EPRI report 1021176 (EPRI, 2010) • Loss of CCW procedure • Key safety system in outage risk monitoring procedure
Loss of NSCW	NSCW is a support system providing heat removal to the CCW heat exchangers and ultimate heat sink for the RHR system. Loss of NSCW will result in loss of the decay heat removal function.	• Loss of service water procedure • Key safety system in outage risk monitoring procedure
Loss of power to class 1E 4,160V AC electrical system	The class 1E 4,160V AC buses provide power to safety-related pumps, valves, and other SSCs. Loss of one bus will result in the loss of one train of the decay heat removal function.	• EPRI report 1021176 (EPRI, 2010) • Loss of class 1E electrical systems procedure • Key safety system in outage risk monitoring procedure
Loss of offsite power	Loss of normal offsite power sources will result in the loss of decay heat removal. Restoration of offsite power, or onsite power from diesel generators, is required to restore decay heat removal.	• EPRI report 1021176 (EPRI, 2010) • Loss of class 1E electrical systems procedure • Key safety system in outage risk monitoring procedure • NUREG-1410 (NRC, 1990)
Loss of inventory	Loss of RCS inventory events have occurred in past operating experience as documented in EPRI report 1021176 (EPRI, 2010).	• EPRI report 1021176 (EPRI, 2010) • RCS leakage procedure • Reference plant operating experience review
Overdraining RCS level	Examples of draining past the target elevation level have occurred in past operating experience, and these events can impact RHR operation. The mid-loop operating procedure includes warnings and cautions about draining to reduced inventory levels.	• EPRI report 1021176 (EPRI, 2010) • Mid-Loop operations procedure

Table 3-5 Shutdown Initiating Event Categories (cont.)

Initiating Event Category	Basis for Selecting IE Category	Reference(s)
Loss of level control	This event results from RCS inventory being displaced or transferred but not lost from the system. The cause can be related to voids in the system and can result in inaccurate level indication.	• EPRI report 1021176 (EPRI, 2010)
Loss of instrument air	Loss of instrument air will cause the charging line flow control valve to fail open. This can lead to loss of inventory control and pressure control, particularly in pressurizer solid conditions. The RHR flow control valves will also fail open. Manual actions may need to be taken to limit the cooldown rate.	• Loss of instrument air procedure
Loss of reactivity control	Reactivity control is a key safety function to maintain the reactor core in a safe state.	• Key safety function
Fuel handling event	The potential for fuel handling events was identified through review of the reference plant procedures and operating experience.	• Fuel handling event procedure • Reference plant operating experience review

3.2.2 Low Power and Hot Standby Initiating Event Identification

Low power operation describes the transition between a shutdown state and at-power operation. This period includes plant operations in Technical Specification Modes 1, 2, and 3 (see Table 2-1 for mode definitions). As defined in Section 2.5, low power operation includes power operation at 50 percent or lower than the nominal operating power.

Many of the operating conditions, including the availability of plant systems, RCS pressure and temperature, are similar to full power operation. The initiating events identified for the at-power model (see Section 2 of (NRC, 2022)) are considered to be applicable to low power operation. The list of at-power initiating events is used as the initial list of low power events. As the plant transitions between full power and low power there are changes that can influence the initiating events. The key differences that distinguish low power and hot standby operations from full power operation are discussed below.

- During power reduction (POS 1):
 - One MFW pump is stopped, and the second MFW pump provides steam generator inventory in manual control.
- After manual plant trip and during cooldown with steam generators (POS 2):
 - Steam generator cooling is provided by AFW pumps.
 - The main turbine is tripped, and steam generator heat removal is accomplished by atmospheric relief valves.

- Two reactor coolant pumps are stopped, while two remain operating.
- The cold over pressure protection system is enabled, which lowers the PORV lift set points.
- The accumulators are isolated to prevent RCS injection.
- The safety injection signals on low pressurizer pressure and main steam line pressure are blocked.
- The safety injection pumps are placed in PTL condition.
- During startup to hot standby (POS 14):
 - RHR is isolated from RCS.
 - AFW is aligned for steam generator cooling with atmospheric relief valves.
 - Cold over pressure protection system is disabled, and PORVs are set to their normal operating setpoints.
 - Accumulators are aligned for service
 - Safety injection and centrifugal charging pumps are aligned for automatic ECCS actuation.
- During reactor startup and low power operation (POS 15A and 15B)
 - Control rods are withdrawn to bring the reactor critical.
 - Steam dumps to the main condenser are operable.
 - Steam generator cooling is transferred from AFW to MFW.
 - Rolling of the main turbine is begun, and the main generator output breakers are closed at 20 percent power.

The list of initiating events considered for low power and hot standby is shown in Table 3-6. The table identifies the potential differences from the corresponding at-power initiating event category due to the unique low power and hot standby conditions.

The reference plant operating experience was reviewed to identify initiating events (or precursors to initiating events) that have occurred during low power operation. The information sources that were reviewed were previously discussed in Section 3.2.1. Most events were identified by searching the LER database or from the initiating event identification section of the reference plant's at-power, internal events PRA.

Table 3-6 Low Power and Hot Standby Initiating Events

Initiating Event Category	Low Power Hot Standby Difference from At-Power
Turbine trip	The frequency of turbine trips is expected to be different than at-power operation as the turbine is placed back into service during reactor startup. Turbine trip is not applicable to POSs where it is not in service (POSs 2–15A).
Reactor trip	The frequency of reactor trips is expected to be different than at-power operation as systems are aligned and re-started during startup. Reactor trip is not applicable for POSs occurring after the reactor has been tripped.
Other transients	The frequency of other transients is expected to be different than at-power operation as systems are aligned and re-started during startup. Other transients (resulting in reactor trip) are not applicable for POSs occurring after the reactor has been tripped.
Loss of main feedwater	The frequency of loss of main feedwater is expected to be different than at-power operation as the system alignment is changed (e.g., manual control, reducing or increasing the number of operating pumps, and transitioning between AFW and MFW) during cooldown and startup. Loss of MFW is not applicable to POSs where AFW is used for steam generator cooling.
Loss of condenser heat sink	The frequency of loss of condenser is expected to be different than at-power operation as the steam generator heat removal is shifted between the main condenser and the atmospheric relief valves during cooldown and startup. Loss of condenser is not applicable to POSs where it is not in service (POSs 2–15A).
Large LOCA	No significant difference from at-power is expected.
Medium LOCA	No significant difference from at-power is expected.
Small LOCA	No significant difference from at-power is expected.
Reactor vessel rupture	No significant difference from at-power is expected.
ISLOCA	Operation at lower than nominal RCS pressure might reduce the likelihood of ISLOCA occurrence, but during low power and hot standby RCS is at or near nominal pressure. No significant difference from at-power is expected.
Steam generator tube rupture	Operation at lower than nominal RCS pressure might reduce the likelihood of SGTR occurrence, but during low power and hot standby RCS is at or near nominal pressure. No significant difference from at-power is expected.
Loss of offsite power	Plant configuration changes and maintenance activities may result in change to initiating event frequency. Plant response modeling expected to be similar to at-power PRA.
Loss of auxiliary component cooling water	No significant difference from at-power is expected.
Loss of nuclear service cooling water	No significant difference from at-power is expected.

Table 3-6 Low Power and Hot Standby Initiating Events (cont.)

Initiating Event Category	Low Power Hot Standby Difference from At-Power
Inadvertent safety injection	The frequency of inadvertent SI is expected to be different than at-power operation as SI signals are blocked and systems isolated during cooldown and systems are restored during startup.
Loss of RCP seal injection	No significant difference from at-power is expected.
Secondary-side break upstream MSIVs/downstream MFIVs	No significant difference from at-power is expected.
Secondary-side break downstream MSIVs/upstream MFIVs	No significant difference from at-power is expected.
Loss of instrument air	No significant difference from at-power is expected.
Loss of safety-related 125V DC buses	No significant difference from at-power is expected.
Loss of safety-related 4,160V AC buses	No significant difference from at-power is expected.
Loss of safety-related 120V AC panels	For low power, no significant difference from at-power is expected. For POSs occurring after the reactor has been tripped, the loss of vital 120V AC panels is expected to have less of an impact but may still cause loss of instrumentation and control of risk significant equipment.
Loss of auxiliary feedwater	Loss of AFW is not an initiating event in the at-power model but is an initiating event during POSs with AFW being used for steam generator cooling (POSs 2, 14, and 15A).
Loss of atmospheric relief valves	Loss of atmospheric relief valves is not an initiating event in the at-power model but is an initiating event during POSs where ARVs are being used for steam generator cooling (POSs 2, 14, and 15A).

3.3 Initiating Event Qualitative Assessment

The initiating event categories identified for shutdown and low power operations were qualitatively assessed to determine how to incorporate them into the LPSD PRA model.

3.3.1 Shutdown Initiating Event Qualitative Assessment

Each of the identified shutdown initiating events was assessed to determine the appropriate modeling approach to be used in the LPSD PRA. The initiating event categories are defined in terms of what types of failures can result in the initiating event. Initiating events are grouped or subsumed where appropriate.

Loss of RHR Cooling

The RHR system is the primary means of providing decay heat removal during shutdown POSs. Loss of RHR cooling will result in RCS heat up and eventual boiling and core damage if decay heat removal is not restored. The loss of RHR cooling initiating event encompasses several different types of failures:

- Failures of the RHR pumps
- Failures of RHR piping and valves that inhibit the RHR flow path
- Inadvertent isolation or diversion of the RHR flow path
- Loss of cooling to the RHR heat exchangers
- Failures of the CCW system resulting loss of cooling to RHR heat exchangers

The plant response to the loss of RHR cooling initiating event is described in Section 4 and its initiating event frequency is estimated in Section 8.1.

Loss of CCW

The CCW system is a support system providing heat removal to the RHR heat exchangers, RHR pump seal coolers, and spent fuel pool system. Loss of CCW will result in loss of the decay heat removal function. The loss of the CCW system function is considered to be subsumed into the loss of RHR cooling initiating event.

Loss of NSCW

Loss of NSCW will lead to loss of cooling capability to the CCW heat exchangers and RHR pump motor coolers and results in a functional loss of decay heat removal. The loss of NSCW also impacts accident mitigating systems, such as emergency diesel generators, centrifugal charging pumps, and safety injection pumps.

The potential for loss of NSCW initiating events is discussed further in the accident scenario prioritization in Section 4.2.

Loss of Electrical Power to Class 1E 4,160V AC Bus

The loss of electrical power to class 1E 4,160V AC bus initiating event represents the loss of one of the two safety-related buses. The bus that powers the operating RHR train is the one that is assumed to be impacted by the initiating event. In POSs where both trains of RHR are operating, either bus could potentially be impacted by the initiating event. As a modeling simplification, the train B bus is assumed to be impacted by the initiating event. The loss of power to a 4,160V AC bus can be caused by failures of relays or breakers that would result in de-energizing the bus. The bus can also be de-energized by inadvertent opening of relays or breakers related to maintenance and testing activities. A review of operating experience shows that most events during shutdown result from errors in maintenance and testing, and the bus can be recovered by the associated emergency diesel generator automatically starting and re-energizing the bus.

The plant response to the loss of power to class 1E 4,160V AC initiating event is described in Section 4 and its initiating event frequency is estimated in Section 8.1.

Loss of Offsite Power

The loss of offsite power event is a simultaneous loss of electrical power to all safety-related buses that causes emergency diesel generators to start and supply power to the buses. The event includes all classes of loss of offsite power initiating events, including plant-centered, switchyard-centered, grid-related, and weather-related events.

The plant response to the loss of offsite power initiating event is described in Section 4 and its initiating event frequency is estimated in Section 8.1.

Loss of Inventory

The loss of inventory initiating event involves the loss of coolant inventory outside of the RCS, reactor vessel, refueling cavity, and the RHR system when connected to the RCS. This initiating event reduces the overall inventory available for cooling the core. Loss of inventory events can occur due to pipe or pressure vessel failures, human errors resulting in inadvertent flow diversions, or lifting of pressure relief valves.

The plant response to the loss of inventory initiating event is described in Section 4 and its initiating event frequency is estimated in Section 8.1.

Overdraining RCS Level

The overdraining initiating event involves draining RCS level below the desired level when entering the mid-loop condition. The event is only applicable when draining RCS to levels near the RHR suction from the RCS hot legs. The lowering of RCS level below the desired elevation can result in RHR pump cavitation and damage.

The plant response to the overdraining initiating event is described in Section 4 and its initiating event frequency is estimated in Section 8.1.

Loss of Level Control

This event results from RCS inventory being displaced or transferred, but not lost, from the system. The cause can be related to voids in the system and can result in inaccurate level indication. Loss of level events have been observed to occur in recent operating experience and are documented in EPRI report 1021176 (EPRI, 2010). The EPRI report notes that the loss of level events generally involves small inventory losses, and the loss of level events are assessed to have the lowest severity of all types of events involving inventory losses or displacements. In most cases the level displacement would not impact the RHR system capability to provide decay heat removal. At some plants, loss of level events have resulted in automatic isolation of the RHR decay heat removal function, but this is not applicable to the reference plant.

While accurate control and indication of RCS level is important for correctly performing plant evolutions and should be considered in the plant response to initiating events, the loss of level control itself is not considered an accident initiator. Therefore, loss of level control is not modeled as a unique initiating event category in the LPSD PRA. Failures related to loss of level control are considered in other initiating event categories. Failures of level indication that can lead to inadvertent RHR isolation are considered in the loss of RHR cooling category. Losses of level control that can result in overdraining the RCS such that the RHR operation is impacted are considered in the overdraining initiating event category.

Loss of Instrument Air

The loss of the instrument air system will result in the failures of air-operated components to perform their functions. The impacts of loss of instrument air during shutdown operations are highlighted in an attachment to the reference plant's loss of instrument air procedure. The

attachment provides cautions for two types of air-operated components and describes the impact of loss of instrument air. The first type of component is the charging line flow control valve. On loss of instrument air, the charging line flow control valve will fail open, causing charging flow to increase. This can lead to loss of inventory and pressure control if the charging flow is not monitored. Of particular concern are pressurizer solid conditions where an uncontrolled increase in charging flow can lead to a rapid pressure increase and lifting of cold over pressure protection system (COPS) relief valves.

The pressure increase and lifting of the RHR relief valves during pressurizer solid operations has been studied to assess the impacts from loss of RHR. A set of MELCOR calculations were performed to support this study and show that the RHR relief valves lift almost immediately resulting from the pressure increase during pressurizer solid conditions. A similar response is expected due to the pressure increase from increased charging flow. The plant response to loss of instrument air with pressurizer solid is expected to be bounded by the loss of RHR and loss of inventory events with the same conditions. The loss of instrument air event is not evaluated further in this project. Including the specific plant response to loss of instrument air with the pressurizer solid and estimating the frequency of loss of instrument air events applicable to pressurizer solid conditions are candidates for future work, as discussed in Appendix A.

The second type of impacted component is the RHR flow control valve. The RHR flow control valves will fail open leading to an increase in RHR flow and cooldown rate. Manual actions may need to be taken to limit the cooldown rate. This failure of RHR flow control does not prevent the decay heat removal function and is not considered as an initiating event.

Other Shutdown Initiating Events

The other shutdown initiating event categories identified in Table 3-5 are reactivity control failures and fuel handling events. Types of events in these categories can include inadvertent control rod movement, inadvertent coolant dilution, dropped fuel assembly, mis-loaded fuel assembly, and other fuel assembly damage. These types of events can have important impacts on plant operations and outage management. Additional study is needed to assess the conditions resulting from these events that could result in significant fuel damage and possible radiological release. These events are suggested as candidates for future work and are documented in Appendix A.

3.3.2 Low Power and Hot Standby Initiating Event Qualitative Assessment

The low power initiating events were qualitatively assessed to determine the appropriate modeling approach to be used in the LPSD PRA. The plant response during low power and hot standby is expected to be similar to the plant response as modeled in the L3PRA project at-power PRA (NRC, 2022). The current practice in modeling of at-power PRAs is to define initiating events to include all reactor critical operations. This includes full operating power and low power states. The risk associated with low power states should be adequately addressed by the at-power PRA, given the use of reactor critical years to define the PRA scope. However, this project gives consideration to the potential impacts of the unique conditions during low power and hot standby states that may warrant deviations from the at-power modeling approach. The list of at-power initiating event categories was reviewed to assess differences in the plant response modeling and initiating event definition and frequency.

The key plant conditions during low power operations that are considered in the initiating event analysis include:

- The increased likelihood of reactor trip or turbine trip due to the reduction in reactor power and the associated reduction in main feedwater and steam flow
- The transition from having two main feedwater pumps in operation to only one main feedwater pump in operation
- The transfer of the source of electrical power to the non-safety electrical systems from the unit auxiliary transformers to the reserve auxiliary transformers
- The blocking of ATWS mitigation system actuation circuitry signals for a portion of low power operation

Table 3-7 shows a qualitative assessment of each at-power initiating event category with consideration for low power conditions.

The key plant conditions during hot standby operations that are considered in the initiating event analysis include:

- In hot standby the reactor is subcritical, consistent with the definition for technical specification Mode 3 – Hot Standby. The reactor trip breakers are open and control rods are fully inserted. At-power initiating events that result in reactor trip are no longer applicable.
- The AFW system is providing feedwater to the steam generators for decay heat removal. The main feedwater system is stopped. The atmospheric relief valves are used for secondary heat removal. The main condenser is not in service.
- The safety injection accumulators are isolated to prevent injection. Safety injection actuation signals are blocked, and safety injection pumps are placed in PTL to prevent automatic actuation.
- Reactor coolant pumps in two of the four RCS loops are stopped.
- COPS is enabled.

Table 3-8 shows a qualitative assessment of each at-power initiating event category with consideration for hot standby conditions. Also, two additional initiating event categories, for loss of the AFW system and loss of atmospheric relief valves, are included in the assessment.

Table 3-7 Low Power Initiating Event Qualitative Assessment

Initiating Event Category	Low Power Qualitative Assessment
Turbine trip	Plant configuration changes may result in change to initiating event frequency. Plant response modeling expected to be similar to at-power PRA.
Reactor trip	Plant configuration changes may result in change to initiating event frequency. Plant response modeling expected to be similar to at-power PRA.
Other transients	Plant configuration changes may result in change to initiating event frequency. Plant response modeling expected to be similar to at-power PRA.
Loss of main feedwater	Plant configuration changes may result in change to initiating event frequency. Plant response modeling expected to be similar to at-power PRA.
Loss of condenser heat sink	Plant configuration changes may result in change to initiating event frequency. Plant response modeling expected to be similar to at-power PRA.
Large LOCA	No significant differences from the at-power PRA were identified for this initiating event category.
Medium LOCA	No significant differences from the at-power PRA were identified for this initiating event category.
Small LOCA	No significant differences from the at-power PRA were identified for this initiating event category.
Reactor vessel rupture	No significant differences from the at-power PRA were identified for this initiating event category.
ISLOCA	No significant differences from the at-power PRA were identified for this initiating event category.
Steam generator tube rupture	No significant differences from the at-power PRA were identified for this initiating event category.
Loss of offsite power	Plant configuration changes and maintenance activities may result in change to initiating event frequency. Plant response modeling expected to be similar to at-power PRA.
Loss of auxiliary component cooling water	No significant differences from the at-power PRA were identified for this initiating event category.
Loss of nuclear service cooling water	No significant differences from the at-power PRA were identified for this initiating event category.
Inadvertent safety injection	Plant configuration changes may result in change to initiating event frequency. Plant response modeling expected to be similar to at-power PRA.
Loss of RCP seal injection	No significant differences from the at-power PRA were identified for this initiating event category.
Secondary-side break	No significant differences from the at-power PRA were identified for this initiating event category.
Loss of instrument air	No significant differences from the at-power PRA were identified for this initiating event category.
Loss of safety-related 125V DC bus	Plant configuration changes may result in change to initiating event frequency, but most activities are expected to occur after plant shutdown. Plant response modeling expected to be similar to at-power PRA.

Table 3-7 Low Power Initiating Event Qualitative Assessment (cont.)

Initiating Event Category	Low Power Qualitative Assessment
Loss of safety-related 4.16kV AC bus	Plant configuration changes may result in change to initiating event frequency, but most activities are expected to occur after plant shutdown. Plant response modeling expected to be similar to at-power PRA.
Loss of safety-related 120V AC panels	Plant configuration changes may result in change to initiating event frequency, but most activities are expected to occur after plant shutdown. Plant response modeling expected to be similar to at-power PRA.

Table 3-8 Hot Standby Initiating Event Qualitative Assessment

Initiating Event Category	Hot Standby Qualitative Assessment
Turbine trip	Not applicable. The main turbine is not in service. Steam load is transferred to the steam dump system
Reactor trip	Not applicable. The reactor is tripped.
Other transients	Not applicable. The reactor is tripped.
Loss of main feedwater	Not applicable. Main feedwater is no longer in service.
Loss of condenser heat sink	Not applicable. The main condenser is not in service.
Large LOCA	No significant differences from the at-power PRA were identified for this initiating event category.
Medium LOCA	No significant differences from the at-power PRA were identified for this initiating event category.
Small LOCA	No significant differences from the at-power PRA were identified for this initiating event category.
Reactor vessel rupture	No significant differences from the at-power PRA were identified for this initiating event category.
ISLOCA	Operation at lower than nominal RCS pressure might reduce the likelihood of ISLOCA occurrence, but during hot standby RCS is at or near nominal pressure. No significant differences from the at-power PRA were identified for this initiating event category.
Steam generator tube rupture	Operation at lower than nominal RCS pressure might reduce the likelihood of SGTR occurrence, but during hot standby RCS is at or near nominal pressure. No significant differences from the at-power PRA were identified for this initiating event category.
Loss of offsite power	Plant configuration changes may result in change to initiating event frequency.
Loss of auxiliary component cooling water	No significant differences from the at-power PRA were identified for this initiating event category.
Loss of nuclear service cooling water	No significant differences from the at-power PRA were identified for this initiating event category.
Inadvertent safety injection	Plant configuration changes may result in change to initiating event frequency.
Loss of RCP seal injection	Plant configuration changes may result in change to initiating event frequency.
Secondary-side break	Due to reduced steam load, the impacts of a secondary break would be less severe than during full power operation. It is not clear if changes in plant conditions would impact event frequency.
Loss of instrument air	Event will result in failures of air-operated components to their fail-safe states. Operator response may be required, but it is not considered an accident condition, as the reactor is already shutdown.
Loss of safety-related 125V DC bus	Event will not result in reactor transient, as the reactor is already shutdown, but can result in loss of control of systems required for safety functions (e.g., AFW, ACCW, and NSCW). The impact can be modeled in the associated initiating event categories for those systems.

Table 3-8 Hot Standby Initiating Event Qualitative Assessment (cont.)

Initiating Event Category	Hot Standby Qualitative Assessment
Loss of safety-related 4.16kV AC bus	Plant configuration changes may result in change to initiating event frequency, but most activities are expected to occur after plant shutdown.
Loss of safety-related 120V AC panels	Event impacts instrumentation and control power for the RPS, ESFAS, and process instrumentation and control systems. The impacts are not considered an accident condition, as the reactor is already shutdown.
Loss of auxiliary feedwater	Loss of AFW is introduced as a new initiating event category, as AFW is in use for providing decay heat removal during hot standby.
Loss of atmospheric relief valves	Loss of ARVs is introduced as a new initiating event category, as ARVs are in use for providing decay heat removal during hot standby.

4 ACCIDENT SEQUENCE ANALYSIS

4.1 Introduction

The objectives of the accident sequence analysis are to ensure that the response of the plant's systems and operators to an initiating event is reflected in the assessment of CDF and support the Level 2 PRA analysis.

The accident sequences are organized into accident scenarios. Accident scenarios are defined by the combination of a plant operating state (as defined in Section 2) and an initiating event (as defined in Section 3). An event tree model was developed for each accident scenario based on the characteristics of the initiating event and POS. Each scenario can contain multiple accident sequences that lead to core damage. The accident sequences represent the different combinations of success and failures of plant systems and operator actions in response to the initiating event. Additional discussion of the accident scenario selection for this project is found in Section 4.2.

An event tree model was developed for the accident scenarios that were quantitatively assessed. The event trees provide a graphical representation of the accident sequences associated with each accident scenario. The event trees model each key safety function that is used in response to an initiating event. The event trees are described in Section 4.3.

The event tree model and associated fault trees were adjusted for each modeled POS to account for the specific plant conditions. Section 4.4 describes each modeled POS with information about the conditions that can influence the accident progression. The POS descriptions include the assumptions about initial conditions (e.g., pressure, temperature, and decay heat level), equipment status, and identification of human failure events.

4.2 Accident Scenario Selection

In defining the accident scenarios all combinations of POSs and initiating events applicable to those POSs were considered. The number of IE and POS combinations can produce a large number of potential scenarios. The risk significance of these scenarios can vary greatly depending on duration exposed to the plant state, the level of vulnerability associated with the state, and other factors.

In order to limit the time and resources required to complete the LPSD PRA, the low power and hot standby accident scenarios were addressed through qualitative analysis. A series of sensitivity studies was performed to provide a bounding estimate of the possible risk contributions from these scenarios. The low power and hot standby accident scenarios are discussed in Section 4.2.1.

For shutdown accident scenarios, a scenario prioritization scheme was developed to focus the analysis on the most risk-significant accident scenarios and sequences. Scenarios with low priority were addressed with simplified analyses or qualitative assessments. The scenario prioritization is discussed in Section 4.2.2. NRC sponsored a separate expert panel ranking exercise to rank expected risk contributions from LPSD POSs and POTs, which is documented in NUREG/CR-7265 (NRC, 2020a). The outcome of the ranking exercise was not completed

prior to the initial model revision of the LPSD PRA and, therefore, was not available to inform the scenario prioritization.

4.2.1 Low Power and Hot Standby Accident Scenario Assessment

The low power and hot standby POSs include POS 1, 2, 14, 15A and 15B. These POSs represent the transition between at-power and shutdown states. The plant conditions and initiating event response associated with these POSs resemble the at-power model. Event trees for the low power and hot standby POSs were not developed and are not included in the base LPSD PRA. The low power and hot standby accident scenarios were addressed through qualitative analysis and by performing a series of sensitivity studies to provide a bounding estimate of the possible risk contribution from these scenarios. This approach was chosen for the low power and hot standby scenarios because: (1) most aspects of low power accident scenarios are addressed by the at-power PRA, (2) the low power and hot standby POSs have relatively short exposure time, and (3) there are limited data available to support low power and hot standby initiating event frequency estimates.

While the initiating event categories for the low power and hot standby POSs are similar to the at-power model, the initiating event frequencies may be different. The changing plant conditions and system alignments, as discussed in Section 3.2.2, could contribute to potentially higher initiating event frequencies during low power and hot standby when compared to at-power. A review of low power and hot standby operating experience was performed, but there are several challenges that make estimating initiating event frequencies difficult. These challenges include:

- Uncertainty in determining the boundary between at-power and low power events.
- Inconsistencies in identifying low power and hot standby events (event frequencies are not routinely collected for these plant states).
- Uncertainty in determining low power and hot standby hours of operation.

Despite these challenges, low power and hot standby events were reviewed to understand the potential impacts on initiating event frequencies during these POSs. A survey of 26 reactor units was performed to identify low power and hot standby initiating events and operating hours. The 26 units included three-loop and four-loop PWRs with similar operating period as the reference plant. The data were collected for operating years 1990 through 2015. The data for low power and hot standby initiating events and operating hours are derived from the NRC Reactor Operating Experience Data database managed by the Idaho National Laboratory (INL, 2013). The database includes failures, unavailability, and operating data that are provided by the Institute of Nuclear Power Operations.

The INL database includes monthly operating reports for each operating reactor unit. The monthly operating reports include hours of operation with the reactor critical and sub-critical and hours with the main generator online and offline. The surveyed units account for 5,811,552 reactor-hours from 1990 to 2015. Figure 4-1 shows the distribution of operating hours for the group of 26 reactor units. The available information does not allow for a detailed evaluation of POS-specific timing, but it can be used to estimate the time spent in low power states. The hours of critical operation with main generator offline roughly align with the low power POSs 1, 15A and 15B.

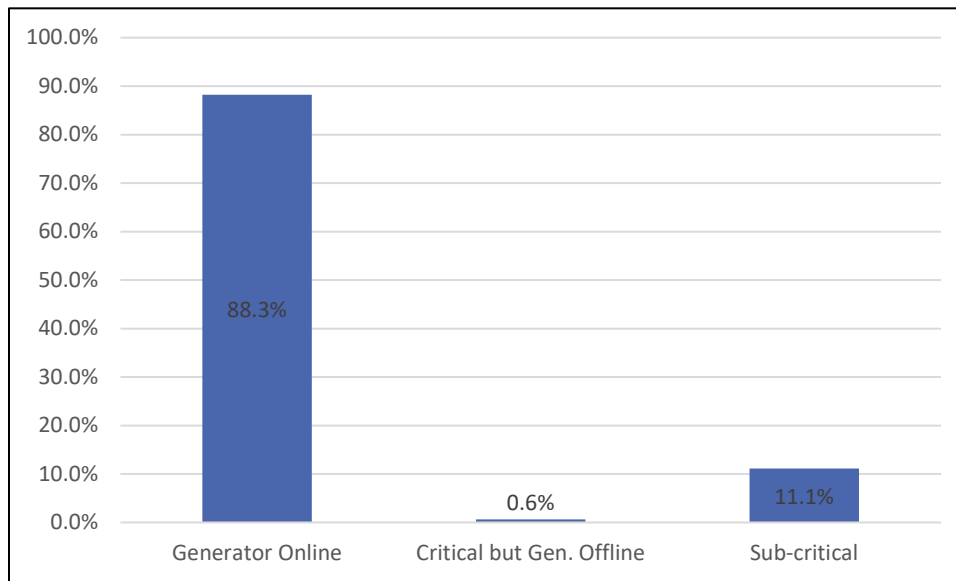


Figure 4-1 Distribution of Operation Hours for Surveyed Reactor Units, 1990-2015

The surveyed reactor units were also reviewed to identify transient initiating events that occurred during low power operations. Events occurring between 0 percent and 50 percent power level were collected in the following categories: transient, main turbine trip, loss of main feedwater, and loss of condenser heat sink. The results of the low power event survey are shown in Table 4-1. Combining the event counts with the hours of critical operation with generator offline gives an estimate of the initiating event rates during low power POSs. The full power initiating event rates that are used in the L3PRA project at-power PRA (NRC, 2022) are provided for reference.

Table 4-1 Low Power Event Count and Event Rate Estimates

IE Category	Low Power Event Count	Low Power Event Rate per Year	Full Power IE Rate per Year	Ratio of Low Power to Full Power Event Rate
Transient	73	1.8E+01	5.6E-01 (1-IE-OTRANS + 1-IE-RTRIP)	32.8
Main Turbine Trip	12	3.0	1.7E-01 (1-IE-TTRIP)	17.6
Loss of Main Feedwater	2	5.0E-01	6.6E-02 (1-IE-LOMFW)	7.6
Loss of Condenser Heat Sink	3	7.5E-01	6.0E-02 (1-IE-LOCHS)	12.5

In assessing the low power initiating events and operating hours, there are several uncertainties that have not been addressed. First, there are no clear boundaries between low power events and full power events. Some events occurring at less than 50 percent power are included in the full power initiating event analyses. There is potential for double counting events that are not clearly distinguished as either low power or full power events. The low power operating hours

are approximated by the reported critical operation with generator offline, but this may not align exactly with the low power POS boundary definitions. Also, there can be much variation between plants and from outage to outage in the specific conditions and time spent in different plant states. This survey of 26 reactor units makes no attempt to account for outlier or abnormal outages that occurred during the observation period from 1990 to 2015. Despite these uncertainties, this simple assessment of the data show that the initiating event rates during low power POSs are potentially much higher than the corresponding full power initiating event rates.

While the available data provide insights about the occurrence of low power events, the hot standby POSs are not represented at all in the current data collection approach. The hot standby POSs involve transitioning to other means of decay heat removal (e.g., transitioning from main feedwater to auxiliary feedwater for steam generator cooling). This introduces new categories of initiating events (e.g., loss of auxiliary feedwater), which are not routinely collected.

The unavailability and uncertainty in the low power and hot standby data collection make it difficult to produce consistent initiating event frequency estimates, which also provides challenges to any quantitative assessment. While the risk contribution from low power and hot standby POSs is expected to be low due to small exposure period, there may be unique plant conditions that pose safety challenges, such as the higher event occurrence rates shown in Table 4-1 above. A quantitative assessment of low power and hot standby risk was not pursued in this project. However, the absence of low power and hot standby states are acknowledged in the uncertainty analysis. A series of sensitivity analyses have been developed to assess the potential contributions from these states. Refer to Section 10.3 for a summary of the sensitivity analyses performed for this project.

4.2.2 Shutdown Accident Scenario Prioritization

For shutdown accident scenarios, a scenario prioritization scheme was developed to focus the accident sequence analysis work on the most risk-significant conditions and accidents. The prioritization scheme considered all combinations of shutdown initiating events and POSs. Shutdown POSs include POS 3, 4, 5A, 5B, 6, 7, 8E, 8L, 9, 10, 11, 12, and 13. The shutdown POSs differ significantly from the at-power model in terms of plant conditions and the types of events that can challenge safety functions.

There are six initiating event categories that are considered for shutdown accident scenarios:

- Loss of RHR
- Loss of inventory
- Overdraining RCS level
- Loss of offsite power
- Loss of power to 4,160V AC bus
- Loss of NSCW

A set of criteria were developed to prioritize the shutdown accident scenarios. This prioritization was developed to increase assurance that the highest priority scenarios would be analyzed first and be supported by realistic, plant-specific thermal hydraulic analyses. The following three factors were considered to prioritize potential LPSD accident scenarios in terms of their risk significance: containment status, time to reach saturation temperature, and event frequency. The criteria for prioritizing the accident scenarios are discussed below.

- **Containment status:** High priority is given to accident scenarios with the containment equipment hatch open at the time of the initiating event. Conditions when the containment is open are expected to be the most significant contributors to releases. Containment openings during shutdown can include the equipment hatch, the personnel airlock, containment ventilation purge system, and other penetrations. The equipment hatch status was selected to represent this criterion, because the time required to close the hatch is expected to be the most limiting factor in establishing containment integrity.
- **Time to reach saturation temperature:** High priority is given to accident scenarios with a time to reach saturation temperature of less than two hours. The time available to respond to and mitigate an initiating event can be limited under some shutdown conditions (e.g., reduced RCS inventory). The time to reach saturation temperature is used as a factor for determining priority because the RHR system will not be placed in operation if boiling is imminent. Once saturation temperature is reached, alternative means of decay heat removal must be pursued.
- **Event frequency:** High priority is given to accident scenarios where the product of the initiating event frequency, the average fraction of the year spent in outage, and the average fraction of the outage spent in the POS, is greater than 10^{-4} per year.⁷ This provides a criterion to focus on the highest frequency events.

The shutdown accident scenarios were organized into two tiers of priority based on the number of criteria met. The first tier includes scenarios that meet all three of the criteria. The second tier includes scenarios that meet two of the three criteria. The accident scenarios that were incorporated into the model were selected from these two groups.

Priority Tier 1: All Three Criteria Are Met

The scenarios that meet all three criteria occur in POS 5B and POS 9. The scenarios are:

- POS 5B scenarios: Loss of RHR, loss of offsite power, and loss of power to 4,160V AC bus
- POS 9 scenarios: Loss of RHR, loss of inventory, loss of offsite power, and loss of power to 4,160V AC bus

Priority Tier 2: Two of Three Criteria Are Met

This tier brings many more scenarios into consideration for modeling. The scenarios are grouped below based on the criteria that are met.

⁷ The prioritization approach was applied prior to completion of the initiating event frequency analysis, which is documented in Section 8. For the prioritization approach, estimated initiating event frequencies are taken from the EPRI analysis of loss of decay heat removal initiating event frequencies (EPRI, 2001). For most initiating events, the updated frequencies in Section 8 do not significantly affect the prioritization approach. However, the updated loss of inventory frequencies do have a significant effect on what POSs are prioritized. With the updated loss of inventory frequencies, POSs 8E, 8L, and 9 would no longer meet the frequency criterion, while POSs 4, 5A, 11, 12, and 13 would meet the criterion.

High priority for “containment status” and “time to reach saturation temperature”:

- POS 5B scenarios: loss of inventory, loss of NSCW
- POS 9 scenarios: loss of NSCW

High priority for “containment status” and “event frequency”:

- POS 5A scenarios: loss of offsite power, loss of power to 4,160V AC bus
- POS 8E scenarios: loss of RHR, loss of inventory, loss of offsite power, loss of power to 4,160V AC bus
- POS 8L scenarios: loss of RHR, loss of inventory, loss of offsite power, loss of power to 4,160V AC bus
- POS 11 scenarios: loss of offsite power, loss of power to 4,160V AC bus
- POS 12 scenarios: loss of RHR, loss of offsite power, loss of power to 4,160V AC bus

High priority for “time to reach saturation temperature” and “event frequency”:

- POS 6 scenarios: overdrain entering mid-loop, loss of offsite power, loss of power to 4,160V AC bus
- POS 10 scenarios: overdrain entering mid-loop, loss of offsite power, loss of power to 4,160V AC bus

The accident scenarios meeting the Tier 1 or Tier 2 criteria are considered as candidates for incorporating into the model. The actual accident scenarios selected for modeling are identified in Table 4-2. The modeled scenarios include all the Tier 1 priority scenarios and many of the Tier 2 priority scenarios. The Tier 2 scenarios that are not modeled are the POS 5A, POS 11, and POS 12 scenarios for loss of power to 4,160V AC bus. Initial model results showed the CDF contributions from loss of power to 4,160V AC bus were not significant for other POSs. Therefore, the loss of power to 4,160V AC bus events were de-emphasized to focus on completing initiating events expected to be more significant contributors, e.g., loss of offsite power. While some scenarios were excluded from the quantitative model, a sensitivity study was completed to provide estimates of their CDF contributions (see Section 10.3.1).

An additional qualitative assessment of potential accident scenarios was performed to increase assurance that no significant scenarios were excluded. The reasons for including additional scenarios are discussed below.

- There is considerable uncertainty in estimating the loss of inventory frequency, which is based on only a small number of events. Loss of inventory is considered for modeling for some POSs (POS 5A and POS 7) that do not meet the event frequency criteria due to the uncertainty in the frequency. After the initial screening analysis, review of the loss of inventory initiating event data revealed that the most common type of loss of inventory is associated with POSs with RCS intact. The POS 5A loss of inventory scenario was modeled to ensure this type of loss of inventory was represented in the model. Also,

Table 4-2 Accident Scenarios Selected for Modeling

IE Categories:	Loss of RHR	Loss of Inventory	Overdrain RCS Level	Loss of 4,160V AC Bus	Loss of Offsite Power	Loss of NSCW
Shutdown POS						
POS 3 Mode 4 cooldown	modeled	not modeled	N/A	not modeled	modeled	not modeled
POS 4 Mode 5 cooldown	modeled	not modeled	N/A	not modeled	modeled	not modeled
POS 5A Pressurizer solid	modeled	modeled	N/A	not modeled	modeled	not modeled
POS 5B Draining RCS	modeled	modeled	modeled	modeled	modeled	modeled
POS 6 Mid-loop (pre-refuel)	modeled	modeled	modeled	modeled	modeled	modeled
POS 7 Filling refuel cavity	modeled	modeled	N/A	modeled	modeled	not modeled
POS 8E Refuel cavity flooded (pre-refuel)	modeled	modeled	N/A	modeled	modeled	modeled
POS 8L Refuel cavity flooded (post-refuel)	modeled	modeled	N/A	modeled	modeled	modeled
POS 9 Draining refuel cavity	modeled	modeled	modeled	modeled	modeled	modeled
POS 10 Mid-loop (post-refuel)	modeled	modeled	modeled	modeled	modeled	modeled
POS 11 Vacuum fill RCS	modeled	not modeled	N/A	not modeled	modeled	not modeled
POS 12 Mode 5 heatup	modeled	not modeled	N/A	not modeled	modeled	not modeled
POS 13 Mode 4 heatup	modeled	not modeled	N/A	not modeled	modeled	not modeled

after modeling the POS 8E loss of inventory scenario, the POS 7 loss of inventory scenario was added because there is less inventory available during POS 7, which could negatively impact the response to the event. While the frequency for large losses of inventory during POS 7 and POS 8E (with the refueling cavity partially or fully flooded) may be rare, the potential consequences of a loss of inventory could be severe.

- Mid-loop is considered a high-risk plant evolution, so all accident scenarios during these POSs (POS 6 and POS 10) were modeled, even though some do not meet the priority criteria due to relatively short durations and containment being closed. Also, all accident scenarios were modeled for the two POSs that involve draining prior to mid-loop (POS 5B and POS 9), since the lowered RCS inventory level, while higher than mid-loop level, still limits the time available for operator response and containment is open at this time.
- The loss of RHR modeling provides a basis for all other shutdown initiating event categories. All the shutdown initiating events have similar impacts on the decay heat removal function. The loss of RHR is modeled for all the shutdown POSs to provide a basis for modeling the impacts of other initiators.

- The prioritization scheme informed the initial versions of the model. However, other factors were considered in later versions of the model. These included results from initial model revisions, feedback from reviewers, and efficiencies that were gained after completing parts of the model. After the initial version of the model, loss of offsite power scenarios were identified as significant contributors, so they were modeled for all shutdown POSs. Some additional accident scenarios were added to the model because similarities to existing modeled scenarios made it easy to add them with adjustments to some of the modeling assumptions. These included accident scenarios for POS 7 (partially flooded refueling cavity), which has conditions similar to POS 8E (refueling cavity filled). Lastly, preliminary results and reviewer comments suggested maintenance configurations during POS 8E and POS 8L could have a significant impact on LONSCW scenarios, so these scenarios were added to the model even though they were not identified in the prioritization scheme.

Another consideration in the scenario prioritization was the NRC-sponsored expert panel ranking exercise on LPSD POS and POT risk contributions, which is documented in NUREG/CR-7265 (NRC, 2020a). The outcome of the ranking exercise was not completed prior to the initial model version of the LPSD PRA and, therefore, was not available to inform the development of the scenario prioritization scheme. However, the expert ranking was compared against the outcome of the scenario prioritization scheme to ensure there were no omissions of potentially important contributors. One of the key similarities between the two ranking approaches is that both identified the importance of POSs with reduced RCS inventory (i.e., POS 5B, POS 6, POS 9, and POS 10).

4.3 Event Tree Models

Event trees are used to model the safety functions, systems, and operator actions that can be used to mitigate the initiating events. The event tree models have been developed for six shutdown initiating event categories: loss of RHR, loss of inventory, overdraining RCS level, loss of offsite power, loss of power to 4,160V AC bus, and loss of NSCW. The loss of inventory and overdraining RCS level initiating event categories are further subdivided to account for variations in the initiating events and POS-specific differences in the plant response. The subdivisions for these categories are described below.

Loss of inventory sub-categories:

- LOINV-OP – loss of inventory due to RCS pressure excursion resulting in lifting of pressure relief valves. These events are applicable only in POSs with the RCS intact. This scenario uses a representative loss of inventory flow rate of 500 gpm. The representative flow rate is informed by an accident scenario simulation, and the basis is discussed in Section 5.3.2.
- LOINV-SP – loss of inventory due to a small break or pressure boundary failure resulting in a spray event with break flow rate up to 100 gpm.
- LOINV – loss of inventory due to a break or pressure boundary failure resulting in a break flow in the range of 100 to 2000 gpm.
- LOINV-LG – loss of inventory due to a large break or pressure boundary failure resulting in break flow greater than 2000 gpm.

Overdraining RCS level sub-categories:

- LOINV-DS – loss of inventory due to human errors resulting in inadvertent flow diversions and reducing the RCS inventory level available for cooling the core.
- ODML-EARLY – Overdraining the RCS level when entering mid-loop condition prior to refueling.
- ODML-LATE – Overdraining the RCS level when entering mid-loop condition after refueling.

A total of 11 event tree models were developed to represent each of the initiating event categories and sub-categories. The event trees are shown in Table 4-3.

Table 4-3 Event Trees for Shutdown Initiating Events

Event Tree Name	Event Tree Description
1-SD-LO4160V	Loss of electrical power to class 1E 4,160V AC bus
1-SD-LOINV	Loss of inventory
1-SD-LOINV-DS	Loss of inventory – displacement event
1-SD-LOINV-LG	Loss of inventory – large leak
1-SD-LOINV-OP	Loss of inventory – RCS at pressure
1-SD-LOINV-SP	Loss of inventory – spray event
1-SD-LONSCW	Loss of NSCW
1-SD-LOOP	Loss of offsite power
1-SD-LORHR	Loss of RHR cooling
1-SD-ODML-EARLY	Overdrain entering mid-loop – before refueling
1-SD-ODML-LATE	Overdrain entering mid-loop – after refueling

The event trees are constructed to model the plant response to the initiating events as applicable to the relevant POTs and POSs. The plant response can vary depending on the specific POT and POS conditions, and the models account for these differences. The duration and frequency of the POTs and POSs are also accounted for in the model structure. The following four levels of information are defined as necessary aspects of the event tree model.

1. Initiating event – Identifies the initiating event category and provides the annual frequency of the event.
2. Plant outage type – Identifies the type of outage being modeled and applies a factor to account for the average annual fractional time spent in the POT.
3. Plant operating state – Identifies the POS that the plant occupies when the initiating event occurs and applies a factor to account for the average fractional duration of the POS during the outage.
4. Plant response tree – Models the systems and operator actions used to respond to the initiating event, accounting for the POT- and POS-specific conditions.

The above information is structured in a series of event trees, transfers between event trees, and rules that outline the linking of accident sequences and POS-specific changes and

substitutions. The entry point for each accident scenario is one of the event trees identified in Table 4-3. These event trees provide the initiating event information and include branches representing each of the modeled POTs. Figure 4-2 shows an example for the loss of RHR initiating event. Note, as previously mentioned in Section 1.2, only POT-4 (refueling outage) is quantitatively modeled in this project. As such, the end states for the other POT branches in Figure 4-2 are preceded by an “@” (in SAPHIRE, end states beginning with an “@” are not included when the model is quantified).

The POT branches transfer to event trees that contain the POS information. An event tree branch is developed for each POS that is modeled for that initiating event. An example POS event tree is shown in Figure 4-3.

The POS event tree transfers to the plant response event trees. The plant response event trees contain the models for the plant systems and operator actions that are used to respond to the initiating event. A standard set of plant response event trees are developed for each modeled initiating event. The plant response event trees are varied for each POS by substituting top events with POS-specific fault trees and adjusting basic event settings using flag sets. Note, in some cases, more than one POS will be transferred to the same plant response event tree, due to similarity in plant response for these POSs. The plant response event trees are discussed further in the sections below.

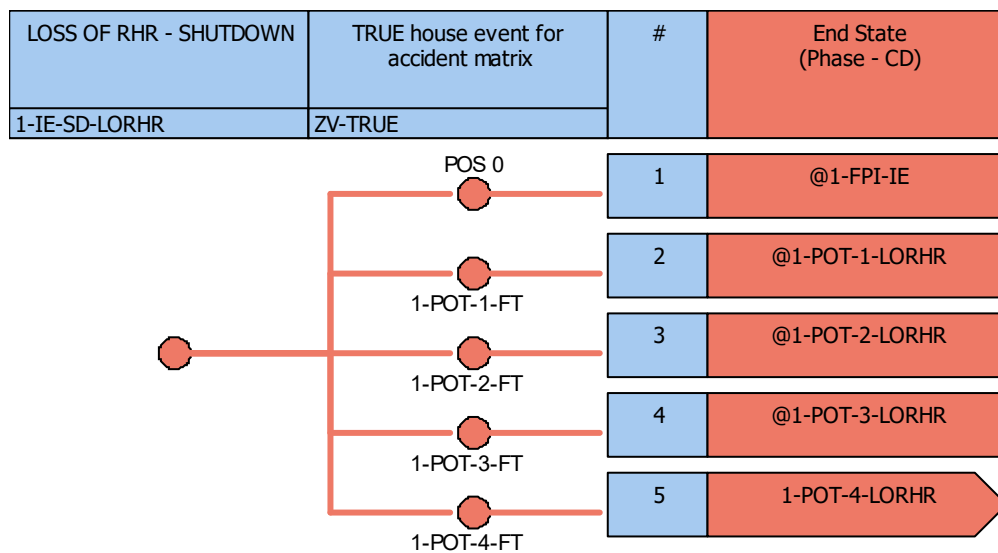


Figure 4-2 LORHR Initiating Event and POT Event Tree

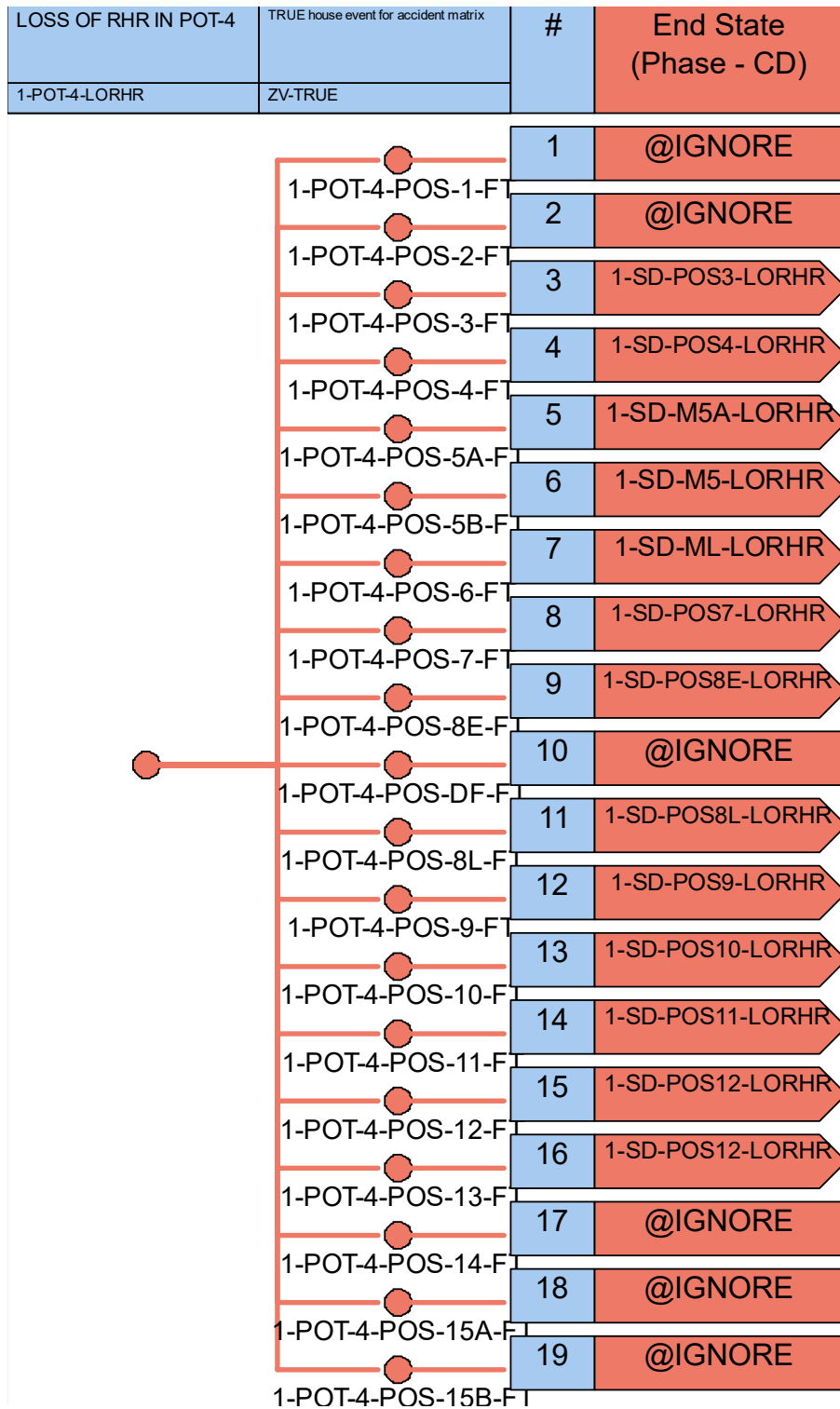


Figure 4-3 LORHR Initiating Event and POT Event Tree

4.3.1 LORHR Plant Response Event Tree

The LORHR event tree models the response to losing RHR cooling during shutdown plant operating states. The event tree is applicable to POSs where one or two RHR trains are operating and providing core cooling. The same base event tree is used for initiating events resulting in loss of RHR cooling due to loss of cooling water to the RHR heat exchanger(s), failure of the RHR pump(s), or isolation or diversion of the RHR flow path.

Figure 4-4 shows the modeled event tree plant response to a loss of RHR initiating event. The primary mitigation functions following an initiating event during RHR system operation include recovery of the failure associated with the initiating event (i.e., recovery of a previously operating RHR train), initiation of the standby RHR train (if one was previously in standby), secondary heat removal, aligning alternate cooling through bleed and feed, RWST replenishment,⁸ and long-term establishment of RHR cooling or recirculation flow from the sump.

The reference plant's AOP governing the operator response to a loss of RHR was reviewed to inform the development of the LORHR event tree. The AOP is divided into the following four sections that are applicable to different POSs:

- The first section of the LORHR AOP is applicable to Mode 4 or Mode 5 with RCS level in the pressurizer range and with no steam generator nozzle dams installed. This section directs operators to restore RHR cooling or use the secondary heat sink for RCS cooling. This section is applicable to POSs 3, 4, 5A, 12, and 13.
- The second section of the LORHR AOP provides the operator response to a loss of RHR occurring in Mode 5 or Mode 6 with RCS level below the pressurizer range. This section is applicable to POSs 5B, 6, 7, 9, and 10.
- The third section of the LORHR AOP directs the operator response to loss of RHR while the reactor cavity is flooded, and the fuel transfer canal is open. This section applies to POSs 7 (for the POS 7 modeled conditions described in Section 4.4.6), 8E, and 8L.
- The fourth section of the LORHR AOP applies to loss of RHR during vacuum refill conditions (POS 11).

The second section of the LORHR AOP is discussed further below. Focus is placed on this section because during the applicable POSs, there are several plant conditions that contribute to elevated risk significance related to a loss of RHR event. These conditions include unavailability of the secondary side heat sink for core cooling, reduced RCS inventory levels, and relatively short time to reach boiling temperature for POSs prior to refueling (POSs 5B and 6) and after refueling (POS 10).

⁸ Top event 1-SD-RWST-REFL-1D represents refill of the RWST. Although the description is worded in terms of failure (i.e., FAILURE TO REFILL RWST), the event tree uses the same logic structure as other top events with the up branch representing success of the function and the down branch representing failure. No credit is given for RWST refill due to lack of procedure-guided direction on aligning sufficient makeup capacity. Top event 1-SD-LPR was included to consider alignment of sump recirculation using RHR pumps. However, there were no shutdown conditions identified where sump recirculation would be aligned. Both of these top events are retained in the event tree as a way of documenting that the represented functions were considered, though ultimately not credited.

In the LORHR AOP, the operators are directed to isolate RHR letdown and to adjust the normal charging to maintain RCS water level and also to isolate any known RCS leakage. The operators must immediately trip the RHR pumps if pump cavitation occurs or if water level is less than one foot above the hot leg centerline. If level is above one foot above the hot leg centerline, then they are directed to start the standby RHR pump, if it is not already operating.

If the time to boil is determined to be less than 60 minutes, then the containment is to be evacuated and isolated, and the containment coolers are to be set to low speed.

If RCS temperature (as indicated by core exit thermocouples) is greater than 200°F, then the operators are directed to refill the RCS using one of the safety injection pumps (SIPs) via hot leg injection or one of the centrifugal charging pumps (CCPs) via cold leg injection either to the 50-percent pressurizer level or until RHR cooling is restored.

If the RCS temperature is less than 200°F and the RCS water level is less than one foot above the hot leg centerline, then the operators are directed to try gravity feed from the RWST, through any of several paths, but preferably through the RHR pumps to the cold legs.

Once RCS refill is accomplished, the operators attempt to restart an RHR pump for shutdown cooling.

Table 4-4 lists the key steps taken in response to the LORHR initiating event, including the reference to the relevant steps from the plant procedure.

Table 4-4 Key Steps in Response to Loss of RHR in Mode 5 or Mode 6 with RCS Level Below the Pressurizer Range

Description of Action or Step Taken
Check RHR Status (RCS leakage) • Are RHR pumps running/should they be stopped? • Is RHR system intact or leaking?
Check Plant Status (time to boil) • Determine time to boiling based on existing condition. • If less than 60 minutes, evacuate personnel from containment, isolate containment, and align containment cooling fans for low speed operations.
Establish makeup to RCS • If RCS temperature is above 200°F, align SIP for hot leg injection, and if temperature continues to rise, align CCP for cold leg injection. • If RCS temperature is below 200°F and level is within one foot of the hot leg centerline, establish RWST gravity feed to RCS.
Restore RHR flow • Put at least one RHR train in service.

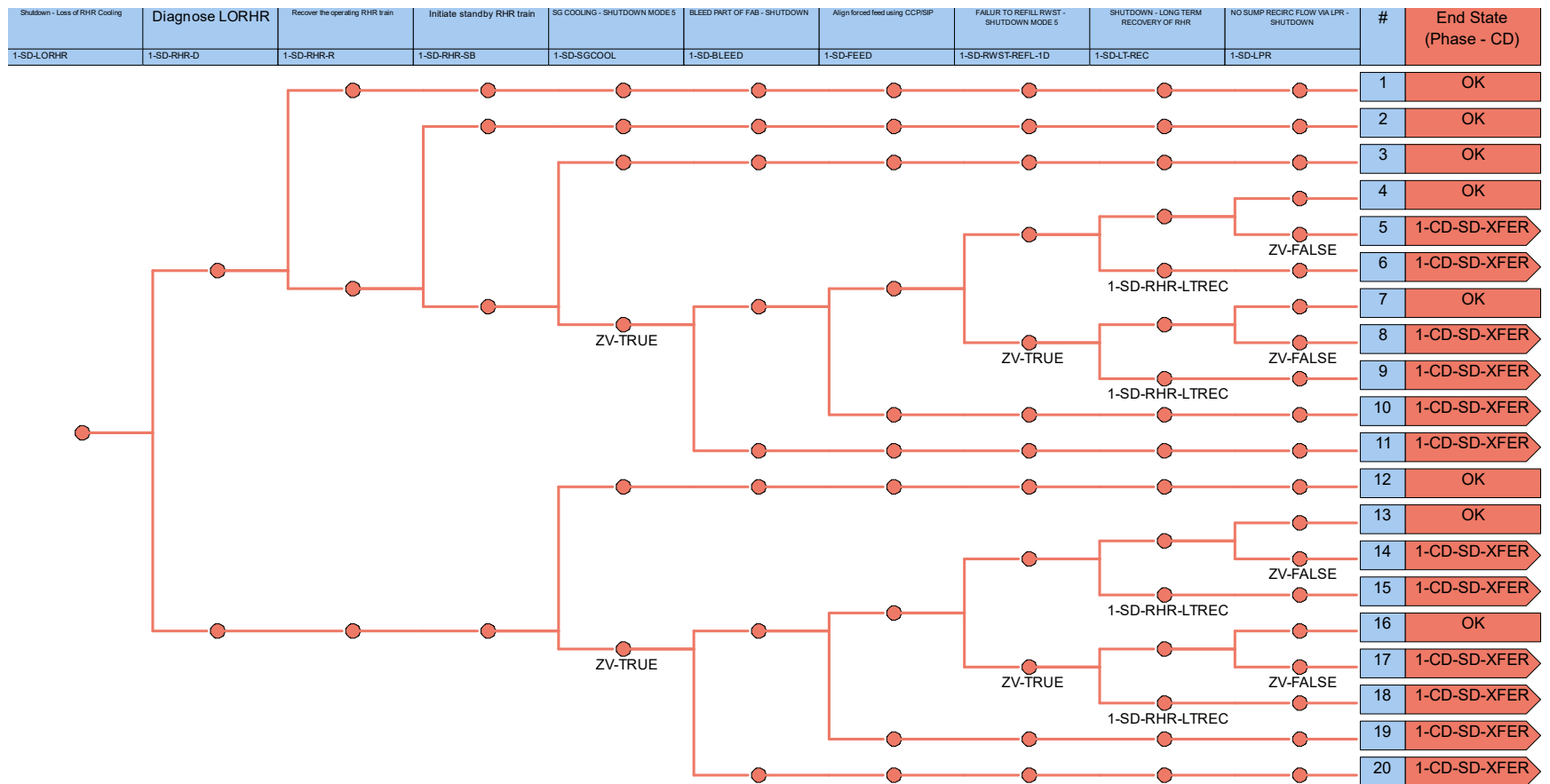


Figure 4-4 LORHR Plant Response Event Tree

LORHR Top Event Descriptions

The following provides a brief description of each LORHR event tree top event and the related fault trees. Each node in the actual event trees in the model is adjusted for the POS-specific conditions, for example, equipment availability, time available for action, and failure criteria. The modeling approach and assumptions for each top event are described below.

1-SD-RHR-D: This top event models the operators' diagnosis of the LORHR event. The operator action basic event models the operator failure to diagnose the initiating event based on the available indications of RHR failure. The diagnosis and the action to restore RHR (modeled in the subsequent event tree node) must be completed prior to RCS level reaching one foot above the hot leg centerline. The time available varies depending on the POS, and a portion of the available time is apportioned for diagnosis. Estimation of human error probabilities for failure to diagnose LORHR is discussed in detail in Section 7.

1-SD-RHR-R: This top event models the recovery of the operating RHR train(s). Failure to recover the RHR train is modeled by an operator failure event or failures of the RHR system, flow path, or support systems. The POS conditions and nature of the initiating event influence the likelihood of recovering the RHR train. If both RHR trains are normally operating prior to the initiating event, then this top event models recovery of either one of the two RHR trains. If one RHR train is operating and one in standby prior to the initiating event, then this top event models recovery of the train that was failed by the initiator. Depending on the nature of the failure, the operating train may or may not be recoverable. The RHR train must be recovered prior to RCS level reaching one foot above the hot leg centerline. Note, RHR train B is assumed to be operating if only one train is in operation.

1-SD-RHR-SB: This top event models aligning the standby RHR train to provide core cooling. The standby train (assumed to be RHR train A) may be aligned for shutdown cooling, available but not aligned, or out of service for maintenance, depending on the POS. The standby RHR train must be recovered prior to RCS level reaching one foot above the hot leg centerline. If both RHR trains are operating and failed by the initiating event, then this top event is not used. Recovery of either one of the two RHR trains is modeled by top event 1-SD-RHR-R.

1-SD-SGCOOL: This top event models using steam generators for RCS cooling. Heat removal using the steam generators is possible during some shutdown POSs. For successful steam generator cooling, secondary inventory must be available, and operators must establish AFW to maintain steam generator level. Once the RCS is vented, then steam generators are no longer effective for heat removal, regardless of steam generator inventory level. This top event is not used for POSs where the RCS is vented or where RCS pressure and level would not be sufficient for heat removal using steam generators (POSs 5B, 6, 7, 8E, 8L, 9, 10, and 11.)

1-SD-BLEED: This top event models venting of the RCS by opening one of two pressurizer PORVs. The venting of a PORV depressurizes the RCS to support core cooling by forced injection, gravity feed from the RWST, or re-establishing RHR cooling. During POSs where the RCS is vented (e.g., pressurizer safety valves removed) this top event is not relevant (i.e., this event is set to 'FALSE' and success is guaranteed). During POSs where RCS is not vented, RCS level is in the pressurizer range, and steam generator loops are not isolated, the reference plant's procedural guidance focuses on restoring RHR cooling and using steam generators for RCS cooling. There is no procedure step for establishing a bleed path. For these POSs (i.e., POSs 3, 4, 5A, 12, and 13), the top event is set to 'TRUE' and failure is guaranteed. For

POS 11 during vacuum refill, the reference plant's relevant procedure does specify steps to break RCS vacuum and establish a vent path. POS 11 is the only POS where the top event fault tree logic is evaluated.

1-SD-FEED: This top event fault tree models establishing injection paths using SIPs or CCPs to control RCS inventory.⁹ The fault tree top event includes an operator action to establish forced feed paths. The success criteria are dependent on the POS-specific decay heat load. The specific criteria are discussed further under system success criteria in Section 5.4. Successful forced feed must be established before the RCS water boils off to two-thirds of the active fuel height.

Once SIP or CCP feed is established, it can continue to operate until the RWST inventory is depleted. The RWST holds approximately 730,000 gallons of water. It is assumed that injection from the RWST will continue until the RWST low-low level alarm occurs when 240,000 gallons are remaining. With the above considerations, the injections with SIPs or CCPs can last for between four and eight hours during the early POSs prior to refueling (POS 3 through POS 6). After refueling, the injection with SIPs or CCPs can be extended up to 24 hours with reduced injection flow.

1-SD-RWST-REFL-1D: This top event fault tree models refilling the RWST to extend the available inventory for injection. The fault tree includes three basic events: an operator action, a random failure (rupture) of the RWST, and an undeveloped event representing component failures (i.e., hardware failures preventing refilling of the RWST). Successful refill of RWST will extend the time of injection beyond 24 hours. However, the makeup flow capability is much less than injection flow requirements needed to maintain core cooling during the early parts of the outage. Also, there is a lack of procedural guidance for emergency makeup for most POSs. Therefore, this action is unlikely to be successful; and the event is set to 'TRUE.' However, this top event is retained in the event tree for future consideration. When the reactor cavity is flooded, there are some conditions where operators may be directed to transfer a limited volume of water from the SFP to the RWST. Credit for RWST refill can be considered under these conditions.

1-SD-LT-REC: This top event represents the late recovery of RHR cooling. The model includes a human failure event that is adjusted based on the POS-specific time available for recovery. The time available for recovery of RHR is based on the time that SIP or CCP injection is available to provide alternate core cooling and maintain RCS temperature less than 200°F.

1-SD-LPR: This top event models sump recirculation given feed and bleed was successful. If RHR cooling with normal suction from the RCS is recovered, then recirculation from the sump is not required to reach a stable state. Therefore, this top event is effectively ignored by setting it to 'FALSE' for all sequences; however, it is retained in the event tree for future modeling consideration if there are unique conditions where sump recirculation would be required. Also, contributing to the decision to bypass this top event is the assumption that the sump is unavailable during much of the outage after protective covers are installed over sump screens during Mode 5.

⁹ Control of RCS inventory is also possible by means of gravity drain from RWST. Gravity feed is not credited for LORHR, LO4160V, and some LOOP sequences due to the procedure giving preference to establishing feed with SIPs or CCPs. Gravity feed is modeled for other sequences (e.g., LOINV and station blackout) where the gravity feed procedure steps are more likely to be relevant under those plant conditions.

4.3.2 LOINV Plant Response Event Tree

The LOINV initiating event is divided into four categories representing different types of inventory loss initiating events applicable under different POS conditions. The four categories are:

- SD-LOINV-OP – loss of inventory due to RCS pressure excursion resulting in lifting of pressure relief valves. These events are applicable only in POSs with the RCS intact. This scenario is modeled for POS 5A.
- SD-LOINV-SP – loss of inventory due to a small break or pressure boundary failure resulting in a spray event with break flow rate up to 100 gpm. This scenario is modeled for POSs 5B, 6, 9, and 10.
- SD-LOINV – loss of inventory due to a break or pressure boundary failure resulting in a break flow in the range of 100 to 2000 gpm. This scenario is modeled for POSs 5B, 6, 9, and 10.
- SD-LOINV-LG – loss of inventory due to a large break or pressure boundary failure resulting in break flow greater than 2000 gpm. This scenario is modeled for POSs 7, 8E, and 8L.

The same LOINV event tree structure is applied for all categories. The modeling difference between the categories is attributed to the different leak flow rates associated with the loss of inventory. The three representative loss of inventory categories are defined to incorporate the POS operating conditions, insights from operating experience data, and consideration for how the failure could challenge key safety functions. For all categories, the representative loss of inventory scenario is associated with a failure in an operating RHR train suction line. In defining the scenarios in this way, these events challenge both the RCS inventory control and decay heat removal functions. This assumption is conservative for POSs where only one RHR train is operating, but ultimately these LOINV scenarios do not have significant contributions to the overall shutdown CDF.

The loss of inventory modeling also considers the potential for losses outside containment, which would result in a containment bypass scenario. All loss of inventory event sequences where the initial action to isolate the loss is failed are considered for potential bypasses. An analysis of the RHR system pipe lengths is used to determine the likelihood of the loss occurring inside or outside of containment (see Section 8.2, Loss of Inventory Outside Containment). If the initial diagnosis and isolation are successful, then the event tree sequences are not considered for potential containment bypass.

Figure 4-5 shows the expected plant response to the loss of inventory initiating event. The primary mitigation functions following a loss of inventory initiating event during RHR system operation include isolation of the leakage, RCS makeup, recovery and restoration of RHR (e.g., starting/recovering one RHR train), secondary heat removal, bleed and feed, RWST replenishment, and long-term establishment of RHR cooling. The event tree sequences with failure of the initial diagnosis and isolation are considered for possible containment bypass scenarios, which are carried forward for analysis in the Level 2 PRA.

The reference plant's AOP governing the operator response to a loss of RCS inventory was reviewed to inform the development of the LOINV event tree. The AOP is divided into the following three sections based on the plant conditions at the time the leakage occurs:

- The first section of the LOINV AOP is applicable during Modes 1, 2, and 3 with RCS pressure greater than 1000 psig.
- The second section of the LOINV AOP is applicable during Mode 3 with RCS pressure less than 1000 psig, and Mode 4.
- The third section of the LOINV AOP is applicable during Mode 5.

The potential impact on RHR operation due to a loss of inventory event may also invoke AOP for loss of RHR. Operators will likely be referencing both AOPs in response to the event. During most shutdown POSs, when RHR is the sole means of shutdown cooling, the response to a loss of inventory event is specified by the third section of the LOINV AOP and the second section of the LORHR AOP.

The first step in the LOINV AOP will direct the operator to check plant conditions. If RCS inventory level is not in the pressurizer indication range, then the operator is directed to go to the LORHR AOP. Other relevant steps related to loss of inventory will direct operators to monitor RHR pumps for cavitation, maintain pressurizer level using one CCP, and monitor RWST level.

In the LORHR AOP, the operators are directed to isolate RHR letdown and to adjust the normal charging to maintain RCS water level and also to isolate any known RCS leakage. The operators must immediately trip the RHR pumps if pump cavitation occurs or if water level is less than one foot above the hot leg centerline. If level is above one foot above the hot leg centerline, then they are directed to start the standby RHR pump, if it is not already operating.

If the time to boil is determined to be less than 60 minutes, then the containment is to be evacuated and isolated, and the containment coolers are to be set to low speed.

If RCS (core exit) temperature is greater than 200°F, then the operators are directed to refill the RCS using one of the SIPs via hot leg injection or one of the CCPs via cold leg injection either to the 50-percent pressurizer level or until RHR cooling is restored.

If the RCS temperature is less than 200°F and the RCS water level is within one foot of the hot leg centerline, then the operators are directed to try gravity feed from the RWST, through any of several paths, but preferably through the RHR pumps to the cold legs.

Once RCS refill is accomplished, the operators attempt to restart an RHR pump for shutdown cooling using the cold legs.

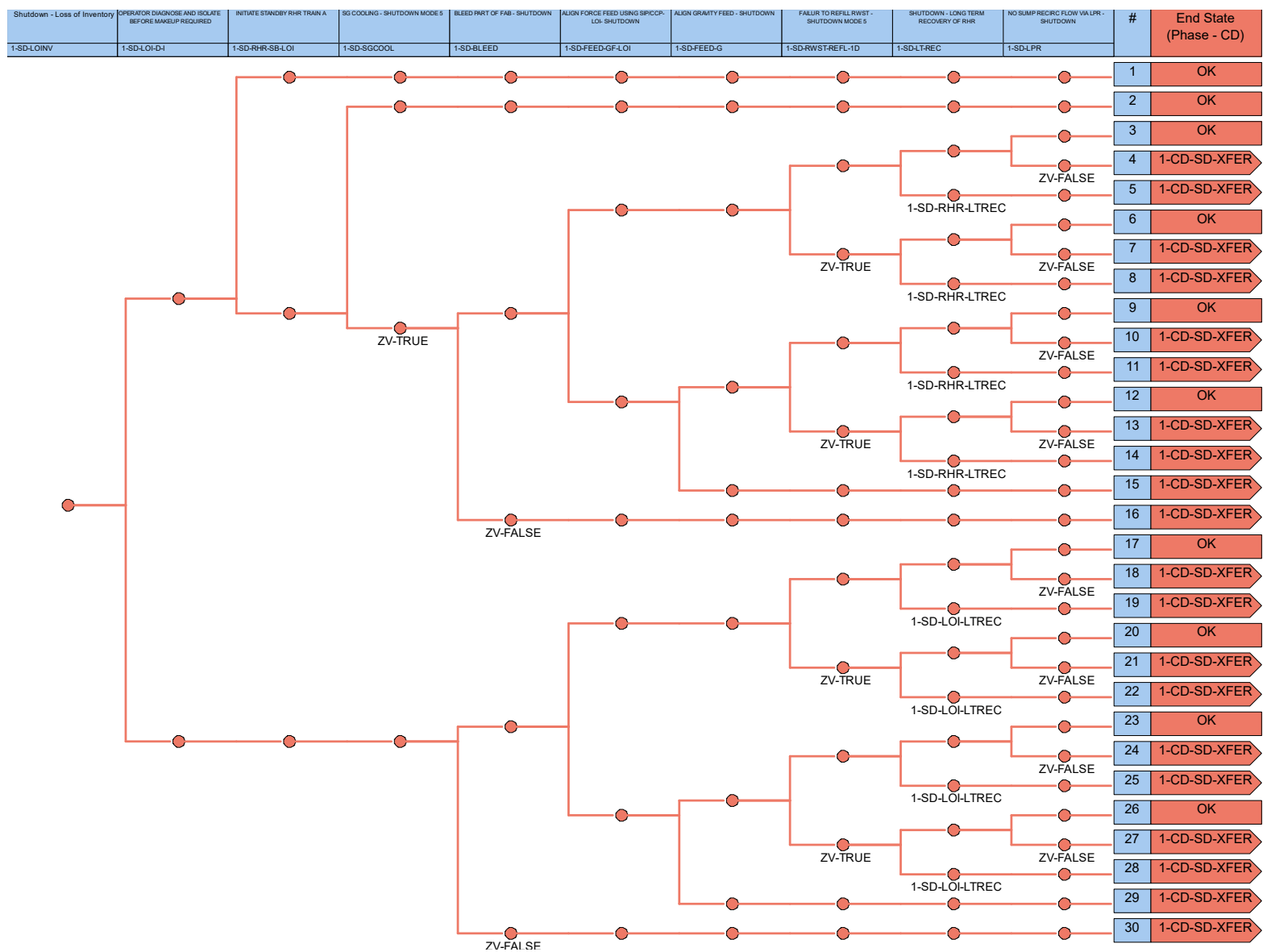


Figure 4-5 LOINV Plant Response Event Tree

Table 4-5 lists the key steps taken in response to the LOINV initiating event, including the reference to the relevant steps from the plant procedure.

Table 4-5 Key Steps in Response to Loss of Inventory

Description of Action or Step Taken	
Check plant conditions in Mode 5 • RCS inventory in pressurizer indication range, and • SG nozzle dams NOT installed • If conditions not met, then go to LORHR AOP	
Monitor operating RHR pump NOT cavitating.	
Isolate letdown, raise charging flow, and align one CCP if needed to maintain pressurizer level greater than 9%.	
Monitor RWST level and makeup as necessary.	
Check RHR status (RCS leakage) • Are RHR pumps running/should they be stopped? • Is RHR system intact or leaking? • Control RCS inventory • Close RHR let down • Verify RHR Recirc. to RWST is isolated • Identify and isolate any RCS leakage	
Check plant status (time to boil) • Determine time to boiling based on existing condition. • If less than 60 minutes, evacuate personnel from containment, isolate containment, and align containment cooling fans for low speed operations.	
Establish makeup to RCS • If RCS temperature is above 200°F, align SIP for hot leg injection, and if temperature continues to rise, align CCP for cold leg injection. • If RCS temperature is below 200°F and level is within one foot of hot leg centerline, establish RWST gravity feed to RCS. • If RCS level is less than one foot below hot leg centerline, refill RCS by aligning SIP for hot leg injection, and if temperature continues to rise, align CCP for cold leg injection.	
Restore RHR flow • Vent any RHR pump that experiences cavitation. • Put at least one RHR train in service.	

LOINV Top Event Descriptions

The following provides a brief description of each LOINV event tree top event and the related fault trees. Each node in the event tree is adjusted for the POS-specific conditions, for example, equipment availability, time available for action, and failure criteria. The modeling approach and assumptions for each top event are described below.

1-SD-LOI-D-I: This top event models the operators' diagnosis of the LOINV event. The operator action basic event models the operator failure to diagnose and isolate the loss of inventory and control RCS level at one foot above the hot leg centerline or higher. The time available varies depending on the POS.

1-SD-RHR-SB-LOI: This top event models recovering the RHR system for shutdown cooling. The fault tree establishes restoration of one operating RHR train, after the RCS level is stabilized. RHR Train B is assumed unavailable due to the initiating event. RHR train A is modeled for recovery.

1-SD-SGCOOL: This top event models using steam generators for RCS cooling. Heat removal using the steam generators is possible during some shutdown POSs. For successful steam generator cooling, secondary inventory must be available, and operators must establish AFW to maintain steam generator level. Once the RCS is vented, then steam generators are no longer effective for heat removal, regardless of steam generator inventory level. This top event is not used for POSs where the RCS is vented or where RCS pressure and level would not be sufficient for heat removal using steam generators (POSs 5B, 6, 7, 8E, 8L, 9, 10, and 11.)

1-SD-BLEED: This top event models venting of the RCS by opening one of two pressurizer PORVs. The venting of a PORV depressurizes the RCS to support core cooling by forced injection, gravity feed from the RWST, or re-establishing RHR cooling. During POSs where the RCS is vented (e.g., pressurizer safety valves removed) this top event is not relevant (i.e., this event is set to 'FALSE' and success is guaranteed). During POSs where RCS is not vented, RCS level is in the pressurizer range, and steam generator loops are not isolated, the reference plant's procedural guidance focuses on restoring RHR cooling and using steam generators for RCS cooling. There is no procedure step for establishing a bleed path. For these POSs (i.e., POSs 3, 4, 5A, 12, and 13), the top event is set to 'TRUE' and failure is guaranteed. For POS 11 during vacuum refill, the reference plant's relevant procedure does specify steps to break RCS vacuum and establish a vent path. POS 11 is the only POS where the top event fault tree logic is evaluated

1-SD-FEED-GF-LOI: This top event fault tree models establishing injection paths using SIPs or CCPs to control RCS inventory. If operators fail to isolate the loss of inventory before makeup is required, then it is assumed that the RCS level is less than one foot above the hot leg centerline. The procedure directs operators to restore RCS level greater than one foot above the hot leg centerline before RHR cooling can be restored. The fault tree top event includes an operator action to establish forced feed paths. The success criteria are dependent on the POS-specific decay heat load. The specific criteria are discussed further under system success criteria in Section 5.4. Successful forced feed must be established before the RCS water boils off to two-thirds of the active fuel height.

1-SD-FEED-G: This top event fault tree models establishing gravity feed using the RHR or SIP injection paths. The fault tree top event includes an operator action to establish gravity feed. The action is considered dependent on the action to establish forced injection, as both actions are directed toward establishing a similar function. It is expected that gravity feed will only be attempted if forced feed has already failed. This action requires that the RHR or safety injection discharge valves be throttled open locally. A controlled gravity feed can extend the RWST flow for up to six hours, when the cause of the inventory loss is isolated. According to AOP direction, the gravity feed is activated if the RCS level is between within one foot of the hot leg centerline.

It is preferable to establish gravity feed from the RWST through the RHR pumps to the cold legs. The AOP also calls for establishing gravity feed through the SIPs to the cold legs.

1-SD-RWST-REFL-1D: This top event fault tree models refilling the RWST to extend the available inventory for injection. The fault tree includes three basic events: an operator action, a random failure (rupture) of the RWST, and an undeveloped event representing component failures (i.e., hardware failures preventing refilling of the RWST). Successful refill of RWST will extend the time of injection beyond 24 hours. However, the makeup flow capability is much less than injection flow requirements needed to maintain core cooling during the early parts of the outage. Also, there is a lack of procedural guidance for emergency makeup for most POSs. Therefore, this action is unlikely to be successful; and the event is set to 'TRUE' (i.e., guaranteed failure). However, this top event is retained in the event tree for future consideration. When the reactor cavity is flooded, there are some conditions where operators may be directed to transfer a limited volume of water from the SFP to the RWST. Credit for RWST refill can be considered under these conditions.

1-SD-LT-REC: This top event represents the late recovery of RHR cooling. Isolation of the loss of inventory is required for successful recovery of RHR. For those sequences where the initial diagnosis and isolation fails, this top event substitutes a fault tree that includes another human failure event modeling the failure to isolate the loss. The time available for recovery of RHR is based on the time that SIP or CCP injection is available to provide alternate core cooling.

1-SD-LPR: This top event models sump recirculation given feed and bleed was successful. If RHR cooling with normal suction from the RCS is recovered, then recirculation from the sump is not required to reach a stable state. Therefore, this top event is effectively ignored by setting it to 'FALSE' for all sequences; however, it is retained in the event tree for future modeling consideration if there are unique conditions where sump recirculation would be required. Also, contributing to the decision to bypass this top event is the assumption that the sump is unavailable during much of the outage after protective covers are installed over sump screens during Mode 5.

4.3.3 Overdrain Plant Response Event Tree

Two different categories of overdrain event trees are modeled: overdraining entering mid-loop (ODML) and inventory displacement events (LOINV-DS), which occur at reduced RCS inventory levels but prior to entering the mid-loop POSs. Both types of events involve the draining or displacement of RCS inventory below the desired level due to human errors, inaccurate level indication, or a combination of both.

The ODML event tree models the response to draining the RCS below the desired level when entering the mid-loop condition. This event tree is applicable to POSs where mid-loop is entered (POS 6 and POS 10). The event tree name ODML-EARLY refers to an overdrain event when entering mid-loop during POS 6, and ODML-LATE refers to an overdrain when entering mid-loop during POS 10. The lowering of RCS level below the desired elevation can result in RHR pump cavitation and damage.

The LOINV-DS event tree models the response to draining the RCS below the desired level during draining operations near the reactor vessel head level. At this level small changes in level would not be expected to impact RHR cooling. However, if the inadvertent draining is not

recognized and stopped, then the RHR function could be impacted. This event tree is modeled for POSs 5B and 9.

Figure 4-6 shows the expected plant response to the overdraining initiating event. Both the ODML and LOINV-DS event trees use the same structure. This event tree is also similar to the event tree for the loss of inventory (LOINV) event. The primary mitigation functions following an overdrain entering mid-loop condition during RHR system operation include isolation of the leakage, RCS makeup, restoration of RHR (e.g., starting/recovering one RHR train), secondary heat removal, bleed and feed, RWST replenishment, and long-term establishment of RHR cooling.

The governing AOP for the operator response is the reference plant's AOP for loss of RHR cooling. This procedure is also used if RCS leakage, or over-draining, is detected.

In the LORHR AOP, the operators are directed to isolate RHR letdown and to adjust the normal charging to maintain RCS water level and also to isolate any known RCS leakage (drain down). The operators must immediately trip the RHR pumps if pump cavitation occurs or if water level is less than one foot above the hot leg centerline. If level is above one foot above the hot leg centerline, then they are directed to start the standby RHR pump, if it is not already operating.

If the time to boil is determined to be less than 60 minutes, then the containment is to be evacuated and isolated, and the containment coolers are to be set to low speed.

If RCS (as indicated by core exit thermocouples)) temperature is greater than 200°F, then the operators are directed to refill the RCS using one of the SIPs via hot leg injection or one of the CCPs via cold leg injection either to the 50-percent pressurizer level or until RHR cooling is restored.

If the RCS temperature is less than 200°F and the RCS water level is less than one foot above the hot leg centerline, then the operators are directed to try gravity feed from the RWST, through any of several paths, but preferably through the RHR pumps to the cold legs.

Once RCS refill is accomplished, the operators attempt to recover and restart an RHR pump for shutdown cooling using the cold legs.

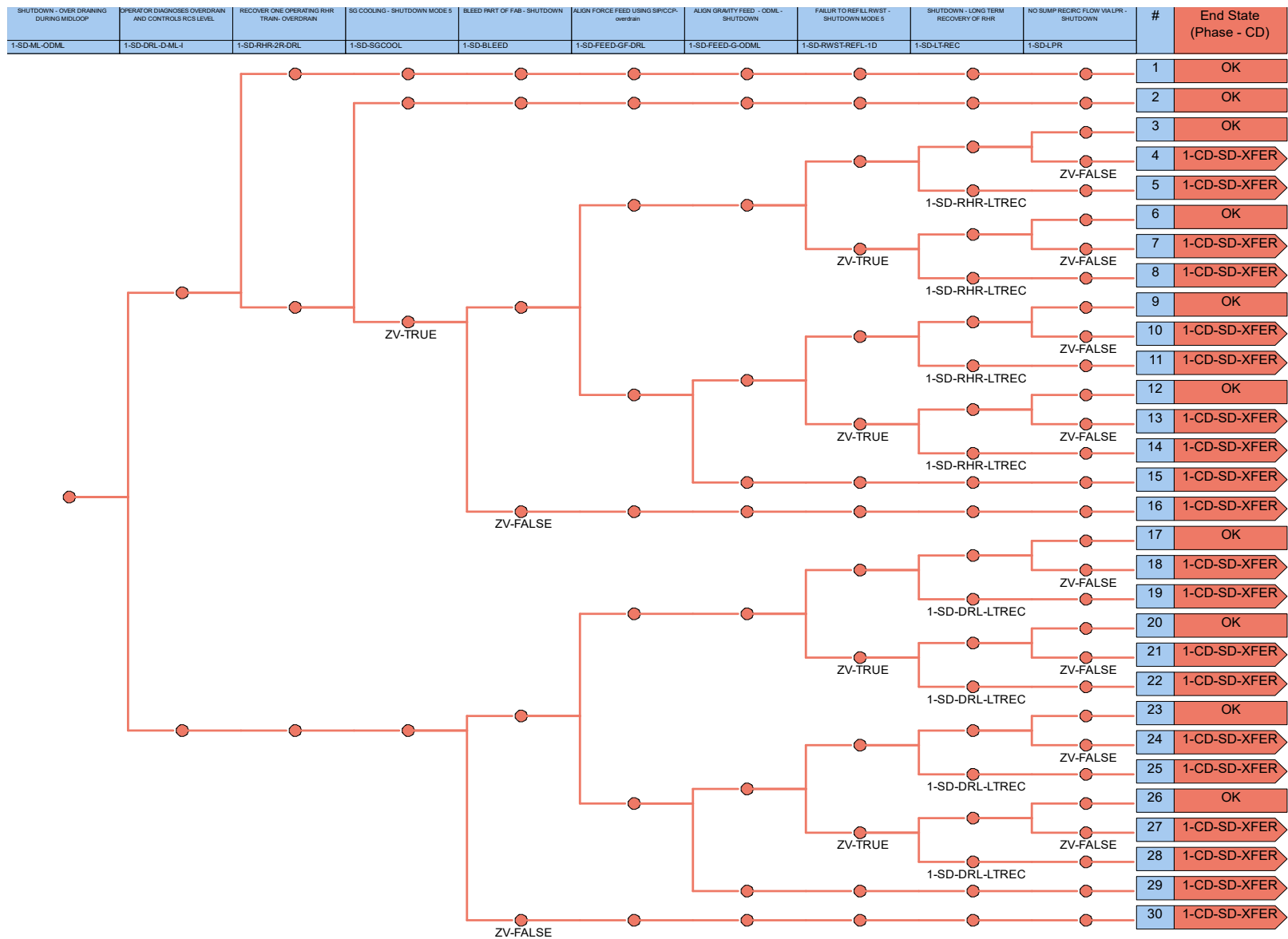


Figure 4-6 ODMIL Plant Response Event Tree

Table 4-6 lists the key steps taken in response to the overdraining initiating event, including the reference to the relevant steps from the plant procedure. These steps are also carried out in response to LOINV. One difference here is that the source of leakage is known, and operators are cautioned and trained to be especially careful during drain down of RCS level.

Table 4-6 Key Steps in Response to Overdraining

Description of Action or Step Taken
Check RHR status (RCS leakage) • Are RHR pumps running/should they be stopped? • Is RHR system intact or leaking? • Control RCS inventory • Close RHR let down • Verify RHR Recirc. to RWST is isolated • Identify and isolate any RCS leakage
Check plant status (time to boil) • Determine time to boiling based on existing condition. • If less than 60 minutes, evacuate personnel from containment, isolate containment, and align containment cooling fans for low speed operations.
Establish makeup to RCS • If RCS temperature is above 200°F, align SIP for hot leg injection, and if temperature continues to rise, align CCP for cold leg injection. • If RCS temperature is below 200°F and level is within one foot of hot leg centerline, establish RWST gravity feed to RCS. • If RCS level is more than one foot below hot leg centerline, refill RCS by aligning SIP for hot leg injection, and if temperature continues to rise, align CCP for cold leg injection.
Restore RHR flow • Vent any RHR pump that experiences cavitation. • Put at least one RHR train in service.

Overdraining Top Event Descriptions

The following provides a brief description of each ODML event tree top event and the related fault trees. Each node in the event tree is adjusted for the POS-specific conditions; for example, equipment availability, time available for action, and failure criteria. The modeling approach and assumptions for each top event are described below.

1-SD-DRL-D-ML-I: This top event fault tree models the operators' diagnosis and isolation of drain down and controlling the RCS level at or higher than one foot above the hot leg centerline.

1-SD-RCS-MAKUP: This top event models the RCS makeup. RCS makeup is accomplished by using one CCP or SIP. The success criteria are one of two CCPs in two of four cold legs or one of two SIPs in two of four hot legs. The fault tree for this event is the same as that for the forced feed top event (1-SD-FEED-GF-DRL). The fault tree models equipment failures based on the equipment that was included in the at-power fault trees. Two CCPs are available, and the SIPs are also available in "pull-to-lock" with their discharge manual values closed.

The probability that the operators fail to establish RCS makeup using the SIP or CCP before the RCS water boils off to two-thirds of the active fuel height is considered to be similar to that for the loss of inventory in POS 5B.

1-SD-RHR-2R-DRL: This top event fault tree models maintaining or restoring the RHR system for decay heat removal after successfully diagnosing the drain down and controlling RCS level. If the initial diagnosis and level control is not successful, then this top event is bypassed.

During the investigation of the drain down, the operators stop the operating RHR train(s). Once the RCS level is maintained higher than one foot above the hot leg centerline, RHR trains can be put in service. The operator action basic event models operator failure to establish one RHR train. The fault tree models RHR equipment failures based on the equipment that was included in the at-power fault trees.

1-SD-SGCOOL: This top event fault tree models RCS cooling using SGs. Because the SGs are isolated (the RCS water level is below the SG tube sheet), no credit is given for RCS cooling. Therefore, this event is set to 'TRUE' (i.e., guaranteed failure).

1-SD-BLEED: This top event fault tree models establishing bleed paths to be used as a part of feed and bleed operation, as an alternate means of providing heat removal. Because the reactor vessel head is in place, but the three pressurizer safety valves are removed and the RCS is vented, this event is assumed to always be successful (i.e., it is set to 'FALSE').

1-SD-FEED-GF-DRL: This top event fault tree models establishing injection paths using SIPs or CCPs as a part of feed and bleed operation, as an alternate means of decay heat removal. The fault tree top event includes an operator action to establish forced feed paths. The success criteria are dependent on the POS-specific decay heat load. The specific criteria are discussed further under system success criteria in Section 5.4.

1-SD-FEED-G-DRL: This top event fault tree models establishing gravity feed using the RHR or SIP injection paths. The fault tree top event includes an operator action to establish gravity feed. The action is considered dependent on the action to establish forced injection, as both actions are directed toward establishing a similar function. It is expected that gravity feed will only be attempted if forced feed has already failed. This action requires that the RHR or safety injection discharge valves be throttled open locally. A controlled gravity feed can extend the RWST flow for up to six hours, when the cause of the inventory loss is isolated. According to AOP direction, the gravity feed is activated if the RCS level is between within one foot of the hot leg centerline. It is preferable to establish gravity feed from the RWST through the RHR pumps to the cold legs. The AOP also calls for establishing gravity feed through the SIPs to the cold legs.

1-SD-RWST-REFL-1D: This top event fault tree models three basic events: an operator action to refill the RWST (1-HPI-XHE-XM-REFILL-1D), a random failure (rupture) of the RWST, and an undeveloped event representing component failures (i.e., hardware failures preventing refilling of the RWST). However, the makeup flow capability is much less than injection flow requirements needed to maintain core cooling during the early parts of the outage. Also, there is a lack of procedural guidance for emergency makeup. Therefore, this event is set to 'TRUE' (i.e., guaranteed failure). However, there may be other conditions that prompt this action, so it is retained in the event tree for possible future consideration.

1-SD-LT-REC: This top event represents the late recovery of RHR cooling. The time available for recovery of RHR is based on the time that SIP or CCP injection is available to provide alternate core cooling. For branches where the RCS is not isolated before makeup is required, an alternate fault tree is substituted to include consideration for possible RHR pump damage related to the drain down. A human failure event is included to model the failure to trip any running RHR pumps in response to the event. If running pumps are not tripped, then air ingestions could cause pump damage preventing any long-term recovery of the pump.

1-SD-LPR: This top event models sump recirculation given feed and bleed was successful. If RHR cooling with normal suction from the RCS is recovered, then recirculation from the sump is not required to reach a stable state. Therefore, this top event is effectively ignored by setting it to 'FALSE' for all sequences; however, it is retained in the event tree for future modeling consideration if there are unique conditions where sump recirculation would be required. Also, contributing to the decision to bypass this top event is the assumption that the sump is unavailable during much of the outage after protective covers are installed over sump screens during Mode 5.

4.3.4 LO4160V Plant Response Event Tree

The LO4160V event tree models the response to a loss of electrical power to a class 1E 4,160V AC bus during shutdown plant operating states. The event tree is applicable to POSs where one or two RHR trains are operating and providing core cooling.¹⁰ The same base event tree is used for initiating events resulting in loss of AC power for the different LPSD operating states.

Figure 4-7 shows the modeled event tree plant response to a LO4160V initiating event. This event tree is very similar to the event tree that models the loss of RHR event. The primary mitigation functions following a LO4160V initiating event during RHR system operation include restoring power to the 4,160V AC bus (i.e., via starting and loading of the corresponding emergency diesel generator) and recovery of shutdown cooling (i.e., restoration of residual heat removal, steam generator cooling, bleed and feed, RWST refill, and long-term recovery of RHR).

The reference plant's AOP governing the operator response to a loss of a 4,160V AC class 1E electrical system was reviewed to inform the development of the LO4160V event tree. The AOP guides the operator response to the affected train by either reenergizing the train by its associated diesel generator or remaining de-energized due to the diesel generator failing to tie to the 4,160V AC bus. There are two relevant AOP sections that are applicable to different modes of operation, as described below:

- The first section of the LO4160V AOP is applicable to operating Modes 1 through 4.
- The second section of the LO4160V AOP (Modes 5 and 6) directs the operator to first recover AC power to the affected bus from its associated diesel generator and then recover RHR cooling.

¹⁰ If two RHR trains are operating, the loss of power to a single class 1E 4,160V AC bus will not directly lead to a loss of RHR cooling. However, this would be an entry condition to the AOP and require human action to adjust/maintain RHR flow for the intact train. Therefore, the loss of power to a class 1E 4,160V AC bus is conservatively assumed to be an initiating event regardless of how many RHR trains are operating, though the overall CDF contribution for these conditions is low.

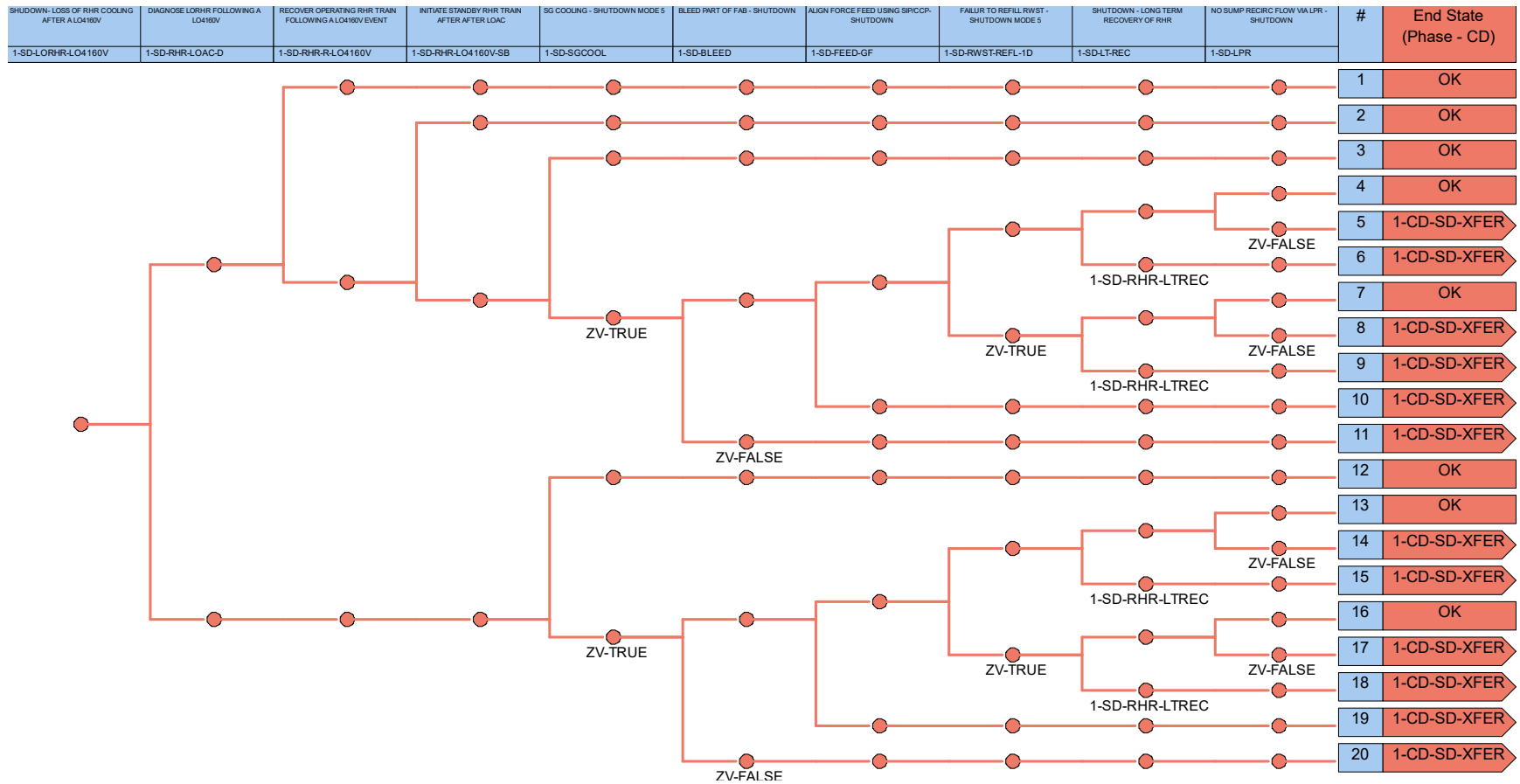


Figure 4-7 LO4160V Plant Response Event Tree

Table 4-7 lists the key steps taken in response to the LO4160V initiating event, including the reference to the relevant steps from the plant procedure.

Table 4-7 Key Steps in Response to LO4160V

Description of Action or Step Taken
Restore power to AC emergency bus from DG • Did the DG start automatically? If not, manually start. • Is the AC emergency bus of the started DG energized? Bus frequency should be at 60Hz, if less adjust frequency. • Is NSCW operating?
Place selected train of RHR in shutdown cooling by initiating RHR system • Check RHR status (RCS leakage) • Check plant status (time to boil) • Establish makeup to RCS • Restore RHR flow
Initiate system procedure for 4,160V AC bus 1E electrical distribution system to energize at least one AC emergency bus using any available power supply. • If offsite power available to either RAT, use normal or emergency incoming feeder breaker. • If power available to standby auxiliary transformer (SAT), initiate SAT operations procedure.
Was power restored to at least one AC emergency bus? • If not, return to above step.
Check proper operation of the NSCW cooling system. Start two CCW pumps in selected train.

Loss of Power to 4,160V AC Bus Top Event Descriptions

The following provides a brief description of each LO4160V event tree top event and the related fault trees. Each node in the event tree is adjusted for the POS-specific conditions, for example, equipment availability, time available for action, and failure criteria. The modeling approach and assumptions for each top event are described below.

1-SD-RHR-LOAC-D: This top event models the operators' diagnosis of a loss of power to a 4,160V AC bus and the impact on RHR operation for decay heat removal. The operator action basic event models the operator failure to diagnose the initiating event. The diagnosis and the action to restore RHR (modeled in the subsequent event tree node) must be completed prior to RCS level reaching one foot above the hot leg centerline. The time available varies depending on the POS, and a portion of the available time is apportioned for diagnosis. Estimation of human error probabilities for failure to diagnose loss of power to a 4,160V AC bus is discussed in detail in Section 7.

1-SD-RHR-R-LO4160V: This top event models the recovery of the operating RHR train(s). Failure to recover the RHR train is modeled by an operator failure event or failures of the RHR flow path. The POS conditions and nature of the initiating event influence the likelihood of recovering the RHR train. If only one RHR train is in operation during the POS, train B is assumed to be operating and the associated class 1E 4,160V AC bus B is assumed to be

impacted by the initiating event. Following loss of power to the bus, emergency power will be initiated automatically. Recovery of RHR train B is dependent on successful operation of the B EDG to restore power to the bus.

1-SD-RHR-LO4160V-SB: This top event models aligning the standby RHR train to provide core cooling. The standby train (assumed to be RHR train A) may be aligned for shutdown cooling, available but not aligned, or out of service for maintenance, depending on the POS. The standby RHR train must be recovered prior to RCS level reaching one foot above the hot leg centerline. If both RHR trains are operating prior to the initiating event, then operation of the RHR train A is not impacted by the loss of power to the opposite train's bus. The recovery action in this case would involve maintaining and verifying adequate cooling flow with the available RHR train.

1-SD -SGCOOL: This top event models using steam generators for RCS cooling. Heat removal using the steam generators is possible during some shutdown POSs. For successful steam generator cooling, secondary inventory must be available, and operators must establish AFW to maintain steam generator level. Once the RCS is vented, then steam generators are no longer effective for heat removal, regardless of steam generator inventory level. This top event is not used for POSs where the RCS is vented or where RCS pressure and level would not be sufficient for heat removal using steam generators (POSs 5B, 6, 7, 8E, 8L, 9, 10, and 11.)

1-SD-BLEED: This top event models venting of the RCS by opening one of two pressurizer PORVs. The venting of a PORV depressurizes the RCS to support core cooling by forced injection, gravity feed from the RWST, or re-establishing RHR cooling. During POSs where the RCS is vented (e.g., pressurizer safety valves removed) this top event is not relevant (i.e., this event is set to 'FALSE' and success is guaranteed). During POSs where RCS is not vented, RCS level is in the pressurizer range, and steam generator loops are not isolated, the reference plant's procedural guidance focuses on restoring RHR cooling and using steam generators for RCS cooling. There is no procedure step for establishing a bleed path. For these POSs (i.e., POSs 3, 4, 5A, 12, and 13), the top event is set to 'TRUE' and failure is guaranteed. For POS 11 during vacuum refill, the reference plant's relevant procedure does specify steps to break RCS vacuum and establish a vent path. POS 11 is the only POS where the top event fault tree logic is evaluated.

1-SD-FEED-GF: This top event fault tree models establishing injection paths using SIPs or CCPs to control RCS inventory. The fault tree top event includes an operator action to establish forced feed paths.¹¹ The success criteria are dependent on the POS-specific decay heat load. The specific criteria are discussed further under system success criteria in Section 5.4. Successful forced feed must be established before the RCS water boils off to two-thirds of the active fuel height.

1-SD-RWST-REFL-1D: This top event fault tree models refilling the RWST to extend the available inventory for injection. The fault tree includes three basic events: an operator action, a random failure (rupture) of the RWST, and an undeveloped event representing component failures (i.e., hardware failures preventing refilling of the RWST). Successful refill of RWST will extend the time of injection beyond 24 hours. However, the makeup flow capability is much less than injection flow requirements needed to maintain core cooling during the early parts of the

¹¹ For the same reasons discussed previously for the LORHR event tree, gravity feed is not credited in response to a loss of power to a class 1E 4,160V AC bus.

outage. Also, there is a lack of procedural guidance for emergency makeup for most POSs. Therefore, this action is unlikely to be successful; and the event is set to 'TRUE' (i.e., guaranteed failure). However, this top event is retained in the event tree for future consideration. When the reactor cavity is flooded, there are some conditions where operators may be directed to transfer a limited volume of water from the SFP to the RWST. Credit for RWST refill can be considered under these conditions.

1-SD-LT-REC: This top event represents the late recovery of RHR cooling. The model includes a human failure event that is adjusted based on the POS-specific time available for recovery. The time available for recovery of RHR is based on the time that SIP or CCP injection is available to provide alternate core cooling.

1-SD-LPR: This top event models sump recirculation given feed and bleed was successful. If RHR cooling with normal suction from the RCS is recovered, then recirculation from the sump is not required to reach a stable state. Therefore, this top event is effectively ignored by setting it to 'FALSE' for all sequences; however, it is retained in the event tree for future modeling consideration if there are unique conditions where sump recirculation would be required. Also, contributing to the decision to bypass this top event is the assumption that the sump is unavailable during much of the outage after protective covers are installed over sump screens during Mode 5.

4.3.5 LOOP Plant Response Event Tree

The LOOP event tree models the response to a loss of offsite power event during shutdown plant operating states. The event tree is applicable to POSs where one or two RHR trains are operating and providing core cooling. The event tree model represents all classes of loss of offsite power initiating events, including plant-centered, switchyard-centered, grid-related, and weather-related events. Since a LOOP event will induce a LORHR scenario with the additional complications resulting from loss of AC power to multiple equipment (with the potential for a station blackout to occur), the modeling approach taken here is to adapt LORHR scenarios for the individual POSs to include the effect of a LOOP. The following types of sequences are assessed: (a) a LOOP-induced LORHR that can be recovered with successful EDGs powering safety-related buses without offsite power, (b) a LOOP-induced LORHR that can be recovered with successful offsite power restoration after onsite AC power failure, and (c) a LOOP-induced LORHR developing into a station blackout (SBO) due to onsite AC power failure and non-recovery of offsite AC power.

Figure 4-8 shows the modeled event tree plant response to the loss of offsite power initiating event showing the three sequences that are assessed. If the onsite emergency power is successful, then the event tree is transferred to a LORHR recovery event tree. The loss of RHR recovery with successful onsite emergency power is modeled in the same way as a loss of RHR event, as shown in Figure 4-4 above. If onsite emergency power is failed, then the restoration of offsite power is considered. Operators have a maximum of 2 hours available for recovery of offsite power based on the most limiting depletion time of plant batteries.¹² During POSs with reduced inventory, there is less time available for offsite power recovery that would support

¹² The assumption of the most limiting depletion time of plant batteries is consistent with the modeling approach used in the L3PRA project Level 1 PRA for at-power operations (NRC, 2022). The turbine building batteries (2-hour battery life) that supply DC power to circuit switchers directly upstream of the RATs are selected as the basis for the most limiting depletion time.

sufficient time for equipment recoveries to prevent core damage.¹³ If offsite power is recovered during POSs with reduced inventory, then RHR cooling cannot be immediately recovered. During the time allotted for offsite power recovery, RCS conditions are degraded to the point where boiling has occurred or is imminent. Alternate cooling and RCS level control must first be established prior to attempting recovery of RHR cooling. Figure 4-9 shows the event tree model for actions associated with restoring RHR cooling after successful recovery of AC power.

If both onsite emergency power and offsite power recovery are failed, then an SBO occurs. Figure 4-10 shows the event tree model for recovering RHR cooling under SBO conditions. Steam generator cooling with the turbine-driven AFW pump providing feedwater is credited for POSs where steam generator cooling is possible. The other possible cooling strategy during SBO is bleed and feed with gravity-driven feed from the RWST. If gravity feed is successful, then additional time is credited for restoring AC power and recovering RHR cooling.

The reference plant's AOP governing the operator response to a loss of class 1E AC electrical systems was reviewed to inform the development of the LOOP event tree. If neither of the class 1E 4,160V AC buses is energized, then the AOP directs operators to other procedures based on the operating mode of the plant.

- If in Modes 1, 2, or 3, then operators refer to the emergency operating procedure for reactor trip or safety injection.
- If in Mode 4, then operators are directed to go to the emergency operating procedure for a loss of all AC power, which directs operators to check that the RCS is isolated (i.e., PORVs closed and letdown isolated), verify adequate AFW flow using the turbine-driven AFW pump, and try to restore power to any AC emergency bus from its associated diesel generator.
- If in Modes 5 or 6, then operators are directed to the appropriate response section of the AOP for loss of class 1E AC electrical systems.

The loss of electrical systems AOP is applicable to most of the shutdown POSs and is discussed further below. Operators are directed to try to restore power to any AC emergency bus from its associated diesel generator. If a bus is energized, then operators will place a train of RHR in service for shutdown cooling. If RHR is not able to be placed into service, then operators are directed to initiate the loss of RHR AOP. If a bus cannot be energized with a diesel generator, then operators will continue their effort to restore power to at least one AC emergency bus using any available power supply. Operators will energize a bus using an incoming power source from either reserve auxiliary transformer (RAT) or from the station auxiliary transformer (SAT). After power is restored, operators will check for proper operation of the NSCW, CCW, and RHR systems.

¹³ During POSs 5B, 9, 10, and 11, 1 hour is available for offsite power recovery. During POS 6, 30 minutes are available for offsite power recovery. All other POSs use the maximum 2-hour time for offsite power recovery.

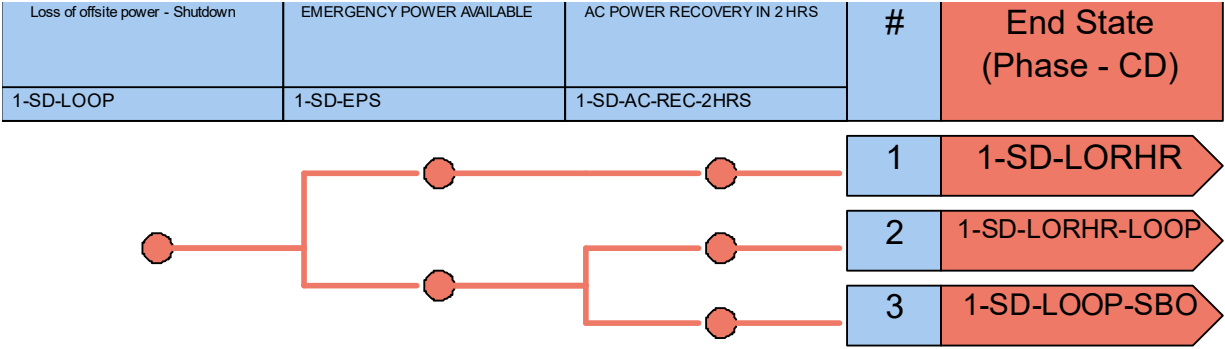


Figure 4-8 LOOP Plant Response Event Tree – AC Power Recovery

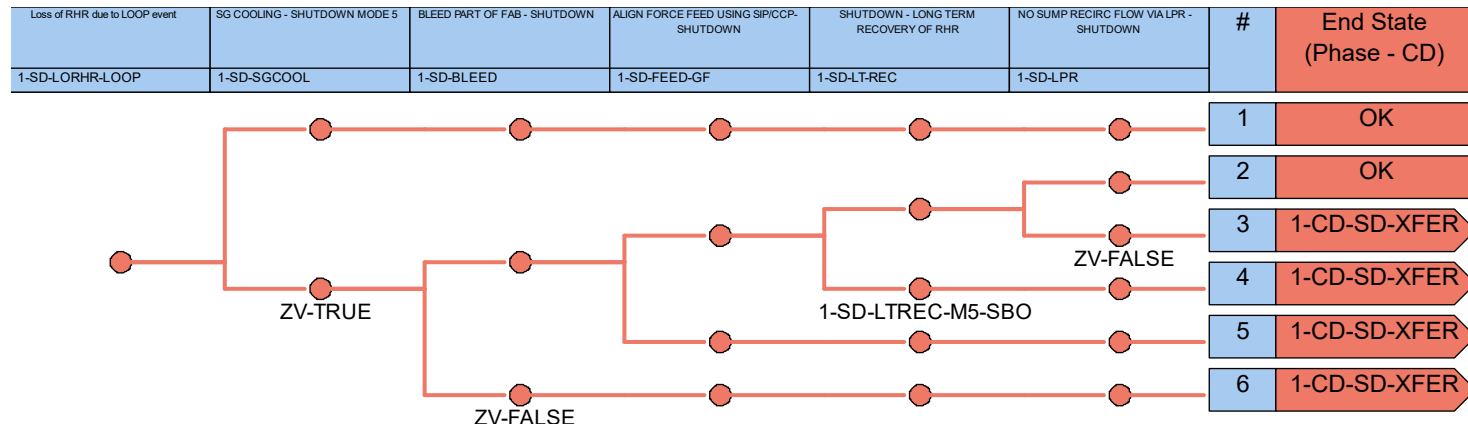


Figure 4-9 LOOP Plant Response Event Tree – RHR Recovery after AC Power Restored

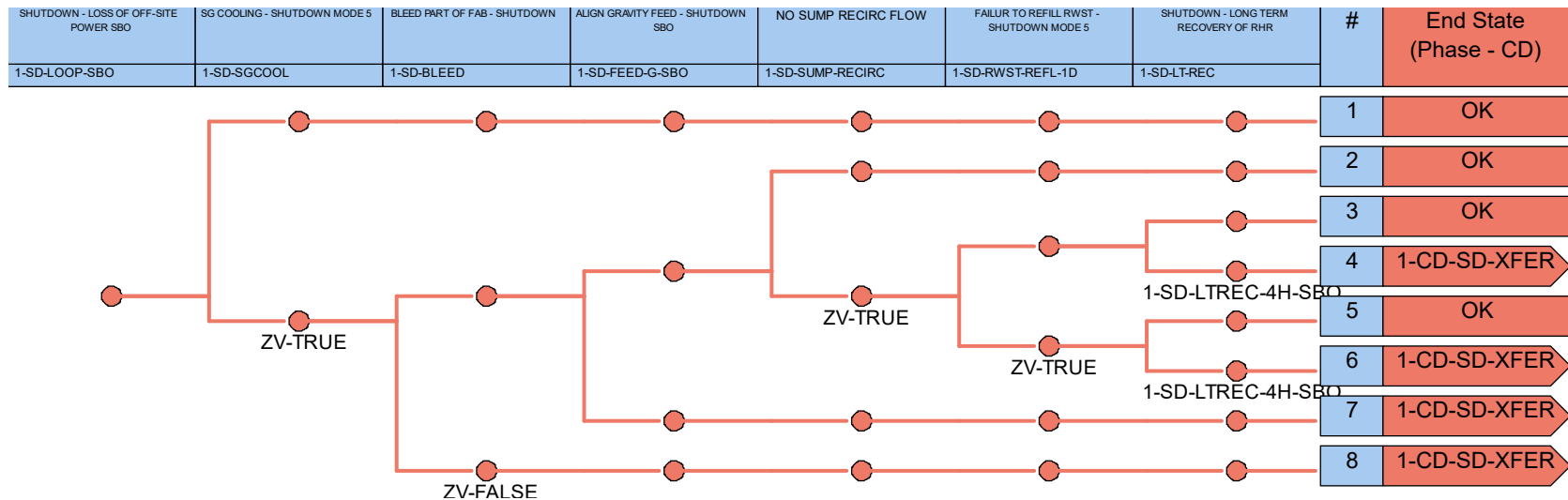


Figure 4-10 LOOP Plant Response Event Tree – RHR Recovery for Station Blackout

Table 4-8 lists the key steps taken in response to a LOOP event.

Table 4-8 Key Steps in Response to LOOP

Description of Action or Step Taken
Restore power to AC emergency bus from DG - Did the DG start automatically? If not, manually start. - Is the AC emergency bus of the started DG energized? - Check NSCW operating. - Place selected train of RHR in shutdown cooling by initiating RHR system.
Place selected train of RHR in shutdown cooling by initiating RHR system - Check RHR status (RCS leakage) - Check plant status (time to boil) - Establish makeup to RCS - Restore RHR flow
Initiate system procedure for 4,160V AC bus 1E electrical distribution system to energize at least one AC emergency bus using any available power supply. - If offsite power available to either RAT, use normal or emergency incoming feeder breaker. - If power available to standby auxiliary transformer (SAT), initiate SAT operations procedure.
Was power restored to at least one AC emergency bus? If not, return to above step.
Check proper operation of the NSCW, CCW, and RHR systems.

LOOP Top Event Descriptions

The following provides a brief description of each LOOP event tree top event and the related fault trees. Each node in the event tree is adjusted for the POS-specific conditions; for example, equipment availability, time available for action, and failure criteria. The modeling approach and assumptions for each top event are described below.

1-SD-EPS: Following the LOOP, emergency power will be initiated automatically. This fault tree models failures of the EDGs. The offsite alternate AC source is assumed to be available and credited. If EDG failures occur, then credit is given for EDG recovery. The EDG recovery credit is considered applicable to shutdown POSs because:

1. The loss of electrical systems AOP directs operators to manually start an EDG if it did not automatically start and load onto its associated bus.
2. Given that EDG maintenance regularly occurs during refueling outages, the appropriate crew, tools, and parts that could be needed to restore an EDG would be readily available.
3. During an outage, the work planning infrastructure would be active and capable of authorizing the appropriate recovery activities.

The time allotted for EDG repair varies based on the POS and sequence conditions. The probability of exceeding the EDG repair time is based on the unplanned EDG outage times provided by utilities (NRC, 2021a). EDG recovery is only applied when both safety-related EDG trains are either failed or unavailable. Successful recovery assumes one EDG train is restored to power its associated emergency bus.

1-SD-AC-REC-2HRS: This fault tree models failure to recover available offsite power sources. The failure to recover probabilities are based on industry-wide LOOP duration data and consistent with the approach used in the at-power internal event LOOP analysis. The time available for recovery is limited to 2 hours based on the most limiting plant batteries required to realign offsite power to a safety-related bus, which are the turbine building batteries (assumed 2-hour depletion time as consistent with the at-power model documentation [NRC, 2022]). In some POSs, the time for power recovery is less than 2 hours when boiling or other RCS conditions would prevent restoring RHR cooling after power recovery. In these cases, the time for power recovery was based on the time for boiling to occur.

1-SD-RHR-D: This top event models the operators' diagnosis of the LOOP event and identification of actions needed to restore RHR cooling. The operator must confirm NSCW, CCW, and RHR systems are operating properly after recovery of AC power. If the sequencer is available, then associated pumps may be automatically loaded when power is recovered to the bus. The diagnosis and the action to restore RHR (modeled in the subsequent event tree node) must be completed prior to RCS level reaching one foot above the hot leg centerline. The time available varies depending on the POS, and a portion of the available time is apportioned for diagnosis. Estimation of human error probabilities for failure to diagnose loss of RHR in response to a LOOP is discussed in detail in Section 7.

1-SD-RHR-R: This top event models the recovery of the operating RHR train(s) after AC power has been restored by either EDGs or offsite power recovery. The RHR recovery actions are the same as those modeled in the LORHR event tree top event. Failure to recover the RHR train is modeled by an operator failure event or failures of the RHR flow path. The POS conditions and nature of the initiating event influence the likelihood of recovering the RHR train. If both RHR trains are normally operating prior to the initiating event, then this top event models recovery of either one of the two RHR trains. If one RHR train is operating and one in standby prior to the initiating event, then this top event models recovery of the train that was failed by the initiator. Depending on the nature of the failure, the operating train may or may not be recoverable. The RHR train must be recovered prior to RCS level reaching one foot above the hot leg centerline. Note, RHR train B is assumed to be operating if only one train is in operation.

1-SD-RHR-SB: This top event models aligning the standby RHR train to provide core cooling after AC power has been restored by either EDGs or offsite power recovery. The RHR standby train actions are the same as those modeled in the LORHR event tree top event. The standby train (assumed to be RHR train A) may be aligned for shutdown cooling, available but not aligned, or out of service for maintenance depending on the POS. The standby RHR train must be recovered prior to RCS level reaching one foot above the hot leg centerline. If both RHR trains are operating and failed by the initiating event, then this top event is not used. Recovery of either one of the two RHR trains is modeled by substituting 1-SD-RHR-2R for the previous top event.

1-SD-SGCOOL: This top event models using steam generators for RCS cooling. Heat removal using the steam generators is possible during some shutdown POSs. For successful steam

generator cooling, secondary inventory must be available, and operators must establish AFW to maintain steam generator level. Under loss of AC power conditions, the turbine-driven AFW pump can place the plant in a stable condition provided AC power is recovered prior to battery depletion (which occurs approximately four hours after the start of the SBO). Once the RCS is vented, then steam generators are no longer effective for heat removal, regardless of steam generator inventory level. This top event is not used for POSs where the RCS is vented or where RCS pressure and level would not be sufficient for heat removal using steam generators (POSs 5B, 6, 7, 8E, 8L, 9, 10, and 11).

1-SD-BLEED: This top event models venting of the RCS by opening one of two pressurizer PORVs. The venting of a PORV depressurizes the RCS to support core cooling by forced injection, gravity feed from the RWST, or re-establishing RHR cooling. During POSs where the RCS is vented (e.g., pressurizer safety valves removed) this top event is not relevant (i.e., this event is set to 'FALSE' and success is guaranteed). During POSs where RCS is not vented, RCS level is in the pressurizer range, and steam generator loops are not isolated, the reference plant's procedural guidance focuses on restoring RHR cooling and using steam generators for RCS cooling. There is no procedure step for establishing a bleed path. For these POSs (i.e., POSs 3, 4, 5A, 12, and 13), the top event is set to 'TRUE' and failure is guaranteed. For POS 11 during vacuum refill, the reference plant's relevant procedure does specify steps to break RCS vacuum and establish a vent path. POS 11 is the only POS where the top event fault tree logic is evaluated.

1-SD-FEED: This top event fault tree models establishing injection paths using SIPs or CCPs to control RCS inventory. The fault tree top event includes an operator action to establish forced feed paths. The success criteria are dependent on the POS-specific decay heat load. The specific criteria are discussed further under system success criteria in Section 5.4. Successful forced feed must be established before the RCS water boils off to two-thirds of the active fuel height. (For sequences where emergency power and restoration of offsite power are failed, the feed function is modeled by 1-SD-FEED-G-SBO – see below.)

1-SD-RWST-REFL-1D: This top event fault tree models refilling the RWST to extend the available inventory for injection. The fault tree includes three basic events: an operator action, a random failure (rupture) of the RWST, and an undeveloped event representing component failures (i.e., hardware failures preventing refilling of the RWST). Successful refill of RWST will extend the time of injection beyond 24 hours. However, the makeup flow capability is much less than injection flow requirements needed to maintain core cooling during the early parts of the outage. Also, there is a lack of procedural guidance for emergency makeup for most POSs. Therefore, this action is unlikely to be successful; and the event is set to 'TRUE.' However, this top event is retained in the event tree for future consideration. When the reactor cavity is flooded, there are some conditions where operators may be directed to transfer a limited volume of water from the SFP to the RWST. Credit for RWST refill can be considered under these conditions.

1-SD-LT-REC: This top event represents the late recovery of RHR cooling. The model includes a human failure event that is adjusted based on the POS-specific time available for recovery. The time available for recovery of RHR is based on the time that SIP or CCP injection is available to provide alternate core cooling.

1-SD-LPR: This top event models sump recirculation given feed and bleed was successful. If RHR cooling with normal suction from the RCS is recovered, then recirculation from the sump is

not required to reach a stable state. Therefore, this top event is effectively ignored by setting it to 'FALSE' for all sequences; however, it is retained in the event tree for future modeling consideration if there are unique conditions where sump recirculation would be required. Also, contributing to the decision to bypass this top event is the assumption that the sump is unavailable during much of the outage after protective covers are installed over sump screens during Mode 5.

1-SD-FEED-G-SBO: This top event fault tree models establishing injection paths using gravity feed from the RWST to control RCS inventory. This top event is used in SBO sequences where forced injection is not available. The fault tree top event includes an operator action to establish a gravity feed path. The success criteria are defined as operators locally throttling open one of the two RWST to RHR pump suction valves. Successful gravity feed must be established before boiling occurs. The flow rate is assumed to be 1600 gpm, which is based on the parametric assessment of gravity feed developed for this project and discussed in Section 5.3. The useable RWST water volume will be exhausted in approximately six hours. During POSs with the refueling cavity flooded (POS 8E and POS 8L), the volume available in the RWST is significantly reduced. The volume will be exhausted in approximately three hours.

4.3.6 LONSCW Plant Response Event Tree

The LONSCW event tree models the response to losing the ultimate cooling source to the CCW heat exchangers and RHR pump motor coolers. The loss of NSCW also impacts accident mitigating systems including the centrifugal charging pumps and safety injection pumps. The loss of NSCW leads to a functional loss of RHR cooling. The NSCW function must be restored before RHR cooling can be restored.

Figure 4-11 shows the modeled event tree plant response to a loss of NSCW initiating event. The event tree includes an initiating event fault tree to model the combinations of failures that can lead to a loss of NSCW event. The primary mitigating functions include diagnosis of the LONSCW event, aligning gravity injection, recovery of an NSCW pump, aligning forced injection after NSCW recovery, and long-term recovery of RHR after NSCW recovery.

The reference plant's AOP governing the operator response to a loss of nuclear service cooling water was reviewed to inform the development of the LONSCW event tree. The AOP is applicable to all plant states. The first several steps of the procedure involve diagnosing and verifying the NSCW systems status and the plant status. If neither train of NSCW is verified to have two pumps running, then the operators are directed to place one train in single pump operation. If required systems, including RHR and CCW, are unavailable, then the procedure directs operators to initiate appropriate AOPs to restore those functions.

The loss of RHR procedure directs operators to assess RHR system status. The procedure directs operators to control RCS inventory, and if RCS temperature is greater than 200°F, then the operators are directed to refill the RCS using safety injection pumps or centrifugal charging pumps. The pumped injection cannot be initiated until NSCW is restored to provide pump cooling.

If the RCS temperature is less than 200°F and the RCS water level is within one foot of the hot leg centerline, then the operators are directed to try gravity feed from the RWST, through any of several paths, but preferably through the RHR pumps to the cold legs. If no NSCW cooling is available, then operators are expected to recognize gravity feed as the only viable option to

prevent RCS boiling. Once RCS refill is accomplished, the operators have additional time to attempt restoration of an NSCW pump and restart RHR for shutdown cooling.

Table 4-9 lists the key steps taken in response to the LORHR initiating event, including reference to the relevant steps from the plant procedure.

Table 4-9 Key Steps in Response to Loss of Nuclear Service Cooling Water System

Description of Action or Step Taken
Verify proper operation of unaffected NSCW train • If neither NSCW train can be placed in normal operation, then • Place one train of NSCW in single pump operation
Verify the following components running if required: • CCP • SIP • RHR Pump • CCW Pumps Initiate the following as appropriate • Loss of CCW AOP • Loss of RHR AOP
Check Plant Status (time to boil) • Determine time to boiling based on existing condition. • If less than 60 minutes, evacuate personnel from containment, isolate containment, and align containment cooling fans for low speed operations.
Establish makeup to RCS • If RCS temperature is above 200°F, align SIP for hot leg injection, and if temperature continues to rise, align CCP for cold leg injection. • If RCS temperature is below 200°F and level is within one foot of the hot leg centerline, establish RWST gravity feed to RCS.
Restore RHR flow Put at least one RHR train in service.

LONSCW Top Event Descriptions

The following provides a brief description of each LONSCW event tree top event and the related fault trees. Each node in the actual event trees in the model is adjusted for the POS-specific conditions, for example, equipment availability, time available for action, and failure criteria. The modeling approach and assumptions for each top event are described below.

1-IEFT-LONSCW: This top event models the initiating event fault tree to include the combination of system failures that result in a LONSCW initiating event. The fault trees are adjusted for POSs depending on the number of NSCW trains that are in operation at the time of the initiating event.

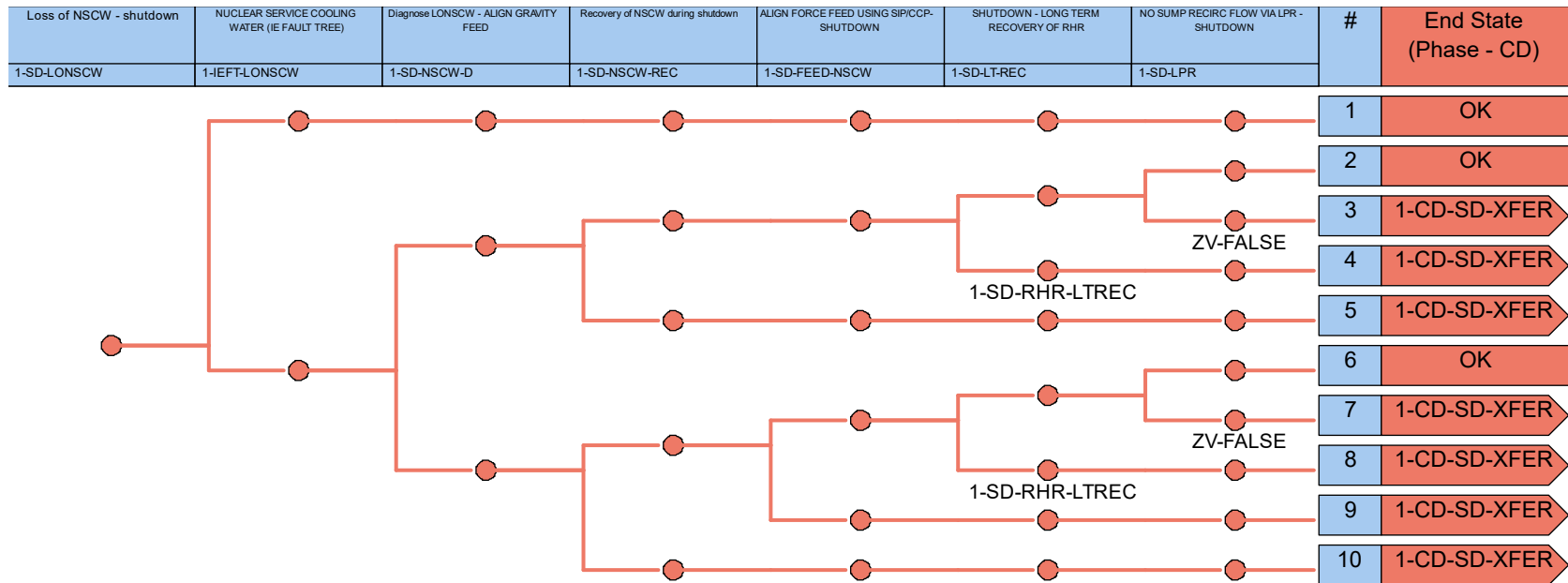


Figure 4-11 LONSCW Plant Response Event Tree

1-SD-NSCW-D: This top event models the operators' diagnosis of the LONSCW event, working through the appropriate procedure steps, and alignment of gravity injection. Alignment of gravity injection is a time-sensitive action, particularly during mid-loop operation. The diagnosis of the event considers the limited time available for the POS conditions.

1-SD-NSCW-REC: This top event models a human reliability event to restore one train of NSCW for single pump operation. The time available for restoration depends on success or failure of the prior top event representing operator diagnosis and aligning gravity injection. Human failure dependencies are evaluated between actions to align gravity injection and actions to restore NSCW.

1-SD-FEED-NSCW: This top event fault tree models establishing injection paths using SIPs or CCPs to control RCS inventory. The fault tree top event includes an operator action and hardware failures that would prevent forced feed injection. This top event is only credited if NSCW is restored.

1-SD-LT-REC: This top event represents the late recovery of RHR cooling. The model includes a human failure event that is adjusted based on the POS-specific time available for recovery. The time available for recovery of RHR is based on the time that SIP/CCP injection or gravity injection is available to provide alternate core cooling.

1-SD-LPR: This top event models sump recirculation given feed and bleed was successful. If RHR cooling with normal suction from the RCS is recovered, then recirculation from the sump is not required to reach a stable state. Therefore, this top event is effectively ignored by setting it to 'FALSE' for all sequences; however, it is retained in the event tree for future modeling consideration if there are unique conditions where sump recirculation would be required. Also, contributing to the decision to bypass this top event is the assumption that the sump is unavailable during much of the outage after protective covers are installed over sump screens during Mode 5.

4.4 Plant Operating State (POS) Initiating Event Response

The POS-specific conditions and response to initiating events are described in the following sections. The following information is provided for each modeled POS:

- Description of the POS-specific response to the initiating event categories LORHR, LOINV, LO4160V, LOOP, and LONSCW, with reference to the applicable procedures. The ODM event tree, which models overdrain when entering mid-loop, is only applicable to POS 6 and POS 10. The LOINV-DS event tree, which models displacements of inventory, is only applicable to POS 5B and POS 9.
- Table of the initial POS conditions and the calculated values for time to reach saturation temperature, core damage, and other conditions relevant to the operator response.
- Table of equipment status and availability during the POS.
- Table of human failure events (HFEs) modeled in the POS accident response.

Note, POSs 1, 2, 14, 15A, and 15B are not discussed in this section, since they were addressed through qualitative analysis, as discussed in Section 4.2.1.

4.4.1 POS 3 Initiating Event Response

One RHR train is first placed into service during POS 3. The first train of RHR is started when RCS pressure is less than 365 psig and RCS temperature drops below 340°F. RHR train B is the preferred train to place into service. RHR train A remains aligned for ECCS injection. The AFW system is also operating during POS 3. At the end of POS 2, the AFW cross-connect valve between the A and B trains is opened to permit one motor-driven AFW pump to feed all four steam generators. One motor-driven AFW pump is secured and one continues to operate. One motor-driven AFW pump and one RHR train are operated in parallel during POS 3. RCS level is maintained in the normal operating range in the pressurizer. Two of four RCPs are running. The containment equipment hatch and personnel airlock are closed during POS 3.

LORHR Response in POS 3:

The LORHR AOP directs operators to attempt to restore RHR cooling and continue using the secondary heat sink for RCS cooling. Operators will check if any RHR pumps are running. Then, the operators check that at least one RCS loop is available for cooling. If steam generator normal range levels are less than 10 percent, operators establish feed to maintain steam generators in the normal range and control RCS temperature with steam dumps or atmospheric relief valves. Operators will place the standby RHR train in service for RHR cooling. Operators will continue actions to monitor RHR and maintain one train of RHR cooling.

LOOP Response in POS 3:

In POS 3 operators are directed to go to emergency operating procedure for loss of all AC power. The procedure directs operators to check that the RCS is isolated (i.e., PORVs closed and letdown isolated), verify adequate AFW flow using the turbine-driven AFW pump, and try to restore power to any AC emergency bus from its associated diesel generator. Turbine-driven AFW is not sufficient to reach a safe, stable state if power is not restored. Control of the AFW turbine-driven pump requires safety-related 125V DC power, which will be lost after safety-related batteries are depleted in four hours. If a bus cannot be energized with a diesel generator, then operators will continue efforts to restore power to at least one AC emergency bus using any available power supply. If power is successfully restored, then operators will restore steam generator cooling or RHR, or refill the RCS with CCPs or SIPs.

The LOINV, LO4160V, and LONSCW initiating events are not modeled for POS 3. The initial conditions and time to boil, equipment status, and modeled HFEs for POS 3 are provided in Table 4-10, Table 4-11, and Table 4-12, respectively.

Table 4-10 POS 3 Initial Conditions and Time to Boil

Plant Condition	Value	Notes and References
Time After Reactor Shutdown	5 hr	Based on recent reference plant refueling outages, time to reach Mode 4 has ranged from 4 to 7 hours.
Decay Heat	21.2 MW	
Initial RCS Temperature	340°F	
Initial RCS Pressure	350 psig	
Initial RCS Water Level	80% level in pressurizer	In preparation for cooling to Cold Shutdown, pressurizer level is raised to 80%.
Time to SG Dryout	5 hr	Conservative estimate used to inform available time for operator actions.
Time to Core Boiling (to T_{sat})	7.5 hr	Conservative estimate used to inform available time for operator actions.
Time to boiloff to within one foot of hot leg centerline	N/A	Not used for this POS. RHR must be recovered before boiling. RCS pressure and temperature must be less than 365 psig and 350°F, respectively.
Time to Core Damage	10 hr	Conservative estimate used to inform available time for operator actions.

Table 4-11 POS 3 Equipment Status

Equipment	Status	Remarks
RHR pumps	1 of 2 running	RHR train B in operation. RHR train A hot leg suction valve closed.
Normal charging pump	Operating	Let down is in auto.
Reactor Coolant Pumps	2 of 4 running	
Centrifugal charging pumps	2 of 2 in standby	
Safety injection pumps	2 of 2 in PTL and isolated	One of two pumps maintained in a state ready to be restored with manual actions from the control room.
Safety injection signals	Blocked	
Accumulators	Isolated	
Containment spray pumps	2 of 2 in standby	
Steam generators and feedwater	Available	Steam generators providing cooling with one AFW MDP running and cross-connected to all four SGs.
Pressurizer PORV/block valves	Armed at COPS setpoints with block valves open	
Pressurizer safety relief valves	Closed	
COPS (cold over-pressurization protection system)	Armed with two PORVs and one RHR suction relief valve	
Reactor vessel head	tensioned	
Emergency diesel generators	2 of 2 in standby	
Main generator and unit auxiliary transformers	Not in service	
Reserve auxiliary transformers	2 of 2 in service	RAT maintenance may occur during outage but is typically performed during Mode 6.
Containment equipment hatch	Closed and bolted	
Containment purge vent	Mini purge (14-inch) operating as needed	
Containment personnel airlock	Closed	
Containment cooler units	All running in high speed	
Containment sump	Available	

Table 4-12 POS 3 Equipment Status

Human Failure Event	Human Failure Description	Estimated Time Available	Modeling Approach
LORHR-Related HFEs			
1-RHR-XHE-XD-POS3	Operator fails to diagnose LORHR prior to loss of secondary cooling due to SG dryout.	~5 hours assuming no feedwater to SGs	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-TRB-POS3	Operator fails to recover the operating RHR train prior to loss of secondary cooling due to SG dryout.	~5 hours assuming no feedwater to SGs	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XM-TRA-POS3	Operator fails to align and initiate the standby RHR train prior to loss of secondary cooling due to SG dryout.	~5 hours assuming no feedwater to SGs	HFE evaluated for quantification (see Section 7)
1-AFW-XHE-XM-SGC-POS3	Operator fails to feed SGs to maintain level and control RCS temperature using steam dumps or ARVs.	5 hours before SGs dry out without maintaining feedwater	HFE evaluated for quantification (see Section 7)
LOOP-Related Repair Events			
1-EPS-XHE-XL-NR04H	Operator Fails to Recover Emergency Diesel in 4 Hours - based on unplanned EDG outage times	4 hours	Quantification based on EDG repair data (NRC, 2021a)
1-OEP-XHE-XL-NR02H	Operator Fails to Recover Offsite Power in 2 Hours	2 hours	Quantification based on LOOP duration data (NRC, 2021a)

4.4.2 POS 4 Initiating Event Response

During POS 4 the RCS cooldown continues, with both RHR cooling loops now in service. The plant enters Mode 5 with RCS temperature at 200°F. RCS level is maintained in the normal operating range in the pressurizer and pressure is maintained at 350 psig. Any running AFW pumps are secured and cooldown proceeds using RHR only. The main steam isolation valves (MSIVs) are closed and condenser vacuum is broken. Therefore, the condenser steam dumps are no longer operable. The steam generator level is at 60–70 percent of the normal operating range. Two of four RCPs are running. The containment equipment hatch and personnel airlock are open during POS 4.

LORHR Response in POS 4:

The LORHR AOP directs operators to attempt to restore RHR cooling and perform the necessary actions to use the secondary heat sink for RCS cooling. Operators will check if any RHR pumps are running. Operators will continue actions to monitor RHR and establish one train of RHR cooling. If RHR flow from either train cannot be established, then operators are directed to establish steam generator cooling. Operators will establish feedwater to maintain level greater than 10 percent of normal operating range in the steam generators. Operators continue

attempts to start one RHR train. If RHR is not available, then operators will maintain RCS temperature using the steam generator atmospheric relief valves.

LOOP Response in POS 4:

The operator response to loss of offsite power is governed by AOPs for loss of electrical systems and loss of RHR. Operators are directed to try to restore power to any AC emergency bus from its associated diesel generator. If RHR is not able to be placed into service, then operators are directed to initiate the LORHR AOP. If a bus cannot be energized with a diesel generator, then operators will continue efforts to restore power to at least one AC emergency bus using any available power supply. If power is successfully restored, then operators will restore steam generator cooling or RHR, or refill the RCS with CCPs or SIPs.

The LOINV, LO4160V, and LONSCW initiating events are not modeled for POS 4. The initial conditions and time to boil, equipment status, and modeled HFEs for POS 4 are provided in Table 4-13, Table 4-14, and Table 4-15, respectively.

Table 4-13 POS 4 Initial Conditions and Time to Boil

Plant Condition	Value	Notes
Time After Reactor Shutdown	8 hr	Based on recent reference plant refueling outages, time to reach Mode 5 has ranged from 6 to 9 hours.
Decay Heat	19.9 MW	
Initial RCS Temperature	200°F	
Initial RCS Pressure	350 psig	
Initial RCS Water Level	80% level in pressurizer	In preparation for cooling to Cold Shutdown, pressurizer level is raised to 80%.
Time to SG Dryout	5 hr	Conservative estimate used to inform available time for operator actions.
Time to Core Boiling (to T_{sat})	7.5 hr	Conservative estimate used to inform available time for operator actions.
Time to boiloff to within one foot of hot leg centerline	N/A	Not used for this POS. RHR must be recovered before boiling. RCS pressure and temperature must be less than 365 psig and 350°F, respectively.
Time to Core Damage	10 hr	Conservative estimate used to inform available time for operator actions.

Table 4-14 POS 4 Equipment Status

Equipment	Status	Remarks
RHR pumps	2 of 2 running	
Normal charging pump	Operating	Let down is in auto.
Reactor Coolant Pumps	2 of 4 running	
Centrifugal charging pumps	2 of 2 in standby	
Safety injection pumps	2 of 2 in PTL and isolated	One of two pumps maintained in a state ready to be restored with manual actions from the control room.
Safety injection signals	Blocked	
Accumulators	Isolated	
Containment spray pumps	Not operable	
Steam generators and feedwater	Available, 2 AFW MDPs and 1 AFW TDP not in service	AFW pumps not in service, but available to be restored.
Pressurizer PORV/block valves	Armed at COPS setpoints with block valves open	
Pressurizer safety relief valves	Closed	
COPS (cold over-pressurization protection system)	Armed with two PORVs and two RHR suction relief valves	
Reactor vessel head	tensioned	
Emergency diesel generators	1 of 2 in standby, 1 available for local operation	EDG A automatic actuation is unavailable due to sequencer testing but could be operated locally.
Main generator and unit auxiliary transformers	Not in service	
Reserve auxiliary transformers	2 of 2 in service	
Containment equipment hatch	Open	
Containment purge vent	Mini purge (14-inch) operating	
Containment personnel airlock	Open	
Containment cooler units	Shutdown	
Containment sump	Sump screens covered	Protective covers are installed in Mode 5 causing sump to be unavailable.

Table 4-15 POS 4 Human Failure Events

Human Failure Event	Human Failure Description	Estimated Time Available	Modeling Approach
LORHR-Related HFEs			
1-RHR-XHE-XD-POS4	Operator fails to diagnose LORHR prior to loss of secondary cooling due to SG dryout.	~5 hours assuming no feedwater to SGs	Use same HFE for similar conditions: 1-RHR-XHE-XD-POS3
1-RHR-XHE-XL-TRN-POS4	Operator fails to recover either RHR train prior to loss of secondary cooling due to SG dryout.	~5 hours assuming no feedwater to SGs	HFE evaluated for quantification (see Section 7)
1-AFW-XHE-FO-SGCOOL	Operator fails to feed SGs to maintain level and control RCS temperature using steam dumps or ARVs.	~20 hours before SGs dry out without maintaining feedwater	HFE evaluated for quantification (see Section 7)
LOOP-Related Repair Events			
1-EPS-XHE-XL-NR04H	Operator Fails to Recover Emergency Diesel in 4 Hours - based on unplanned EDG outage times	4 hours	Quantification based on EDG repair data (NRC, 2021a)
1-OEP-XHE-XL-NR02H	Operator Fails to Recover Offsite Power in 2 Hours – based on LOOP duration data	2 hours	Quantification based on LOOP duration data (NRC, 2021a)

4.4.3 POS 5A Initiating Event Response

During POS 5A the RCS cooldown continues using both RHR cooling loops. The plant enters a pressurizer water solid condition for reducing gas concentrations in the RCS. RCS temperature is maintained between 190°F and 170°F, and RCS pressure is maintained between 350 psig and 225 psig. Steam generators are available as a heat sink with inventory levels initially at 95 percent. Any RCS pressure increase resulting from an initiating event will result in reaching the RHR relief valve set point (450 psig) almost immediately due to the water solid condition. However, the plant-specific simulations of this condition (for details of these calculations see Section 5.3 of this report) show that sufficient RCS cooling and level can be maintained as long as heat removal by the steam generators and atmospheric relief valves is underway. During POS 5A two of four RCPs are running. The containment equipment hatch and personnel airlock are open.

LORHR Response in POS 5A:

The LORHR AOP directs operators to attempt to restore RHR cooling and continue using the secondary heat sink for RCS cooling. Operators will check if any RHR pumps are running. Operators will continue actions to monitor RHR and establish one train of RHR cooling. If RHR flow from either train cannot be established, then operators are directed to establish steam generator cooling. Operators will establish feedwater to maintain level greater than 10 percent of normal operating range in the steam generators. Operators continue attempts to start one

RHR train. If RHR is not available, then operators will maintain RCS temperature using the steam generator atmospheric relief valves.

LOINV-OP Response in POS 5A:

The operator response to loss of inventory is governed by the AOPs for RCS leakage and loss of RHR. If the inventory loss is diagnosed and level maintained in the pressurizer range, then operators may establish steam generator cooling for decay heat removal. If the RHR relief valve fails to close after lifting and losses result in reduced RCS inventory level, then RHR pumps are monitored for indication of cavitation. Operators are directed to stop RHR pumps if RCS level is less than one foot above the hot leg centerline. If RCS level can be maintained greater than one foot above the hot leg centerline, then RHR cooling can be recovered using either available RHR train. If operators fail to trip RHR pumps prior to level reaching one foot below the hot leg centerline, then pump damage is assumed to occur. If early diagnosis and RHR recovery are not successful, then operators can establish gravity or forced feed to control RCS level. Once RCS refill is accomplished, the operators establish conditions to restart an RHR pump for shutdown cooling. This scenario uses a representative loss of inventory flow rate of 500 gpm. The selection of flow rate is informed by plant-specific deterministic analysis of a loss of RHR scenario during POS 5A leading to lifting of RHR relief valves (see Section 5.3 for additional discussion).

LOOP Response in POS 5A:

The operator response to loss of offsite power is governed by AOPs for loss of electrical systems and loss of RHR. Operators are directed to try to restore power to any AC emergency bus from its associated diesel generator. If RHR is not able to be placed into service, then operators are directed to initiate the LORHR AOP. If a bus cannot be energized with a diesel generator, then operators will continue efforts to restore power to at least one AC emergency bus using any available power supply. If power is successfully restored, then operators will restore steam generator cooling or RHR, or refill the RCS with CCPs or SIPs.

LO4160V and LONSCW event trees are not modeled for POS 5A. The initial conditions and time to boil, equipment status, and modeled HFEs for POS 5A are provided in Table 4-16, Table 4-17, and Table 4-18, respectively.

Table 4-16 POS 5A Initial Conditions and Time to Boil

Plant Condition	Value	Notes
Time After Reactor Shutdown	12 hr	
Decay Heat	18.7 MW	
Initial RCS Temperature	180°F	
Initial RCS Pressure	350 psig	
Initial RCS Water Level	pressurizer solid	
Time to SG Dryout	22 hr	Based on plant-specific MELCOR calculations.
Time to Core Boiling (to T_{sat})	23.7 hr	Timing assumes no operator intervention.
Time to boiloff to within one foot of hot leg centerline	23.9 hr	Timing assumes no operator intervention.
Time to Core Damage	27.1 hr	Timing assumes no operator intervention.

Table 4-17 POS 5A Equipment Status

Equipment	Status	Remarks
RHR pumps	2 of 2 running	
Normal charging pump	Operating	Let down is in manual.
Reactor Coolant Pumps	2 of 4 running	
Centrifugal charging pumps	2 of 2 in PTL	
Safety injection pumps	2 of 2 in PTL and isolated	One of two pumps maintained in a state ready to be restored with manual actions from the control room.
Safety injection signals	Blocked	
Accumulators	Isolated	
Containment spray pumps	Not operable	
Steam generators and feedwater	Available, 2 AFW MDPs and 1 AFW TDP not in service	AFW pumps not in service, but available to be restored.
Pressurizer PORV/block valves	Armed at COPS setpoints with block valves open	
Pressurizer safety relief valves	Closed	
COPS (cold over-pressurization protection system)	Armed with two PORVs and two RHR suction relief valves	
Reactor vessel head	tensioned	
Emergency diesel generators	1 of 2 in standby, 1 available for local operation	EDG A automatic actuation is unavailable due to sequencer testing but could be operated locally.
Main generator and unit auxiliary transformers	Not in service	
Reserve auxiliary transformers	2 of 2 in service	
Containment equipment hatch	Open	
Containment purge vent	Mini purge (14-inch) operating	
Containment personnel airlock	Open	
Containment cooler units	Shutdown	
Containment sump	Sump screens covered	Protective covers are installed in Mode 5 causing sump to be unavailable.

Table 4-18 POS 5A Human Failure Events

Human Failure Event	Human Failure Description	Estimated Time Available	Modeling Approach
LORHR-Related HFEs			
1-RHR-XHE-XD-3H-5A	Operator fails to diagnose LORHR prior to loss of secondary cooling due to SG dryout.	~20 hours assuming no feedwater to SGs	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-TRN-POS5A	Operator fails to recover either RHR train prior to loss of secondary cooling due to SG dryout. Similar action to POS 4; same screening value is selected.	~20 hours assuming no feedwater to SGs	Use same HFE for similar conditions: 1-RHR-XHE-XL-TRN-POS4
1-AFW-XHE-FO-SGCOOL	Operator fails to feed SGs to maintain level and control RCS temperature using steam dumps or ARVs.	~20 hours	HFE evaluated for quantification (see Section 7)
1-HPI-XHE-XM-FFFEED-6H-5A	Operator fails to establish forced feed injection. Action is similar to POS 5B and same value is used.		HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-LTREC-LT5H	Operator fails to re-establish normal RHR cooling within 5 hours. RCS level can be maintained for approximately 5 hours with controlled injection from RWST using CCPs or SIPs.	5 hours before RWST water volume is exhausted	HFE evaluated for quantification (see Section 7)
LOINV-Related HFEs			
1-LOI-XHE-XD-POS5A	Operator fails to diagnose loss of inventory before level reaches 1 foot below hot leg centerline and isolate leak.	40 minutes	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XM-TRIP-M5LOI	Operator fails to diagnose LOI and trip RHR pumps after being damaged by air ingestion. Procedure will prompt operator to monitor RCS level. Assume pump damage occurs if level reaches 1 foot below hot leg centerline.	~10 minutes	Conservatively use HFE for more restrictive conditions assessed for POS 5B
1-RHR-XHE-XL-TRA-POS5A	Operator fails to align standby RHR train	~2 hours	Conservatively use HFE for more restrictive conditions: 1-RHR-XHE-XL-TRN-M5LOI
1-HPI-XHE-XM-FFFEED-LOI5A	Operator fails to establish forced feed injection.	~2 hours	HFE evaluated for quantification (see Section 7)
1-LOI-XHE-ISO2-LT	Operators fail late isolation of inventory loss after successful forced feed.	~5 hours	HFE evaluated for quantification (see Section 7)

Table 4-18 POS 5A Human Failure Events (cont.)

Human Failure Event	Human Failure Description	Estimated Time Available	Modeling Approach
LOOP-Related Repair Events			
1-EPS-XHE-XL-NR04H	Operator Fails to Recover Emergency Diesel in 4 Hours - based on unplanned EDG outage times	4 hours	Quantification based on EDG repair data (NRC, 2021a)
1-OEP-XHE-XL-NR02H	Operator Fails to Recover Offsite Power in 2 Hours – based on LOOP duration data	2 hours	Quantification based on LOOP duration data (NRC, 2021a)

4.4.4 POS 5B Initiating Event Response

In POS 5B, both RHR trains are operating. The RCS temperature is assumed to be 110°F. During this POS, the RCS is being drained from water solid conditions in preparation for draining to mid-loop. About two-thirds of the POS duration is spent with RCS level about 2 feet below the reactor vessel head flange, and this RCS level is assumed at the time of the initiating event. The containment equipment hatch and personnel airlock are open during POS 5B.

LORHR Response in POS 5B:

If level is greater than one foot above the hot leg centerline, then the LORHR AOP directs operators to start an available RHR pump. If RCS core exit temperature is greater than 200°F, then the operators are directed to refill the RCS using SIPs via hot leg injection or the CCPs via cold leg injection either to the 50-percent pressurizer level or until RHR cooling is restored. Once RCS refill is accomplished, the operators establish conditions to restart an RHR pump for shutdown cooling.

LOINV Response in POS 5B:

The operator response is guided by AOPs for RCS leakage and loss of RHR. Operators will monitor RCS inventory level and RHR pumps for indication of cavitation. Operators are directed to stop RHR pumps if RCS level is less than one foot above the hot leg centerline. If RCS level can be maintained above that level, then RHR cooling can be recovered using either available RHR train. If operators fail to trip RHR pumps prior to level reaching one foot below the hot leg centerline, then pump damage is assumed to occur. If early diagnosis and RHR recovery are not successful, then operators can establish gravity or forced feed to control RCS level. Once RCS refill is accomplished, the operators establish conditions to restart an RHR pump for shutdown cooling. This POS involves draining the RCS, which could result in an inadvertent draining or displacement of inventory that could impact RHR cooling.

LO4160V in POS 5B:

The operator response is guided by the AOP for loss of power to a class 1E 4,160V AC bus. If diesel generator B does not start automatically, the operators are directed to start the diesel generator manually. After aligning the diesel generator, operators are directed to proceed to recover RHR following the steps listed above for the loss of RHR response in POS 5B. If RHR

train B cannot be restored, then RHR train A is available and not affected by the initiating event. Operators are directed to verify an available RHR train is providing adequate shutdown cooling. RHR train A flow and heat exchanger outlet and bypass valves may require adjusting to establish adequate shutdown cooling.

LOOP in POS 5B:

The operator response is guided by AOPs for loss of class 1E electrical systems and loss of RHR. Operators will monitor RCS inventory level and RHR pumps for indication of cavitation. Operators are directed to try to restore power to any AC emergency bus from its associated diesel generator. If RHR is not able to be placed into service, then operators are directed to initiate the LORHR AOP. If a bus cannot be energized with a diesel generator, then operators will continue efforts to restore power to at least one AC emergency bus using any available power supply. If power is successfully restored, then operators will restore RHR or refill the RCS with CCPs or SIPs. If a station blackout condition occurs, then operators may establish gravity feed from the RWST to control RCS inventory. Successful gravity feed extends the sequence to allow additional time for power recovery and restoration of RHR.

LONSCW in POS 5B:

The operator response is guided by AOPs for loss of nuclear service cooling water system is and loss of RHR. Operators are directed to diagnose and verify the NSCW system status and plant status. If neither train of NSCW is verified to be operating, then the operators are directed to place one NSCW train in single pump operation. If required systems, including RHR and CCW, are unavailable, then the procedure directs operators to initiate appropriate AOPs for loss of CCW and loss of RHR. The LORHR AOP directs operators to control RCS level and provides steps to restore RHR cooling with verification that NSCW and CCW cooling functions have been restored. If RHR cooling cannot be restored, then operators are directed to makeup RCS level using gravity feed from the RWST. Successful gravity feed extends the sequence to allow additional time for restoring NSCW and RHR cooling.

The initial conditions and time to boil, equipment status, and modeled HFEs for POS 5B are provided in Table 4-19, Table 4-20, and Table 4-21, respectively.

Table 4-19 POS 5B Initial Conditions and Time to Boil

Plant Condition	Value	Notes
Time After Reactor Shutdown	55 hr	POS 5B is defined to occur between 42 and 82 hours after reactor shutdown. Fifty-five (55) hours is estimated as the time after draining from pressurizer solid conditions to a level just below the reactor head flange.
Decay Heat	13.6 MW	
Initial RCS Temperature	110°F	
Initial RCS Pressure	0 psig	
Initial RCS Water Level	Two feet below the reactor vessel head flange	
Time to Core Boiling (to T_{sat})	0.47 hr	Estimate based on plant-specific calculation of inventory heat up, assuming no inventory losses.
Time to boiloff to within one foot of hot leg centerline	2.3 hr	Timing assumes no operator intervention.
Time to Core Damage	5.3 hr	Timing assumes no operator intervention.

Table 4-20 POS 5B Equipment Status

Equipment	Status	Remarks
RHR pumps	2 of 2 running	
Normal charging pump	Operating	Let down is in manual.
Reactor Coolant Pumps	0 of 4 running	
Centrifugal charging pumps	2 of 2 in standby	
Safety injection pumps	2 of 2 in PTL and isolated	One of two pumps maintained in a state ready to be restored with manual actions from the control room.
Safety injection signals	Blocked	
Accumulators	Isolated	
Containment spray pumps	0 of 2 operable	In Mode 5, containment spray pumps are in pull-to-lock (PTL).
Steam generators and feedwater	Not available	Since the RCS is vented, the unavailability of the steam generators during this period has no effect.
Pressurizer PORV/block valves	Not applicable (3 pressurizer safety valves are removed)	
Pressurizer safety relief valves	3 of 3 are removed	
COPS (cold over-pressurization protection system)	Armed with two PORVs and RHR suction relief valves	
Reactor vessel head	Tensioned	
Emergency diesel generators	2 of 2 in standby	
Main generator and unit auxiliary transformers	Not in service	
Reserve auxiliary transformers	2 of 2 in service	RAT maintenance may occur during outage but is typically performed during Mode 6.

Table 4-20 POS 5B Equipment Status (cont.)

Equipment	Status	Remarks
Containment equipment hatch	Open	
Containment purge vent	Mini purge (14-inch) operating	
Containment personnel airlock	Open	
Containment cooler units	Shutdown	
Containment sump	Sump screens covered	Protective covers are installed in Mode 5 causing sump to be unavailable.

Table 4-21 POS 5B Human Failure Events

Human Failure Event	Human Failure Description	Estimated Time Available	Modeling Approach
LORHR-Related HFEs			
1-RHR-XHE-XD-3H	Operator fails to diagnose LORHR prior to RCS level reaching one foot above hot leg centerline. AOP directs operators to start an available RHR pump if level is greater than one foot above hot leg centerline.	0.5 hours prior to boiling	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-TRN-3H	Operator fails to recover an available RHR train prior to RCS level reaching one foot above hot leg centerline.	2 hours prior to boiloff to one foot above hot leg centerline	HFE evaluated for quantification (see Section 7)
1-HPI-XHE-XM-FFEEED-6H	Operator fails to establish injection to control RCS level prior to core damage.	5 hours prior to core damage	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-LTREC	Operator fails to re-establish normal RHR cooling within 6 hours. RCS level can be maintained for approximately 6 to 8 hours with controlled injection from RWST using CCPs or SIPs.	6 hours before RWST water volume is exhausted	HFE evaluated for quantification (see Section 7) (see Section 8.3.1)
LOINV-Related HFEs			
1-RHR-XHE-XM-M5LOI-I	Operator fails to diagnose and isolate LOINV prior to RCS level reaching one foot above hot leg centerline for drain down rate of 100 gpm	1.75 hours for drain down to one foot above hot leg centerline	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XM-LOI-POS5B	Operator fails to diagnose and isolate LOINV prior to RCS level reaching one foot above hot leg centerline for drain down rate of 1000 gpm	10 minutes for drain down to one foot above hot leg centerline	HFE evaluated for quantification (see Section 7)
1-HPI-XHE-XM-FFEEED-LOI	Operator fails to establish forced feed prior to core damage.	3 hours prior to core damage	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XM-TRIP-M5LOI	Operator fails to diagnose LOI and trip RHR pumps after being damaged by air ingestion. Procedure will prompt operator to monitor RCS level. Assume pump damage occurs if level reaches one foot below hot leg centerline.	10 min to reach level one foot below hot leg centerline after cues	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-TRN-M5LOI	Operator fails to recover an available RHR train prior to RCS boiling.	16 minutes before boiling	HFE evaluated for quantification (see Section 7)
1-LPI-XHE-XM-GFEEED-LOI	Operator fails to establish gravity feed prior to boiling.	16 minutes before boiling	HFE evaluated for quantification (see Section 7)

Table 4-21 POS 5B Human Failure Events (cont.)

Human Failure Event	Human Failure Description	Estimated Time Available	Modeling Approach
1-LOI-XHE-ISO2-LT	Operators fail late isolation of inventory loss after successful forced feed.	~5 hours	HFE evaluated for quantification (see Section 7)
LOOP-Related HFEs			
1-RHR-XHE-XD-XX	Operator fails to diagnose LORHR prior to RCS level reaching one foot above hot leg centerline. AOP directs operators to start an available RHR pump if level is greater than one foot above hot leg centerline. Assuming offsite power or EDGs are available, this event should be similar to LORHR action 1-RHR-XHE-XD-3H.	2 hours prior to boiloff to one foot above hot leg centerline	Use same HFE for similar conditions: 1-RHR-XHE-XD-3H
1-RHR-XHE-XL-TRN-XX	Operator fails to recover an available RHR train prior to RCS level reaching one foot above hot leg centerline. Assuming offsite power or EDGs are available this event should be similar to LORHR action 1-RHR-XHE-XL-TRN-3H.	2 hours prior to boiloff to one foot above hot leg centerline	Use same HFE for similar conditions: 1-RHR-XHE-XL-TRN-3H
1-HPI-XHE-XM-FFEEED-XX	Operator fails to establish injection to control RCS level prior to core damage. Assuming power is recovered this event should be similar to LORHR action 1-HPI-XHE-XM-FFEEED-6H.	5 hours prior to core damage	Use same HFE for similar conditions: 1-HPI-XHE-XM-FFEEED-6H
1-LPI-XHE-XM-GFEEED-LOOP	Fail to locally establish gravity feed before boiling during SBO.	~15 minutes	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-TRN-LOOP	Fail to recover one train of RHR after recovery of offsite power.	6 hours to core damage after successful refill by gravity feed	HFE evaluated for quantification (see Section 7)
1-OEP-XHE-XL-NR01H	Operator Fails to Recover Offsite Power in 1 Hour – based on LOOP duration data	1 hour	Quantification based on LOOP duration data (NRC, 2021a)
LO4160V-Related HFEs			
1-RHR-XHE-XL-TRN-STBY-2H	Operator fails to confirm status of one RHR pump to provide shutdown cooling and verify adequate flow to maintain RCS temperature.	2 hours is assumed if no heat removal is being provided by the unaffected RHR train	HFE evaluated for quantification (see Section 7)

Table 4-21 POS 5B Human Failure Events (cont.)

Human Failure Event	Human Failure Description	Estimated Time Available	Modeling Approach
LONSCW-Related HFEs			
1-LPI-XHE-XM-GFEED-NSCW	Operator fails to establish gravity feed before boiling	30 minutes	HFE evaluated for quantification (see Section 7)
1-HPI-XHE-XM-FFEED-NSCW	Operator fails to establish forced feed after NSCW recovery	2 hours	HFE assessed for more restrictive conditions in POS 6
1-OA-OSW-----H	Operators fail to establish single pump operation after loss of NSCW	30 minutes	HFE adopted from at-power PRA (NRC, 2022)

4.4.5 POS 6 Initiating Event Response

In POS 6, one RHR train is operating, and one RHR train is in standby and aligned for shutdown cooling. The RCS temperature is assumed to be 110°F. During this POS, the RCS is being drained from a level near the reactor vessel flange to just above the center line of the hot leg. The level remains near mid-loop for most of the POS while steam generator nozzle dams are installed. The specific RCS level is one foot above the hot leg centerline, which is noted in the instructions for draining to mid-loop in procedure. The containment equipment hatch is closed and secured with a minimum of four bolts. The emergency and personnel airlocks are closed during POS 6. All other containment penetrations are capable of being closed or isolated according to the containment procedure verification checklist.

LORHR Response in POS 6:

If level is greater than one foot above the hot leg centerline, then the LORHR AOP directs operators to start an available RHR pump. Although the initial level is assumed at one foot above the hot leg centerline (not greater than this level), it is assumed that operators will attempt to restore an available RHR pump prior to RCS temperature reaching 200°F. If RCS core exit temperature is greater than 200°F, then the operators are directed to refill the RCS using SIPs via hot leg injection or the CCPs via cold leg injection either to the 50-percent pressurizer level or until RHR cooling is restored. If RCS level cannot be restored greater than one foot above the hot leg centerline, then operators are directed to align gravity feed from the RWST to restore level. Once RCS refill is accomplished, the operators establish conditions to restart an RHR pump for shutdown cooling.

LOINV Response in POS 6:

The operator response is guided by AOPs for RCS leakage and loss of RHR. Operators will monitor RCS inventory level and RHR pumps for indication of cavitation. Operators are directed to stop RHR pumps if RCS level is less than one foot above the hot leg centerline. If RCS level can be maintained above that level, then RHR cooling can be recovered using either available

RHR train. If operators fail to trip operating RHR pumps prior to level reaching one foot below the hot leg centerline, then pump damage is assumed to occur. If early diagnosis and RHR recovery are not successful, then operators can establish gravity or forced feed to control RCS level. Once RCS refill is accomplished, the operators establish conditions to restart an RHR pump for shutdown cooling.

ODML Response in POS 6:

The operator response to overdraining the RCS when entering mid-loop is governed by the relevant section of the loss of RHR AOP. Operators may diagnose and stop the overdrain before RHR cooling is lost. If that is not successful, then RCS level will need to be raised before RHR can be restored. Injection with CCPs or SIPs can provide alternate decay heat removal and delay the onset of core damage until the RWST inventory is exhausted. If operators fail to trip operating RHR pumps in response to the overdrain, then pump damage is assumed to occur.

LO4160V in POS 6:

The operator response is guided by the AOP for loss of power to a class 1E 4,160V AC bus. If diesel generator B does not start automatically, the operators are directed to start the diesel generator manually. After aligning the diesel generator, and adjusting bus frequency to 60HZ, operators are directed to proceed to recover RHR following the steps listed above for the loss of RHR response in POS 6. If RHR train B cannot be restored, then RHR train A is available and not affected by the initiating event. Operators are directed to verify an available RHR train is providing adequate shutdown cooling. RHR train A flow and heat exchanger outlet and bypass valves may require adjusting to establish adequate shutdown cooling.

LOOP Response in POS 6:

The operator response is guided by AOPs for loss of class 1E electrical systems and loss of RHR. If neither of the class 1E 4,160V AC buses are energized, then the AOP directs operators to other procedures based on the operating mode of the plant. If a bus is energized, then operators will place a train of RHR in service for shutdown cooling. If RHR is not able to be placed into service, then operators are directed to initiate the LORHR AOP. If a bus cannot be energized with a diesel generator, then operators will continue efforts to restore power to at least one AC emergency bus using any available power supply (i.e., either of the two RATs or the SAT). After power is restored, operators will check for proper operation of the NSCW, CCW, and RHR systems.

If emergency diesel generators successfully start and power the safety buses, then there is minimal impact on the time available for operators to restore the decay heat removal function. The event tree sequences with successful diesel generators are treated in the same way as the LORHR response. If diesel generators fail, then 30 minutes allotted for restoration of offsite power. If offsite power restoration is successful, then it is assumed that 30 minutes have elapsed, and RCS boiling has begun. At this time no credit is given for restoring RHR or aligning gravity feed. A success path can be achieved by establishing forced injection and successful late recovery of RHR cooling.

LONSCW in POS 6:

The operator response is guided by AOPs for loss of nuclear service cooling water system is and loss of RHR. Operators are directed to diagnose and verify the NSCW system status and plant status. If neither train of NSCW is verified to be operating, then the operators are direct to place one NSCW train in single pump operation. If required systems, including RHR and CCW, are unavailable, then the procedure directs operators to initiate appropriate AOPs for loss of CCW and loss of RHR. The LORHR AOP directs operators to control RCS level and provides steps to restore RHR cooling with verification that NSCW and CCW cooling functions have been restored. If RHR cooling cannot be restored, then operators are directed to makeup RCS level using gravity feed from the RWST. Successful gravity feed extends the sequence to allow additional time for restoring NSCW and RHR cooling.

The initial conditions and time to boil, equipment status, and modeled HFEs for POS 6 are provided in Table 4-22, Table 4-23, and Table 4-24, respectively.

Table 4-22 POS 6 Initial Conditions and Time to Boil

Plant Condition	Value	Notes
Time After Reactor Shutdown	85 hr	POS 6 begins at 82 hours after reactor shutdown. Three hours are assumed for draining from near the flange to mid-loop level with drain flow of 40 gpm.
Decay Heat	11.8 MW	
Initial RCS Temperature	110°F	
Initial RCS Pressure	0 psig	
Initial RCS Water Level	One foot above hot leg centerline	Normal operating level for mid-loop.
Time to Core Boiling (to T_{sat})	0.31 hr	Estimate based on plant-specific calculation of inventory heat up, assuming no inventory losses.
Time to boiloff to within one foot of hot leg centerline	0.31 hr	
Time to Core Damage	3.3 hr	Timing assumes no operator intervention.

Table 4-23 POS 6 Equipment Status

Equipment	Status	Remarks
RHR pumps	1 of 2 running	Standby RHR train aligned for shutdown cooling.
Normal charging pump	Operating	Letdown is in manual.
Centrifugal charging pumps	2 of 2 in standby	
Safety injection pumps	2 of 2 in PTL and isolated	One of two pumps maintained in a state ready to be restored with manual actions from the control room.
Safety injection signals	Blocked	
Accumulators	Isolated	
Containment spray pumps	0 of 2 operable	
Steam generators and feedwater	Not available	Since the RCS is vented, the unavailability of the steam generators during this period has no effect.
Pressurizer PORV/block valves	Not applicable (3 pressurizer safety valves are removed)	
Pressurizer safety relief valves	3 of 3 are removed	
COPS (cold over-pressurization protection system)	Armed with two PORVs and RHR suction relief valves	
Reactor vessel head	Preparing to de-tension first reactor head bolt	Reactor head disassembly and removal does not occur until RCS level is restored to just below the vessel head flange, but de-tensioning of the first reactor head bolt may occur with reduced inventory during POS 6.
Emergency diesel generators	2 of 2 in standby	
Main generator and unit auxiliary transformers	Not in service	
Reserve auxiliary transformers	2 of 2 in service	
Containment equipment hatch	Closed and secured with a minimum of four bolts	
Containment purge vent	Off and closed	
Containment personnel airlock	One door closed	
Containment cooler units	Shutdown	
Containment sump	Sump screens covered	Protective covers are installed in Mode 5 causing sump to be unavailable.

Table 4-24 POS 6 Human Failure Events

Human Failure Event	Human Failure Description	Time Available for Action	Modeling Approach
LORHR-Related HFEs			
1-RHR-XHE-XD-ML	Operator fails to diagnose LORHR prior to RCS level reaching one foot above hot leg centerline (initial level at time of IE). AOP directs operators to start an available RHR pump if level is greater than one foot above hot leg centerline. RHR recovery will be attempted until reaching 200°F, then operators will be directed to establish injection to control level.	5 minutes after LORHR occurs	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-TRN-POS6	Operator fails to recover an available RHR train prior to RCS temperature reaching 200°F.	20 minutes prior to boiling with level at one foot above hot leg centerline	HFE evaluated for quantification (see Section 7)
1-HPI-XHE-XM-FFEE-MLDRL	Operator fails to establish pumped injection to control RCS level prior to core damage.	3 hours prior to core damage	HFE evaluated for quantification (see Section 7)
1-LPI-XHE-XM-GFEE-MLDRL	Operator fails to establish gravity feed to control RCS level prior to core damage.	3 hours prior to core damage	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-LTREC	Operator fails to re-establish normal RHR cooling within 6 hours. RCS level can be maintained for approximately 6 to 8 hours with controlled injection from RWST using CCPs or SIPs.	~8 hours before RWST water volume is exhausted	HFE evaluated for quantification (see Section 7)
ODML Overdrain-Related HFEs			
1-RHR-XHE-XM-MLDRL-I	Operator fails to diagnose and isolate drain down prior to level below one foot above hot leg centerline.	15 minutes	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XM-TRIP-MLDRL	Operator fails to identify impact of reduced RCS level and trip RHR pumps before they are damaged.	15 minutes	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-TRN-MLDRL	Operator fails to recover an available RHR train prior to boiling.	15 minutes prior to boiling	HFE evaluated for quantification (see Section 7)
LOINV-Related HFEs			
1-RHR-XHE-XM-LOISP-POS6	Operator fails to diagnose and isolate LOINV prior to RCS level reaching one foot above hot leg centerline for drain down rate of 100 gpm	20 minutes	HFE evaluated for quantification (see Section 7)

Table 4-24 POS 6 Human Failure Events (cont.)

Human Failure Event	Human Failure Description	Time Available for Action	Modeling Approach
1-RHR-XHE-XM-LOI-POS6	Operator fails to diagnose and isolate LOINV prior to RCS level reaching one foot above hot leg centerline for drain down rate of 1000 gpm	Few minutes	HFE evaluated for quantification (see Section 7)
1-HPI-XHE-XM-FFEE-LOI-6	Operator fails to establish forced feed prior to core damage.	3 hours prior to core damage	HFE evaluated for quantification (see Section 7)
1-LPI-XHE-XM-GFEE-LOI-6	Operator fails to establish gravity feed prior to boiling.	16 minutes before boiling	HFE evaluated for quantification (see Section 7)
LO4160V-Related HFEs			
1-RHR-XHE-XL-TRN-STBY-ML	Operator fails to confirm status of one RHR pump to provide shutdown cooling and verify adequate flow to maintain RCS temperature.	15 min. is assumed if no heat removal is being provided by the unaffected RHR train	HFE evaluated for quantification (see Section 7)
LOOP-Related Repair Events			
1-EPS-XHE-XL-NR04H	Operator Fails to Recover Emergency Diesel in 4 Hours - based on unplanned EDG outage times	4 hours	Quantification based on EDG repair data (NRC, 2021a)
1-OEP-XHE-XL-NR30M	Operator Fails to Recover Offsite Power in 30 min. – based on LOOP duration data	30 min.	Quantification based on LOOP duration data (NRC, 2021a)
1-LPI-XHE-XM-GFEE-LOOP	Fail to locally establish gravity feed before boiling during SBO.	~15 minutes	HFE evaluated for quantification (see Section 7)
LONSCW-Related HFEs			
1-LPI-XHE-XM-GFD-ML-NSCW	Operator fails to establish gravity feed before boiling	30 minutes	HFE evaluated for quantification (see Section 7)
1-HPI-XHE-XM-FFEE-NSCW	Operatory fails to establish forced feed after NSCW recovery	2 hours	HFE evaluated for quantification (see Section 7)
1-OA-OSW-----H	Operators fail to establish single pump operation after loss of NSCW	30 minutes	HFE adopted from at-power PRA (NRC, 2022)

4.4.6 POS 7 Initiating Event Response

POS 7 covers the transition from reduced inventory to refueling cavity fully flooded, and plant conditions are changing throughout as the reactor vessel head is removed and the containment equipment hatch is re-opened. POS 7 begins with RCS level increasing from mid-loop in POS 6. Level is increased to just below the reactor flange for disassembly and removal of the reactor vessel head. After the reactor vessel head is removed and placed on its stand, the containment equipment hatch is re-opened. The conditions with the containment hatch opened are the focus

for modeling POS 7. The total POS duration is approximately 45 hours and the first approximately 30 hours involve preparation and removal of the reactor vessel head. Therefore, the modeled duration is assumed to be 15 hours to represent only the portion of POS 7 with the containment equipment hatch open.¹⁴ After head removal the refueling cavity continues to be filled. The representative inventory level for this condition is selected to be refueling cavity flooded to 13 feet above the reactor vessel head flange.

RHR train B is operating to provide shutdown cooling during this period, with RHR train A being used to gravity fill the refueling cavity from the RWST. RHR train A is assumed available for shutdown cooling if needed. The fuel transfer tube valve has been opened prior to the start of filling the lower reactor cavity. Both the refueling cavity and the transfer canal volumes are filled together.

With the refueling cavity filled to 13 feet above the reactor vessel head flange, approximately 500,000 gallons remain in the RWST for injection to RCS. The RWST empty level alarm occurs when 105,000 gallons are remaining. This leaves 395,000 gallons available for injection. The established injection flow rate is assumed to be 200 gpm. At this injection rate, the RWST could provide injection for approximately 33 hours before refill of the RWST would be required. The LORHR AOP directs operators to transfer water from the SFP to the RWST if cooling cannot be established. Approximately 20,000 gallons would be available before reaching the SFP low level alarm setpoint.

Loss of RHR Response in POS 7:

The operator response is governed by the AOP for loss of RHR cooling. In general, the operators will attempt to maintain RCS temperature and level, address containment-related concerns, and restore RHR cooling. If the CCPs are available, the operators are directed to use them to maintain temperature by taking suction from the RWST and injecting via the normal charging path to an intact cold leg or injecting via the alternate charging path to an intact cold leg or using the boron injection tank (BIT) flow path. If the CCPs are not available, the operators are directed to use SI pumps to maintain temperature by taking suction from the RWST and injecting via a cold leg. The operators are then directed to transfer water from the SFP to the RWST to stabilize SFP level. During POS 7, the refueling cavity/transfer canal and SFP are isolated by the fuel pool to transfer canal gate. Transferring water to the RWST would only be effective if water is being lost outside of the refueling cavity. It is assumed gravity fill from the RWST would continue until the refueling cavity/transfer canal level is equal to the SFP level. Therefore, transferring water from the SFP to the RWST is not credited. The operators will attempt to restore RHR and/or CCW.

Loss of Inventory Response in POS 7:

The operator response is guided by AOPs for RCS leakage and loss of RHR. If operators diagnose and isolate the leak with inventory level still in the refueling cavity, then RHR can be restored by aligning the standby train of RHR, which is aligned for gravity filling the refueling cavity. If the leak is not diagnosed before reaching RCS reduced inventory level, then operators will have additional cues and indication of RHR pump cavitation. Operators are directed to stop

¹⁴ The risk contribution from the first 30 hours of POS 7 is not accounted for in the study, though it is expected to be substantially less than for the 15 hours of POS 7 that are modeled. Nonetheless, modeling of the additional 30 hours of POS 7 is identified as a potential area for future work in Appendix A

the running RHR pump if RCS level is less than one foot above the hot leg centerline. If RCS level can be maintained higher than one foot above the hot leg centerline, then RHR cooling can be recovered using the available RHR train. If early diagnosis and RHR recovery are not successful, then operators can establish gravity or forced feed to control RCS level. Once RCS refill is accomplished, the operators establish conditions to restart an RHR pump for shutdown cooling.

LO4160V in POS 7:

The operator response is guided by the AOP for loss of power to a class 1E 4,160V AC bus. If diesel generator B does not start automatically, the operators are directed to start the diesel generator manually. After aligning the diesel generator, operators are directed to proceed to recover RHR following the steps listed above for the loss of RHR response in POS 7. If RHR train B cannot be restored, then RHR train A is available to provide shutdown cooling. RHR train A is aligned for gravity filling the refueling cavity. It is assumed the gravity fill will continue until the refueling cavity flooding is complete. Then, RHR train A can be re-aligned for shutdown cooling.

LOOP in POS 7:

The operator response is guided by AOPs for loss of class 1E electrical systems and loss of RHR. Operators are directed to try to restore power to any AC emergency bus from its associated diesel generator. If RHR is not able to be placed into service, then operators are directed to initiate the LORHR AOP. If a bus cannot be energized with a diesel generator, then operators will continue efforts to restore power to at least one AC emergency bus using any available power supply. If power is successfully restored, then operators will restore RHR or refill the RCS with CCPs or SIPs. If a station blackout condition occurs, then operators may establish gravity feed from the RWST to control RCS inventory. Successful gravity feed extends the sequence to allow additional time for power recovery and restoration of RHR.

The initial conditions and time to boil, equipment status, and modeled HFEs for POS 7 are provided in Table 4-25, Table 4-26, and Table 4-27, respectively.

Table 4-25 POS 7 Initial Conditions and Time to Boil

Plant Condition	Value	Notes
Time After Reactor Shutdown	126 hr	POS 7 is defined to occur between 96 and 141 hours after reactor shutdown. It is estimated that the containment equipment hatch is re-opened about two-thirds of the way into the POS.
Decay Heat	10.3 MW	
Initial RCS Temperature	110°F	
Initial RCS Pressure	0 psig	The first reactor vessel flange stud is de-tensioned in POS 6.
Initial RCS Water Level	13 feet above the reactor vessel head flange	
Time to Core Boiling (to T_{sat})	5.7 hr	This estimate is based on the rapid temperature increase approaching, but not reaching, saturation. The refueling cavity level will begin to fall due to evaporation without ever becoming saturated.
Time to Core Damage	46.4 hr	Based on plant-specific MELCOR calculation
LOI Time to Core Boiling (to T_{sat})	Approximately 5 hr	RCS water volume reduced by 10,000 gallons, which is assumed to be lost before isolation occurs.
LOI Time to Core Damage	From 4.4 hr to ~80 hr	The time to core damage varies depending if the inventory loss is diagnosed and isolated. The timing is bounded by the time to drain to mid-loop with maximum break flow rate (1 hr), plus the time to core damage for mid-loop conditions (3.4 hr).

Table 4-26 POS 7 Equipment Status

Equipment	Status	Remarks
RHR pumps	2 of 2 available	RHR train B is operating; RHR train A is aligned for gravity feed, but available if needed.
Normal charging pump	Not in service	
Centrifugal charging pumps	1 of 2 trains available, both pumps in PTL	One of two CCP trains available for injection.
Safety injection pumps	2 of 2 in PTL and isolated	Two of two pumps maintained in a state ready to be restored with manual actions from the control room.
Safety injection signals	Blocked	
Accumulators	Blocked; depressurized to <150 psig	
Containment spray pumps	0 of 2 operable	In Mode 5, containment spray pumps are in PTL.
Steam generators and feedwater	Unavailable - SGs in wet layup (filled with N2 at 95% WR)	Since the RCS is vented, the unavailability of the steam generators during this period has no effect.
Pressurizer PORV/block valves	Not applicable (3 pressurizer safety valves are removed and reactor vessel head removed)	
Pressurizer safety relief valves (3)	3/3 safeties removed	
COPS (cold over-pressurization protection system)	N/A	Based on technical specification bases for COPS
Reactor vessel head	De-tensioned/removed	
Emergency diesel generators	2 of 2 in standby	
Main generator and unit auxiliary transformers	Not in service	
Reserve auxiliary transformers	2 of 2 in service	RAT maintenance begins after refueling cavity is flooded.
Containment equipment hatch	Open	
Containment purge vent	Not in service, closed	
Containment personnel airlock	Open	
Containment cooler units	Shutdown	
Containment sump	Sump screens covered	Protective covers are installed in Mode 5 causing sump to be unavailable.

Table 4-27 POS 7 Human Failure Events

Human Failure Event	Human Failure Description	Time Available for Action	Modeling Approach
LORHR-Related HFEs			
1-RHR-XHE-XD-POS7	Operator fails to diagnose LORHR prior to RCS temperature reaching 200F. AOP directs operators to start an available RHR pump if temperature is less than 200F.	6 hours prior to boiling	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-TRB-POS7	Operator fails to recover an operating RHR train prior to reaching 200F.	6 hours prior to boiling	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-TRA-POS7	Operator fails to align standby train of RHR for shutdown cooling prior to reaching 200F.	6 hours prior to boiling	HFE evaluated for quantification (see Section 7)
1-HPI-XHE-XM-FFEEED-POS7	Operator fails to establish injection to control RCS level prior to core damage.	2 days prior to core damage	Use same HFE for similar conditions: 1-HPI-XHE-XM-FFEEED-POS8E
1-RHR-XHE-XL-LTREC-LT5H	Operator fails to re-establish normal RHR cooling within 5 hours. RCS level can be maintained for approximately 5 hours with controlled injection from RWST using CCPs or SIPs.	5 hours before RWST water volume is exhausted	HFE evaluated for quantification (see Section 7)
LOINV-Related HFEs			
1-RHR-XHE-XM-LOI-POS7	Operator fails to diagnose and isolate loss of inventory before loss of RHR.	1 hour	HFE evaluated for quantification (see Section 7)
LO4160V-Related HFEs			
1-RHR-XHE-XL-STBY-POS7	Fail to align shutdown cooling flow with available RHR train.	6 hours prior to boiling	HFE evaluated for quantification (see Section 7)
LOOP-Related Repair Events			
1-EPS-XHE-XL-NR04H	Operator Fails to Recover Emergency Diesel in 4 Hours - based on unplanned EDG outage times	4 hours	Quantification based on EDG repair data (NRC, 2021a)
1-OEP-XHE-XL-NR02H	Operator Fails to Recover Offsite Power in 2 Hours – based on LOOP duration data	2 hours	Quantification based on LOOP duration data (NRC, 2021a)

4.4.7 POS 8E Initiating Event Response

POS 8E (early or old core) begins at the start of fuel offload with level at 24.5 feet above the reactor vessel head flange. The entire core is offloaded to the SFP. During this period of approximately 68 hours, one RHR loop is running with RHR letdown isolated. The second RHR loop may be unavailable due to maintenance and its hot leg suction valve is closed. One SI pump and one centrifugal charging pump are available, and the others may be unavailable due

to maintenance. The equipment hatch is open but closeable in a short time. The mini-purge system is operating.

One out of the two Unit 1 spent fuel pool cooling trains is running. The other may be, and often is, in maintenance. Both spent fuel pool cooling trains from Unit 2 must be operable.

With the refueling cavity filled, approximately 399,000 gallons remain in the RWST for injection to the RCS. The RWST empty level alarm occurs when 105,000 gallons are remaining. This leaves 294,000 gallons available for injection. The established injection flow rate is assumed to be 200 gpm. At this injection rate, the RWST could provide injection for approximately 24 hours before refill of the RWST would be required. The LORHR AOP directs operators to transfer water from the SFP to the RWST if cooling cannot be established. Approximately 20,000 gallons would be available before reaching the SFP low level alarm setpoint.

Loss of RHR Response in POS 8E:

The operator response is governed by the AOP for loss of RHR cooling. In general, the operators will attempt to maintain RCS temperature and level, address containment-related concerns, and restore RHR cooling. If the CCPs are available, the operators are directed to use them to maintain temperature by taking suction from the RWST and either injecting via the normal charging path to an intact cold leg or injecting via the alternate charging path to an intact cold leg or using the BIT flow path. If the CCPs are not available, the operators are directed to use SI pumps to maintain temperature by taking suction from the RWST and injecting via a cold leg. The operators are then directed to transfer water from the SFP to the RWST to stabilize SFP level. During POS 8E, the refueling cavity and SFP are connected by opening of the transfer tube gate valve and the fuel pool to transfer canal gate. Transferring water to the RWST would only be effective if water is being lost outside of the refueling cavity. Therefore, transferring water from the SFP to the RWST is not credited. The operators will also attempt to restore RHR and/or CCW.

Loss of Inventory Response in POS 8E:

The operator response is guided by AOPs for RCS leakage and loss of RHR. If operators diagnose and isolate the leak with inventory level still in the refueling cavity, then RHR can be restored by aligning the standby train of RHR. Given the POS conditions assume RAT A maintenance is underway, operators must align alternate offsite power to train A before recovering RHR. If the leak is not diagnosed before reaching RCS reduced inventory level, then operators will have additional cues and indication of RHR pump cavitation. Operators are directed to stop the running RHR pump if RCS level is less than one foot above the hot leg centerline. If RCS level can be maintained higher than one foot above the hot leg centerline, then RHR cooling can be recovered using the available RHR train. If early diagnosis and RHR recovery are not successful, then operators can establish gravity or forced feed to control RCS level. Once RCS refill is accomplished, the operators establish conditions to restart an RHR pump for shutdown cooling.

LO4160V in POS 8E:

The operator response is guided by the AOP for loss of power to a class 1E 4,160V AC bus. If diesel generator B does not start automatically, the operators are directed to start the diesel generator manually. After aligning the diesel generator, operators are directed to proceed to

recover RHR following the steps listed above for the loss of RHR response in POS 8E. If RHR train B cannot be restored, then RHR train A is available to provide shutdown cooling. RHR train A is aligned for gravity filling the refueling cavity. It is assumed the gravity fill will continue until the refueling cavity flooding is complete. Then, RHR train A can be re-aligned for shutdown cooling.

LOOP in POS 8E:

The operator response is guided by AOPs for loss of class 1E electrical systems and loss of RHR. Operators are directed to try to restore power to any AC emergency bus from its associated diesel generator. If RHR is not able to be placed into service, then operators are directed to initiate the LORHR AOP. If a bus cannot be energized with a diesel generator, then operators will continue efforts to restore power to at least one AC emergency bus using any available power supply. Restoration of power to a safety-related bus by repair of an EDG is credited with available repair time of four hours. Core damage is estimated to occur at 82 hours, but there is no credit for closing the EDG output breakers after DC power is lost (safety-related batteries are assumed depleted after four hours). If power is successfully restored, then operators will restore RHR or refill RCS with CCPs or SIPs. If a station blackout condition occurs, then operators may establish gravity feed from the RWST to control RCS inventory. Successful gravity feed extends the sequence to allow additional time for power recovery and restoration of RHR.

The initial conditions and time to boil, equipment status, and modeled HFEs for POS 8E are provided in Table 4-28, Table 4-29, and Table 4-30, respectively.

Table 4-28 POS 8E Initial Conditions and Time to Boil

Plant Condition	Value	Notes
Time After Reactor Shutdown	141 hr	POS 8E is defined to occur between 141 and 209 hours after reactor shutdown.
Decay Heat	9.9 MW	
Initial RCS Temperature	110°F	
Initial RCS Pressure	0 psig	The first reactor vessel flange stud was de-tensioned in POS 6.
Initial RCS Water Level	24.5 feet above reactor vessel head flange	
Time to 200°F	6.9 hr	Per LORHR AOP, establish conditions to start RHR. Check hot leg temperature less than 200°F.
Time to Core Boiling (to T_{sat})	7.8 hr	
Time to Core Damage	82 hr	
LOI Time to Core Boiling (to T_{sat})	6.3 hr	RCS water volume reduced by 70,000 gallons, which is assumed to be lost before isolation occurs.
LOI Time to Core Damage	From 4.6 hr to ~80 hr	The time to core damage varies depending if the inventory loss is diagnosed and isolated. The timing is bounded by the time to drain to mid-loop with maximum break flow rate (1.2 hr), plus the time to core damage for mid-loop conditions (3.4 hr).

Table 4-29 POS 8E Equipment Status

Equipment	Status	Remarks
RHR pumps	1 of 2 available	RHR train B is running, RHR train A is unavailable for maintenance.
Normal charging pump	Not in service	
Centrifugal charging pumps	1 of 2 in PTL	One of two pumps available for injection.
Safety injection pumps	1 of 2 in PTL and isolated	One of two pumps maintained in a state ready to be restored with manual actions from the control room.
Safety injection signals	Blocked	
Accumulators	Blocked; depressurized to <150 psig	
Containment spray pumps	0 of 2 operable	In Mode 5, containment spray pumps are in PTL.
Steam generators and feedwater	Unavailable - SGs in wet layup (filled with N2 at 95% WR)	Since the RCS is vented, the unavailability of the steam generators during this period has no effect.
Pressurizer PORV/block valves	Not applicable (3 pressurizer safety valves are removed and reactor vessel head removed)	
Pressurizer safety relief valves (3)	3/3 safeties removed	
COPS (cold over-pressurization protection system)	N/A	Based on technical specification bases for COPS
Reactor vessel head	De-tensioned/removed	
Emergency diesel generators	1 of 2 in standby	
Main generator and unit auxiliary transformers	Not in service	
Reserve auxiliary transformers	1 of 2 in service	RAT maintenance begins after refueling cavity is flooded.
Containment equipment hatch	Open	
Containment purge vent	Not in service, but open	
Containment personnel airlock	Open	
Containment cooler units	Shutdown	
Containment sump	Sump screens covered	Protective covers are installed in Mode 5 causing sump to be unavailable.

Table 4-30 POS 8E Human Failure Events

Human Failure Event	Human Failure Description	Time Available for Action	Modeling Approach
LORHR-Related HFEs			
1-RHR-XHE-XD-POS8E	Operator fails to diagnose LORHR prior to RCS temperature reaching 200F. AOP directs operators to start an available RHR pump if temperature is less than 200F.	~7 hours prior to boiling	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-TRB-POS8E	Operator fails to recover an available RHR train prior to reaching 200F.	~7 hours prior to boiling	HFE evaluated for quantification (see Section 7)
1-HPI-XHE-XM-FFED-POS8E	Operator fails to establish injection to control RCS level prior to core damage.	~3 days prior to core damage	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-LTREC-LT5H	Operator fails to re-establish normal RHR cooling. RCS temperature can be maintained for approximately 3 hours based on decay heat level and reduced RWST inventory (see Table 5-2).	~3 hours for RHR recovery before RWST water volume is exhausted.	HFE evaluated for quantification (see Section 7)
LOINV-Related HFEs			
1-LOI-XHE-XD-POS8E	Fail to diagnose break before level at one foot above hot leg centerline.	1.25 hours	HFE evaluated for quantification (see Section 7)
1-LOI-XHE-XM-ISO-POS8E	Fail to trip RHR and isolate leak.	1.25 hours	HFE evaluated for quantification (see Section 7)
1-ACP-XHE-XL-ALTAC-6H	Fail to align alternate offsite power via SAT, or opposite train RAT, in 6 hours.	6 hours	HFE evaluated for quantification (see Section 7)
LOOP-Related HFEs			
1-EPS-XHE-XL-NR04H	Operator Fails to Recover Emergency Diesel in 4 Hours - based on unplanned EDG outage times	4 hours	Quantification based on EDG repair data (NRC, 2021a)
1-OEP-XHE-XL-NR02H	Operator Fails to Recover Offsite Power in 2 Hours – based on LOOP duration data	2 hours	Quantification based on LOOP duration data (NRC, 2021a)
LONSCW-Related HFEs			
1-HPI-XHE-XM-FFD-PRF-NSCW	Operatory fails to establish forced feed after NSCW recovery	>8 hours	HFE evaluated for quantification (see Section 7)
1-OA-OSW-----H	Operators fail to establish single pump operation after loss of NSCW	30 minutes	HFE adopted from at-power PRA (NRC, 2022)

4.4.8 POS 8L Initiating Event Response

POS 8L (late or new core) begins at the start of fuel reload with level at 24.5 feet above the reactor vessel head flange. The plant is in Mode 6 when the first fuel assembly is loaded. The entire core is loaded into the reactor from the SFP. Decay heat levels are substantially reduced in this POS compared to earlier POSs for two reasons. One is the time spent during the defueled POS. A second reason is the replacement of a fraction of the core with new fuel. It is assumed that 89 out of 193 fuel assemblies are replaced with new fuel. POS 8L is assumed to begin at 329 hours after reactor shutdown. At this time after shutdown, the decay heat power with the new fuel load is estimated to be 4.2 MW.

During the defueled window, the plant will typically swap the protected safety trains and perform testing and maintenance activities on the train that was previously protected in POS 8E. The system models assume the two trains are symmetric, so the swap is not expected to have an impact on the modeled plant response. As a modeling simplification, the same train unavailability is used for modeling POS 8E and POS 8L.

The initial conditions and time to boil, equipment status, and modeled HFEs for POS 8E are provided in Table 4-31, Table 4-32, and Table 4-33, respectively.

Table 4-31 POS 8L Initial Conditions and Time to Boil

Plant Condition	Value	Notes
Time After Reactor Shutdown	329 hr	POS 8L is defined to occur between 329 and 381 hours after reactor shutdown.
Decay Heat	4.2 MW	
Initial RCS Temperature	110°F	
Initial RCS Pressure	0 psig	
Initial RCS Water Level	24.5 feet above reactor vessel head flange	
Time to Core Boiling (to T_{sat})	18 hr	
Time to Core Damage	214 hr	

Table 4-32 POS 8L Equipment Status

Equipment	Status	Remarks
RHR pumps	1 of 2 available	One RHR train is running, other RHR train is unavailable for maintenance.
Normal charging pump	Not in service	
Centrifugal charging pumps	1 of 2 in PTL	One of two pumps available for injection.
Safety injection pumps	1 of 2 in PTL and isolated	One of two pumps maintained in a state ready to be restored with manual actions from the control room.
Safety injection signals	Blocked	
Accumulators	Blocked; depressurized to <150 psig	
Containment spray pumps	0 of 2 operable	In Mode 5, containment spray pumps are in PTL.
Steam generators and feedwater	Unavailable - SGs in wet layup (filled with N2 at 95% WR)	Since the RCS is vented, the unavailability of the steam generators during this period has no effect.
Pressurizer PORV/block valves	Not applicable (3 pressurizer safety valves are removed and reactor vessel head removed)	
Pressurizer safety relief valves (3)	3/3 safeties removed	
COPS (cold over-pressurization protection system)	N/A	Based on technical specification bases for COPS
Reactor vessel head	De-tensioned/removed	
Emergency diesel generators	1 of 2 in standby	
Main generator and unit auxiliary transformers	Not in service	
Reserve auxiliary transformers	1 of 2 in service	RAT maintenance begins after refueling cavity is flooded.
Containment equipment hatch	Open	
Containment purge vent	Not in service, but open	
Containment personnel airlock	Open	
Containment cooler units	Shutdown	
Containment sump	Sump screens covered	Protective covers are installed in Mode 5 causing sump to be unavailable.

Table 4-33 POS 8L Human Failure Events

Human Failure Event	Human Failure Description	Time Available for Action	Modeling Approach
LORHR-Related HFEs			
1-RHR-XHE-XD-POS8L	Operator fails to diagnose LORHR prior to RCS temperature reaching 200F. AOP directs operators to start an available RHR pump if temperature is less than 200F.	~18 hours prior to boiling	Use same HFE for similar conditions: 1-RHR-XHE-XD-POS8E
1-RHR-XHE-XL-TRN-POS8L	Operator fails to recover an available RHR train prior to reaching 200F.	~18 hours prior to boiling	Use same HFE for similar conditions: 1-RHR-XHE-XL-TRB-POS8E
1-HPI-XHE-XM-FFEED-POS8L	Operator fails to establish injection to control RCS level prior to core damage.	several days available prior to core damage	Use same HFE for similar conditions: 1-HPI-XHE-XM-FFEED-POS8E
1-RHR-XHE-XL-LTREC	Operator fails to re-establish normal RHR cooling within 6 hours. RCS level can be maintained for approximately 6 to 8 hours with controlled injection from RWST using CCPs or SIPs.	~6 hours before RWST water volume is exhausted	HFE evaluated for quantification (see Section 7)
LOINV-Related HFEs			
1-LOI-XHE-XD-POS8L	Fail to diagnose break before level at one foot above hot leg centerline.	1.25 hours	Use same HFE for similar conditions: 1-LOI-XHE-XD-POS8E
1-LOI-XHE-XM-ISO-POS8L	Fail to trip RHR and isolate leak.	1.25 hours	Use same HFE for similar conditions: 1-LOI-XHE-XM-ISO-POS8E
1-ACP-XHE-XL-ALTAC-6H	Fail to align alternate offsite power via SAT, or opposite train RAT, in 6 hours.	6 hours	HFE evaluated for quantification (see Section 7)
LOOP-Related HFEs			
1-EPS-XHE-XL-NR04H	Operator Fails to Recover Emergency Diesel in 4 Hours - based on unplanned EDG outage times	4 hours	Quantification based on EDG repair data (NRC, 2021a)
1-OEP-XHE-XL-NR02H	Operator Fails to Recover Offsite Power in 2 Hours – based on LOOP duration data	2 hours	Quantification based on LOOP duration data (NRC, 2021a)
LONSCW-Related HFEs			
1-HPI-XHE-XM-FFD-PRF-NSCW	Operator fails to establish forced feed after NSCW recovery	>8 hours	HFE evaluated for quantification (see Section 7)

Table 4-33 POS 8L Human Failure Events (cont.)

Human Failure Event	Human Failure Description	Time Available for Action	Modeling Approach
1-OA-OSW-----H	Operators fail to establish single pump operation after loss of NSCW	30 minutes	HFE adopted from at-power PRA (NRC, 2022)

4.4.9 POS 9 Initiating Event Response

POS 9 covers the activities during draining of the refueling cavity to 2 feet below the reactor vessel head flange. During this POS, either train of RHR is operating in shutdown cooling mode, with the opposite train of RHR operable and used for draining the RCS to the RWST until level is within one foot of the reactor vessel head flange. RHR letdown to the RWST is used to complete the draining.

POS 5B also represents the plant state with RCS level 2 feet below the reactor vessel head flange. The plant response for POS 9 after draining the RCS level is similar to POS 5B. The total duration of POS 9 is estimated to be 120 hours; however, only the latter part of POS 9 is similar to POS 5B, so the modeled exposure for POS 9 is reduced to include only the final 30 hours of the POS.¹⁵ This portion of POS 9 begins at 471 hours after shutdown. The decay heat power level is estimated to be 3.6 MW. Although the plant conditions and modeled plant response are similar to POS 5B, the lower decay heat level in POS 9 allows additional time for operator actions and could reduce the estimated human error probabilities.

LORHR Response in POS 9:

If level is greater than one foot above the hot leg centerline, then the operators are directed to start an available RHR pump. If RCS core exit temperature is greater than 200°F, then the operators are directed to refill the RCS using SIPs via hot leg injection or the CCPs via cold leg injection either to the 50-percent pressurizer level or until RHR cooling is restored. Once RCS refill is accomplished, the operators establish conditions to restart an RHR pump for shutdown cooling.

LOINV Response in POS 9:

The operator response is guided by AOPs for RCS leakage and loss of RHR. Operators will monitor RCS inventory level and RHR pumps for indication of cavitation. Operators are directed to stop RHR pumps if RCS level is less than one foot above the hot leg centerline. If RCS level can be maintained above that level, then RHR cooling can be recovered using either available RHR train. If operators fail to trip RHR pumps prior to level reaching one foot below the hot leg centerline, then pump damage is assumed to occur. If early diagnosis and RHR recovery are not successful, then operators can establish gravity or forced feed to control RCS level. Once RCS refill is accomplished, the operators establish conditions to restart an RHR pump for

¹⁵ The risk contribution from the first 90 hours of POS 9 is not accounted for in the study, though it is expected to be substantially less than for the 30 hours of POS 9 that are modeled. Nonetheless, modeling of the additional 90 hours of POS 9 is identified as a potential area for future work in Appendix A.

shutdown cooling. This POS involves draining the RCS, which could result in an inadvertent draining or displacement of inventory that could impact RHR cooling.

LO4160V in POS 9:

The operator response is guided by the AOP for loss of power to a class 1E 4,160V AC bus. If the diesel generator does not start automatically, the operators are directed to start the diesel generator manually. After aligning the diesel generator, operators are directed to proceed to recover RHR following the steps listed above for the loss of RHR response in POS 9. If the previously operating RHR train cannot be restored, then the standby RHR train is available and not affected by the initiating event. Operators are directed to verify an available RHR train is providing adequate shutdown cooling. RHR train flow and heat exchanger outlet and bypass valves may require adjusting to establish adequate shutdown cooling.

LOOP in POS 9:

The operator response is guided by AOPs for loss of class 1E electrical systems and loss of RHR. Operators are directed to try to restore power to any AC emergency bus from its associated diesel generator. If RHR is not able to be placed into service, then operators are directed to initiate the LORHR AOP. If a bus cannot be energized with a diesel generator, then operators will continue efforts to restore power to at least one AC emergency bus using any available power supply. If power is successfully restored, then operators will restore RHR or refill the RCS with CCPs or SIPs. If a station blackout condition occurs, then operators may establish gravity feed from the RWST to control RCS inventory. Successful gravity feed extends the sequence to allow additional time for power recovery and restoration of RHR.

LONSCW in POS 9:

The operator response is guided by AOPs for loss of nuclear service cooling water system is and loss of RHR. Operators are directed to diagnose and verify the NSCW system status and plant status. If neither train of NSCW is verified to be operating, then the operators are direct to place one NSCW train in single pump operation. If required systems, including RHR and CCW, are unavailable, then the procedure directs operators to initiate appropriate AOPs for loss of CCW and for loss of RHR. The LORHR AOP directs operators to control RCS level and provides steps to restore RHR cooling with verification that NSCW and CCW cooling functions have been restored. If RHR cooling cannot be restored, then operators are directed to makeup RCS level using gravity feed from the RWST . Successful gravity feed extends the sequence to allow additional time for restoring NSCW and RHR cooling.

The initial conditions and time to boil, equipment status, and modeled HFEs for POS 9 are provided in Table 4-34, Table 4-35, and Table 4-36, respectively.

Table 4-34 POS 9 Initial Conditions and Time to Boil

Plant Condition	Value	Notes
Time After Reactor Shutdown	471 hr	POS 9 begins at 381 hours after shutdown with draining refueling cavity. The last 30 hours of POS 9 are modeled.
Decay Heat	3.6 MW	
Initial RCS Temperature	110°F	
Initial RCS Pressure	0 psig	
Initial RCS Water Level	Two feet below the reactor vessel head flange	
Time to Core Boiling (to T_{sat})	1.3 hr	
Time to boiloff to within one foot of hot leg centerline	5 hr	
Time to Core Damage	15 hr	

Table 4-35 POS 9 Equipment Status

Equipment	Status	Remarks
RHR pumps	1 of 2 running	RHR train B is operating; RHR train A is aligned for gravity feed, but available if needed.
Normal charging pump	Operating	Let down is in manual.
Centrifugal charging pumps	2 of 2 in PTL	Two of two pumps available for injection.
Safety injection pumps	2 of 2 in PTL and isolated	Two of two pumps maintained in a state ready to be restored with manual actions from the control room.
Safety injection signals	Blocked	
Accumulators	Isolated	
Containment spray pumps	0 of 2 operable	In Mode 5, containment spray pumps are in PTL.
Steam generators and feedwater	Not available	RCS is vented for the majority of POS duration. The unavailability of the steam generators during this period has no effect.
Pressurizer PORV/block valves	Not applicable (3 pressurizer safety valves are removed)	
Pressurizer safety relief valves (3)	3 of 3 are removed	
COPS (cold over-pressurization protection system)	Armed with two PORVs and RHR suction relief valves	Based on technical specification bases for COPS
Reactor vessel head	De-tensioned/removed	
Emergency diesel generators	2 of 2 in standby	
Main generator and unit auxiliary transformers	Not in service	
Reserve auxiliary transformers	2 of 2 in service	
Containment equipment hatch	Open	
Containment purge vent	Not in service, closed	
Containment personnel airlock	Open	
Containment cooler units	Shutdown	
Containment sump	Sump screens covered	Protective covers are installed in Mode 5 causing sump to be unavailable.

Table 4-36 POS 9 Human Failure Events

Human Failure Event	Human Failure Description	Estimated Time Available	Modeling Approach
LORHR-Related HFEs			
1-RHR-XHE-XD-POS9	Operator fails to diagnose LORHR prior to RCS level reaching one foot above hot leg centerline. AOP directs operators to start an available RHR pump if level is greater than one foot above hot leg centerline.	1 hours prior to boiling	Use same HFE for similar conditions: 1-RHR-XHE-XD-3H
1-RHR-XHE-XL-TRN-POS9	Operator fails to recover an available RHR train prior to RCS level reaching one foot above hot leg centerline.	5 hours prior to boiloff to one foot above hot leg centerline	HFE evaluated for quantification (see Section 7)
1-HPI-XHE-XM-FFEEED-POS9	Operator fails to establish injection to control RCS level prior to core damage.	15 hours prior to core damage	Use same HFE for similar conditions: 1-HPI-XHE-XM-FFEEED-6H
1-RHR-XHE-XL-LTREC	Operator fails to re-establish normal RHR cooling within 6 hours. RCS level can be maintained for approximately 6 to 8 hours with controlled injection from RWST using CCPs or SIPs.	6 hours before RWST water volume is exhausted	HFE evaluated for quantification (see Section 7)
LOINV-Related HFEs			
1-RHR-XHE-XM-M5LOI-I	Operator fails to diagnose and isolate LOINV prior to RCS level reaching one foot above hot leg centerline for drain down rate of 100 gpm	1.75 hours for drain down to one foot above hot leg centerline	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XM-LOI-POS9	Operator fails to diagnose and isolate LOINV prior to RCS level reaching one foot above hot leg centerline for drain down rate of 1000 gpm	10 minutes for drain down to one foot above hot leg centerline	Use same HFE for similar conditions: 1-RHR-XHE-XM-LOI-POS5B
1-HPI-XHE-XM-FFEEED-LOI	Operator fails to establish forced feed prior to core damage.	3 hours prior to core damage	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XM-TRIP-M5LOI	Operator fails to diagnose LOI and trip RHR pumps after being damaged by air ingestion. Procedure will prompt operator to monitor RCS level. Assume pump damage occurs if level reaches one foot below hot leg centerline.	10 min to reach level one foot below hot leg centerline after cues after cues	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-TRN-M5LOI	Operator fails to recover an available RHR train prior to RCS boiling.	16 minutes before boiling	HFE evaluated for quantification (see Section 7)
1-LPI-XHE-XM-GFEEED-LOI	Operator fails to establish gravity feed prior to boiling.	16 minutes before boiling	HFE evaluated for quantification (see Section 7)

Table 4-36 POS 9 Human Failure Events (cont.)

Human Failure Event	Human Failure Description	Estimated Time Available	Modeling Approach
1-LOI-XHE-ISO2-LT	Operators fail late isolation of inventory loss after successful forced feed.	~5 hours	HFE evaluated for quantification (see Section 7)
LOOP-Related HFEs			
1-RHR-XHE-XD-LOOP	Operator fails to diagnose LORHR prior to RCS level reaching one foot above hot leg centerline. AOP directs operators to start an available RHR pump if level is greater than one foot above hot leg centerline. Assuming offsite power or EDGs are available, this event should be similar to LORHR action 1-RHR-XHE-XD-3H.	1 hour prior to boiling	Use same HFE for similar conditions: 1-RHR-XHE-XD-3H
1-RHR-XHE-XL-TRN-LOOP	Operator fails to recover an available RHR train prior to RCS level reaching one foot above hot leg centerline. Assuming offsite power or EDGs are available this event should be similar to LORHR action 1-RHR-XHE-XL-TRN-POS9.	2 hours prior to boiloff to one foot above hot leg centerline	Use same HFE for similar conditions: 1-RHR-XHE-XL-TRN-POS9
1-HPI-XHE-XM-FFEEED-LOOP	Operator fails to establish injection to control RCS level prior to core damage. Assuming power is recovered this event should be similar to LORHR action 1-HPI-XHE-XM-FFEEED-6H.	5 hours prior to core damage	Use same HFE for similar conditions: 1-HPI-XHE-XM-FFEEED-6H
1-LPI-XHE-XM-GFEEED-LOOP	Fail to locally establish gravity feed before boiling during SBO.	~15 minutes	HFE evaluated for quantification (see Section 7)
1-OEP-XHE-XL-NR01H	Operator Fails to Recover Offsite Power in 1 Hour – based on LOOP duration data	1 hour	Quantification based on LOOP duration data (NRC, 2021a)
LO4160V-Related HFEs			
1-RHR-XHE-XL-TRN-STBY-POS9	Operator fails to restore standby RHR pump to provide shutdown cooling and verify adequate flow to maintain RCS temperature.	2 hours is assumed if unaffected RHR train is in standby	Use same HFE for similar conditions: 1-RHR-XHE-XL-TRN-STBY-2H
LONSCW-Related HFEs			
1-LPI-XHE-XM-GFD-PRF-NSCW	Operator fails to establish gravity feed before boiling	~1 hour	HFE assessed for more restrictive conditions in POS 10
1-HPI-XHE-XM-FFD-PRF-NSCW	Operatory fails to establish forced feed after NSCW recovery	2 hours	HFE assessed for more restrictive

Table 4-36 POS 9 Human Failure Events (cont.)

Human Failure Event	Human Failure Description	Estimated Time Available	Modeling Approach
			conditions in POS 10
1-OA-OSW-----H	Operators fail to establish single pump operation after loss of NSCW	30 minutes	HFE adopted from at-power PRA (NRC, 2022)

4.4.10 POS 10 Initiating Event Response

POS 10 covers the activities when the RCS level is less than 2 feet below the reactor vessel head flange. POS 10 begins when draining to mid-loop using RHR letdown. The RCS is drained to a mid-loop elevation of one foot above the hot leg centerline to remove the nozzle dams and install SG manway covers. The plant response is similar to POS 6, pre-refueling mid-loop operations. However, in POS 10 only one of the RHR trains is in operation, while the other train is in standby and available, if needed.

POS 10 is assumed to begin at 501 hours after shutdown. The decay heat power level is estimated to be 3.5 MW. The lower decay heat level in POS 10 allows additional time for operator actions and reduces the estimated human error probabilities with respect to POS 6. The time to reach saturation temperature is estimated to be 1.11 hours, as outlined in Section 5.3 and Table 5-1. The estimated time to reach saturation temperature is based on an initial RCS temperature of 110°F, the same initial temperature that is used for POS 6. During post-refueling POSs it is possible that the RCS has a lower initial temperature, possibly in the 70–80°F range, but the authors could not identify any reference to POS 10 initial temperature conditions in the reference plant's reference materials. Accounting for the lower initial temperature would provide more time for modeling operator response to accident conditions. However, the additional minutes are not expected to have a significant impact on the human failure events that are applied to POS 10.

The longer time to boil in POS 10 can also have an impact on containment status. The reference plant's outage risk monitoring procedure compares the time to boil with the time required to close containment (usually limited by closure of the equipment hatch). Recent outage reports suggest that the regular plant practice is to close the equipment hatch during fueled mid-loop operations, both pre-refueling and post-refueling. However, the plant may choose to keep the equipment hatch open if the time to boil is significantly long enough compared to the time required to close containment, and this could be an acceptable condition according to the outage risk monitoring procedure. For this project, the assumption that the equipment hatch is closed prior to entering fueled mid-loop operations is consistently applied for both POS 6 and POS 10. Potential for entering a fueled mid-loop condition with the equipment hatch open should be considered if other variations of a refueling outage or other outage types are modeled.

LORHR Response in POS 10:

If level is greater than one foot above the hot leg centerline, then the operators are directed to start an available RHR pump. Although the initial level is assumed at one foot above the hot leg centerline (not greater than this level), it is assumed that operators will attempt to restore an available RHR pump prior to RCS temperature reaching 200°F. If RCS core exit temperature is greater than 200°F, then the operators are directed to refill the RCS using one of the SIPs via hot leg injection or one of the CCPs via cold leg injection either to the 50-percent pressurizer level or until RHR cooling is restored. If RCS level cannot be restored greater than one foot above the hot leg centerline, then operators are directed to align gravity feed from the RWST to restore level. Once RCS refill is accomplished, the operators establish conditions to restart an RHR pump for shutdown cooling.

LOINV Response in POS 10:

The operator response is guided by AOPs for RCS leakage and loss of RHR. Operators will monitor RCS inventory level and RHR pumps for indication of cavitation. Operators are directed to stop RHR pumps if RCS level is less than one foot above the hot leg centerline. If RCS level can be maintained above that level, then RHR cooling can be recovered using either available RHR train. If operators fail to trip the RHR pump prior to level reaching one foot below the hot leg centerline, then pump damage is assumed to occur. Repair is required before RHR can be recovered. If early diagnosis and RHR recovery are not successful, then operators can establish gravity or forced feed to control RCS level. Once RCS refill is accomplished, the operators establish conditions to restart an RHR pump for shutdown cooling.

ODML Response in POS 10:

The operator response to overdraining the RCS when entering mid-loop is governed by the relevant section of the loss of RHR AOP. Operators may diagnose and stop the overdrain before RHR cooling is lost. If that is not successful, then RCS level will need to be raised before the standby RHR train can be restored. Injection with CCPs or SIPs can provide alternate decay heat removal and delay the onset of core damage until the RWST inventory is exhausted. If operators fail to trip operating RHR pumps in response to the overdrain, then pump damage is assumed to occur.

LO4160V in POS 10:

The operator response is guided by the AOP for loss of power to a class 1E 4,160V AC bus. If diesel generator B does not start automatically, the operators are directed to start the diesel generator manually. After aligning the diesel generator, and adjusting bus frequency to 60HZ, operators are directed to proceed to recover RHR following the steps listed above for the loss of RHR response in POS 10. If the operating RHR train cannot be restored, then the standby RHR train is available and not affected by the initiating event. Operators are directed to verify an available RHR train is providing adequate shutdown cooling. Standby RHR train flow and heat exchanger outlet and bypass valves may require adjusting to establish adequate shutdown cooling.

LOOP Response in POS 10:

The operator response is guided by AOPs for loss of class 1E electrical systems and loss of RHR. If neither of the class 1E 4,160V AC buses are energized, then the AOP directs operators to other procedures based on the operating mode of the plant. If a bus is energized, then operators will place a train of RHR in service for shutdown cooling. If RHR is not able to be placed into service, then operators are directed to initiate the LORHR AOP. If a bus cannot be energized with a diesel generator, then operators will continue efforts to restore power to at least one AC emergency bus using any available power supply (i.e., either of the two RATs or the SAT). After power is restored, operators will check for proper operation of the NSCW, CCW, and RHR systems.

The initial conditions and time to boil, equipment status, and modeled HFEs for POS 10 are provided in Table 4-37, Table 4-38, and Table 4-39, respectively.

Table 4-37 POS 10 Initial Conditions and Time to Boil

Plant Condition	Value	Notes
Time After Reactor Shutdown	509 hr	POS 10 begins at 501 hours after reactor shutdown. Eight hours are assumed for draining from near the flange to mid-loop level.
Decay Heat	3.5 MW	
Initial RCS Temperature	110°F	
Initial RCS Pressure	0 psig	
Initial RCS Water Level	One foot above hot leg centerline	Normal operating level for mid-loop.
Time to 200°F	Approx. 1 hr	
Time to Core Boiling (to T_{sat})	1.1 hr	Assuming level at one foot above hot leg centerline.
Time to boiloff to within one foot of hot leg centerline	1.1 hr	
Time to Core Damage	11.4 hr	

Table 4-38 POS 10 Equipment Status

Equipment	Status	Remarks
RHR pumps	1 of 2 running	Standby train is available, if needed.
Normal charging pump	Operating	Letdown is in manual.
Centrifugal charging pumps	2 of 2 in standby	
Safety injection pumps	2 of 2 in PTL and isolated	One of two pumps maintained in a state ready to be restored with manual actions from the control room.
Safety injection signals	Blocked	
Accumulators	Isolated	
Containment spray pumps	0 of 2 operable	
Steam generators and feedwater	Not available	RCS is vented for the majority of POS duration. The unavailability of the steam generators during this period has no effect.
Pressurizer PORV/block valves	Not applicable (3 pressurizer safety valves are removed)	
Pressurizer safety relief valves	3 of 3 are removed	Safety valves are reinstalled at conclusion of this POS.
COPS (cold over-pressurization protection system)	Armed with two PORVs and RHR suction relief valves	
Reactor vessel head	Tensioned	Reactor vessel head installed during POS 9.
Emergency diesel generators	2 of 2 in standby	
Main generator and unit auxiliary transformers	Not in service	
Reserve auxiliary transformers	2 of 2 in service	
Containment equipment hatch	Closed and secured with a minimum of four bolts	
Containment purge vent	Off and closed	
Containment personnel airlock	One door closed	
Containment cooler units	Shutdown	
Containment sump	Sump screens covered	Protective covers are installed in Mode 5 causing sump to be unavailable.

Table 4-39 POS 10 Human Failure Events

Human Failure Event	Human Failure Description	Time Available for Action	Modeling Approach
LORHR-Related HFEs			
1-RHR-XHE-XD-ML	Operator fails to diagnose LORHR prior to RCS level reaching one foot above hot leg centerline (initial level at time of IE). AOP directs operators to start an available RHR pump if level is greater than one foot above hot leg centerline. RHR recovery will be attempted until reaching 200°F, then operators will be directed to establish injection to control level.	5 minutes after LORHR occurs	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-TRN-POS10	Operator fails to recover an available RHR train prior to RCS temperature reaching 200°F.	1.11 hours prior to boiling with level at one foot above hot leg centerline	HFE evaluated for quantification (see Section 7)
1-HPI-XHE-XM-FFEEED-MLDRL	Operator fails to establish pumped injection to control RCS level prior to core damage.	10 hours prior to core damage	HFE evaluated for quantification (see Section 7)
1-LPI-XHE-XM-GFEEED-MLDRL	Operator fails to establish gravity feed to control RCS level prior to core damage.	10 hours prior to core damage	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-LTREC-2D	Operator fails to re-establish normal RHR cooling within 1 day. Non-recovery probability is based on pump recovery data. RCS level can be maintained for approximately 24 hours with controlled injection from RWST using CCPs or SIPs.	~1 day before RWST water volume is exhausted	Use same HFE for similar conditions: 1-RHR-XHE-XL-LTREC
ODML Overdrain-Related HFEs			
1-RHR-XHE-XM-POS10-I	Operator fails to diagnose and isolate drain down prior to level below one foot above hot leg centerline.	15 minutes	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XM-TRIP-MLDRL	Operator fails to identify impact of reduced RCS level and trip RHR pumps before they are damaged.	15 minutes	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-TRN-ML-P10	Operator fails to recover an available RHR train prior to boiling.	50 minutes prior to boiling	HFE evaluated for quantification (see Section 7)
1-RHR-XHE-XL-TRN-AIR-DRL	Operator fails to recover an available RHR train after air ingestion, requiring system venting.	1 day	HFE evaluated for quantification (see Section 7)
LO4160V-Related HFEs			
1-RHR-XHE-XL-TRN-POS10	Operator fails to recover an available RHR train prior to RCS temperature reaching 200°F.	1.11 hours prior to boiling with level at one foot above hot leg centerline	Same HFE as LORHR conditions

Table 4-39 POS 10 Human Failure Events (cont.)

Human Failure Event	Human Failure Description	Time Available for Action	Modeling Approach
LOOP-Related Repair Events			
1-EPS-XHE-XL-NR04H	Operator Fails to Recover Emergency Diesel in 4 Hours - based on unplanned EDG outage times	4 hours	Quantification based on EDG repair data (NRC, 2021a)
1-OEP-XHE-XL-NR30M	Operator Fails to Recover Offsite Power in 1 Hour – based on LOOP duration data	30 min.	Quantification based on LOOP duration data (NRC, 2021a)
LO4160V-Related HFEs			
1-LPI-XHE-XM-GFD-PRF-NSCW	Operator fails to establish gravity feed before boiling	~1 hour	HFE evaluated for quantification (see Section 7)
1-HPI-XHE-XM-FFD-PRF-NSCW	Operatory fails to establish forced feed after NSCW recovery	2 hours	HFE evaluated for quantification (see Section 7)
1-OA-OSW-----H	Operators fail to establish single pump operation after loss of NSCW	30 minutes	HFE adopted from at-power PRA (NRC, 2022)

4.4.11 POS 11 Initiating Event Response

During POS 11, a skid-mounted vacuum pump is connected to the reactor head vent, and a vacuum is established to maximize the air removed from the steam generator tubes as the RCS is filled. The idle train of RHR (train A) or a CCP is used to fill the RCS to the top of the reactor vessel head. Then, filling is continued with the idle train of RHR until desired level in the pressurizer is reached. At the time of the initiating event, the initial conditions are assumed to be RCS level at the top of the reactor vessel head, RCS temperature is 125°F, and a vacuum is drawn to 20" Hg as indicated on the vacuum pump gauge. The equipment hatch is assumed open after raising RCS level greater than 2 feet below the reactor vessel head flange.

Loss of RHR Response in POS 11:

The operator response is governed by the AOP for loss of RHR cooling. The procedure directs operators to break RCS vacuum by opening the vacuum bleed valve on the vacuum skid. Operators are to confirm RCS level is greater than one foot above the hot leg centerline and establish charging flow to control level. Operators are directed to place one train of RHR in shutdown cooling and control temperature using RHR.

The LOINV, LO4160V, and LOOP initiating events are not modeled for POS 11. The initial conditions and time to boil, equipment status, and modeled HFEs for POS 11 are provided in Table 4-40, Table 4-41, and Table 4-42, respectively.

Table 4-40 POS 11 Initial Conditions and Time to Boil

Plant Condition	Value	Notes
Time After Reactor Shutdown	526 hr	
Decay Heat	3.4 MW	
Initial RCS Temperature	125°F	
Initial RCS Pressure	vacuum drawn to 20" Hg	
Initial RCS Water Level	Top of reactor vessel head	
Time to Core Boiling (to T_{sat})	1.3 hr	Conservative estimate for assigning screening HEPs.
Time to boiloff to within one foot of hot leg centerline	5 hr	Conservative estimate for assigning screening HEPs.
Time to Core Damage	15 hr	Conservative estimate for assigning screening HEPs.

Table 4-41 POS 11 Equipment Status

Equipment	Status	Remarks
RHR pumps	2 of 2 available	RHR train B is operating; RHR train A is aligned for gravity feed, but available if needed.
Normal charging pump	Operating	Let down is in manual.
Centrifugal charging pumps	2 of 2 in PTL	Two of two pumps available for injection.
Safety injection pumps	2 of 2 in PTL and isolated	Two of two pumps maintained in a state ready to be restored with manual actions from the control room.
Safety injection signals	Blocked	
Accumulators	Blocked; filled and pressurized with nitrogen	
Containment spray pumps	0 of 2 operable	In Mode 5, containment spray pumps are in PTL.
Steam generators and feedwater	Unavailable - SGs in wet layup (filled with N2 at 95% WR)	Steam generator tubes are not filled. Some heat removal may be possible but is not credited.
Pressurizer PORV/block valves	Armed at COPS setpoints with block valves open	
Pressurizer safety relief valves (3)	Closed	
COPS (cold over-pressurization protection system)	Armed with two PORVs and RHR suction relief valves	Based on technical specification bases for COPS
Reactor vessel head	Tensioned	
Emergency diesel generators	2 of 2 in standby	
Main generator and unit auxiliary transformers	Not in service	
Reserve auxiliary transformers	2 of 2 in service	
Containment equipment hatch	Open	
Containment purge vent	Not in service, closed	
Containment personnel airlock	Open	
Containment cooler units	4 of 4 units running	
Containment sump	Sump screens covered	Protective covers are installed in Mode 5 causing sump to be unavailable.

Table 4-42 POS 11 Human Failure Events

Human Failure Event	Human Failure Description	Time Available for Action	Modeling Approach
LORHR-Related HFEs			
1-RHR-XHE-XD-POS11	Operator fails to diagnose LORHR prior to RCS level reaching one foot above hot leg centerline. AOP directs operators to start an available RHR pump if level is greater than one foot above hot leg centerline. This event is similar to POS 5B action 1-RHR-XHE-XD-3H.	~1 hours	Use same HFE for similar conditions: 1-RHR-XHE-XD-3H
1-RHR-XHE-XL-TRN-POS11	Operator fails to recover an available RHR train prior to RCS level reaching one foot above hot leg centerline. This event is similar to POS 5B action 1-RHR-XHE-XL-TRN-3H.	~5 hours	Use same HFE for similar conditions: 1-RHR-XHE-XL-TRN-3H
1-HPI-XHE-XM-FFEED-POS11	Operator fails to establish injection to control RCS level prior to core damage. This event is similar to POS 5B action 1-HPI-XHE-XM-FFEED-6H.	~15 hours	Use same HFE for similar conditions: 1-HPI-XHE-XM-FFEED-6H
1-ADS-XHE-XM-LOINV-SD	Operator fails to establish bleed path for feed and bleed cooling.	~15 hours	HFE evaluated for quantification (see Section 7)

4.4.12 POS 12 Initiating Event Response

During POS 12 the initial RCS heatup begins. The pressurizer heaters are used to raise the RCS pressure from vacuum to 300 psig and RHR heat exchanger cooling is throttled to raise average RCS temperature to 175°F. The initial conditions at the time of the initiating event are assumed to be RCS pressure of 150 psig and RCS temperature of 150°F. Train B of RHR is running with the other train in standby but operable. Both RHR hot leg suction valves are open. The RCPs remain off for this POS. At the conclusion of POS 12, the equipment hatch is fully bolted closed and containment purge is shutdown. The equipment hatch is assumed open at the time of the initiating event.

Loss of RHR Response in POS 12:

The LORHR AOP directs operators to attempt to restore RHR cooling and perform the necessary actions to use the secondary heat sink for RCS cooling. Operators will check if any RHR pumps are running. Operators will place the standby RHR train in service for RHR cooling. Operators will continue actions to monitor RHR and establish one train of RHR cooling. If RHR flow from either train cannot be established, then operators are directed to establish steam generator cooling. Operators will establish feedwater to maintain level greater than 10 percent of normal operating range in the steam generators. Operators continue attempts to start one RHR train. If RHR is not available, then operators will maintain RCS temperature using the steam generator atmospheric relief valves.

The LOINV, LO4160V, and LOOP initiating events are not modeled for POS 12. The initial conditions and time to boil, equipment status, and modeled HFEs for POS 12 are provided in Table 4-43, Table 4-44, and Table 4-45, respectively.

Table 4-43 POS 12 Initial Conditions and Time to Boil

Plant Condition	Value	Notes
Time After Reactor Shutdown	550 hr	
Decay Heat	3.4 MW	
Initial RCS Temperature	150°F	
Initial RCS Pressure	150 psig	
Initial RCS Water Level	40% level in pressurizer	
Time to SG Dryout	150 hr	This estimate is based on calculation of plant-specific conditions.
Time to Core Boiling (to T_{sat})	162 hr	This estimate is based on calculation of plant-specific conditions.
Time to boiloff to within one foot of hot leg centerline	163 hr	This estimate is based on calculation of plant-specific conditions.
Time to Core Damage	176 hr	This estimate is based on calculation of plant-specific conditions.

Table 4-44 POS 12 Equipment Status

Equipment	Status	Remarks
RHR pumps	2 of 2 available	RHR train B is operating; RHR train A is in standby.
Normal charging pump	Operating	Let down is in manual.
Centrifugal charging pumps	2 of 2 in standby	Two of two pumps available for injection.
Safety injection pumps	2 of 2 in PTL and isolated	Two of two pumps maintained in a state ready to be restored with manual actions from the control room.
Safety injection signals	SSPS is restored	
Accumulators	Blocked; filled and pressurized with nitrogen	
Containment spray pumps	0 of 2 operable	In Mode 5, containment spray pumps are in PTL.
Steam generators and feedwater	Available - SGs in wet layup (filled with N2 at 95% WR)	AFW pumps not in service, but available to be restored.
Pressurizer PORV/block valves	Armed at COPS setpoints with block valves open	
Pressurizer safety relief valves (3)	Closed	
COPS (cold over-pressurization protection system)	Armed with two PORVs and RHR suction relief valves	Based on technical specification bases for COPS
Reactor vessel head	Tensioned	
Emergency diesel generators	2 of 2 in standby	
Main generator and unit auxiliary transformers	Not in service	
Reserve auxiliary transformers	2 of 2 in service	
Containment equipment hatch	Open	
Containment purge vent	Not in service and closed	Close containment penetrations for integrity checks prior to entering Mode 4.
Containment personnel airlock	Open	
Containment cooler units	4 of 4 units running	
Containment sump	Sump screens covered	Protective covers are removed during POS 12 prior to entering Mode 4.

Table 4-45 POS 12 Human Failure Events

Human Failure Event	Human Failure Description	Time Available for Action	Modeling Approach
LORHR-Related HFEs			
1-RHR-XHE-XD-PRF	Operator fails to diagnose LORHR prior to loss of secondary cooling due to SG dryout.	~5 hours assuming no feedwater to SGs	Use same HFE for similar conditions: 1-RHR-XHE-XD-POS3
1-RHR-XHE-XL-TRN-PRF	Operator fails to recover either RHR train prior to loss of secondary cooling due to SG dryout.	~5 hours assuming no feedwater to SGs	Use same HFE for similar conditions: 1-RHR-XHE-XL-TRN-POS4
1-AFW-XHE-XM-SGC-PRF	Operator fails to feed SGs to maintain level and control RCS temperature using steam dumps or ARVs.	5 hours before SGs dry out without maintaining feedwater	Use same HFE for similar conditions: 1-AFW-XHE-XM-SGC-POS3

4.4.13 POS 13 Initiating Event Response

In POS 13, the plant heatup continues and Mode 4 is entered with RCS temperature above 200°F. The RCPs are also started during this POS. RCS pressure is raised to 300 psig. RHR train B is aligned for shutdown cooling, and RHR train A is in standby. During POS 13, the motor-driven AFW pumps are started on mini-flow in preparation for transitioning to using the ARVs for RCS temperature control.

As a modeling simplification, the same plant response model that is used for POS 12 is used for POS 13. POS 13 is assumed to begin at 604 hours after shutdown. The decay heat power level is estimated to be 3.2 MW. The decay heat level is slightly lower than POS 12, and the initial RCS pressure and temperature are higher. The plant response is expected to be similar. Refer to Section 4.4.12 for further details on the plant conditions, equipment status, and human failure events that are applied to POS 13.

5 SUCCESS CRITERIA

5.1 Introduction

The objectives of the success criteria element are to define the plant-specific measures of success and failure that support the other technical elements of the LPSD PRA in such a way that

- overall success criteria are defined (i.e., core damage and large early release)
- success criteria are defined for critical safety functions, supporting systems, structures, components, and operator actions necessary to support accident sequence development
- the methods and approaches have a firm technical basis
- the resulting success criteria are referenced to the specific deterministic calculations.

The definitions for the LPSD PRA end states are discussed in Section 5.2. The criteria for defining core damage and for reaching a safe, stable state are defined. Section 5.3 presents the success criteria for the safety functions that are modeled in the event trees. Section 5.4 gives the system-level success criteria for the plant systems that are used to mitigate accident scenarios.

5.2 End State Success Criteria

Each accident sequence is assigned an end state. For the Level 1 LPSD PRA, two end states are defined: core damage and safe stable state. The end states are denoted by “1-CD-SD” for core damage and “OK” for safe, stable state. The definitions of both states are discussed in further detail below.

5.2.1 Core Damage Definition

The definition of core damage here is the same as the definition provided in the Level 1 reactor at-power PRA standard, “uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage are anticipated and involving enough of the core, if released, to result in offsite public health effects.” (ASME/ANS, 2009)

Of more practical importance is the definition of the core damage surrogate used to represent the above condition. The issue of specifying core damage can be subjective, and even in at-power PRAs, numerous different core damage surrogates have been used. This issue is discussed in other places (see (NRC, 2014), for example) and not elaborated further here.

For the present shutdown analyses, a single surrogate will be used across all POSs as a simplifying assumption, though it is acknowledged that some of the underlying phenomena (most notably the relationship between adequate steam cooling and core water level) do vary. Further, a surrogate is chosen that does not require modeling of the actual fuel heatup, because this allows sequence timing analysis to be performed by either mechanistic codes (e.g., MELCOR) that do provide this information, as well as more simple mass balance tools (e.g., Excel spreadsheets) that do not. The core damage surrogate used is:

*Core damage surrogate = core water level equal to **two-thirds of the active fuel height** (i.e., ~ 2 feet above the fuel mid-plane)*

In the case of mechanistic computer code calculations, either the swollen or collapsed water level in the core is acceptable, but the one chosen should be specified. Typically, the difference in these two does not lead to significant differences in the estimated timing of core damage for analysis of shutdown accident scenarios. However, the use of the downcomer water level could result in significant differences.

For a wide range of conditions, two-thirds active fuel height will precede the time at which early fuel damage occurs (e.g., clad burst), and is expected to almost always precede any significant fuel damage (e.g., autocatalytic oxidation of the exposed sections of numerous assemblies). It is chosen here over a more aggressive surrogate (e.g., one-half or one-third active fuel height) for four reasons:

- It is being applied across a broad range of decay heat and RCS configuration combinations.
- It is a reasonable balance of being realistic while not relying on simple mass balance tool results well-beyond their validity (due to heat transfer and heat generation mechanisms not being captured).
- It is a reasonable estimate of when exposed control material may start to melt (it typically has a lower melting temperature than zircaloy).
- This is the same surrogate as is being used in the SFP portion of the L3PRA project.

5.2.2 Safe and Stable End State

The standard for Level 1 PRA (ASME/ANS, 2009) defines a safe, stable state as a plant condition, following an initiating event, in which RCS conditions are controllable at or near desired values.

This definition is used as guidance in developing the specific criteria for a safe, stable end state for shutdown accident sequences. Another reference that was used as guidance is the Idaho National Laboratory's report, *Safe and Stable State in SPAR Model Event Trees* (Buell, 2016). That report describes a method for modeling a safe stable state. The steps of the method are:

1. identify the key safety functions that are necessary to reach a safe stable state
2. specify success criteria for each identified key safety function
3. identify the systems and operator actions that can mitigate the initiator and achieve the defined success criteria
4. perform appropriate thermal-hydraulic analyses or other engineering evaluations to determine accident progression parameters
5. define the end state of the accident progression as occurring when safe stable state or core damage state has been reached

This method is adapted to model a safe stable state for shutdown plant operating states. The key safety functions for shutdown operations are identified in Section 3.2.1.1. Two of these safety functions are challenged in the modeled shutdown initiating events, and these safety functions must be re-established for a safe stable state to be reached.

- **Decay Heat Removal:** The decay heat removal function must be established for a safe stable state to be reached.
- **Inventory Control:** Sources of RCS inventory loss must be isolated for a safe stable state to be reached.

The success criteria for these two safety functions are defined for reaching a safe stable state. The decay heat removal function must be established by successful operation of RHR cooling or steam generator cooling. Other alternate means of cooling are credited in the event tree models. The alternate cooling involves injecting coolant from the RWST to the RCS using charging or safety injection pumps or by gravity-driven injection. However, these alternate cooling methods are not sufficient to reach a safe stable state because the RWST inventory will eventually be exhausted, and another heat removal method must be established. The success criteria for loss of inventory events require that inventory loss be isolated before a safe stable state can be achieved. The modeling approach for loss of inventory events assumes a break in one train of the RHR system. A safe stable state requires isolating the affected RHR train.

The systems and operator actions that are needed to achieve the safety functions are discussed below. As discussed above, the safe stable state requires decay heat removal by RHR cooling or steam generator cooling. To establish RHR cooling the following systems and support systems are required: RHR, CCW, NSCW, and electrical power. Operator actions to establish RHR cooling can include:

- Aligning and starting an idle RHR train for shutdown cooling
- Recovery of RCS conditions necessary for RHR operation
- Restoration of a previously running RHR train
- Actions to establish necessary support systems for RHR cooling

Steam generator cooling requires use of the AFW system to maintain steam generator inventory and heat removal through atmospheric relief valves. Steam generator cooling also requires plant conditions that would allow for effective heat removal with steam generators. This model assumes that to credit steam generator heat removal, the RCS must not be vented. Operation of the reactor coolant pumps is not required. Operator actions to establish steam generator cooling include actions to operate AFW to maintain steam generator inventory. Additional details on the system success criteria are provided in Section 5.4, and additional information on modeled operator actions is found in Section 7.

The next step in modeling a safe stable state involves performing thermal-hydraulic analyses or other engineering evaluations. Thermal-hydraulic analyses and evaluations of RCS heatup and boiling have been performed to support assessments of the timing and conditions needed to re-establish decay heat removal. Additional discussion of these analyses is provided in Section 5.3.

The final step in the safe stable state analysis is to define the modeled end states. As described above, two end states are defined: core damage and safe stable state. The surrogate condition

for reaching a core damage end state is core water level reaching two-thirds of the active fuel height. A safe stable state is reached when a stable decay heat removal function is established (via RHR or steam generator cooling) and when RCS inventory losses (if applicable) are isolated.

5.3 Safety and Mitigation Function Success Criteria

This section describes the success criteria for each safety and mitigation function that is modeled as a top event in the plant response event trees. The success criteria evaluations are supported by plant-specific calculations to assess the plant conditions and time available for actions. The calculations include analyses using a plant-specific MELCOR model for shutdown conditions and a calculator developed for estimating the decay heat levels, time to reach saturation temperature, and boil-off of RCS level to different elevations. The plant-specific calculations performed for the LPSD PRA are summarized in Table 5-1. Plant-specific calculations are not available for all the modeled accidents and conditions. The heatup and boil-off calculation approach applied to POS 5B through POS 10 (with RCS vented or open) is not applicable to POSs where the RCS is intact and there is the possibility of heat removal by the steam generators. In cases where no POS-specific analyses were performed, the differences between the analyzed cases and the unanalyzed cases were evaluated, and limiting assumptions were applied. These assumptions are expected to have a limited impact on the results, since the unanalyzed cases are not significant risk contributors.

- The timing for POS 3 and POS 4 is informed by the evaluation for POS 5A. For POS 5A, the time to reach saturation temperature is approximately 24 hours, and the time to reach core damage is approximately 27 hours. The evaluation assumes no operator intervention but includes credit for passive steam generator cooling. For POS 3 and POS 4, the time available for responding to the initiating event is expected to be similar to POS 5A, though decay heat level and initial RCS temperature are higher. A conservative assumption of 5 hours is applied for the time to heat to saturation temperature. The conservative assumption is expected to have a minimal impact on the results because the assumed time is greater than the time required for actions to restore cooling.
- The timing for POS 11 is modeled assuming the same timing as POS 9. This is expected to be a conservative assumption because POS 11 occurs at a later time after shutdown and initial RCS inventory level is higher than the inventory level in POS 9.
- The timing for POS 13 is modeled assuming the same timing as POS 12. POS 13 has a slightly lower decay heat level. The initial RCS temperature and pressure are higher than POS 12, but the modeled plant response is expected to be similar for POS 12 and POS 13.

The approaches for assessing the success criteria of mitigation functions that are modeled in the event trees are discussed in the subsections below. The discussion includes the criteria, assumptions, timing analysis, and summary of the plant-specific calculations that were used to develop the criteria.

5.3.1 Diagnosis of Loss of RHR

The diagnosis of the initiating event is the first mitigation top event modeled in each event tree. The diagnosis function is modeled as a separate top event because the success or failure of the

diagnosis can dictate what other mitigating functions are available to respond to the event. If diagnosis fails, then some recovery actions may not be possible. The time available for diagnosis is based on the time for RCS temperature to reach saturation temperature.

A summary of the plant-specific calculations to support the success criteria analyses is presented in Table 5-1. The timing primarily impacts the evaluation of human failure events and the time available for performing the required actions. The timing is not evaluated for every modeled POS. The assumptions described above in Section 5.3 are applied to the POSs that are not directly assessed.

Diagnosis of the initiating event is modeled as a single human failure event. The time available for diagnosis is apportioned between time for diagnosis and time for action to restore core cooling. The time depends on the specific POS and initiating event category. The assumptions for each individual human failure event are documented in Section 7.

5.3.2 Diagnosis and Isolation of Loss of Inventory

The timing analysis for loss of inventory events requires additional evaluation to account for the reduction of RCS inventory that would result in decreasing the times to reach boiling and uncovering the core. The key operator response is to diagnose the event and isolate the inventory loss. If this can be successfully accomplished, then subsequent actions can be taken to restore RHR cooling. If the diagnosis and isolation is successful prior to the RCS level dropping too low, then RHR cooling can be restored. Otherwise, additional actions are required to restore RCS level before RHR cooling restoration is attempted. For mid-loop conditions (POSs 6 and 10), the initial level is assumed to be at one foot above the hot leg centerline. If the diagnosis and isolation is successfully completed prior to level being reduced by 6 inches or less, then restoration of the operating RHR train is assumed to be possible. For all other POSs, the successful diagnosis and isolation is based on preventing the level from reaching one foot above the hot leg centerline.

The POS initial conditions and the type of initiating event influence the time available for acting. A representative loss flow rate is selected for each loss of inventory initiating event scenario. Industry operating experience and plant-specific procedures were reviewed to inform the selection of representative flow rates. The following summarizes relevant information for selecting the representative flow rates.

Loss of inventory events in the past indicate that wide ranges of leak rates and total leakage are possible. An EPRI report on loss of RHR and loss of inventory event trends (EPRI, 2010) identifies loss of inventory events with leak rates ranging from around 10 gpm to a maximum of 700 gpm. With the exception of one event with a total inventory loss of 14,000 gallons, the remainder of events had leak rates less than 300 gpm and total leakage of less than or equal to 10,000 gallons. All large inventory losses (>10,000 gallons) occurred before the year 2000.

Table 5-1 Summary of Plant-Specific Calculations to Support Success Criteria Analyses

Plant State Analyzed	Time After Reactor Shutdown (hr)	Decay Heat (MW)	Time to Reach Saturation Temperature (hr)	Time to Boil Off Level to Mid-loop (hr)	Time to Core Damage (hr)	Analysis Description
POS 5A (pressurizer solid) event leading to LORHR	12	18.7	23.7	23.9	27.1	MELCOR analysis with consideration for inventory losses due to lifting of RHR relief valves and steam generator cooling.
POS 5B event leading to LORHR	55	13.6	0.47	2.3	5.3	Plant-specific heatup and boil-off calculations.
POS 5B LORHR with Gravity Feed	55	13.6	6.6	Not calculated	10.1	MELCOR analysis with sensitivities on different times to initiate gravity feed.
POS 6 event leading to LORHR	82	11.8	0.31	N/A ⁽¹⁾	3.3	Plant-specific heatup and boil-off calculations.
POS 7 event leading to LORHR	126	10.3	5.7 ⁽²⁾	47.2	52.1	MELCOR analysis of configuration with refuel cavity partially flooded.
POS 8E event leading to LORHR	141	9.9	7.8	85.0	90.0	Plant-specific heatup and boil-off calculations.
POS 8L event leading to LORHR	329	4.2	18.2	202.2	214.3	Plant-specific heatup and boil-off calculations.
POS 9 event leading to LORHR	381	4.0	1.3	4.9	15.4	Plant-specific heatup and boil-off calculations.
POS 10 event leading to LORHR	509	3.5	1.1	N/A ⁽¹⁾	11.4	Plant-specific heatup and boil-off calculations.

Table 5-1 Summary of Plant-Specific Calculations to Support Success Criteria Analyses (cont.)

Plant State Analyzed	Time After Reactor Shutdown (hr)	Decay Heat (MW)	Time to Reach Saturation Temperature (hr)	Time to Boil Off Level to Mid-loop (hr)	Time to Core Damage (hr)	Analysis Description
POS 12 event leading to LORHR	533	3.4	161.6	162.5	176.5	MELCOR analysis with consideration for steam generator cooling and post-refuel decay heat.
(1) Initial RCS inventory level for the POS is at or below normal mid-loop level. Therefore, the boil-off calculation is not applicable. (2) During POS 7 through POS 9, the water level of the refueling cavity falls due to evaporation without ever becoming saturated. The time recorded in the column labeled "Time to reach saturation temperature" is the time when the refueling cavity water rapidly heats up to approach its maximum temperature.						

The loss of inventory leakages considered for this event could occur through the connected systems to RHR lines as specified by the loss of RHR AOP. These systems mainly include the RHR, systems connected to RHR, and the charging system.

These connections are all of different sizes, elevation differences, and different pressure drops. Therefore, the expected leak rate due to loss of inventory will vary depending on the location of failure.

For the SD-LOINV-OP event tree, the leakage is assumed equivalent to that associated with a stuck open RHR relief valve on one of the two operating trains of RHR. Recent operating experience (as discussed in Section 8.1.2) shows that the most likely cause of a loss of inventory during POSs with the RCS pressurized is related to lifting of RHR relief valves. The assumed RCS conditions are based on POS 5A, when two reactor coolant pumps are running, and the initial RCS pressure is around 350 psig. The leakage flow is based on a plant-specific MELCOR calculation for a loss of RHR accident during POS 5A. The scenario models a pressure transient causing the RHR relief valves to open. The period just prior to the pressurizer becoming saturated is referenced to estimate flow rate through the RHR relief valves. During this period, the pressurizer level begins to drop, and the average flow rate through the RHR relief valves is approximately 405 gpm. In the simulated scenario, a significant drop in RCS level occurs only after the RCS becomes saturated. Prior to that, the leak flow is expected to continue through the RHR relief valves, but the flow rate would be reduced as the RCS depressurizes. As a bounding assumption, a leak flow of 500 gpm is assumed for the SD-LOINV-OP event tree. Also, as the RCS depressurizes, the relief valve should reseal and stop the inventory loss. However, the valve could fail to reseal. To account for the possible failure of the relief valves to close, the model requires an operator action to isolate the impacted RHR suction line(s) in order to stop the inventory loss.

For the SD-LOINV-SP event tree, the flow rate is based on a small leak or pipe break resulting in a break flow up to 100 gpm. The representative flow rate is assumed to be 100 gpm.

For the SD-LOINV-DS event tree, the loss of inventory is due to an inadvertent displacement of inventory during draining operations. The guiding reference plant operating procedure indicates that draindown rates should be limited to less than 100 gpm when level is above the reactor vessel head elevation and 40 gpm when level is below that elevation. This event tree applies to POS 5B and POS 9 where initial conditions begin with RCS level above reactor vessel head elevation and level is drained to near the reactor vessel head flange. The representative flow rate for this event is assumed to be 100 gpm.

For the SD-LOINV event tree, the flow rate is based on a pipe break resulting in a break flow in the range of 100 to 2000 gpm. The representative flow rate is assumed to be 1000 gpm.

For the SD-LOINV-LG event tree with the refueling cavity flooded, operators would have ample time to diagnose and isolate leaks even when leak rates exceed several hundred gpm. For these conditions a large break resulting in break flow greater than 2000 gpm is modeled. The RHR flow rate is typically about 3500 gpm during POS 8E. If the RHR flow is lost through a break down-stream of the pump or diverted outside the RCS, then the inventory loss rate is assumed to be equal to the nominal 3500 gpm RHR flow rate. This flow rate is used for determining the amount of time available to trip the RHR pump and isolate the break.

The representative flow rates and initial RCS level conditions are used to calculate the time available for operator actions. The timing evaluations are summarized in Table 5-2.

5.3.3 Restoration of RHR

If the initial diagnosis of the initiating event is successful, then early restoration of RHR is possible to reach a safe stable state. Success requires operation of one of two RHR trains, including associated trains of support systems (CCW, NSCW, and electric power). Depending on the POS conditions, both trains may be aligned for shutdown cooling, or they may be in different states. If both trains are available and aligned for shutdown cooling, then a single top event is used to model restoration of either train. If the trains are not in a symmetrical alignment, then each train is modeled as a separate top event.

The time available for restoration of RHR is based on the time to reach RCS boiling or RCS level to drop to one foot above the hot leg centerline, whichever is most limiting for the initiating event scenario. The abnormal operating procedure for loss of RHR directs operators to continue to attempt restoration of RHR with core exit temperature less than 200°F and RCS level greater than one foot above the hot leg centerline. The total time is apportioned between time for diagnosis and time for action to restore RHR. The assumptions for each individual human failure event are documented in Section 7.

Table 5-2 Timing of Loss of Inventory Draindown

POS	Initiator Type	Representative flow rate (gpm)	Time of draindown (min.)	Notes
POS 5A	SD-LOINV-OP	500	44.4	Volume estimated to be 22,200 gal., draining from level in pressurizer to mid-loop.
POS 5, POS 9	SD-LOINV-SP, SD-LOINV-DS	100	104.6	Volume estimated to be 10,455 gal., draining from 2 feet below reactor vessel head flange to mid-loop.
	SD-LOINV	1,000	10.5	
POS 6, POS 10	SD-LOINV-SP	100	22.7	Volume estimated to be 2,266 gal., draining 6 inches below normal mid-loop level.
	SD-ODML	40	56.7	
	SD-LOINV	1,000	2.3	No credit for operator action given the very short time.
POS 8E/L	SD-LOINV-LG	3,500	81.2	Volume estimated to be 284,219 gal., draining refueling cavity flooded level to mid-loop.

5.3.4 Steam Generator Cooling

Steam generator cooling requires use of the AFW system to maintain steam generator inventory and heat removal through atmospheric relief valves. Steam generator cooling also requires plant conditions that would allow for effective heat removal with steam generators. This model assumes that the RCS must not be vented to credit steam generator heat removal. Once a large vent in the RCS is opened, this provides a path for heat removal from the RCS. Heat transfer to the steam generators is expected to be minimal in this configuration. Operation of the reactor coolant pumps is not required. The impact of reactor coolant pump operation on steam generator cooling was explored in a sensitivity calculation as part of the supporting MELCOR analyses. Sufficient heat removal is achieved by natural circulation as long as steam generator inventory is maintained. Operator actions to establish steam generator cooling include actions to operate AFW to maintain steam generator inventory.

5.3.5 Pumped Injection

The criteria for successful pumped injection are informed by a series of MELCOR analyses and supporting calculations, which cover a range of different conditions. A total of four ECCS pumps is considered for injection, two centrifugal charging pumps (CCPs) and two safety injection pumps (SIPs). The availability of these pumps can vary depending on the POS conditions. The number of pumps required for successful pumped injection depends on the plant state and decay heat level. Successful forced injection must be established before the RCS water boils off to two-thirds of the active fuel height.

Pumped injection will be initiated in response to the continuous actions specified in the loss of RHR AOP. The continuous actions are initiated to control RCS inventory higher than one foot above the hot leg centerline and to maintain core exit temperature less than 200 °F. Criteria were developed in the MELCOR model to start and stop the ECCS injection based on filling and maintaining level in the pressurizer and maintaining RCS temperature. The RCS level criteria are defined to maintain level greater than one foot above the hot leg centerline and continue filling the RCS until above the 50% level in the pressurizer. Injection is restarted if level falls below the low-level range of the pressurizer cold calibrated level indication. The temperature criteria are defined to start pumped injection when temperature reaches 200 °F as indicated at the core exit thermocouples, and pumps are stopped when temperature reaches 190 °F. In all the simulation cases that were run, the temperature criteria were the dominant factors in controlling the injection.

Estimating the duration of the injection is important to establish the time available for long-term recovery of RHR cooling. Controlling RCS level and temperature by pumped injection is not sufficient to achieve a safe and stable state because the available injection inventory will eventually be depleted. Although several MELCOR simulations with pumped injection were analyzed, not all POS conditions were evaluated. A set of simplified calculations were used to estimate the time of injection, and these show consistent results with the MELCOR analyses. The duration of the injection depends on the volume available in the RWST and the decay heat level for the POS. The injection time is given by the following equation:

$$t_{inject} = M \frac{C_p \Delta T}{Q}$$

The variables are defined as follows: t_{inject} is the duration of the pumped injection, M is the mass of RWST inventory available for injection, C_p is the specific heat of water [4189 J/(kg K)], ΔT is the change in temperature of the injected water, and Q is the decay heat of the reactor core. The change in temperature of the injected water is based on the assumption that the initial temperature in the RWST is 116 °F, which is the maximum allowed temperature. The injected water is assumed to be heated to an average temperature of 195 °F, as the injection is assumed to occur when coolant temperature approaches 200 °F and continues until temperature drops below 200 °F.

The mass of RWST inventory available for injection is summarized in Table 5-3. The total RWST inventory is 715,000 gallons. The inventory available for injection is assumed to be the volume of water down to the level where the low-low level alarm occurs. The low-low level alarm would typically signify the switchover from RWST injection to containment emergency sump recirculation. However, for most shutdown POSs, the sump is considered unavailable because the sump screens are blocked with protective covers when Mode 5 is entered. Additional inventory could be available if the injection continues until the empty level alarm, but this additional inventory is not credited.

Table 5-3 RWST Inventory Available for Injection

	POSS with RWST Full	POS 7 Refueling Cavity Partially Flooded	POS 8E and 8L Refueling Cavity Flooded
RWST initial inventory	2.70E+06 kg [715,000 gal.]	1.89E+06 kg [500,000 gal.]	1.45E+06 kg [383,000 gal.]
Inventory to reach low- low level alarm	1.79E+06 kg [474,000 gal.]	9.79E+05 kg [260,000 gal.]	5.37E+05 kg [142,000 gal.]
Inventory to reach empty level alarm	2.30E+06 kg [609,000 gal.]	1.49E+06 kg [395,000 gal.]	1.05E+06 kg [278,000 gal.]
Note: RWST inventory is provided by both mass (kg) and volume (gal.).			

The duration time for pumped injection is estimated in Table 5-4. The injection times are estimated for all POSs where the RCS is vented and level is reduced below the pressurizer level. Steam generator cooling is assumed to be available for POSs where the RCS is intact, and pumped injection is not relied upon as an alternate means of decay heat removal for those plant configurations. These approximate values are used as input to the time available for actions to restore RHR cooling. It may be possible to extend the injection time by allowing the RCS to heat up to saturation temperature and even boiling off inventory before injecting. Such a strategy could extend the duration by several hours, or even up to a few days. However, this type of injection strategy is not credited due to lack of procedural guidance. The procedural guidance indicates the injection flow should be established to maintain core temperature below 200 °F. MELCOR simulations show similar results to the estimates in Table 5-4. A MELCOR calculation for a loss of RHR cooling scenario during POS 6 shows that the RWST inventory is depleted in 8.2 hours with two charging pumps being cycled on and off to maintain RCS temperature below 200 °F, which is similar to the duration of 7.8 hours predicted in Table 5-4.

Table 5-4 Estimated Duration of Pumped Injection

Plant Operating State	Decay Heat (MW)	t_{inject} (hours)	Minimum Injection Rate Required	Number of ECCS Pumps Required
POS 5B	13.6	6.7	74 kg/s [1173 gal./min.]	2
POS 6	11.8	7.8	64 kg/s [1017 gal./min.]	2
POS 7	10.3	4.9	56 kg/s [888 gal./min.]	2
POS 8E	9.9	2.8	54 kg/s [852 gal./min.]	2
POS 8L	4.2	6.5	23 kg/s [362 gal./min.]	1
POS 9	3.6	25.5	20 kg/s [311 gal./min.]	1
POS 10	3.5	26.2	19 kg/s [302 gal./min.]	1
POS 11	3.4	26.9	18 kg/s [293 gal./min.]	1

Table 5-4 includes the minimum required injection flow rate to maintain RCS temperature below 200 °F, which is given by $Q / (C_p \Delta T)$. The CCPs and SIPs are assumed to be at or near their maximum flow rate. The CCP maximum flow rate is 34 kg/s [541 gal./min.], and the SIP maximum flow rate is 41 kg/s [652 gal./min.]. The number of pumps to meet the minimum flow rates is two pumps for POSs prior to refueling and one pump for POSs after refueling.

RCS Pressure during Pumped Injection and Possible Nozzle Dam Failure

Steam generator nozzle dams are installed during mid-loop operation in POS 6 to allow access for steam generator surveillance and maintenance activities. The nozzle dams are in place until removal during mid-loop operation in POS 10 after refueling. Failure of one of the nozzle dams in the four RCS loops would result in a large opening in the RCS and draining of RCS inventory. While there are no documented cases of catastrophic nozzle dam failures causing draindowns,¹⁶ the potential impacts of such failures are considered here and discussed further in a sensitivity study in Section 10.3.9.

The nozzle dams are designed to maintain RCS integrity under the hydrostatic pressure from the flooded refueling cavity. However, the nozzle dams could fail if an RCS pressure transient exceeding the design limits of the dams were to occur. Two possible scenarios that could result in RCS pressure transients are considered. First, if the RCS is not adequately vented prior to nozzle dam installation, then the RCS will pressurize if cooling is lost. In particular, adequate venting on the hot leg is necessary prior to establishing an opening in the cold leg. Without adequate venting, an RCS pressure increase could cause rapid loss of inventory from the cold leg opening. However, the reference plant has clear procedural guidance for establishing adequate RCS vent paths prior to entering a reduced RCS inventory state. Therefore, the failure to establish adequate RCS vent paths is considered to be unlikely.

The second scenario considered for RCS pressurization is related to pumped injection in response to loss of RHR cooling. The simulated loss of RHR cooling scenarios performed for this study show that the injection flow required to maintain RCS temperature below 200 °F will quickly fill the pressurizer and spill out of the removed pressurizer safety valve openings. The RCS pressure will increase when the ECCS pumps are running, and the pressurizer is filled. While operators could stop the injection prior to overfilling the pressurizer, maintaining the RCS temperature below 200 °F is expected to take precedence based on the abnormal operating procedure response to loss of RHR cooling.

Two variations of the loss of RHR simulation scenarios were run, one with all (four) CCPs and SIPs running for injection and one with just two CCPs running. The peak RCS pressure achieved with all four pumps running was approximately 6.5 bar-abs [80 psig]. The peak RCS pressure with two CCPs running was approximately 3.4 bar-abs [35 psig]. No definitive references for the nozzle dam failure pressure were identified for this project, but the peak RCS pressures observed in these simulations are expected to challenge the nozzle dam integrity. NUREG-1449 (NRC, 1993) refers to earlier studies that assumed a nozzle dam failure pressure of 25 psi and states “the actual failure pressure is not well known and likely varies among different designs.” Other nozzle dam failure pressure values have been reported in the range of 35 psig to 45 psig. The operation of all four CCPs and SIPs would be expected to fail the nozzle

¹⁶ Operating experience data associated with nozzle dams are not readily available, though there are some reports of operational issues and leakage with nozzle dams. These issues have typically been resolved by extending mid-loop operation to correct or repair the nozzle dam installation before proceeding with the outage.

dams, but operation of all four pumps is not expected. Flow from all four pumps is not required to maintain cooling, and all four pumps may not be available for injection under some POS conditions. However, the operation of two pumps is more likely to occur, and this would also present a challenge to the nozzle dams with peak RCS pressures reaching levels similar to the expected failure pressure.

5.3.6 Gravity Injection

Gravity injection requires aligning the RWST to the cold legs via the RHR system. Successful gravity injection must be established before the RCS reaches saturation temperature. After boiling has begun, the pressure in the reactor vessel will rise and eventually will exceed the gravitational head provided by the RWST. If gravity injection is successfully aligned before RCS boiling, then the injection will continue for approximately six hours. This assumes no operator actions to control the injection flow rate. The gravity injection success criteria analysis has been supported by a series of plant-specific MELCOR calculations.

5.3.7 Long-term Recovery of RHR

The long-term recovery of RHR is applied to sequences where successful RCS injection can delay the onset of core damage for several hours. The additional time provides opportunity for actions to restore RHR operation. The time available is based on the duration of the injection (pumped or gravity feed), and considers the inventory of the RWST during the POS. See Table 5-4 above for durations of pumped injection. The actions for restoring RHR operation involve responding to the steps in the applicable abnormal operating procedure(s), verifying conditions to restore RHR operation, venting the RHR train (if applicable), and following the applicable system procedures to perform these actions. The restoration actions are expected to take one hour to perform. The human error probabilities (HEPs) are developed based on the evaluation of performance shaping factors, as discussed in Section 7.5. The HEP values depend on the available time for injection but are not sensitive to specific POS-to-POS variations. Two human failure events are defined for RHR restoration, one for POSs with less than five hours available for injection and one for POSs with five hours or more available. Non-recovery probabilities are based on data for repair times in response to RHR failures. The non-recovery probabilities are discussed further in Section 8.2.

If the initiating event involves a loss of electrical power (i.e., LOOP or LO4160V), then the long-term recovery of RHR also requires restoration of power to at least one RHR train.

In addition to the human failure aspects, the following conditions must be met for long-term restoration of RHR cooling.

- AC power is available to at least one train of RHR.
- RHR system equipment failures would not prevent the recovery of at least one train of RHR. RHR restoration and repair is discussed in Section 8.4. RHR failures that require repair are expected to exceed the time available for restoring RHR cooling. Therefore, no credit is given for repair of RHR equipment failures.
- The accident sequence does not include operator failures to trip RHR pumps in response to loss of RCS level and pump cavitation. Failures to protect the RHR pumps is assumed to result in non-recoverable pump failure.

5.3.8 Emergency Power

Emergency power is provided by one of two emergency diesel generators (EDGs) in response to a loss of electrical power event (i.e., LOOP or LO4160V). Each EDG is associated with one class 1E 4,160V AC bus. There is no credit for cross-tying an EDG from the opposite train or from the other unit.

Recovery of an EDG after failure is only credited for those POS conditions where one EDG train is unavailable for maintenance and testing. It is assumed that restoration of an EDG is more likely to be successful given the availability of maintenance crews during these conditions. The time available for repair is specified as follows:

- Time available for EDG recovery is assumed to be two hours if none of the conditions below apply. Two hours is chosen to be consistent with the time available for restoring offsite power (see discussion of “Recovery of Offsite Power” below).
- For POSs with reduced inventory, the time to reach saturation temperature can be less than two hours. No credit is given for EDG recovery for POSs with reduced inventory.
- For station blackout sequences, establishing gravity injection or turbine-drive AFW (as applicable to the POS conditions) extends the time available for EDG recovery to four hours.
- EDG recovery is not credited if there are failures-to-open of the offsite power supply circuit breakers to the RATs.

5.3.9 Recovery of Offsite Power

The recovery of offsite power is based on industry-wide LOOP duration data and consistent with the approach used in the at-power internal event LOOP analysis. The time available for recovery is limited to two hours based on the most limiting plant batteries required to realign offsite power to a safety-related bus, which are the turbine building batteries (see Level 1 at-power PRA documentation (NRC, 2022)). For POSs where power recovery is required before 2 hours due to short times to reach boiling, then the following timing assumptions are applied. For POS 6 mid-loop operations, the time for offsite power recovery is 30 minutes. For POS 5B, POS 9 and POS 10, with RCS level near the reactor vessel flange, the time for offsite power recovery is 1 hour.

5.4 System Success Criteria

The success criteria for plant systems that are used for accident mitigation are presented in this section. The success criteria assumptions use information from the L3PRA project at-power PRA model (NRC, 2022), plant information references (e.g., FSAR, procedures), and plant-specific calculations and analyses. The modeled plant systems, the success criteria assumptions, and the bases for those assumptions are shown in Table 5-5.

Table 5-5 System Success Criteria

System	Function	Success Criteria	Discussion / Basis
RHR	Shutdown cooling	1 of 2 RHR trains (a train includes pump, heat exchanger, hot leg suction and cold leg injection flow paths)	Based on system description in FSAR, one RHR train is sufficient for cooldown.
RHR	Cold leg injection	1 of 2 injection paths (each RHR train feeds 2 cold leg injection paths)	RHR injection paths are only modeled for decay heat removal. Each RHR train feeds two injection paths. One path is assumed sufficient if check valve failure or blockage occurs in the other path.
	Gravity drain	1 of 2 RWST to RHR suction paths with 1 of 2 cold leg injection paths in each train.	AOP for loss of RHR lists four options for gravity draining from RWST: through RHR pumps to cold legs, through SI pumps to cold legs, through RHR pumps to hot legs, and through SI pumps to hot legs. Only the first (and preferred) method is modeled.
CCW	RHR heat exchanger and pump seal cooler	2 of 3 pumps in each train	Same criterion as used in at-power model based on reference plant's system notebook.
NSCW	NSCW pump trains	2 of 3 pumps in each train	Same criterion as used in at-power model based on reference plant's system notebook.
	NSCW cooling tower fans	1 of 4 fans in each cooling tower	Criteria applies when cooling tower is not bypassed. This is the same criterion that is used in the L3PRA at-power modeling for accident scenarios that do not result in a safety injection signal (NRC, 2022).
	NSCW cooling tower bypass mode	0 of 4 fans required in bypass mode	Cooling tower is bypassed in cold weather conditions and is modeled as a fraction of time in bypass mode. The same modeling approach is applied in the at-power model. See Section 5.2.7 of the at-power model report (NRC, 2022).
RCS vent	Bleed path for alternate cooling	1 of 2 PORVs and associated block valves	1 of 2 PORVs required for RCS venting during non-vented POSs. The same criteria is applied for feed and bleed sequences in the L3PRA at-power Level 1 PRA (NRC, 2022).
Centrifugal Charging Pumps	Injection for alternate cooling	1 of 2 CCPs for injection	AOP for loss of RHR specifies alignments for SIPs for hot leg injection or CCPs for cold leg injection.
	Cold leg injection paths	2 of 4 cold leg injection paths	AOP for loss of RHR specifies alignments for SIPs for hot leg injection or CCPs for cold leg injection.

Table 5-5 System Success Criteria (cont.)

System	Function	Success Criteria	Discussion / Basis
Safety Injection	Injection for alternate cooling	1 of 2 SIPs for injection	AOP for loss of RHR specifies alignments for SIPs for hot leg injection or CCPs for cold leg injection.
	Hot leg injection paths	2 of 4 hot leg injection paths	AOP for loss of RHR specifies alignments for SIPs for hot leg injection or CCPs for cold leg injection.
Emergency Power	Onsite power in response to loss of offsite power	1 EDG is required to power its associated 4,160V AC bus	Cross ties to other buses or from the other unit are not credited.
Offsite Power	Normal offsite power sources	1 reserve auxiliary transformer (RAT) or the standby auxiliary transformer (SAT)	Each class 1E 4,160V AC bus is normally powered by its associated RAT. The SAT may be aligned as an alternate power source, if available.
	Alternate AC sources with refueling cavity flooded	Both 4,160V AC buses powered from same RAT, or the SAT	During RAT maintenance, alternate AC alignments are credited. Both class 1E 4,160V AC buses can be powered from the same RAT. The SAT may be aligned as an alternate power source, if available.

6 SYSTEM MODELS

The objectives of the systems analysis element are to identify and quantify the causes of failure for each plant system represented in the initiating event analysis and accident sequence analysis. Each system that is modeled in response to shutdown accident scenarios is discussed below.

6.1 Residual Heat Removal (RHR)

Description: The RHR system is used in all shutdown cooling modes for removing decay heat.

RHR Fault Tree Use: The RHR system is necessary as a cooldown system for all initiating events (i.e., LOOP, LO4160V, LORHR, and LOINV) in shutdown cooling.

Required Support Systems: The following support systems are required for successful operation of the RHR system in shutdown cooling mode:

- AC power is required for motive power to pump motors and for motive and control power to motor-operated valve (MOV) operators, while DC power is required for control power to the pump motors and circuit breakers. All the pumps, valves, and other equipment which are required for cooling of safety-related components are supplied from safety-related buses. These safety-related buses are supplied by emergency power in the event that primary power sources are lost.
- The NSCW system cools the RHR pump motors and lube oil.
- The CCW system cools the RHR pump seals. However, this dependency is not modeled in the PRA based on manufacturer test results that indicate the seals will remain operable under worst case operating temperature and pressure conditions without CCW cooling to the seals.
- The RHR pump rooms receive room cooling which is provided by the normal chilled water system and the essential chilled water system, but the room cooling is not required for pump operation based on the plant's room heatup analysis.

Associated Fault Trees

1-SD-RHR-D: The top event models the operators' diagnosis of a LORHR event. The operator action basic event models the operator failure to diagnose the initiating event based on the available indications of RHR failure. The diagnosis and the action to restore RHR (modeled in the subsequent event tree node) must be completed prior to RCS level reaching one foot above the hot leg centerline. The time available varies depending on the POS, and a portion of the available time is apportioned for diagnosis. Estimation of human error probabilities for failure to diagnose LORHR is discussed in detail in Section 7.

1-SD-RHR-R: The top event models the recovery of the operating RHR train(s). Failure to recover the RHR train is modeled by an operator failure event or failures of the RHR system, flow path, or support systems. The POS conditions and nature of the initiating event influence the likelihood of recovering the RHR train. If both RHR trains are normally operating prior to the initiating event, then this top event models recovery of either one of the two RHR trains. If one

RHR train is operating and one is in standby prior to the initiating event, then this top event models recovery of the train that was failed by the initiator. Depending on the nature of the failure, the operating train may or may not be recoverable. The RHR train must be recovered prior to RCS level reaching one foot above the hot leg centerline. Note that the RHR B train is assumed to be operating if only one train is in operation.

1-SD-RHR-SB: The top event models aligning the standby RHR train to provide core cooling. The standby train (assumed to be RHR train A) may be aligned for shutdown cooling, operable but not aligned, or out of service for maintenance, depending on the POS. The standby RHR train must be recovered prior to RCS level reaching one foot above the hot leg centerline. If both RHR trains are operating and failed by the initiating event, then this top event is not used. Instead, recovery of either one of the two operating RHR trains is modeled by top event 1-SD-RHR-R.

1-SD-LT-REC: This top event represents the late recovery of RHR cooling. The time available for recovery of RHR with successful injection is two days (or three days if RWST refill is credited). The RHR non-recovery probabilities are based on the time available for repair/restoration activities and RHR repair data. The RHR non-recovery probabilities are discussed further in Section 8.

6.2 Steam Generator Cooling

Steam generators are not used and may not be available during many POSs. For those POSs where steam generator cooling is available, success requires 1 of 4 SGs to operate and remove decay heat. Steam generators are used in case of failure of the RHR system to provide shutdown cooling for all initiating events (i.e., LOOP, LO4160V, LORHR, and LOINV) in shutdown cooling.

Associated Fault Trees

1-SD-SGCOOL: The top event models using steam generators for RCS cooling. Heat removal using the steam generators is possible during some shutdown POSs. For successful steam generator cooling, secondary inventory must be available and operators must establish AFW to maintain steam generator level. Once the RCS is vented, then steam generators are no longer effective for heat removal, regardless of steam generator inventory level. This top event is not used for POSs where the RCS is vented or where RCS pressure and level would not be sufficient for heat removal using steam generators (POSs 5B, 6, 7, 8E, 8L, 9, 10, and 11).

6.3 Auxiliary Feedwater System (AFW)

Description: The AFW system is designed to supply feedwater from the condensate storage tanks (CSTs) to the SGs whenever the reactor coolant temperature is above 350°F and the main feedwater (MFW) system is not in operation (i.e., during start-up, cooldown, or emergency conditions resulting in a loss of MFW).

The AFW system automatically provides feedwater for the removal of reactor core decay heat following a loss of MFW. MFW may be lost due to a loss of offsite AC power, or a secondary-side piping or component failure. The AFW system prevents damage to the reactor core until the reactor coolant temperature is brought from a condition of full power to the condition at which the RHR system may be placed in operation. The AFW system supplies

feedwater to the SGs at a flow rate sufficient to support normal low power transients, such as start-up, cooldown, and hot standby.

Each unit has two train-oriented motor-driven AFW pumps and one turbine-driven AFW pump that take suction from one of the two CSTs per unit. Either CST 1 or CST 2 can be used, but the pumps are normally aligned to CST 1, and only one CST is in service at a given time. Upon depletion of the in-service CST, the alternate CST can be manually aligned to the suction of the pumps. Each pump has a suction line from each CST. An additional source of water is available from the demineralized water tank, which is automatically used for makeup to the CSTs, if available. If AFW is used for long-term decay heat removal, makeup to CST 1 or operator alignment of CST 2 is required to achieve a 72-hour safe/stable end-state.

Each motor-driven auxiliary feedwater pump is sized to supply the feedwater flow required for removal of 100 percent of the decay heat from the reactor. The turbine-driven pump is sized to supply up to twice the capacity of a motor-driven pump. The nominal success criteria for the AFW system are 1 of 2 motor-driven AFW pumps or the turbine-driven AFW pump delivers at least 570 gpm to at least 2 of 4 SGs. Steam removal from the SGs fed with AFW is required by either: (1) three TBVs or (2) an ARV or 1 of 5 safety relief valves (SRVs) for 2 of 4 SGs.

The two motor-driven AFW pumps are automatically started by the reactor protection system (RPS), engineered safety features actuation system (ESFAS), anticipated transient without scram (ATWS) mitigation system actuation circuitry (AMSAC), or loss of MFW upon receipt of the following signals:

- Two of four low-low level signals from any one SG (RPS)
- Any safety injection (SI) signal (ESFAS)
- Any loss of, or degraded, safety-related 4.16 kV AC bus voltage signal (ESFAS)
- An AMSAC signal
- A signal resulting from the trip of both MFW pumps

The two motor-driven pumps can also be started from the main control board and the remote shutdown panels.

The turbine-driven pump is automatically started by the RPS, ESFAS, or AMSAC, upon receipt of the following signals:

- Two of four low-low level signals from any two SGs (RPS)
- Any loss of, or degraded, safety-related 4.16 kV AC bus voltage signal (ESFAS)
- An AMSAC signal

The turbine-driven pump can also be started and controlled from the main control board and the remote turbine-driven pump AFW panel. In addition, the MOVs in the steam supply line to the turbine can be operated from the main control board and the remote turbine-driven pump AFW panel.

The AFW actuation signals are inhibited during shutdown POSs. Effective heat removal with the steam generators is not possible after establishing a large vent path in the RCS. Therefore, RHR cooling is the preferred method for heat removal for most of the shutdown POSs.

AFW Fault Tree Use: AFW is used as a subtree for SG cooling.

Required Support Systems: The following support systems are required for successful operation of the AFW system:

- The AFW system is dependent upon the electric power system for the operation of the motor-driven pumps and MOVs, for instrument power for equipment control, and for monitoring and indication of system parameters. AC power is required for the motor-driven pumps and their associated suction and discharge valves. Each motor-driven pump and its associated motor-operated discharge valve are powered from the corresponding diesel generator, in the event normal AC power is lost. DC power is required for the motor-driven pump start control circuitry, the turbine-driven pump steam supply MOVs, and the turbine-driven pump discharge valves to each SG.
- The main steam system provides steam from the SG 1 and 2 steam lines to the turbine of the turbine-driven pump. The turbine can perform its intended function for steam inlet pressures ranging from 90 to 1130 psig.
- The AFW pumps are located in separate rooms in the AFW pump house. The turbine-driven pump room is cooled by natural circulation with outside air for emergency conditions. The operation of the turbine-driven pump actuates a signal to open the engineered safety feature (ESF) damper allowing ventilation by natural circulation. However, the turbine-driven AFW pump does not need room heating, ventilation and air conditioning (HVAC) (including the opening of the ESF pneumatically operated damper) for its operation for at least 24 hours. Thus, the loss of turbine-driven AFW pump room cooling is not modeled in the current L3PRA Level 1 model. The motor-driven AFW pump rooms are cooled by outside air through the use of a fan and a damper in each room. These are part of the ESF system. The fan is started when the room temperature reaches 90°F. Based on room heatup analysis, the motor-driven AFW pump rooms would reach a temperature of 161°F in 24 hours without room cooling, which is below that which would result in failure of the pumps.
- The SG ARVs require 480V AC supply power and 125V DC control power to open.¹⁷ The TBVs require 125V DC control power and instrument air to open. In addition, the condenser heat sink must be available for the TBVs to be operable.

6.4 Emergency Power System (EPS)

Description: After a loss of offsite power (LOOP), the emergency diesel generators (EDGs) start and supply power for the mitigating systems. If at least one AC emergency bus is energized with an EDG supplying power, the event progression would be similar to a transient. If both EDGs fail to start or run after a LOOP event, a SBO occurs.

The EDGs provide standby onsite power required by the safety-related AC power systems in the event of a loss of preferred power sources, for powering the essential loads necessary to safely shut down the reactor under any operating and accident condition.

¹⁷ The SG ARVs can also be operated using local manual operators. However, this manual operation is not credited for cooldown and depressurization during non-SBO scenarios in the reviewed version of the reference plant's PRA model. Due to its negligible risk significance, manual opening the SG ARVs locally is also not included in the L3PRA, at-power and LPSD, Level 1 models.

The system has two independent trains for each unit. Each train has an emergency diesel generator connected exclusively to a single safety-related 4.16 kV AC bus of a load group. Each safety-related 4.16 kV AC bus has similar safety-related equipment on it. One bus is adequate to satisfy minimum ESF demands caused by a LOCA and a simultaneous loss of offsite power supply. Power and control cables for the EDGs and associated switchgear are routed to maintain physical separation.

The EDG fuel oil storage and transfer system is used to transfer diesel fuel oil from the DG fuel oil storage tank using pumps to replenish the EDG day tank as fuel oil is being consumed. The EDG fuel oil day tanks each contain 1,250 gallons, to provide approximately 2.6 hours of operation for its associated diesel engine at the maximum operating load without resupply from an EDG fuel oil storage tank. The design of the diesel fuel oil storage system allows replenishment of fuel oil without interrupting operation of the EDG. Should continuous operation of the generator for more than seven days be required, it will be necessary to refill the diesel fuel oil storage tank. A fuel oil transfer pump is automatically started and stopped by a day tank level switch. The second pump is started automatically in the event of low discharge pressure from the lead pump or day tank low-low level. Should a pump be run continuously or fail to stop, any overflow is returned to the storage tank via the recirculation line.

The EDG building ESF HVAC system is required for EDG operation. Each EDG has two fans along with their associated dampers. Failures of both fans (or their associated dampers) will cause the EDG to fail.

EPS Fault Tree Use: This top event fault tree is used in the shutdown event trees for LOOP and loss of 4.16kV AC bus.

Required Support Systems: The NSCW system provides cooling water for the EDG jacket water cooling system, which also removes heat from the lube oil system.

6.5 Feed and Bleed (FAB)

Description: The feed and bleed mode of reactor cooling is needed if AFW/MFW are unavailable, and requires successful high pressure injection (HPI) to provide flow to the RCS cold legs and opening of the PORV(s) to remove RCS decay heat. Operators must initiate feed and bleed cooling by manually initiating a SI signal (if an automatic SI actuation has not already been generated). After initiation of the SI signal, the operator will then open the PORV(s). In addition, operators must trip all the RCPs before initiating feed and bleed.

Required Support Systems:

- The plant electrical power systems are required for the operation of HPI. AC power is required for motive power to pump motors and for motive and control power to MOV operators, while DC power is required for control power to the pump motors and circuit breakers. All of the pumps, valves, and other equipment which are required for cooling of safety-related components are supplied from safety-related buses. These safety-related buses are supplied by emergency power in the event that primary power sources are lost.
- The ESFAS sends start signals to safety injection pumps (SIPs), centrifugal charging pumps (CCPs), and several valves for safety injection. In addition, it starts the CCPs on

a loss of power signal. Safety injection signals are inhibited, and SIPs placed in pull-to-lock during cooldown to shutdown in POS 2. During shutdown POSs, manual actions, as guided by abnormal operating procedures, are required to align SIPs or CCPs for injection.

- NSCW provides cooling water to the CCPs and SI pump motor coolers and lube oil coolers.

Associated Fault Trees

1-SD-BLEED: The top event models venting of the RCS by opening one of two pressurizer PORVs. The venting of a PORV depressurizes the RCS to support core cooling by forced injection, gravity feed from the RWST, or re-establishing RHR cooling. During POSs where the RCS is vented (e.g., pressurizer safety valves removed), this top event is not relevant. The top event is set to 'FALSE' and success is guaranteed for RCS vented conditions.

1-SD-FEED: This top event fault tree models establishing injection paths using SIPs or CCPs to control RCS inventory. The fault tree top event includes an operator action to establish forced feed paths. The success criteria are defined as: one of two SIPs through two of four hot legs, one of two SIPs through two of four cold legs, or one of two CCPs through two of four cold legs. Successful forced feed must be established before the RCS water boils off to two-thirds of the active fuel height.

Once SIP or CCP feed is established, it can continue to operate until the RWST inventory is depleted. The RWST holds approximately 730,000 gallons of water. It is assumed that injection from the RWST will continue until the RWST empty level alarm occurs when 105,000 gallons are remaining. There would still be approximately 80,000 gallons of usable water in the tank. The assumptions about injection flow rates and durations are summarized in Table 5-4.

7 HUMAN RELIABILITY ANALYSIS

7.1 Introduction

The purpose of this section is to describe the approach used in performing the human reliability analysis (HRA) for internal events at low power and shutdown (LPSD) conditions. The various plant operating states (POSs) associated with LPSD can be divided into those POSs that occur during low power and those POSs that occur during shutdown. Moreover, these two groups can be further divided into POSs with old fuel prior to refueling (POS_{SD-PR}) and new fuel after refueling (POS_{SD-AR}).

The human failure events (HFEs) were identified and their HEPs quantified using the SPAR-H method (NRC, 2005a; NRC, 2006; Whaley, 2011). However, generally only the HFEs for POS_{SD-PR} were quantified using the SPAR-H method. For those HFEs occurring during POS_{SD-AR} that were identified as similar to those occurring during POS_{SD-PR}, the HEPs for the POS_{SD-PR} were used. However, this approach was limited to HFEs in situations where the associated HEPs would not be significantly influenced by the available time, as the after refueling HFEs typically have much longer times available for operator response. If any of the HFEs for POS_{SD-AR} were considered to be unique and not similar to any of the POS_{SD-PR} HFEs, then their HEPs were quantified using the SPAR-H method.

The following sections provide a more detailed discussion of the approach and evaluation of the HRA performed for LPSD conditions:

- Section 7.2 – Information Collection
- Section 7.3 – Identification and Selection of HFEs
- Section 7.4 – SPAR-H Method
- Section 7.5 – SPAR-H Performance Shaping Factors
- Section 7.6 – Dependency Evaluation
- Section 7.7 – Quantification and Uncertainty Analysis

7.2 Information Collection

Information was collected to ensure an understanding of the various human activities performed during LPSD conditions. The procedures, conditions, etc., associated with LPSD operations differ significantly as compared to at-power operations. Consequently, information was collected as specifically related to LPSD.

The information collection included the following:

- visit to the reference plant in November 2014 to observe a refueling outage (including interviews with operators and senior training staff, and a visit to the plant simulator)
- review of plant procedures used during LPSD operations
- review of LPSD events for commercial NPPs
- interviews with current operators at the reference plant
- review of past LPSD HRAs conducted for other plants.

A primary objective of the reference plant visit was to observe plant conditions and equipment locations during shutdown conditions, as conditions and activities during plant shutdown can differ significantly from at-power operation. An example of a difference is the several hundred

additional workers that are on site to perform testing and maintenance activities during shutdown. Notable observations were made during the walkdowns in the following areas:

- protected equipment markings
- temporary modifications to the plant
- maintenance activities in progress
- potential impacts on the operating unit.

Additionally, during the plant visit, the operations coordinator and an operator trainer were interviewed. These interviews allowed the team to clarify their understanding of reference plant's LPSD operations, as well as gain better understanding of the general hazards that may occur during LPSD operations. Evaluating operator actions during specific accident scenarios sometimes required additional information beyond what was discussed earlier. The additional information generally included an estimate of the expected required time to perform a set of procedurally directed actions, the clarity of the procedural steps, and the actual work environment under which the actions (including diagnosis) are performed. The project staff consulted a former Senior Reactor Operator with over 15 years of experience in five different commercial PWR nuclear power plants. This consultation was valuable for the HRA analysts to better understand nominal operator training regarding the specific scenario, the level of stress that operators may experience, the difficulty of performing the required tasks, and the general coordination and communication during the task performance.

Finally, a high-level review of operating experience with LPSD events for commercial NPPs was conducted. This effort focused on major qualitative insights rather than performing a detailed quantitative review. This review had the following objectives:

- ensure the project has accounted for the major insights obtained from prior efforts
- include more recent events that may not be part of past reviews of operating experience
- look for unique initiating events and plant configurations
- review events where the initiating event is human-induced to identify any impact on operator mitigation actions.

Appendix D provides more details of the LPSD operating experience information that was collected.

7.3 Identification and Selection of HFEs

The HFEs are identified as part of the accident scenario development, which is discussed in more detail in Section 4. For each accident scenario, the applicable plant procedures were reviewed. The necessary actions and procedural steps to respond to the initiating event are identified in a series of tables in Section 4.3. Also, as part of the accident scenario development, an event tree for each scenario was created. The event tree top events are defined for the different functions that can be credited in response to the accident. The operator action (or actions) required to perform each top event function were identified. As most automatic actuation functions are disabled during shutdown, every function is expected to require some operator action. In this way, the HFEs were identified as part of the accident scenario development. After the HFEs were identified for each accident scenario, they were reviewed to

consider if they are applicable across multiple accident scenarios (in cases where the accident scenario conditions are the same or similar, then the same HFE may be applied to multiple accident scenarios).

The following steps were performed in identifying the HFEs:

1. The accident scenarios to be modeled were identified and defined.
2. The applicable abnormal operating procedure(s) for each accident scenario were reviewed.
3. An event tree model was developed for each accident scenario, with top events defined for each function that is modeled in the accident response.
4. An HFE (or set of HFEs) was identified for each top event function modeled in the event tree, considering the specific conditions of the POS and initiating event category being modeled in the scenario.
 - a. These HFEs are identified for each POS in Section 4.4.
 - b. The HFEs were defined at a functional level. If operators have multiple options for performing a modeled function, then a single HFE was defined, which could encompass multiple actions. For example, providing forced injection with either the charging system or safety injection system was modeled as a single HFE.
5. If required, additional sequence-specific HFEs were considered. In some cases, a human failure early in an accident sequence can introduce the need for additional actions. For example, if the initial diagnosis of a loss of RHR event fails, the restoration of RHR may require an additional action to vent the system. This action would not be necessary if the initial diagnosis was successful and RHR pump(s) were tripped prior to air being ingested into the system.
6. HFEs were reviewed for applicability to other accident scenarios. The same HFE was used in multiple scenarios if the initiating event response and POS conditions were the same or similar.

Table 7-1 provides the list of identified HFEs.

Table 7-1 List of Modeled Human Failure Events

HFE Name	HFE Description	Accident Scenarios
1-ACP-XHE-XL-ALTAC-6H	FAIL TO ALIGN ALTERNATE OFFSITE POWER VIA SAT OR OPPOSITE TRAIN RAT IN 6 HOURS	POS 8E 1-SD-LO4160V, 1-SD-LOINV
1-AFW-XHE-FO-SGCOOL	OPERATORS FAIL TO ESTABLISH AFW/SG COOLING - SHUTDOWN	POS 4, 5A 1-SD-LORHR
1-AFW-XHE-XM-SGC-POS3	OPERATORS FAIL TO MAINTAIN SG LEVEL AND CONTROL RCS TEMPERATURE	POS 3 1-SD-LORHR
1-AFW-XHE-XM-SGC-PRF	OPERATOR FAILS TO MAINTAIN SG LEVEL AND CONTROL RCS TEMPERATURE	POS 12, 13 1-SD-LORHR
1-DEP-XHE-XM-BLEED	OPERATOR FAILS TO DEPRESSURIZE RCS FOR FORCED INJECTION	POS 11 1-SD-LORHR
1-HPI-XHE-XM-FFD-PRF-NSCW	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP AFTER LOSS OF NSCW	POS 8E-13 1-SD-LONSCW
1-HPI-XHE-XM-FFEED-6H	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP	POS 5B, 9, 11 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOOP
1-HPI-XHE-XM-FFEED-6H-5A	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP POS 5A	POS 3-5A, 12-13 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOOP
1-HPI-XHE-XM-FFEED-LOI	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-LOI	POS 5B, 9, 11 1-SD-LOINV
1-HPI-XHE-XM-FFEED-LOI5A	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-LOI POS5A	POS 5A 1-SD-LOINV
1-HPI-XHE-XM-FFEED-LOI-6	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-LOI POS6	POS 6 1-SD-LOINV
1-HPI-XHE-XM-FFEED-LOI-7	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-LOI POS7	POS 7 1-SD-LOINV
1-HPI-XHE-XM-FFEED-LOI8E	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-LOI POS8E	POS 8E, 8L 1-SD-LOINV
1-HPI-XHE-XM-FFEED-MLDRL	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-MLDRL	POS 6, 10 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOOP, 1-SD-ODML
1-HPI-XHE-XM-FFEED-NSCW	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP - LONSCW	POS 5B, 6 1-SD-LONSCW
1-HPI-XHE-XM-FFEED-POS8E	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP - POS8E	POS 7, 8E, 8L 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOOP
1-LOI-XHE-ISO2-LT	FAIL TO ISOLATE LOI AFTER SUCCESSFUL INJECTION	All POSs 1-SD-LOINV
1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY	POS 5A 1-SD-LOINV
1-LOI-XHE-XD-POS8E	OPERATOR FAILS TO DIAGNOSE LOI EARLY	POS 8E, 8L 1-SD-LOINV

Table 7-1 List of Modeled Human Failure Events (cont.)

HFE Name	HFE Description	Accident Scenarios
1-LOI-XHE-XM-ISO-POS8E	OPERATOR FAILS TO TRIP AND ISOLATE RHR TRAIN IMPACTED BY LOI	POS 8E, 8L 1-SD-LOINV
1-LPI-XHE-XM-GFD-ML-NSCW	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOI	POS 6 1-SD-LONSCW
1-LPI-XHE-XM-GFD-PRF-NSCW	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOI	POS 9, 10 1-SD-LONSCW
1-LPI-XHE-XM-GFEED-LOI	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOI	POS 5B, 9, 11 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOOP
1-LPI-XHE-XM-GFEED-LOI5A	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOI POS 5A	POS 5A 1-SD-LOINV
1-LPI-XHE-XM-GFEED-LOI-6	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOI POS 6	POS 6 1-SD-LOINV
1-LPI-XHE-XM-GFEED-LOI-7	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOI POS 7	POS 7 1-SD-LOINV
1-LPI-XHE-XM-GFEED-LOI8E	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOI POS 8E	POS 8E, 8L 1-SD-LOINV
1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP	All POSs 1-SD-LOOP
1-LPI-XHE-XM-GFEED-MLDRL	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-DRL	POS 6, 10 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOOP, 1-SD-ODML
1-LPI-XHE-XM-GFEED-NSCW	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED - LONSCW	POS 5B 1-SD-LONSCW
1-RHR-XHE-XD-3H	OPERATORS FAIL TO DIAGNOSE LORHR BEFORE BOILING STARTS (SHUTDOWN)	POS 5B, 9, 11 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOOP
1-RHR-XHE-XD-3H-5A	OPERATORS FAIL TO DIAGNOSE LORHR BEFORE BOILING STARTS (SHUTDOWN) POS 5A	POS 4, 5A 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOOP
1-RHR-XHE-XD-ML	OPERATORS FAIL TO DIAGNOSE LORHR BEFORE BOILING DURING MID-LOOP	POS 6, 10 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOOP
1-RHR-XHE-XD-POS3	FAIL TO DIAGNOSE LORHR BEFORE BOILING STARTS (SHUTDOWN)	POS 3 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOOP
1-RHR-XHE-XD-POS7	OPERATORS FAIL TO DIAGNOSE LORHR BEFORE BOILING	POS 7 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOOP

Table 7-1 List of Modeled Human Failure Events (cont.)

HFE Name	HFE Description	Accident Scenarios
1-RHR-XHE-XD-POS8E	OPERATORS FAIL TO DIAGNOSE LORHR BEFORE BOILING POS 8E	POS 8E, 8L 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOOP
1-RHR-XHE-XD-PRF	FAIL TO DIAGNOSE LORHR BEFORE BOILING STARTS (SHUTDOWN)	POS 12, 13 1-SD-LORHR
1-RHR-XHE-XL-LTREC	SHUTDOWN - FAIL TO RE-ESTABLISH NORMAL RHR COOLING AFTER SUCCESSFUL INJECTION	Sequence specific 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOINV, 1-SD-LOOP
1-RHR-XHE-XL-LTREC-LT5H	SHUTDOWN - FAIL TO RE-ESTABLISH NORMAL RHR COOLING AFTER SUCCESSFUL INJECTION	Sequence specific 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOINV, 1-SD-LOOP
1-RHR-XHE-XL-STBY-POS7	FAIL TO ALIGN SHUTDOWN COOLING FLOW WITH AVAILABLE RHR TRAIN	POS 7 1-SD-LO4160V
1-RHR-XHE-XL-TRA-POS3	OPERATORS FAIL TO ALIGN & INITIATE STANDBY RHR TRAIN A	POS 3 1-SD-LORHR, 1-SD-LOOP
1-RHR-XHE-XL-TRA-POS7	OPERATORS FAIL TO RECOVER OPERATING RHR TRAIN A	POS 7 1-SD-LORHR, 1-SD-LOOP
1-RHR-XHE-XL-TRA-POS8E	FAIL TO RECOVER OPERATING RHR TRAIN A	POS 8E, 8L 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOOP
1-RHR-XHE-XL-TRA-PRF	FAIL TO ALIGN AND INITIATE STANDBY RHR TRAIN A	POS 12, 13 1-SD-LORHR
1-RHR-XHE-XL-TRB-POS3	OPERATORS FAIL TO RECOVER OPERATING RHR TRAIN B	POS 3 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOOP
1-RHR-XHE-XL-TRB-POS7	OPERATORS FAIL TO RECOVER OPERATING RHR TRAIN B	POS 7 1-SD-LORHR, 1-SD-LOOP
1-RHR-XHE-XL-TRB-POS8E	OPERATORS FAIL TO RECOVER OPERATING RHR TRAIN B	POS 8E, 8L 1-SD-LORHR, 1-SD-LO4160V, 1-SD-LOOP
1-RHR-XHE-XL-TRB-PRF	FAIL TO RECOVER OPERATING RHR TRAIN B	POS 12, 13 1-SD-LORHR
1-RHR-XHE-XL-TRN-3H	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN	POS 5B, 9, 11 1-SD-LORHR, 1-SD-LOOP
1-RHR-XHE-XL-TRN-LOOP	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN AFTER POWER RECOVERY	All POSs 1-SD-LOOP
1-RHR-XHE-XL-TRN-M5LOI	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN - LOI	POS 5B, 9, 11 1-SD-LOINV
1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN - MLDRL	POS 6 1-SD-ODML-EARLY

Table 7-1 List of Modeled Human Failure Events (cont.)

HFE Name	HFE Description	Accident Scenarios
1-RHR-XHE-XL-TRN-ML-P10	OPERATORS FAIL TO RECOVER ONE RHR TRAIN	POS 10 1-SD-ODML-LATE
1-RHR-XHE-XL-TRN-POS4	OPERATORS FAIL TO RECOVER ONE RHR TRAIN	POS 4, 5A 1-SD-LORHR
1-RHR-XHE-XL-TRN-POS6	OPERATORS FAIL TO RECOVER ONE RHR TRAIN	POS 6 1-SD-LORHR, 1-SD-LOOP
1-RHR-XHE-XL-TRN-POS10	OPERATORS FAIL TO RECOVER ONE RHR TRAIN	POS 10 1-SD-LORHR, 1-SD-LOOP
1-RHR-XHE-XL-TRN-POS9	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN	POS 9 1-SD-LORHR, 1-SD-LOOP
1-RHR-XHE-XL-TRN-STBY-2H	OPERATORS FAIL TO VERIFY ADEQUATE SHUTDOWN COOLING FLOW WITH AVAILABLE RHR TRAIN	POS 5B, 9 1-SD-LO4160V
1-RHR-XHE-XL-TRN-STBY-ML	FAIL TO VERIFY ADEQUATE SHUTDOWN COOLING FLOW WITH AVAILABLE RHR TRAIN	POS 6, 10 1-SD-LO4160V
1-RHR-XHE-XM-LOI-POS5B	OPERATORS FAIL TO DIAGNOSE & ISOLATE LOI BEFORE LOSS OF SD COOLING - POS5B	POS 5B 1-SD-LOINV
1-RHR-XHE-XM-LOI-POS6	OPERATORS FAIL TO DIAGNOSE & ISOLATE LOI BEFORE LOSS OF SD COOLING - POS6	POS 6 1-SD-LOINV
1-RHR-XHE-XM-LOI-POS7	OPERATORS FAIL TO DIAGNOSE & ISOLATE LOI BEFORE LOSS OF SD COOLING - POS7	POS 7 1-SD-LOINV
1-RHR-XHE-XM-LOISP-POS6	OPERATORS FAIL TO DIAGNOSE & ISOLATE LOI BEFORE LOSS OF SD COOLING - POS6	POS 6 1-SD-LOINV-SP
1-RHR-XHE-XM-M5LOI-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE LOI BEFORE LOSS OF SD COOLING - MODE 5	POS 5B, 9 1-SD-LOINV-SP, 1-SD-LOINV-DS
1-RHR-XHE-XM-MLDRL-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML	POS 6 1-SD-ODML-EARLY
1-RHR-XHE-XM-POS10-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML	POS 10 1-SD-ODML-LATE
1-RHR-XHE-XM-TRIP-LOI-POS6	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS IN AN LOI POS 6	POS 6, 10 1-SD-LOINV
1-RHR-XHE-XM-TRIP-M5LOI	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS IN AN LOI	POS 5B, 9 1-SD-LOINV
1-RHR-XHE-XM-TRIP-MLDRL	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS IN MLDRL	POS 6, 10 1-SD-ODML

7.4 SPAR-H Method

The SPAR-H method uses a calculational tool whereby the analyst enters certain information for each specified HFE into a “worksheet,” and the analyst then quantifies an associated HEP. For each HFE, a diagnostic and execution aspect are separately evaluated. The former is the failure to properly diagnose the accident (HFE_D) and consequently failing to properly identify the needed action. The latter is the failure of the human to properly execute the needed action (HFE_E).

The first step of the SPAR-H method is the identification of the required human events. Once the human events are identified, the conditions under which the action is performed are examined. An understanding of the PRA models and accident conditions is necessary to identify the applicable performance shaping factors (PSFs), both positive and negative. This understanding can be facilitated by considering the eight SPAR-H PSFs. The PSFs involve consideration of both (1) the cues and the cognitive processes for diagnosis and (2) the procedural steps involved in executing the actions and the conditions under which the execution is performed. In some cases, the diagnosis may be self-revealing or obvious to operators such that it can be judged not to be a contributor to the estimated HEP value. The evaluation of the HEPs would then be limited to only the execution portion.

The diagnostic and execution actions both rely on available indications and alarms. Available indications and alarms are generally identified in abnormal operating procedures, with their associated thresholds. Depending on the specific accident condition, some instrumentation/alarms may not be available for diagnostic purposes. During execution, the crew and the operator also rely on indications for confirming and verifying that the actions taken have been successful. An example is assuring that the flow is fully established after system alignment is completed. The indications that are available during the accident sequence condition, the clarity (or, on the contrary, the ambiguity) of information provided, the precision and the range of the indications, the fidelity of the instrumentation, and the inherent redundancy and diversity were all examined for both the diagnostic and execution parts of the HRA characterization. This information was considered for evaluating the various PSFs in the SPAR-H worksheets.

Another important source of information for evaluating the PSFs in SPAR-H was verifying the existence and clarity of the procedures for the tasks required to be performed during a specific accident sequence. Also, given that procedures are often written for a generic class of accident sequences, their applicability to and compatibility with the specific accident sequence being considered were also examined. The diagnostic portion of HRA generally deals with the operator ability to identify the right procedures to follow. The diagnostic and execution portion of HRA also depends on the type of the procedure; that is, whether it is symptom based, rule based, or specific to an accident.

Additional steps in the characterization of the human events included reviewing pertinent plant-specific and industry-wide historical events, performing walk-throughs to gain practical understanding of the ergonomic/human machine interaction (HMI) related to the HRA actions, and conducting interviews with the plant crew and operators to gain insights and obtain informed input to support completion of the SPAR-H worksheet.

The SPAR-H method starts with a nominal HEP for HFE_D and HFE_E, which are adjusted depending on whether the PSFs are negative or positive. Negative PSFs are those that result in

a negative influence (when the PSF is greater than 1), such as extreme stress, highly complex task, high stress situations, or poor procedures. Positive PSFs are those that result in a positive influence (when the PSF is less than 1), such as extensive experience or good ergonomics. The nominal HEPs for diagnosis and execution are 10^{-2} and 10^{-3} , respectively.

Once the HEPs have been adjusted by considering the various PSFs, it is also necessary to evaluate the dependency among the HFEs.

Section 7.5 provides discussions on the types of information to be documented as a part of characterizing the human actions – both diagnosis and execution. This is done by providing a summary discussion for characterizing each of the HRA PSFs, using the types of information collected and discussed above, and the rules used for incorporating dependency effects on the various HFEs for both diagnosis and execution.

7.5 SPAR-H Performance Shaping Factors

The following are the PSFs used in the SPAR-H method and are subsequently discussed in this section for both HFE_D and HFE_E:

- timing
- stress
- complexity
- training
- procedures
- human machine interface
- fitness for duty
- work process

For each PSF, SPAR-H assigns factors to adjust the nominal HEP. It is possible in some cases in which multiple negative PSFs are applied, to calculate an HEP greater than 1.0. To correct for these cases, an adjustment factor was applied (discussed further in Section 7.5.9).

When there is insufficient information to evaluate a specific PSF, SPAR-H includes a factor for “insufficient information.” This specific factor was not used in the LPSD PRA, since sufficient information was available to evaluate each PSF.

7.5.1 Time Required and Time Available

The PSF of time examines the time available for the human action and the time needed to diagnose and execute the action. These times will influence the probability of the human successfully performing the action. It is necessary to evaluate the time available and the time required for both diagnosis and execution.

The time available (T_A) to diagnose and execute the action is dependent on the actual accident scenario and is defined through the following (see Section 4.6 and Appendix A of NUREG-1921 (NRC, 2012b)):

T_0	=	start time (e.g., when the initiating event occurs)
T_{sw}	=	system time window is the time from start of the event until the action is no longer beneficial

T_{delay}	=	time from initiating event until cue for operator action occurs
T_A	=	time available for action is then $T_{\text{sw}} - T_{\text{delay}}$
T_{cog} (or T_D)	=	cognition time consisting of detection, diagnosis, and decision making
T_{exe} (or T_E)	=	execution time including any time needed for communication, travel to action location, collection of tools, etc., as well as time to manipulate components

Note that the adverse consequence used to define the system time window may include reaching a certain core exit temperature, time to start boiling, time to reach a certain level in the vessel, time to SG dryout, time to a certain low level in RWST, time to initiation of core uncover, etc. The accident analysis generally provides system time windows for the various accident scenarios. Typically, the HRA analyst must determine which system time windows are applicable to each scenario being analyzed and the associated specific human actions. It also should be noted that the estimate of the available time does not usually start at the time of the initiator, but when a cue occurs for operator action. For example, the initiator occurs at t_0 and operator intervention must occur in 4 hours. This scenario could be interpreted with an available time of 4 hours. However, if the symptom (or cue) does not occur until 3 hours after the initiator, the available time is actually 1 hour, and not 4 hours.

The time needed to execute the human event (T_E) is the time the operator needs to perform the necessary actions, once the event is diagnosed, to prevent the adverse consequences. This time includes, where applicable, the time needed to direct the operator, time to travel to the location, time to collect any necessary tools, time to dress out (including breathing devices), time to perform the actual action, time for peer check (if required by procedure), etc. This time also considers other factors, such as accessibility and availability of sufficient workspace. These times were generally informed by historical events, reviews of plant procedures used in the event response, layout and locations of equipment needed, and subjective estimates made by plant personnel project analysts. In addition, the steps in the procedure were reviewed to understand the tasks involved and approximate proportion of time required for each step.

The time needed to diagnose the event and identify the needed human response (T_D) is the time the operator needs to acknowledge the occurrence of the event and identify the task needed to be executed by the operator. These times were also generally informed by historical events, reviews of plant procedures used in the event response, layout and locations of indications and alarms, and subjective estimates made by plant personnel project analysts. However, the time estimated for diagnosis was more subjective than that estimated for execution. While "time measurements" could be performed in some manner for execution, this was generally not possible for diagnosis. Moreover, some diagnosis can be self-revealing, where others may require examining multiple symptoms and conditions, as specified in abnormal operating procedures. Therefore, the diagnostic time can vary for different actions. A simple diagnosis is defined by occurrence of a single event (not concurrent with other events) accompanied by unique symptoms such that the operator can decide on the course of action in a straightforward manner. As a rule of thumb, for simple diagnostic actions within the control room, when multiple compelling indications and alarms are available, the required time is judged to take less than 5 minutes.

Using the SPAR-H model, the time required for diagnosis and execution (T_R) is compared to the available time (T_A), then evaluated and classified as either:

- Nominal
- Inadequate time
- Barely adequate time
- Extra time
- Expansive time

In assigning the classification, an assessment of T_R as compared to T_A was performed. The required time ($T_R = T_D + T_E$) may either be greater or less than the T_A . Therefore, the Δt between T_A and T_R may allow extra time to diagnose and execute or may not allow adequate time to diagnose and execute. When evaluating the Δt , a nominal T_D (T_{D-N}) and T_E (T_{E-N}) was computed. This nominal time was based on the average time estimated to diagnose, accounting for variations in conditions that may impact the diagnosis, and the average time estimated to execute all the procedural steps associated with the action (as discussed above). The nominal time does not include extra time to check performance and make corrections. The following general guidelines were used:

- When T_A is comparable to T_R , nominal times are assigned for both T_D and T_E
- When T_A is less than T_R , but at least two-thirds of T_R , a “barely adequate” time is assigned for T_D .¹⁸ If the remaining time ($T_A - 2/3 \cdot T_D$) is equal to the minimum required execution time, a “barely adequate” time is assigned for T_E , otherwise the execution HFE probability is set to 1.0.
- When T_A is less than two-thirds of T_R , an “inadequate” time is assigned and a failure probability of 1.0 is assigned to that HFE.
- When T_A exceeds the nominal T_R by at least a factor of two, then the analyst selects a T_D that is twice the T_{D-N} . This allocation allows maximum credit to be taken for T_D . T_E is then the difference between T_A and T_D , which may allow more time for T_E than estimated for T_{E-N} . For example:
 - Total T_A in loss of inventory is 90 minutes
 - T_{D-N} and T_{E-N} are estimated at 5 and 12 minutes, respectively (total of 17 minutes)
 - There is a Δt of 73 minutes
 - Given the large Δt , it is possible (and preferable) to allocate a T_D of ($2 \times T_{D-N}$ and > 30 minutes) so that T_D can be classified as “expansive”
 - The remaining Δt ($90 - 30 = 60$ minutes) is then allocated for execution
 - T_E can be classified as “extra” since 60 minutes is 5 times the T_{E-N}
- When T_A exceeds T_R within a factor of two, a T_E is selected that equals the T_{E-N} and allocate the remaining Δt for diagnosis. For example:

¹⁸ Even if the available time is less than the nominal time for diagnosis and execution, the actions still may be taken if they are performed with increased haste.

- Total T_A for loss of RHR is 40 minutes
 - T_{D-N} and T_{E-N} are estimated at 5 and 18 minutes, respectively (total of 23 minutes)
 - The T_{E-N} of 18 minutes is allocated for T_E , and T_E is classified as “nominal”
 - The remaining Δt ($40-18=22$ minutes) is assigned for T_D
 - The 22 minutes is at least twice the T_{D-N} , so T_D is classified as “expansive time”
- When a single diagnosis covers several actions, the T_{D-N} is used as the diagnosis time for all the related actions.

The guidelines used to evaluate the PSF for time are summarized below in Table 7-2.

Table 7-2 Time PSF Evaluation

Evaluation	Evaluation Guidelines (Diagnosis and Execution)
Nominal	Assigned when the T_A is slightly greater than the T_R . There is sufficient time available for both diagnosis and execution without having to “hurry” through the process. However, the nominal time does not include any additional time for rechecking, correcting, or recovering from “mistakes.”
Inadequate	Assigned when T_A is less than two-thirds of T_R (based on a nominal performance). There is not enough time to fully and successfully diagnose and execute (complete) the required activities associated with the action. Failure to diagnose and execute is considered certain. The human error probability (HEP) is set to 1.0 and no further PSF is considered for this action.
Barely adequate	Assigned when T_A is at least equal to two-thirds of T_{R-N} . This situation occurs when the Δt between T_A and two-thirds of T_{D-N} is at least the minimum time required for the execution. For this situation, the nominal HEPs for T_D and T_E are increased by a factor of 10.
Extra time	Assigned when the available time is greater than the required time. SPAR-H assigns an “extra time” classification for diagnosis when the available time is up to twice the nominal time and at least 30 minutes, and for execution when the available time is up to five times the nominal time. There is time to check the performance and make corrections. For this evaluation, a factor of 0.1 is applied to the nominal HEP for T_D and T_E .
Expansive time	Assigned when the available time is much greater than the required time. SPAR-H assigns an expansive time classification generally when the available time is at least twice what is needed for diagnosis and greater than 30 minutes, and 50 times what is needed for execution. There is time to both check the performance, ensuring the response is consistent with what is expected, and make corrections with help/verification from other personnel, including SROs and shift technical advisors (STAs). For this evaluation, a factor of 0.1 to 0.01 is applied to the nominal HEP for T_D (based on analyst judgment) and a factor of 0.01 is applied to the nominal HEP for T_E .

7.5.2 Stress and Stressors

The PSF for stress and stressors (i.e., the causes of stress) examines the level of undesirable conditions and circumstances that impede the human in completing an action. The stress PSF is applicable to both diagnosis and execution. Stressors can be physical or mental.

Physical stressors can be the result of work conditions, such as high temperature, high humidity, and use of protective clothing or breathing apparatus; work load; and availability of sufficient resources. Mental stressors can be caused by disruptive and unexpected events, unfamiliarity with the specific situation, working in high hazard conditions, and the consequences of the operator failing to accomplish the task.

Examples of physical stresses that are most applicable to post-initiator operator actions considered for LPSD are:

- Working under harsh accident environmental conditions
 - Fear of electrical shock, explosion, harsh environment, or radiological exposure
- High workload and inadequate staffing¹⁹

Examples of mental stresses that are most applicable to post-initiator operator actions considered for LPSD are:

- Fear of the potential consequence of failing to do the task actions
 - Fear that failure results in imminent possibility of core damage
 - Fear of failure when under scrutiny; for example, when reporting required NRC notifications
- Non-routine and unexpected situation caused by unlikely accidents²⁰
 - Working during SBO conditions which have lasted beyond 30 minutes²¹
 - Response to a LOCA, or large loss of inventory, when the cause is not known
- Unanticipated situation causing confusion
 - Lack of sufficient information and cues for deciding on a course of action
 - Anomalous readings and alarms, or conflicting indications
 - Spurious and unanticipated actuations

It is also important to note that the stress is caused by the collective and synergistic effect of several PSFs. Determining the PSF category for stress within SPAR-H relies significantly on subjective judgment of the analysts, as noted in SPAR-H step-by-step guidance (Whaley, 2011), and given below:

“It is important to note that stress is not independent of other PSFs. Often stress results from limited time, lengthy duration of performing a task, high complexity, poor procedures, poor training, poor work processes, or poor crew dynamics. However, the analyst should make an effort to avoid any “double counting” of specific influencing factors. If time constraints are being accounted for in the Available Time PSF, then the effect of time on the Stress/Stressors PSF should be minimized (note that other details of the context, not explicitly accounted for in other PSFs might still need to be accounted for in the Stress/Stressors PSF).”

It is important to note that the effect of stress can either enhance or degrade performance. In SPAR-H, only high and extreme levels of stress will negatively affect human performance.

The guidelines used to evaluate the PSF for stress are summarized below in Table 7-3.

¹⁹ High workload and inadequate staffing are generally considered for event assessment, and could also impact the time PSF.

²⁰ Operators generally do not show signs of stress for familiar situations, such as routine operations.

²¹ Station blackout imposes a unique demand on operators, since all actions are done locally and coordinated through control room operators with limited emergency lighting and possibly difficulties in access and egress.

Table 7-3 Stress/Stressors PSF Evaluation

PSF Attribute	Evaluation Guidelines	
	Diagnosis	Execution
Extreme	A level of disruptive stress in which performance will deteriorate drastically: Lack of sufficient information and cues for deciding on a course of action For this evaluation, a factor of 5 is applied to the nominal HEP.	A level of disruptive stress in which performance will deteriorate drastically: - Fear that failure results in imminent possibility of core damage - Working under life-threatening conditions For this evaluation, a factor of 5 is applied to the nominal HEP.
High	A level of stress higher than the nominal level: Anomalous readings and alarms, or conflicting indications For this evaluation, a factor of 2 is applied to the nominal HEP.	A level of stress higher than the nominal level: - Fear of failure when under scrutiny; for example, when reporting required NRC notifications - Working during extended SBO conditions - Spurious and unanticipated actuations - High workload and urgency of the action For this evaluation, a factor of 2 is applied to the nominal HEP.
Nominal	A level of stress that is conducive to good performance: Stress levels considered neither high nor extreme	A level of stress that is conducive to good performance: Stress levels considered neither high nor extreme

7.5.3 Complexity

The PSF for complexity examines how difficult the task is to perform in the given context. The factors that can affect the complexity are the scope (the size of the task), the mental effort required for comprehension and decision making, and the confirmatory indications that progress is made or alerting indications that a deviation potentially has occurred. The following attributes were considered for characterizing the diagnosis and execution portions of the complexity PSF.

Diagnosis Attributes

- Parameters to be monitored (pressure, temperature, level, etc.)
- Indicators and alarms (level of redundancy and diversity)
- Direct indication versus interpreted symptoms for diagnostic
- Indirect indications from reading curves or tables
- Unique and obvious cues not common to other conditions
- Communication and coordination between staffing in different locations

Execution Attributes

- Number of steps, tasks, procedures
- Diversity in tasks; different equipment types and operations
- Number of locations
- Availability of confirmatory indications to alert the operator of potential deviations
- Number of staff involved

The guidelines used to evaluate the PSF for complexity are summarized below in Table 7-4.

Table 7-4 Complexity PSF Evaluation

Evaluation	Evaluation Guidelines	
	Diagnosis	Execution
Highly complex	Very difficult to perform – There is much ambiguity in what needs to be diagnosed. • The need to decipher, interpret, and combine several indicators and alarms in order to assess the situation • The occurrence of several unrelated failures which may mask each other, creating a unique difficulty in performing diagnosis. This could happen in an event or in some accident sequence minimal cutset generated by PRAs.²² For this evaluation, a factor of 5 is applied to the nominal HEP.	Very difficult to perform – There is much ambiguity in what needs to be executed. • Several human actions, involving the participation of several individuals at different locations, which are carried out in a particular sequence to achieve several goals dictated by concurrent use of more than one procedure. For this evaluation, a factor of 5 is applied to the nominal HEP.
Moderately complex	Somewhat difficult to perform – There is some ambiguity in what needs to be diagnosed. • There are concurrent diagnoses which involve several variables. • The indications are not direct and the needed parameters have to be calculated or read from pre-generated curves. For this evaluation, a factor of 2 is applied to the nominal HEP.	Somewhat difficult to perform – There is some ambiguity in what needs to be executed. • There are concurrent actions which may involve several controls. • Several human actions that are carried out in a particular sequence, possibly in different locations, to achieve one specific goal using one specific procedure. For this evaluation, a factor of 2 is applied to the nominal HEP.
Nominal	Not difficult to perform – There is little ambiguity in what needs to be diagnosed. • A single or a few variables are involved • Multiple indicators and alarms directly related to the parameters are available	Not difficult to perform – There is little ambiguity in what needs to be executed • A single or a few actions are involved • A single action or a set of interrelated actions performed in an orderly fashion in a single location by one crew member.

²² This is generally considered for event assessments, not for PRA analysis. The postulated accident could involve failures that mask each other. As an example, a rising SG level due to a small SGTR (steam generator tube rupture) may be masked by a SG stuck open relief valve.

Table 7-4 Complexity PSF Evaluation (cont.)

Evaluation	Evaluation Guidelines	
	Diagnosis	Execution
Obvious diagnosis	Diagnosis becomes greatly simplified – It would be difficult for an operator to misdiagnose. • Diagnostics that cannot be confused with something else. A single indication for situation assessment, generally shown, among other things, with a status light. For this evaluation, a factor of 0.1 is applied to the nominal HEP.	There may be actions that are quick and very easy to perform (e.g., flipping a switch on the main control panel), where execution complexity could be evaluated as “obvious execution”; however, SPAR-H does not include this evaluation.

7.5.4 Training

The PSF for training examines the combined effect of formal training (simulator training, classroom training, and drills), experience (encountering the scenario and performing response during operation), and knowledge (deep technical understanding of the scenario progression and response).

The experience and training of the plant personnel depend on the extent of experience of the individual or crew, and whether the staff has been trained on the particular type of accident being evaluated, the amount of time passed since training, the frequency of training, the fidelity of simulator or drills, and the systems involved in the task and scenario. The training PSF considers the applicability of the training to the specific scenario; that is, how helpful the training received will be in responding to the scenario. It is quite plausible that a crew may be rated as “Nominal” or “High” for one set of accident sequences and “Poor” for others. Another consideration is whether the scenario is novel or unique to the staff (i.e., whether the staff has been involved in a similar scenario, in either a training or an operational setting). In many cases, significant accident initiators are anticipated, and the diagnosis and execution are pre-planned.

The following attributes were considered for characterizing the diagnosis and execution portions of the training PSF.

Diagnosis Attributes

- Expected crew knowledge and training
- Routine and frequently drilled
- Pre-planned actions and regular briefings of the crew²³
- Simulator and drill fidelity

²³ In general, at US nuclear power plants, scenarios such as loss of RHR and inventory during LPSD are well exercised on simulators. In addition, during LPSD conditions, briefings are provided at the beginning of each shift that involve discussing the plant configuration at the time, the status of the various safety functions, and important activities to take place during the shift. The briefings also generally include addressing loss of RHR or inventory during the shutdown POS that may be related to planned operations.

Execution Attributes

- Expected crew knowledge and training
- Routine and frequently drilled
- Pre-planned actions and regular briefing of the crew
- Realistic and high-fidelity training (e.g., table-top discussions versus in location training)

The guidelines used to evaluate the PSF for training are summarized below in Table 7-5.

Table 7-5 Training PSF Evaluation

Evaluation	Evaluation Guidelines	
	Diagnosis	Execution
Low	Less than 6 months of experience or training – This level of experience or training does not provide the level of knowledge and deep understanding required to adequately perform the required diagnosis; i.e., does not expose individuals to various abnormal conditions. This generally means: • The crew has limited knowledge, experience, and training on how to diagnose such accidents • The accident scenario is not routine and not frequently drilled • There is generally no pre-planned briefing at the beginning of each shift • Limited simulator exercise or drill; potentially not covering the variety of abnormal conditions that could occur (including low fidelity simulations/drills) For this evaluation, a factor of 10 is applied to the nominal HEP.	Less than 6 months of experience or training – This level of experience or training does not provide the level of knowledge and deep understanding required to adequately perform the required tasks; i.e., does not expose individuals to various abnormal conditions. This generally means: • The crew has limited knowledge, experience, and training with the required actions • The accident scenario is not routine and not frequently drilled • There is generally no pre-planned briefing at the beginning of each shift • Limited simulator exercise or drill; potentially not covering the variety of abnormal conditions that could occur (including low fidelity simulations/drills) For this evaluation, a factor of 3 is applied to the nominal HEP.
Nominal	More than 6 months of experience or training – This level of experience or training provides an adequate amount of formal schooling and instruction to ensure that individuals are proficient in diagnosis of abnormal conditions and have been exposed to a variety of abnormal conditions. This generally means: • Adequate (expected based on industry average) training, experience, and knowledge • There is generally a pre-planned briefing at the beginning of each shift	More than 6 months of experience or training – This level of experience or training provides an adequate amount of formal schooling and instruction to ensure that individuals are proficient in performing the required tasks and have been exposed to abnormal conditions. This generally means: • Adequate (expected based on industry average) training, experience, and knowledge • There is generally a pre-planned briefing at the beginning of each shift
High	Extensive experience (a demonstrated master) – This level of experience or training means extensive knowledge and experience in a wide range of potential scenarios. Formal pre-planning and regular briefing of the operators at the beginning of each shift. For this evaluation, a factor of 0.5 is applied to the nominal HEP.	Extensive experience (a demonstrated master) – This level of experience/training means extensive knowledge and experience in a wide range of potential scenarios. Formal pre-planning and regular briefing of the operators at the beginning of each shift. For this evaluation, a factor of 0.5 is applied to the nominal HEP.

7.5.5 Procedures

The PSF for procedures examines the existence and clarity of the procedures for the tasks required to be performed during a specific accident sequence. For procedures that are written for a generic class of accident sequences, their applicability and compatibility with the specific accident being considered were also examined.

Abnormal operating procedures (AOPs) rather than emergency operating procedures (EOPs), are used during shutdown operation. The diagnostic portion of HRA generally deals with the operator ability to identify the right AOPs to follow. Some AOPs are similar to EOPs and can be considered as symptom-based procedures.²⁴ In these cases, diagnosis is considered to be more straightforward (as opposed to procedures that are more event based), and consequently the failure to diagnose is reduced. In some instances, an AOP is generically written (e.g., for Mode 5 operation) and the AOP is used for all shutdown POSs that fall within that mode of operation.²⁵ In these situations, operator understanding of the POS-specific diagnostic steps and actions is required to compensate for the generic procedures. When the procedure is generic, the assigned PSF attribute is "available, but poor."

Another consideration for evaluating the PSF attributes for plant procedures is the number of procedures involved. Some accident sequences may require more than one procedure to be followed, either sequentially or in parallel. The operators will focus on one procedure initially (event initiator) and then they shift emphasis to the other procedures as needed. For example, for LOOP initiators during LPSD conditions, the operators may initially focus on the AOP for LOOP characterization, but shortly after, they shift their focus to core protection (e.g., the AOP for loss of RHR). This could impact the available time for the operators to take actions in the second procedure and will put additional demand on the first procedure for clear transition to the other procedures. The execution portion for such cases is generally demanding and could also affect both the complexity and work process PSFs, even if the procedure PSF is considered to be nominal.

The following attributes were considered for characterizing the diagnosis and execution portions of the procedure PSF.

Diagnosis Attributes

- Availability of a clear symptom-based procedure(s)
- Compatibility of any generic procedures with the specific accident sequence
- Ambiguity in the procedure(s)
- Incomplete or lack of procedure(s)

Execution Attributes

- Number of procedures involved and clear transitions between them
- Clear unambiguous procedures

²⁴ Symptom-oriented procedures (sometimes called function-oriented procedures) provide the means to maintain critical safety functions. These procedures allow operators to maintain the plant in a safe condition, without the need to diagnose the exact event.

²⁵ Another example is a consequential loss of inventory or loss of RHR during shutdown following an SBO event. The loss of inventory or loss of RHR procedure could be generic and does not account for SBO conditions.

- Compatibility with the accident sequence (generic versus accident specific)
- Incomplete or lack of procedure

The guidelines used to evaluate the PSF for procedures are summarized below in Table 7-6.

Table 7-6 Procedure PSF Evaluation

Evaluation	Evaluation Guidelines	
	Diagnosis	Execution
Not available	The procedure needed for a particular task or tasks in the event is not available. For this evaluation, a factor of 50 is applied to the nominal HEP.	The procedure needed for a particular task or tasks in the event is not available. For this evaluation, a factor of 50 is applied to the nominal HEP.
Incomplete	Information is needed that is not in the procedure or procedure sections; sections or task instructions (or other needed information) are absent. For this evaluation, a factor of 20 is applied to the nominal HEP.	Information is needed that is not in the procedure or procedure sections; sections or task instructions (or other needed information) are absent. For this evaluation, a factor of 20 is applied to the nominal HEP.
Available, but poor	A procedure is available, but it is difficult to use because of factors such as being generic, or has ambiguity, such that it impedes performance: • Generic procedure lacks steps in identifying specific indications and alarms applicable to the specific shutdown POS For this evaluation, a factor of 5 is applied to the nominal HEP.	A procedure is available, but it is difficult to use because of factors such as being generic, or has ambiguity, such that it impedes performance: • Procedure lacks sufficient level of details; e.g., written at a functional action level • Transition from one procedure to others is not clearly identified when multiple procedures are used For this evaluation, a factor of 5 is applied to the nominal HEP.
Nominal	Procedures are available and support diagnostics: • Procedures have sufficient level of detail and steps are clearly written and can be followed. The operators use a single procedure or several procedures one at a time with clear transitions	Procedures are available and support executions: • Procedures have sufficient level of detail and steps are clearly written and can be followed. The operators use a single procedure or several procedures one at a time with clear transitions
Diagnostic, symptom-oriented	Symptom-based procedure that enhances the accident class diagnosis. For this evaluation, a factor of 0.5 is applied to the nominal HEP.	Not applicable

7.5.6 Ergonomic/HMI (Human Machine Interface)

The ergonomic/HMI PSF examines the design of the controls, indications, working conditions, etc., that the operator uses to diagnose and execute the action. The ergonomic/HMI PSFs for operator diagnostics and execution actions are specific to a plant and the accident sequence under consideration. Plant walk-through of the related areas and the layout of the systems, controls, actuations, and diagnostic aids are generally needed for determining the PSFs associated with this element. Other factors to consider are the quality and quantity of information available from instrumentation, and the interaction of the operator/crew with the equipment to carry out tasks. In general, most diagnosis and execution actions carried out in the control room are considered “good” (PSF=0.5) when no plant-specific walk-through has been performed.

The following attributes were considered for characterizing the diagnosis and execution portions of the ergonomic/HMI PSF.

Diagnosis Attributes

- All diagnostic aids are within a modern control room
- Poor local panel design layout²⁶
- Confusing labeling
- Too many non-critical alarms and indications

Execution Attributes

- All actions taken within control room
- Poor local control/actuation panel design layout
- Confusing or unclear labels on control handles
- Appropriate tools for manual actions

The guidelines used to evaluate the PSF for ergonomics/HMI are summarized below in Table 7-7.

²⁶ There have been several observations of improper panel lay out across industry operating experience. A typical example is putting the flow indicator for train A next to the discharge valve for train B.

Table 7-7 Ergonomics/HMI PSF Evaluation

Evaluation	Evaluation Guidelines	
	Diagnosis	Execution
Missing, misleading	The instrumentation provides misleading indication for diagnosis • Inaccurate reading • Highly unreliable For this evaluation, a factor of 50 is applied to the nominal HEP.	The controls required for execution are either misleading or not functioning • Confusing labeling • Poor or misleading layout For this evaluation, a factor of 50 is applied to the nominal HEP.
Poor	Plant design could negatively impact the diagnostic performance • Too many non-critical alarms and indications • Poor lighting and visibility • Poor computer interface and display (e.g., font size, picture size, amount of information, etc.) For this evaluation, a factor of 10 is applied to the nominal HEP.	Plant design could negatively impact the execution performance • Poor local control/actuation panel design layout • Tools are not fully designed to support the local manual actions • Poor lighting and visibility For this evaluation, a factor of 10 is applied to the nominal HEP.
Nominal	The design of the plant supports correct performance; neither misleading nor poor.	The design of the plant supports correct performance; neither misleading nor poor.
Good	The design of the plant positively impacts diagnosis, providing needed information in a manner that lessens the opportunities for error. Most of the control room designs which meet the human factor engineering design requirements fall in this category. For this evaluation, a factor of 0.5 is applied to the nominal HEP.	The design of the plant positively impacts execution of the tasks, providing needed controls in a manner that lessens the opportunities for error. Most of the control room designs which meet the human factor engineering design requirements fall in this category. For this evaluation, a factor of 0.5 is applied to the nominal HEP.

7.5.7 Fitness for Duty

The PSF for fitness for duty examines whether the individual performing the task is physically and mentally fit to perform the task at the time the accident is progressing. Different tasks require different levels of mental and physical fitness for duty. Fitness for duty is generally considered to be met as long as the specific tasks requirements (as relates to fitness for duty) are considered in selecting the crews (i.e., selected crews possess appropriate capabilities to perform the task) and the crew has previously performed similar tasks. Other factors that may affect fitness include fatigue, sickness, drug use (legal or illegal), overconfidence, personal problems, and distractions. An existing licensee's program for fitness for duty may also be examined for scoring this factor. Many factors related to fitness for duty can vary from time to time which makes this PSF more suitable for event analysis, rather than PRA HRA evaluation. For example, the individual could be experiencing fatigue from long duty hours and taking medications (e.g., cold medicine) that leave the individual drowsy. Other factors considered are the individual's overconfidence, family problems, lack of sleep, and personal life stresses.

The guidelines used to evaluate the PSF for fitness for duty are summarized below in Table 7-8.

Table 7-8 Fitness for Duty PSF Evaluation

Evaluation	Evaluation Guidelines	
	Diagnosis	Execution
Unfit	The individual is unable to carry out the required diagnosis, due to physical or mental incapacitation. A failure probability of 1.0 is assigned for the HEP.	The individual is unable to carry out the required actions, due to physical or mental incapacitation. A failure probability of 1.0 is assigned for the HEP.
Degraded fitness	The individual is able to carry out the diagnosis, although mental performance is degraded due to fever or fatigue from long duty hours. For this evaluation, a factor of 5 is applied to the nominal HEP.	The individual is able to carry out the tasks, although mental and physical performance are degraded due to fever or fatigue from long duty hours. For this evaluation, a factor of 5 is applied to the nominal HEP.
Nominal	No known performance degradation is anticipated. Nominal is also used when this PSF is not a performance driver.	No known performance degradation is anticipated. Nominal is also used when this PSF is not a performance driver.
Note: Based on interactions with staff at the site, review of procedures, and compliance with regulatory requirements, the reference plant fitness for duty program was determined to meet the nominal PSF attribute for both diagnosis and execution for all of the HFEs examined, with no notable indications of either unfit or degraded fitness.		

7.5.8 Work Processes

The PSF for work processes examines the effect of plant performance in the areas of inter-organizational interactions, safety culture, work planning, good practices, communication, and management support and policies. These are considered to be important for successfully executing an action. Work planning includes the planning and execution of all the prerequisite tasks needed to perform an action, including all preparatory steps, such as ensuring the ease of access, availability of the needed tools, lighting, and all other work area requirements. There may be aspects of the work process PSF that can influence diagnosis HFEs (e.g., a degraded safety culture), but there were no indications that the reference plant's work processes would systematically influence operator diagnoses. Consequently, this PSF was considered not applicable for diagnosis.

Although some of these aspects may overlap with other PSF attributes, the overlap is considered to be minimal. As an example, the sufficiency of emergency lighting installed in a work area is considered in the ergonomic/HMI PSF evaluation. However, the PSF for work processes considers whether the emergency lights have poor performance or high failure rates. Also considered under this PSF is a review of the plant-specific operating experience, that may indicate repeated problems with coordination, communication, planning, or general schisms among plant staff (operators, maintenance crews, and management).

In some events, such as SBO, the control room operators have to supervise, guide, and monitor several different actions that have to be performed locally in an orderly fashion. In such cases, the control room operators' functions would include initiation, coordination, and verification of the work performed by crew members locally (similar to a command and control center). Aspects such as communication and work planning for local actions become quite important; however, these aspects are not directly addressed under other PSFs. It should be noted that the required work processes in SBO scenarios are now included in some simulation training of control room operators, identifying the work process areas that may need better planning. However, such simulation training has limited applicability for work planning for the crew who perform the local actions.

When an execution is performed locally, the ease of access (e.g., restricted space or obstacles that could inhibit entrance to an area), and the required tools for the crew to perform their tasks (e.g., fully-charged portable fire extinguishers) are assured and maintained using administrative controls. Administrative controls have been shown historically to lead to high failure probabilities, unless they are supplemented by other means, such as routine inspections. Overreliance on administrative controls which are not strictly enforced could increase the HEP values for local actions. Some enhancement in plant operational activities, which are reported by industry as a part of "good practices," could also positively affect the HEP estimates. Utility cognizance of lessons learned by industry, and implementing these lessons, would enhance the PSF for work processes. Repeated problems, as identified by the operational events, could also be indicative of an inadequate corrective action program (CAP).

The PSF for work processes is plant-specific and time-dependent (this PSF could change within the plant life). In most cases, this PSF is assigned as nominal, unless there is an indication that the plant has better or worse than average performance in one or more of the items described above. Generally, the following attributes are considered for good work processes:

- Well-enforced administrative controls
- Administrative controls are supported by other means
- Operating experience indicates better than average performance in the related area
- Effective CAP
- Review of work processes employed at the plant indicates good communication
- Well-maintained safety culture
- Well-understood and supportive policies
- Cohesive crew
- Clear lines of communication

The guidelines used to evaluate the PSF for work processes are summarized below in Table 7-9.

Table 7-9 Work Processes PSF Evaluation

PSF Attribute	Evaluation Guidelines
Poor	Performance is negatively affected by the work processes at the plant - Deficiencies in one or more noted attributes based on actual recent observations or operational data - Deficiencies in two or more of the noted attributes judged by the review of plant activities, interview of personnel, or other documents For this evaluation, a factor of 5 is applied to the nominal HEP.
Nominal	Performance is not significantly affected by work processes at the plant - No deficiencies were found in the noted attributes - Judged to be neither poor nor good performer
Good	Work processes employed at the plant enhance performance and lead to a more successful outcome - Good or above average performance in one or more of the noted attributes based on actual recent observations or operational data - Good or above average performance in two or more of the noted attributes judged by the review of plant activities, interview of personnel, or other documents For this evaluation, a factor of 0.5 is applied to the nominal HEP for execution.

7.5.9 Adjustment Factor

In the SPAR-H method, when there are three or more negative PSFs, the HEP is calculated (or “adjusted”) differently than simply multiplying the nominal HEP by the evaluated PSF values. In these cases, an adjustment factor is applied to preclude an adjusted HEP that exceeds 1.0. To obtain the adjustment factor, a “composite” PSF (PSF_C) is first calculated. This PSF_C is derived by multiplying together all the evaluated PSF values. The HEP is then calculated using the following equation:

$$HEP = \frac{HEP_N * PSF_C}{[HEP_N * (PSF_C - 1) + 1]}$$

7.6 HRA Dependency Characterization

An approach was developed to ensure that all key HFE dependency combinations were identified. The same approach was applied in the L3PRA project Level 1 at-power model (NRC, 2022). The L3PRA Level 1 LPSD model was solved by setting the HEPs for applicable HFEs to 0.9.²⁷ It was determined that a cut set contribution cutoff of 10⁻⁶ per year was sufficient to prevent any potential HFE combinations that could significantly affect the LPSD CDF from being eliminated from the evaluation prior to the review. This resulted in 386 cut sets (with elevated HEPs) with CDFs greater than or equal 10⁻⁶ per year.²⁸ Using a lower cut set CDF screening frequency threshold could add a significant number of additional HFE combinations; however, it is not believed that the additional combinations will result in many new HFE pairs (i.e., most, if

²⁷ HEPs that were currently set to 1.0 (or TRUE) were left at that value, while all others were set to 0.9.

²⁸ The corresponding CDFs for these cutsets will be at least an order of magnitude lower even if complete dependency existed, because the 1st HFE in the combination is independent (i.e., its HEP would not be elevated).

not all, will contain HFE pairs already included in this evaluation). From these cut sets, 101 distinct HFE pairs were identified. The HFE pairs are shown in Table 7-10. The remainder of this section describes the HFE dependency evaluation that was performed for the HFE pairs and a second approach to account for potential non-conservatism related to combined HEP values.

A second approach was developed to supplement the HFE dependency analysis and apply a threshold failure probability to combinations of HFEs. This threshold approach does not explicitly model dependency between HFEs. The threshold approach applies a minimum combined HEP value to combinations of HFEs. The approach is intended to account for potential unrealistically low combined human failure probabilities in cut sets containing multiple HFEs. This is particularly relevant to the LPSD PRA due to the human actions that may be required during low power and shutdown operations while many automatic actuations are disabled. In identifying the cut sets with multiple HFEs, as described above, the combined HEP values were calculated at their nominal values, as specified in Section 7.7. All HFE combinations of two, three, or four HFEs were evaluated. There were no cut sets that contained more than four HFEs. Ninety-five HFE combinations were found that had combined HEP values less than 10^{-6} . A set of substitution rules were applied to replace the HFE combination with a single basic event with probability of 10^{-6} and log normal uncertainty distribution with an error factor of 10. The combined HEP threshold substitutions are shown in Table 7-11.

There are several limitations of the HFE dependency analysis for the Level 1 PRA LPSD model. First, the HFEs that are modeled in the Level 1 PRA LPSD model do not include all possible actions that the operating crew might perform during an accident sequence. The Level 1 PRA HFEs model actions that are relevant to mitigating or preventing core damage. However, there are additional actions that are relevant to the Level 2 PRA and can occur during the time frame of the pre-core damage accident sequences. Such actions can include containment isolation actions and starting the containment cooling units. These actions are not modeled in the Level 1 PRA LPSD model. The dependency analysis does not explicitly consider the Level 2 PRA-related HFEs, even when they can occur in between the time sequence of a Level 1 PRA HFE pair. However, these types of intervening actions are considered in evaluating the Level 1 PRA HFE pair in that the intervening action can influence several of the PSFs (e.g., timing, procedures, and complexity). The explicit modeling of dependency between Level 1 PRA HFEs and Level 2 PRA HFEs is addressed in the shutdown Level 2 PRA model and documentation.

A second limitation is that the dependency approach was not applied for cases involving potential dependency between initiating events and subsequent HFEs. The initiating events are based on a statistical evaluation of operating experience (see Section 8.1), rather than modeled using HRA methods. However, it is acknowledged that many shutdown initiating events could involve human actions and failures that might influence the subsequent HFEs in the event response. This is addressed by including an initial diagnosis HFE that considered the context of the event occurrence, but the dependency model is not applied. The potential for dependency between shutdown initiating events and subsequent HFEs is a source of model uncertainty.

After the dependent HFEs were identified, the level of dependency was determined. NUREG-1792 (NRC, 2005b) presents guidance on determining the level of dependency between HFEs. NUREG-1792 presents a few specific elements that should be evaluated for determining level of dependency, such as:

- The same crew member(s) are responsible for the actions.
- The actions can be considered to take place relatively close in time, such that a common crew mindset may carry over from one action to the next.
- There are similar plant conditions between the actions, and they are being directed by identical (or nearly so) procedure and cue.
- The actions are performed in the same location and performed in similar ways.

More generally, NUREG-1792 suggests evaluating if the actions have similar PSFs and if there is reason to believe that the crew's interpretation of the need or decision for an action might influence the crew's decisions for actions later in the scenario.

The LPSD model considers these elements in determining the level of dependency and used the same approach as used during the L3PRA project at-power analysis (NRC, 2022).²⁹ The approach for determining the dependency level specifically considered the elements outlined in NUREG-1792 of same crew, timing of cues and action, same procedures and cues, and same location. The approach applied for the LPSD HRA also assessed if enough resources are available for the successful completion of both actions. The combination of all these elements starts to address the more general suggestion by NUREG-1792 to evaluate similar PSFs across the actions. In addition, the evaluation of same procedures and cues for the actions is asserted to address an element termed "common cognitive," referring to the crew having a common mindset or belief in approaching the actions. This "common cognitive" is likely to also influence the crew's decisions for actions later in the scenario. Specifically, the following criteria are evaluated through this approach:

- **Same Crew.** If the actions are assumed to be performed by different crews, the HFEs may be considered to be independent. If the difference in time between the cues for each of the HFEs is greater than the length of the shift (e.g., 12 hour), a new crew can be assumed to be responding to the cue of the second HFE.
- **Time.** This element assesses the amount of time that is estimated to have elapsed between the cues for each of the HFEs. The two HFEs are either occurring simultaneously or separated in time by one of the following intervals:

²⁹ Note that the LPSD PRA dependency evaluation differs in two ways from the method employed for the L3PRA at-power analysis. First, in the at-power analysis, no credit is currently given for an intervening success. However, for the LPSD dependency evaluation, intervening success was assessed on a case-by-case basis. This assessment is consistent with the approach used in the EPRI HRA Calculator (as described in (NRC, 2006)) and states that an intervening success would break dependency between a pair of HFEs. Second, given the lack of guidance on evaluating stress between HFEs using the dependency tree from the EPRI HRA Calculator, the dependency evaluation of HFE pairs in the at-power model made the conservative assumption to use the higher branch for the stress node in the decision tree (i.e., high/moderate stress), without evaluating stress explicitly. However, based on further evaluation, the decision was made for the LPSD PRA (and all other L3PRA project models currently being developed, or that will be developed) to use the lower branch for the stress node (i.e., low stress). Previously completed L3PRA project models will not be revised to effect this change due to project resource limitations and the current expectation that this change would not significantly impact the results.

$0 < \text{Time} \leq 15 \text{ minutes}$
 $15 < \text{Time} \leq 30 \text{ minutes}$
 $30 < \text{Time} \leq 60 \text{ minutes}$
 $\text{Time} > 60 \text{ minutes}$

- **Common Cognitive / Same Cues and Procedures.** If the crew can be assumed to be in a common cognitive mindset while responding to both HFEs, complete dependency is assigned. This element is assessed by evaluating if the cue and/or procedure steps being used are identical for the HFEs being evaluated. NUREG-1921 defines a cue as “a change in condition or signal that triggers the need for an action” (pg. A-2); therefore, a cue can be thought of as either an indication, procedure step, or alarm that alerts the crew member to the need for a response.
- **Adequate Resources.** This element assesses whether an adequate number of staff is available to support the required actions. For the dependency determination, this assessment is only relevant if the actions are required to be completed during the same time. If staffing is found to be insufficient, complete dependency is assumed.
- **Same Location.** The location refers to the room or general area in which the actions will be executed. If the actions are executed in the same location, a higher level of dependency is typically assessed.

Considering these criteria, a level of dependency (i.e., complete, high, moderate, low, or zero) was assigned to each HFE pair using the dependency decision tree shown in Figure 7-1.³⁰

Following the assignment of the dependency level, the HEPs for the dependent HFEs were calculated by applying the following dependency formulas given in THERP (NRC, 1983):

Dependence Level	Equation
Zero	HEP
Low	$(1 + 19 \times \text{HEP}) / 20$
Moderate	$(1 + 6 \times \text{HEP}) / 7$
High	$(1 + \text{HEP}) / 2$
Complete	1.0

If more than two HFEs are considered to be dependent in a sequence, the dependency calculation for each successive HFE is calculated based only on the immediately preceding HFE. In other words, if three HFEs termed HFE₁, HFE₂, and HFE₃ were found to be dependent, the calculation of the dependent HEP for HFE₂ would be based on the dependency level determined between HFE₁ and HFE₂. The calculation for the dependency HEP for HFE₃ would be based on the dependency level determined between HFE₂ and HFE₃. This calculation is based on guidance given in THERP.

³⁰ This figure differs in some areas from the dependency evaluation tree included in the L3PRA at-power report, for the reasons stated previously (i.e., credit in the LPSD PRA for intervening success and the assumption of “low stress” for the dependency evaluation, versus the assumption of “high stress” in the at-power analysis).

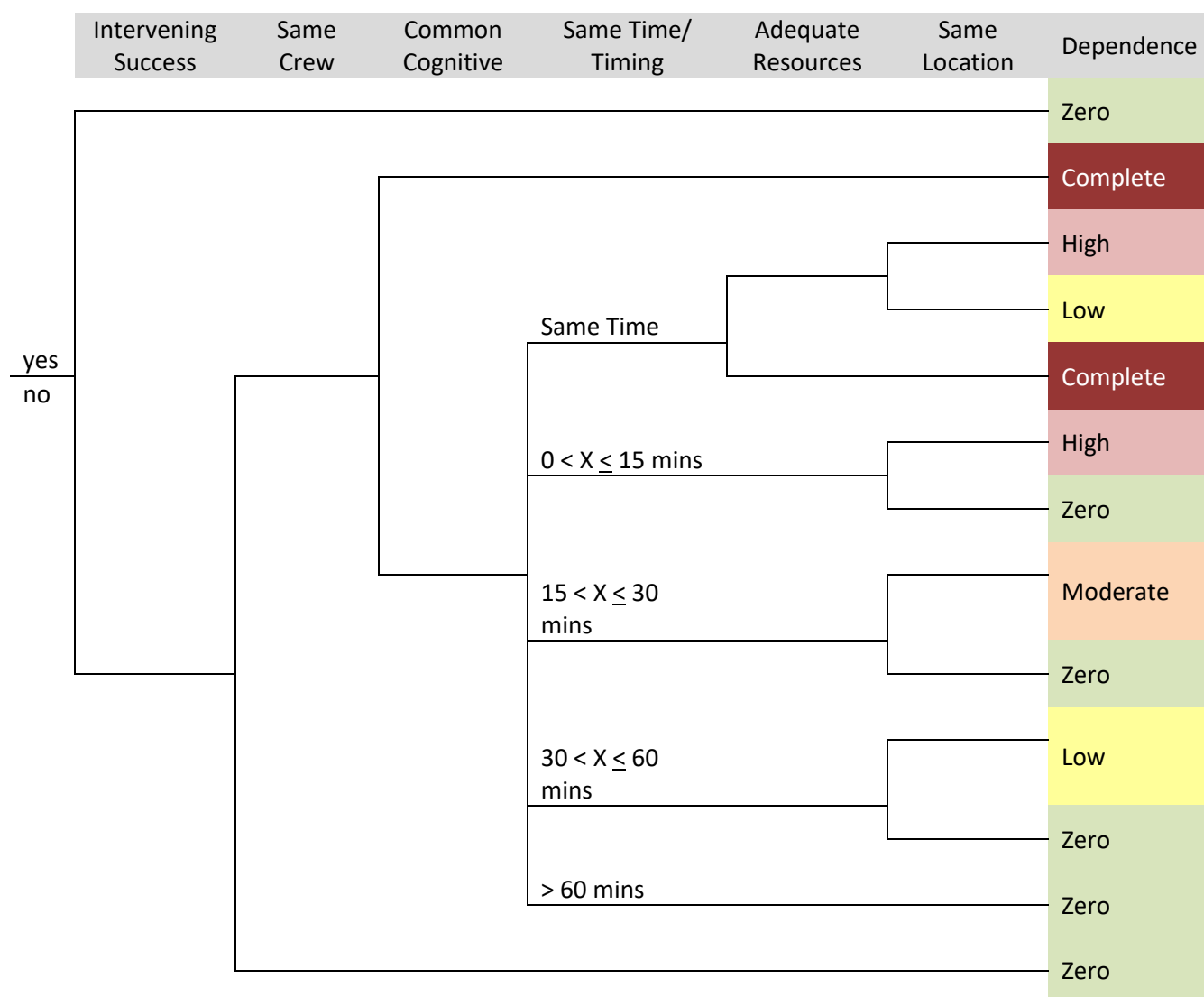


Figure 7-1 L3PRA Modified Dependency Decision Tree

Table 7-10 lists those HFE pairs evaluated for dependency. For all HFE pairs that were assessed to have non-zero dependency, a post-processing rule was developed to substitute a dependent HFE into the cut set results in place of HFE₂. The substituted dependent HFE is shown in Table 7-10. The substituted dependent event is listed as “N/A” for HFE pairs where the dependency level is zero. The HEP values and error factors for the dependent events are discussed in Section 7.7.

Table 7-10 LPSD HFE Dependency Pairs and Evaluation

#	HFE ₁	HFE ₂	Dependency Level	Substituted Dependent HEP
1	1-ACP-XHE-XL-ALTAC-6H	1-HPI-XHE-XM-FFEED-POS8E	Zero	N/A
2	1-ACP-XHE-XL-ALTAC-6H	1-LPI-XHE-XM-GFEED-LOI8E	Zero	N/A
3	1-AFW-XHE-FO-SGCOOL	1-HPI-XHE-XM-FFEED-6H-5A	Zero	N/A
4	1-AFW-XHE-FO-SGCOOL	1-HPI-XHE-XM-FFEED-LOI5A	Zero	N/A
5	1-AFW-XHE-XM-SGC-POS3	1-RHR-XHE-XL-TRB-POS3	Zero	N/A
6	1-HPI-XHE-XM-FFEED-LOI	1-LPI-XHE-XM-GFEED-LOI	Low	1-LPI-XHE-XM-GFEED-LD
7	1-HPI-XHE-XM-FFEED-LOI	1-LPI-XHE-XM-GFEED-LOI5A	Low	1-LPI-XHE-XM-GFEED-LD
8	1-HPI-XHE-XM-FFEED-LOI	1-LPI-XHE-XM-GFEED-LOI-6	Low	1-LPI-XHE-XM-GFEED-LD
9	1-HPI-XHE-XM-FFEED-LOI	1-LOI-XHE-XM-ISO2-LT	Zero	N/A
10	1-HPI-XHE-XM-FFEED-LOI5A	1-LPI-XHE-XM-GFEED-LOI5A	Low	1-LPI-XHE-XM-GFEED-LD
11	1-HPI-XHE-XM-FFEED-LOI-6	1-LPI-XHE-XM-GFEED-LOI-6	Low	1-LPI-XHE-XM-GFEED-LD
12	1-HPI-XHE-XM-FFEED-LOI-7	1-LPI-XHE-XM-GFEED-LOI-7	Low	1-LPI-XHE-XM-GFEED-P7-LD
13	1-HPI-XHE-XM-FFEED-MLDRL	1-LPI-XHE-XM-GFEED-MLDRL	Low	1-LPI-XHE-XM-GFEED-LD
14	1-HPI-XHE-XM-FFEED-POS8E	1-LPI-XHE-XM-GFEED-LOI8E	Low	1-LPI-XHE-XM-GFEED-LD
15	1-HPI-XHE-XM-FFEED-POS8E	1-LOI-XHE-XM-ISO2-LT	Zero	N/A
16	1-LOI-XHE-XD-POS5A	1-RHR-XHE-XM-TRIP-M5LOI	Moderate	1-RHR-XHE-XM-TRIP-MD
17	1-LOI-XHE-XD-POS5A	1-LOI-XHE-XM-ISO2-LT	Zero	N/A
18	1-LOI-XHE-XD-POS5A	1-HPI-XHE-XM-FFEED-LOI	Zero	N/A
19	1-LOI-XHE-XD-POS5A	1-RHR-XHE-XL-TRN-AIR-LOI	Zero	N/A
20	1-LOI-XHE-XD-POS5A	1-AFW-XHE-FO-SGCOOL	Zero	N/A
21	1-LOI-XHE-XD-POS8E	1-RHR-XHE-XM-TRIP-M5LOI	Low	1-RHR-XHE-XM-TRIP-LD
22	1-LOI-XHE-XD-POS8E	1-LOI-XHE-XM-ISO2-LT	Zero	N/A
23	1-LOI-XHE-XD-POS8E	1-RHR-XHE-XL-TRN-AIR-LOI	Zero	N/A
24	1-LOI-XHE-XD-POS8E	1-HPI-XHE-XM-FFEED-POS8E	Zero	N/A
25	1-LOI-XHE-XM-ISO-POS8E	1-RHR-XHE-XM-TRIP-M5LOI	Low	1-RHR-XHE-XM-TRIP-LD
26	1-LOI-XHE-XM-ISO-POS8E	1-LOI-XHE-XM-ISO2-LT	Zero	N/A
27	1-LOI-XHE-XM-ISO-POS8E	1-RHR-XHE-XL-TRN-AIR-LOI	Zero	N/A

Table 7-10 LPSD HFE Dependency Pairs and Evaluation (cont.)

#	HFE ₁	HFE ₂	Dependency Level	Substituted Dependent HEP
28	1-LOI-XHE-XM-ISO-POS8E	1-HPI-XHE-XM-FFEED-POS8E	Zero	N/A
29	1-LPI-XHE-XM-GFEED-LOI-7	1-HPI-XHE-XM-FFEED-LOI	Zero	N/A
30	1-OA-ALIGNPW-G-H	1-LPI-XHE-XM-GFEED-LOOP	Zero	N/A
31	1-OA-NSCWCT-MV-H	1-HPI-XHE-XM-FFEED-POS8E	Zero	N/A
32	1-RHR-XHE-XD-3H	1-DEP-XHE-XM-BLEED	Zero	N/A
33	1-RHR-XHE-XD-3H	1-HPI-XHE-XM-FFEED-6H	Zero	N/A
34	1-RHR-XHE-XD-3H-5A	1-AFW-XHE-FO-SGCOOL	Zero	N/A
35	1-RHR-XHE-XD-ML	1-HPI-XHE-XM-FFEED-MLDRL	Zero	N/A
36	1-RHR-XHE-XD-ML	1-HPI-XHE-XM-FFEED-6H	Zero	N/A
37	1-RHR-XHE-XD-POS3	1-AFW-XHE-XM-SGC-POS3	Zero	N/A
38	1-RHR-XHE-XD-POS7	1-HPI-XHE-XM-FFEED-6H	Zero	N/A
39	1-RHR-XHE-XD-POS7	1-HPI-XHE-XM-FFEED-POS8E	Zero	N/A
40	1-RHR-XHE-XD-POS8E	1-HPI-XHE-XM-FFEED-POS8E	Zero	N/A
41	1-RHR-XHE-XD-PRF	1-AFW-XHE-XM-SGC-PRF	Zero	N/A
42	1-RHR-XHE-XL-STBY-POS7	1-HPI-XHE-XM-FFEED-6H	Zero	N/A
43	1-RHR-XHE-XL-STBY-POS7	1-RHR-XHE-XL-TRB-POS7	Zero	N/A
44	1-RHR-XHE-XL-TRA-POS7	1-HPI-XHE-XM-FFEED-LOI-7	Zero	N/A
45	1-RHR-XHE-XL-TRA-POS8E	1-HPI-XHE-XM-FFEED-POS8E	Zero	N/A
46	1-RHR-XHE-XL-TRA-POS8E	1-LPI-XHE-XM-GFEED-LOI8E	Zero	N/A
47	1-RHR-XHE-XL-TRA-PRF	1-AFW-XHE-XM-SGC-PRF	Zero	N/A
48	1-RHR-XHE-XL-TRB-POS3	1-RHR-XHE-XL-TRA-POS3	Zero	N/A
49	1-RHR-XHE-XL-TRB-POS7	1-HPI-XHE-XM-FFEED-6H	Zero	N/A
50	1-RHR-XHE-XL-TRB-POS7	1-HPI-XHE-XM-FFEED-POS8E	Zero	N/A
51	1-RHR-XHE-XL-TRB-POS8E	1-HPI-XHE-XM-FFEED-POS8E	Zero	N/A
52	1-RHR-XHE-XL-TRB-PRF	1-RHR-XHE-XL-TRA-PRF	Zero	N/A
53	1-RHR-XHE-XL-TRB-PRF	1-AFW-XHE-XM-SGC-PRF	Zero	N/A
54	1-RHR-XHE-XL-TRN-3H	1-DEP-XHE-XM-BLEED	Zero	N/A
55	1-RHR-XHE-XL-TRN-3H	1-HPI-XHE-XM-FFEED-6H	Zero	N/A
56	1-RHR-XHE-XL-TRN-3H	1-AFW-XHE-FO-SGCOOL	Zero	N/A

Table 7-10 LPSD HFE Dependency Pairs and Evaluation (cont.)

#	HFE ₁	HFE ₂	Dependency Level	Substituted Dependent HEP
57	1-RHR-XHE-XL-TRN-M5LOI	1-LPI-XHE-XM-GFEED-LOI-6	Low	1-LPI-XHE-XM-GFEED-LD
58	1-RHR-XHE-XL-TRN-M5LOI	1-LPI-XHE-XM-GFEED-LOI	Zero	N/A
59	1-RHR-XHE-XL-TRN-M5LOI	1-HPI-XHE-XM-FFEED-LOI	Zero	N/A
60	1-RHR-XHE-XL-TRN-M5LOI	1-HPI-XHE-XM-FFEED-LOI-6	Zero	N/A
61	1-RHR-XHE-XL-TRN-MLDRL	1-LPI-XHE-XM-GFEED-MLDRL	Moderate	1-LPI-XHE-XM-GFEED-MD
62	1-RHR-XHE-XL-TRN-MLDRL	1-HPI-XHE-XM-FFEED-MLDRL	Zero	N/A
63	1-RHR-XHE-XL-TRN-MLDRL	1-RHR-XHE-XL-TRN-AIR-DRL	Zero	N/A
64	1-RHR-XHE-XL-TRN-POS4	1-AFW-XHE-FO-SGCOOL	Zero	N/A
65	1-RHR-XHE-XL-TRN-POS6	1-HPI-XHE-XM-FFEED-MLDRL	Zero	N/A
66	1-RHR-XHE-XL-TRN-STBY-2H	1-RHR-XHE-XL-TRN-3H	High	1-RHR-XHE-XL-TRN-HD
67	1-RHR-XHE-XL-TRN-STBY-2H	1-HPI-XHE-XM-FFEED-6H	Zero	N/A
68	1-RHR-XHE-XM-LOI-POS6	1-RHR-XHE-XM-TRIP-M5LOI	High	1-RHR-XHE-XM-TRIP-HD
69	1-RHR-XHE-XM-LOI-POS6	1-LOI-XHE-XM-ISO2-LT	Zero	N/A
70	1-RHR-XHE-XM-LOI-POS6	1-HPI-XHE-XM-FFEED-LOI	Zero	N/A
71	1-RHR-XHE-XM-LOI-POS6	1-RHR-XHE-XL-TRN-AIR-LOI	Zero	N/A
72	1-RHR-XHE-XM-LOI-POS7	1-LPI-XHE-XM-GFEED-LOI-7	Moderate	1-LPI-XHE-XM-GFEED-P7-MD
73	1-RHR-XHE-XM-LOI-POS7	1-RHR-XHE-XM-TRIP-M5LOI	Low	1-RHR-XHE-XM-TRIP-LD
74	1-RHR-XHE-XM-LOI-POS7	1-LOI-XHE-XM-ISO2-LT	Zero	N/A
75	1-RHR-XHE-XM-LOI-POS7	1-HPI-XHE-XM-FFEED-LOI	Zero	N/A
76	1-RHR-XHE-XM-LOI-POS7	1-RHR-XHE-XL-TRN-AIR-LOI	Zero	N/A
77	1-RHR-XHE-XM-M5LOI-I	1-RHR-XHE-XM-TRIP-M5LOI	Low	1-RHR-XHE-XM-TRIP-LD
78	1-RHR-XHE-XM-M5LOI-I	1-LOI-XHE-XM-ISO2-LT	Zero	N/A
79	1-RHR-XHE-XM-M5LOI-I	1-HPI-XHE-XM-FFEED-LOI	Zero	N/A
80	1-RHR-XHE-XM-M5LOI-I	1-RHR-XHE-XL-TRN-AIR-LOI	Zero	N/A
81	1-RHR-XHE-XM-MLDRL-I	1-HPI-XHE-XM-FFEED-MLDRL	Zero	N/A

Table 7-10 LPSD HFE Dependency Pairs and Evaluation (cont.)

#	HFE ₁	HFE ₂	Dependency Level	Substituted Dependent HEP
82	1-RHR-XHE-XM-MLDRL-I	1-RHR-XHE-XM-TRIP-MLDRL	Moderate	1-RHR-XHE-XM-TRIP-MD
83	1-RHR-XHE-XM-MLDRL-I	1-RHR-XHE-XL-TRN-AIR-DRL	Zero	N/A
84	1-RHR-XHE-XL-TRN-POS10	1-HPI-XHE-XM-FFEED-MLDRL	Zero	N/A
85	1-RHR-XHE-XL-TRN-ML-P10	1-LPI-XHE-XM-GFEED-MLDRL	Low	1-LPI-XHE-XM-GFEED-LD
86	1-RHR-XHE-XL-TRN-ML-P10	1-HPI-XHE-XM-FFEED-MLDRL	Zero	N/A
87	1-RHR-XHE-XL-TRN-ML-P10	1-RHR-XHE-XL-TRN-AIR-DRL	Zero	N/A
88	1-RHR-XHE-XM-POS10-I	1-HPI-XHE-XM-FFEED-MLDRL	Zero	N/A
89	1-RHR-XHE-XM-POS10-I	1-RHR-XHE-XM-TRIP-MLDRL	Moderate	1-RHR-XHE-XM-TRIP-MD
90	1-RHR-XHE-XM-POS10-I	1-RHR-XHE-XL-TRN-AIR-DRL	Zero	N/A
91	1-RHR-XHE-XM-LOI-POS5B	1-HPI-XHE-XM-FFEED-LOI	Zero	N/A
92	1-RHR-XHE-XM-LOI-POS5B	1-RHR-XHE-XM-TRIP-M5LOI	High	1-RHR-XHE-XM-TRIP-HD
93	1-RHR-XHE-XM-LOI-POS5B	1-RHR-XHE-XL-TRN-AIR-LOI	Zero	N/A
94	1-RHR-XHE-XM-LOI-POS5B	1-LOI-XHE-ISO2-LT	Zero	N/A
95	1-RHR-XHE-XM-LOISP-POS6	1-HPI-XHE-XM-FFEED-LOI	Zero	N/A
96	1-RHR-XHE-XM-LOISP-POS6	1-RHR-XHE-XM-TRIP-M5LOI	High	1-RHR-XHE-XM-TRIP-HD
97	1-RHR-XHE-XM-LOISP-POS6	1-RHR-XHE-XL-TRN-AIR-LOI	Zero	N/A
98	1-RHR-XHE-XM-LOISP-POS6	1-LOI-XHE-ISO2-LT	Zero	N/A
99	1-LPI-XHE-XM-GFEED-NSCW	1-OA-OSW-----H	Zero	N/A
100	1-LPI-XHE-XM-GFD-ML-NSCW	1-OA-OSW-----H	High	1-OA-OSW-----H-HD
101	1-LPI-XHE-XM-GFD-PRF-NSCW	1-OA-OSW-----H	Zero	N/A

Table 7-11 HFE Cut Set Combination Substitutions with Minimum Combined HEP Value

1 st HFE	2 nd HFE	3 rd HFE	4 th HFE	Combined HEP	Basic Event Substitution
1-AFW-XHE-FO-SGCOOL	1-HPI-XHE-XM-FFEED-LOI5A	1-LPI-XHE-XM-GFEED-LOI5A	1-RHR-XHE-XL-TRN-M5LOI	1.1E-10	1-SD-HEP-JOINT-01
1-ACP-XHE-XL-ALTAC-6H	1-HPI-XHE-XM-FFEED-POS8E	1-LPI-XHE-XM-GFEED-LOI8E		4.5E-08	1-SD-HEP-JOINT-02
1-AFW-XHE-FO-SGCOOL	1-HPI-XHE-XM-FFEED-6H-5A	1-RHR-XHE-XD-3H-5A		1.6E-12	1-SD-HEP-JOINT-03
1-AFW-XHE-FO-SGCOOL	1-HPI-XHE-XM-FFEED-6H-5A	1-RHR-XHE-XL-TRN-3H		1.6E-09	1-SD-HEP-JOINT-04
1-AFW-XHE-FO-SGCOOL	1-HPI-XHE-XM-FFEED-6H-5A	1-RHR-XHE-XL-TRN-POS4		3.1E-11	1-SD-HEP-JOINT-05
1-AFW-XHE-FO-SGCOOL	1-LOI-XHE-XD-POS5A	1-RHR-XHE-XM-TRIP-M5LOI		4.6E-07	1-SD-HEP-JOINT-06
1-AFW-XHE-FO-SGCOOL	1-LOI-XHE-XD-POS5A	1-RHR-XHE-XL-TRN-AIR-LOI		6.3E-09	1-SD-HEP-JOINT-07
1-AFW-XHE-XM-SGC-POS3	1-RHR-XHE-XL-TRA-POS3	1-RHR-XHE-XL-TRB-POS3		3.1E-13	1-SD-HEP-JOINT-08
1-AFW-XHE-XM-SGC-PRF	1-RHR-XHE-XL-TRA-PRF	1-RHR-XHE-XL-TRB-PRF		3.1E-13	1-SD-HEP-JOINT-09
1-HPI-XHE-XM-FFEED-6H	1-RHR-XHE-XL-STBY-POS7	1-RHR-XHE-XL-TRB-POS7		2.1E-11	1-SD-HEP-JOINT-10
1-HPI-XHE-XM-FFEED-6H	1-RHR-XHE-XL-TRN-3H	1-RHR-XHE-XL-TRN-STBY-2H		2.1E-07	1-SD-HEP-JOINT-11
1-HPI-XHE-XM-FFEED-LOI	1-LOI-XHE-XM-ISO2-LT	1-LOI-XHE-XD-POS5A		2.4E-06	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-LOI	1-LOI-XHE-XM-ISO2-LT	1-RHR-XHE-XM-LOI-POS6		9.4E-05	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-LOI	1-LOI-XHE-XM-ISO2-LT	1-RHR-XHE-XM-LOI-POS7		4.8E-07	1-SD-HEP-JOINT-14
1-HPI-XHE-XM-FFEED-LOI	1-LOI-XHE-XM-ISO2-LT	1-RHR-XHE-XM-M5LOI-I		1.0E-06	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-LOI	1-LOI-XHE-XD-POS5A	1-LPI-XHE-XM-GFEED-LOI5A		4.4E-05	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-LOI	1-LOI-XHE-XD-POS5A	1-RHR-XHE-XM-TRIP-M5LOI		1.4E-05	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-LOI	1-LOI-XHE-XD-POS5A	1-RHR-XHE-XL-TRN-AIR-LOI		1.9E-07	1-SD-HEP-JOINT-18
1-HPI-XHE-XM-FFEED-LOI	1-LPI-XHE-XM-GFEED-LOI-6	1-RHR-XHE-XM-LOI-POS6		1.7E-03	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-LOI	1-LPI-XHE-XM-GFEED-LOI-7	1-RHR-XHE-XM-LOI-POS7		3.1E-06	Combined HEP value exceeds 1E-06 (note 1)

Table 7-11 HFE Cut Set Combination Substitutions with Minimum Combined HEP Value (cont.)

1 st HFE	2 nd HFE	3 rd HFE	4 th HFE	Combined HEP	Basic Event Substitution
1-HPI-XHE-XM-FFEED-LOI	1-LPI-XHE-XM-GFEED-LOI	1-RHR-XHE-XL-TRN-M5LOI		4.4E-07	1-SD-HEP-JOINT-21
1-HPI-XHE-XM-FFEED-LOI	1-LPI-XHE-XM-GFEED-LOI	1-RHR-XHE-XM-M5LOI-I		1.9E-05	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-LOI	1-RHR-XHE-XM-LOI-POS6	1-RHR-XHE-XM-TRIP-M5LOI		1.9E-03	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-LOI	1-RHR-XHE-XM-LOI-POS7	1-RHR-XHE-XM-TRIP-M5LOI		1.0E-06	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-LOI	1-RHR-XHE-XM-M5LOI-I	1-RHR-XHE-XM-TRIP-M5LOI		2.1E-06	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-LOI	1-RHR-XHE-XL-TRN-AIR-LOI	1-RHR-XHE-XM-LOI-POS6		7.7E-06	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-LOI	1-RHR-XHE-XL-TRN-AIR-LOI	1-RHR-XHE-XM-LOI-POS7		3.9E-08	1-SD-HEP-JOINT-27
1-HPI-XHE-XM-FFEED-LOI	1-RHR-XHE-XL-TRN-AIR-LOI	1-RHR-XHE-XM-M5LOI-I		2.9E-08	1-SD-HEP-JOINT-28
1-HPI-XHE-XM-FFEED-LOI-6	1-LPI-XHE-XM-GFEED-LOI-6	1-RHR-XHE-XL-TRN-M5LOI		4.4E-07	1-SD-HEP-JOINT-29
1-HPI-XHE-XM-FFEED-LOI-7	1-LPI-XHE-XM-GFEED-LOI-7	1-RHR-XHE-XL-TRA-POS7		1.3E-07	1-SD-HEP-JOINT-30
1-HPI-XHE-XM-FFEED-MLDRL	1-LPI-XHE-XM-GFEED-MLDRL	1-RHR-XHE-XD-ML		4.5E-07	1-SD-HEP-JOINT-31
1-HPI-XHE-XM-FFEED-MLDRL	1-LPI-XHE-XM-GFEED-MLDRL	1-RHR-XHE-XL-TRN-MLDRL		5.0E-05	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-MLDRL	1-LPI-XHE-XM-GFEED-MLDRL	1-RHR-XHE-XL-TRN-POS6		4.5E-05	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-MLDRL	1-LPI-XHE-XM-GFEED-MLDRL	1-RHR-XHE-XM-MLDRL-I		7.2E-05	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-MLDRL	1-RHR-XHE-XM-MLDRL-I	1-RHR-XHE-XM-TRIP-MLDRL		2.3E-05	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-MLDRL	1-RHR-XHE-XL-TRN-AIR-DRL	1-RHR-XHE-XM-MLDRL-I		3.2E-08	1-SD-HEP-JOINT-36
1-HPI-XHE-XM-FFEED-POS8E	1-LOI-XHE-XM-ISO2-LT	1-LOI-XHE-XD-POS8E		6.1E-11	1-SD-HEP-JOINT-37
1-HPI-XHE-XM-FFEED-POS8E	1-LOI-XHE-XM-ISO2-LT	1-LOI-XHE-XM-ISO-POS8E		6.1E-09	1-SD-HEP-JOINT-38
1-HPI-XHE-XM-FFEED-POS8E	1-LOI-XHE-XD-POS8E	1-LPI-XHE-XM-GFEED-LOI8E		1.1E-09	1-SD-HEP-JOINT-39

Table 7-11 HFE Cut Set Combination Substitutions with Minimum Combined HEP Value (cont.)

1 st HFE	2 nd HFE	3 rd HFE	4 th HFE	Combined HEP	Basic Event Substitution
1-HPI-XHE-XM-FFEED-POS8E	1-LOI-XHE-XD-POS8E	1-RHR-XHE-XM-TRIP-M5LOI		1.3E-10	1-SD-HEP-JOINT-40
1-HPI-XHE-XM-FFEED-POS8E	1-LOI-XHE-XD-POS8E	1-RHR-XHE-XL-TRN-AIR-LOI		5.0E-12	1-SD-HEP-JOINT-41
1-HPI-XHE-XM-FFEED-POS8E	1-LOI-XHE-XM-ISO-POS8E	1-LPI-XHE-XM-GFEED-LOI8E		1.1E-07	1-SD-HEP-JOINT-42
1-HPI-XHE-XM-FFEED-POS8E	1-LOI-XHE-XM-ISO-POS8E	1-RHR-XHE-XM-TRIP-M5LOI		1.3E-08	1-SD-HEP-JOINT-43
1-HPI-XHE-XM-FFEED-POS8E	1-LOI-XHE-XM-ISO-POS8E	1-RHR-XHE-XL-TRN-AIR-LOI		5.0E-10	1-SD-HEP-JOINT-44
1-HPI-XHE-XM-FFEED-POS8E	1-LPI-XHE-XM-GFEED-LOI8E	1-OA-NSCWCT-MV-H		2.5E-07	1-SD-HEP-JOINT-45
1-HPI-XHE-XM-FFEED-POS8E	1-LPI-XHE-XM-GFEED-LOI8E	1-RHR-XHE-XL-TRA-POS8E		1.1E-08	1-SD-HEP-JOINT-46
1-LPI-XHE-XM-GFEED-MLDRL	1-RHR-XHE-XL-TRN-AIR-DRL	1-RHR-XHE-XL-TRN-MLDRL		1.3E-06	Combined HEP value exceeds 1E-06 (note 1)
1-ACP-XHE-XL-ALTAC-6H	1-HPI-XHE-XM-FFEED-POS8E			1.0E-07	1-SD-HEP-JOINT-48
1-ACP-XHE-XL-ALTAC-6H	1-LPI-XHE-XM-GFEED-LOI8E			8.5E-04	Combined HEP value exceeds 1E-06 (note 1)
1-AFW-XHE-FO-SGCOOL	1-LOI-XHE-XD-POS5A			3.1E-06	Combined HEP value exceeds 1E-06 (note 1)
1-AFW-XHE-FO-SGCOOL	1-RHR-XHE-XD-3H-5A			1.6E-10	1-SD-HEP-JOINT-51
1-AFW-XHE-FO-SGCOOL	1-RHR-XHE-XL-TRN-M5LOI			3.1E-08	1-SD-HEP-JOINT-52
1-AFW-XHE-FO-SGCOOL	1-RHR-XHE-XL-TRN-3H			1.6E-07	1-SD-HEP-JOINT-53
1-AFW-XHE-FO-SGCOOL	1-RHR-XHE-XL-TRN-POS4			3.1E-09	1-SD-HEP-JOINT-54
1-AFW-XHE-XM-SGC-POS3	1-RHR-XHE-XD-POS3			3.1E-10	1-SD-HEP-JOINT-55
1-AFW-XHE-XM-SGC-POS3	1-RHR-XHE-XL-TRN-POS4			6.3E-10	1-SD-HEP-JOINT-56
1-AFW-XHE-XM-SGC-PRF	1-RHR-XHE-XD-PRF			3.1E-10	1-SD-HEP-JOINT-57
1-AFW-XHE-XM-SGC-PRF	1-RHR-XHE-XL-TRA-PRF			1.3E-08	1-SD-HEP-JOINT-58
1-AFW-XHE-XM-SGC-PRF	1-RHR-XHE-XL-TRB-PRF			6.3E-10	1-SD-HEP-JOINT-59
1-DEP-XHE-XM-BLEED	1-RHR-XHE-XD-3H			7.5E-07	1-SD-HEP-JOINT-60

Table 7-11 HFE Cut Set Combination Substitutions with Minimum Combined HEP Value (cont.)

1 st HFE	2 nd HFE	3 rd HFE	4 th HFE	Combined HEP	Basic Event Substitution
1-DEP-XHE-XM-BLEED	1-RHR-XHE-XL-TRN-3H			7.5E-06	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-6H	1-RHR-XHE-XD-3H			2.1E-07	1-SD-HEP-JOINT-62
1-HPI-XHE-XM-FFEED-6H	1-RHR-XHE-XD-ML			2.1E-07	1-SD-HEP-JOINT-63
1-HPI-XHE-XM-FFEED-6H	1-RHR-XHE-XD-POS7			4.3E-09	1-SD-HEP-JOINT-64
1-HPI-XHE-XM-FFEED-6H	1-RHR-XHE-XL-STBY-POS7			8.5E-07	1-SD-HEP-JOINT-65
1-HPI-XHE-XM-FFEED-6H	1-RHR-XHE-XL-TRN-3H			2.1E-06	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-6H	1-RHR-XHE-XL-TRN-STBY-2H			4.3E-07	1-SD-HEP-JOINT-67
1-HPI-XHE-XM-FFEED-MLDRL	1-LPI-XHE-XM-GFEED-MLDRL			3.6E-03	Combined HEP value exceeds 1E-06 (note 1)
1-HPI-XHE-XM-FFEED-POS8E	1-OA-NSCWCT-MV-H			5.6E-07	1-SD-HEP-JOINT-69
1-HPI-XHE-XM-FFEED-POS8E	1-RHR-XHE-XD-POS7			5.6E-07	1-SD-HEP-JOINT-70
1-HPI-XHE-XM-FFEED-POS8E	1-RHR-XHE-XD-POS8E			1.3E-10	1-SD-HEP-JOINT-71
1-HPI-XHE-XM-FFEED-POS8E	1-RHR-XHE-XL-TRA-POS8E			2.5E-08	1-SD-HEP-JOINT-72
1-HPI-XHE-XM-FFEED-POS8E	1-RHR-XHE-XL-TRB-POS7			1.3E-09	1-SD-HEP-JOINT-73
1-HPI-XHE-XM-FFEED-POS8E	1-RHR-XHE-XL-TRB-POS8E			1.3E-09	1-SD-HEP-JOINT-74
1-LOI-XHE-XM-ISO2-LT	1-LOI-XHE-XD-POS5A			6.1E-04	Combined HEP value exceeds 1E-06 (note 1)
1-LOI-XHE-XM-ISO2-LT	1-LOI-XHE-XD-POS8E			1.2E-06	Combined HEP value exceeds 1E-06 (note 1)
1-LOI-XHE-XM-ISO2-LT	1-LOI-XHE-XM-ISO-POS8E			1.2E-04	Combined HEP value exceeds 1E-06 (note 1)
1-LOI-XHE-XM-ISO2-LT	1-RHR-XHE-XM-LOI-POS6			2.4E-02	Combined HEP value exceeds 1E-06 (note 1)
1-LOI-XHE-XM-ISO2-LT	1-RHR-XHE-XM-LOI-POS7			1.2E-04	Combined HEP value exceeds 1E-06 (note 1)
1-LOI-XHE-XM-ISO2-LT	1-RHR-XHE-XM-M5LOI-I			2.7E-04	Combined HEP value exceeds 1E-06 (note 1)

Table 7-11 HFE Cut Set Combination Substitutions with Minimum Combined HEP Value (cont.)

1 st HFE	2 nd HFE	3 rd HFE	4 th HFE	Combined HEP	Basic Event Substitution
1-LOI-XHE-XD-POS8E	1-RHR-XHE-XM-TRIP-M5LOI			2.7E-06	Combined HEP value exceeds 1E-06 (note 1)
1-LOI-XHE-XD-POS8E	1-RHR-XHE-XL-TRN-AIR-LOI			1.0E-07	1-SD-HEP-JOINT-82
1-LOI-XHE-XM-ISO-POS8E	1-RHR-XHE-XM-TRIP-M5LOI			2.7E-04	Combined HEP value exceeds 1E-06 (note 1)
1-LOI-XHE-XM-ISO-POS8E	1-RHR-XHE-XL-TRN-AIR-LOI			1.0E-05	Combined HEP value exceeds 1E-06 (note 1)
1-LPI-XHE-XM-GFEED-LOI	1-RHR-XHE-XL-TRN-M5LOI			1.1E-04	Combined HEP value exceeds 1E-06 (note 1)
1-LPI-XHE-XM-GFEED-LOI-6	1-RHR-XHE-XL-TRN-M5LOI			1.1E-04	Combined HEP value exceeds 1E-06 (note 1)
1-LPI-XHE-XM-GFEED-LOI8E	1-RHR-XHE-XL-TRA-POS8E			2.1E-04	Combined HEP value exceeds 1E-06 (note 1)
1-LPI-XHE-XM-GFEED-MLDRL	1-RHR-XHE-XL-TRN-AIR-DRL			8.5E-05	Combined HEP value exceeds 1E-06 (note 1)
1-OA-ALIGNPW-G-H	1-LPI-XHE-XM-GFEED-LOOP			5.1E-02	Combined HEP value exceeds 1E-06 (note 1)
1-RHR-XHE-XL-STBY-POS7	1-RHR-XHE-XL-TRB-POS7			1.3E-08	1-SD-HEP-JOINT-90
1-RHR-XHE-XL-TRN-3H	1-RHR-XHE-XL-TRN-STBY-2H			3.1E-07	1-SD-HEP-JOINT-91
1-RHR-XHE-XD-POS7	1-RHR-XHE-XL-LTREC-LT5H			3.8E-07	1-SD-HEP-JOINT-92
1-RHR-XHE-XD-POS8E	1-RHR-XHE-XL-LTREC-LT5H			3.8E-07	1-SD-HEP-JOINT-93
1-RHR-XHE-XL-TRB-POS8E	1-RHR-XHE-XL-LTREC			3.8E-07	1-SD-HEP-JOINT-94
1-RHR-XHE-XD-POS8E	1-RHR-XHE-XL-LTREC			3.8E-08	1-SD-HEP-JOINT-95
Note 1. Some HFE combinations were identified with combined HEP values <i>potentially</i> less than 10 ⁻⁶ ; however, these combinations were found to be greater than 10 ⁻⁶ after final evaluation of HEP values was completed. Therefore, no substitution for the minimum combined HEP threshold was performed.					

7.7 HFE Quantification and Uncertainty Analysis

The final quantification and uncertainty distribution for each independent HFE is shown in Table 7-12. The HEP values were developed using the methods and guidance described in the preceding sections. A SPAR-H analysis worksheet was developed for each HFE. The parameter uncertainty of each HFE is characterized by a lognormal distribution with an error factor assigned following the guidelines that are summarized in Table 7-13, which are an expanded version of rules originally found in THERP guidance. The error factor guidelines expand on the THERP guidance by establishing revised error factors for high failure probability events to minimize the fraction of the cumulative distribution that would otherwise fall above a value of 1.0. The THERP guidance recommended higher error factors for smaller HEPs to reflect the greater uncertainties associated with infrequently occurring events and recommended that the nominal HEP be assumed to be the median of the lognormal distribution. Nevertheless, the HEP point estimate here is assumed to be the mean value, a lognormal distribution is prescribed, and the error factor is determined based on the magnitude of the mean in accordance with the following rules summarized in Table 7-13.

Table 7-12 HFE Quantification and Uncertainty

HFE Name	HEP Value	Error Factor	Median	95th
1-ACP-XHE-XL-ALTAC-6H	2.00E-03	5.00E+00	1.24E-03	6.20E-03
1-AFW-XHE-FO-SGCOOL	1.25E-04	1.00E+01	4.69E-05	4.69E-04
1-AFW-XHE-XM-SGC-POS3	2.50E-05	1.00E+01	9.39E-06	9.39E-05
1-AFW-XHE-XM-SGC-PRF	2.50E-05	1.00E+01	9.39E-06	9.39E-05
1-DEP-XHE-XM-BLEED	6.00E-03	5.00E+00	3.72E-03	1.86E-02
1-HPI-XHE-XM-FFD-PRF-NSCW	9.93E-03	5.00E+00	6.15E-03	3.08E-02
1-HPI-XHE-XM-FFEED-6H	1.70E-03	5.00E+00	1.05E-03	5.27E-03
1-HPI-XHE-XM-FFEED-6H-5A	9.93E-03	5.00E+00	6.15E-03	3.08E-02
1-HPI-XHE-XM-FFEED-LOI	3.86E-03	5.00E+00	2.39E-03	1.20E-02
1-HPI-XHE-XM-FFEED-LOI5A	7.70E-03	5.00E+00	4.77E-03	2.39E-02
1-HPI-XHE-XM-FFEED-LOI-6	3.86E-03	5.00E+00	2.39E-03	1.20E-02
1-HPI-XHE-XM-FFEED-LOI-7	3.86E-03	5.00E+00	2.39E-03	1.20E-02
1-HPI-XHE-XM-FFEED-LOI8E	7.75E-04	1.00E+01	2.91E-04	2.91E-03
1-HPI-XHE-XM-FFEED-MLDRL	7.95E-03	5.00E+00	4.93E-03	2.46E-02
1-HPI-XHE-XM-FFED-NSCW	7.23E-02	5.00E+00	4.48E-02	2.24E-01
1-HPI-XHE-XM-FFEED-POS8E	5.00E-05	1.00E+01	1.88E-05	1.88E-04
1-LOI-XHE-XM-ISO2-LT	2.44E-02	5.00E+00	1.51E-02	7.56E-02
1-LOI-XHE-XD-POS5A	2.50E-02	5.00E+00	1.55E-02	7.75E-02
1-LOI-XHE-XD-POS8E	5.00E-05	1.00E+01	1.88E-05	1.88E-04
1-LOI-XHE-XM-ISO-POS8E	5.00E-03	5.00E+00	3.10E-03	1.55E-02
1-LPI-XHE-XM-GFD-ML-NSCW	4.23E-01	2.60E+00	3.57E-01	9.29E-01
1-LPI-XHE-XM-GFD-PRF-NSCW	8.00E-03	5.00E+00	4.96E-03	2.48E-02
1-LPI-XHE-XM-GFEED-LD	4.52E-01	2.40E+00	3.92E-01	9.42E-01
1-LPI-XHE-XM-GFEED-LOI	4.23E-01	2.60E+00	3.57E-01	9.29E-01
1-LPI-XHE-XM-GFEED-LOI5A	4.23E-01	2.60E+00	3.57E-01	9.29E-01
1-LPI-XHE-XM-GFEED-LOI-6	4.23E-01	2.60E+00	3.57E-01	9.29E-01
1-LPI-XHE-XM-GFEED-LOI-7	1.75E-02	5.00E+00	1.08E-02	5.42E-02
1-LPI-XHE-XM-GFEED-LOI8E	4.23E-01	2.60E+00	3.57E-01	9.29E-01
1-LPI-XHE-XM-GFEED-LOOP	5.50E-01	1.80E+00	5.16E-01	9.29E-01
1-LPI-XHE-XM-GFEED-MD	5.05E-01	2.00E+00	4.62E-01	9.24E-01
1-LPI-XHE-XM-GFEED-MLDRL	4.23E-01	2.60E+00	3.57E-01	9.29E-01
1-LPI-XHE-XM-GFEED-NSCW	2.15E-01	5.00E+00	1.33E-01	6.66E-01
1-LPI-XHE-XM-GFEED-P7-LD	6.66E-02	5.00E+00	4.13E-02	2.06E-01

Table 7-12 HFE Quantification and Uncertainty (cont.)

HFE Name	HEP Value	Error Factor	Median	95th
1-LPI-XHE-XM-GFEED-P7-MD	1.58E-01	5.00E+00	9.79E-02	4.90E-01
1-RHR-XHE-XD-3H	1.25E-04	1.00E+01	4.69E-05	4.69E-04
1-RHR-XHE-XD-3H-5A	1.25E-06	1.00E+01	4.69E-07	4.69E-06
1-RHR-XHE-XD-ML	1.25E-03	5.00E+00	7.75E-04	3.87E-03
1-RHR-XHE-XD-POS3	1.25E-05	1.00E+01	4.69E-06	4.69E-05
1-RHR-XHE-XD-POS7	2.50E-06	1.00E+01	9.39E-07	9.39E-06
1-RHR-XHE-XD-POS8E	2.50E-06	1.00E+01	9.39E-07	9.39E-06
1-RHR-XHE-XD-PRF	1.25E-05	1.00E+01	4.69E-06	4.69E-05
1-RHR-XHE-XL-LTREC	1.50E-02	5.00E+00	9.29E-03	4.65E-02
1-RHR-XHE-XL-LTREC-LT5H	1.50E-01	5.00E+00	9.29E-02	4.65E-01
1-RHR-XHE-XL-STBY-POS7	5.00E-04	1.00E+01	1.88E-04	1.88E-03
1-RHR-XHE-XL-TRA-POS3	5.00E-04	1.00E+01	1.88E-04	1.88E-03
1-RHR-XHE-XL-TRA-POS7	5.00E-04	1.00E+01	1.88E-04	1.88E-03
1-RHR-XHE-XL-TRA-POS8E	5.00E-04	1.00E+01	1.88E-04	1.88E-03
1-RHR-XHE-XL-TRA-PRF	5.00E-04	1.00E+01	1.88E-04	1.88E-03
1-RHR-XHE-XL-TRB-POS3	2.50E-05	1.00E+01	9.39E-06	9.39E-05
1-RHR-XHE-XL-TRB-POS7	2.50E-05	1.00E+01	9.39E-06	9.39E-05
1-RHR-XHE-XL-TRB-POS8E	2.50E-05	1.00E+01	9.39E-06	9.39E-05
1-RHR-XHE-XL-TRB-PRF	2.50E-05	1.00E+01	9.39E-06	9.39E-05
1-RHR-XHE-XL-TRN-3H	1.25E-03	5.00E+00	7.75E-04	3.87E-03
1-RHR-XHE-XL-TRN-LOOP	2.50E-04	1.00E+01	9.39E-05	9.39E-04
1-RHR-XHE-XL-TRN-HD	5.01E-01	2.00E+00	4.58E-01	9.17E-01
1-RHR-XHE-XL-TRN-M5LOI	2.50E-04	1.00E+01	9.39E-05	9.39E-04
1-RHR-XHE-XL-TRN-MLDRL	1.25E-02	5.00E+00	7.75E-03	3.87E-02
1-RHR-XHE-XL-TRN-ML-P10	1.25E-03	5.00E+00	7.75E-04	3.87E-03
1-RHR-XHE-XL-TRN-POS4	2.50E-05	1.00E+01	9.39E-06	9.39E-05
1-RHR-XHE-XL-TRN-POS6	1.25E-02	5.00E+00	7.75E-03	3.87E-02
1-RHR-XHE-XL-TRN-POS10	1.25E-03	5.00E+00	7.75E-04	3.87E-03
1-RHR-XHE-XL-TRN-POS9	1.25E-04	1.00E+01	4.69E-05	4.69E-04
1-RHR-XHE-XL-TRN-STBY-2H	2.50E-04	1.00E+01	9.39E-05	9.39E-04

Table 7-12 HFE Quantification and Uncertainty (cont.)

HFE Name	HEP Value	Error Factor	Median	95th
1-RHR-XHE-XL-TRN-STBY-ML	2.50E-04	1.00E+01	9.39E-05	9.39E-04
1-RHR-XHE-XM-LOI-POS6	1.00E+00	1.00E+00	1.00E+00	1.00E+00
1-RHR-XHE-XM-LOISP-POS6	2.98E-02	5.00E+00	1.85E-02	9.23E-02
1-RHR-XHE-XM-LOI-POS5B	1.84E-01	5.00E+00	1.14E-01	5.70E-01
1-RHR-XHE-XM-LOI-POS7	5.05E-03	5.00E+00	3.13E-03	1.56E-02
1-RHR-XHE-XM-M5LOI-I	1.20E-03	5.00E+00	7.44E-04	3.72E-03
1-RHR-XHE-XM-MLDRL-I	1.02E-02	5.00E+00	6.32E-03	3.16E-02
1-RHR-XHE-XM-POS10-I	1.20E-03	5.00E+00	7.44E-04	3.72E-03
1-RHR-XHE-XM-TRIP-HD	5.01E-01	2.00E+00	4.59E-01	9.19E-01
1-RHR-XHE-XM-TRIP-LD	5.36E-02	5.00E+00	3.32E-02	1.66E-01
1-RHR-XHE-XM-TRIP-LOI-POS6	3.75E-03	5.00E+00	2.32E-03	1.16E-02
1-RHR-XHE-XM-TRIP-M5LOI	3.75E-03	5.00E+00	2.32E-03	1.16E-02
1-RHR-XHE-XM-TRIP-MLDRL	3.75E-03	5.00E+00	2.32E-03	1.16E-02
1-RHR-XHE-XM-TRIP-MD	1.46E-01	5.00E+00	9.05E-02	4.52E-01

Table 7-13 Guidelines for Assigning Error Factors for Human Failure Events

HEP (mean)	Error Factor
$HEP < 0.001$	10
$0.001 \leq HEP < 0.3$	5
$0.3 \leq HEP < 0.4$	3
$0.4 \leq HEP < 0.45$	2.95
$0.45 \leq HEP < 0.5$	2.45
$0.5 \leq HEP < 0.55$	2.10
$0.55 \leq HEP < 0.6$	1.86
$0.6 \leq HEP < 0.65$	1.66
$0.65 \leq HEP < 0.7$	1.51
$0.7 \leq HEP < 0.75$	1.38
$0.75 \leq HEP < 0.8$	1.28
$0.8 \leq HEP < 0.85$	1.20
$0.85 \leq HEP < 0.9$	1.12
$HEP = 0.90$	1.05
$HEP > 0.90$	1.00 (i.e., no uncertainty prescribed)

8 DATA ANALYSIS

The objectives of the data analysis element are to provide estimates of the parameters used to determine the probabilities of the basic events representing equipment failures and unavailabilities modeled in the PRA. The basic events from the L3PRA project at-power model (NRC, 2022) are used to represent most of the equipment failures in the low power and shutdown (LPSD) PRA. The at-power failure rates are used if there is no reason for a unique failure rate for LPSD conditions. This section describes the unique basic events that are developed for the LPSD PRA and the approach for applying plant operating state (POS)-specific conditions to testing and maintenance unavailabilities. The shutdown initiating event frequencies, testing and maintenance unavailabilities, and unique basic events developed for residual heat removal (RHR) repair and recovery are described in Sections 8.1, 8.3, and 8.4, respectively. Section 8.2 addresses the special case of losses of inventory outside of containment.

8.1 Initiating Event Frequency Estimates

A combination of different data sources was used for estimating initiating event frequencies. An EPRI report published in 2001 (EPRI, 2001) is the most recent available reference for shutdown initiating event frequencies. The data from this report covers the years 1989 through 2000. More recent data was desired to better represent the operating experience from more recent outages. A subsequent EPRI report published in 2010 (EPRI, 2010) includes operating experience data for the years 1990 through 2009. However, this report does not estimate initiating event frequencies, it only collects and catalogues the events that occurred during the analyzed period. Therefore, the operating experience data from the 2010 EPRI report was supplemented by information from the INL database (INL, 2013), to estimate the shutdown observation time needed for the shutdown initiating event frequency calculations. In summary, the 2001 EPRI report was used as the source for prior distributions that were then updated with the more recent data obtained from the 2010 EPRI report and the INL database. Note, that since the prior distributions include data through 2000, only the data from 2001 through 2009 in the 2010 EPRI report were used for the update.

One of the challenges in performing the initiating event frequency analysis is the limited availability of initiating event data for shutdown events. The previously referenced EPRI reports are the most recent comprehensive collections of shutdown operating experience and frequency estimates. Investigations into more recent operating experience have shown that the information needed for estimating initiating event frequencies is not readily available from routinely reported information. Event occurrences could not always be corroborated from multiple sources. For example, a licensee event report (LER) could not always be identified for events recorded in the EPRI reports. In some cases, the event description was available, but the plant where it occurred could not be identified. Also, only limited information was available about the exposure period for plant operating states where an initiating event would be relevant. Given these uncertainties in the available data, the initiating event frequency approach for this project is to develop generic, industry-wide shutdown initiating event frequencies, rather than plant-specific frequencies.

Section 8.1.1 describes the approach employed for estimating the frequencies for the loss of offsite power (LOOP), loss of 4,160V AC bus (LO4160V), and loss of RHR (LORHR) initiators. The approaches employed to estimate the frequencies of the loss of inventory, overdraining,

and loss of nuclear service cooling water (NSCW) scenarios are described in Sections 8.1.2, 8.1.3, and 8.1.4, respectively.

8.1.1 Frequency Estimates for LOOP, LO4160V and LORHR

The initiating event frequency estimates for initiating event categories LOOP, LO4160V and LORHR were prepared by the Idaho National Laboratory. This section summarizes the initiating event data and approach for the frequency estimates.

Eighty-three shutdown initiating events in INL's database were analyzed to produce estimated frequencies for: loss of offsite power (all plants), loss of AC bus (all plants), and loss of residual heat removal capacity (PWRs only).

It is assumed that LOOP can be pooled across BWRs and PWRs, since plant type has no relationship with the reliability of offsite power. The total non-critical time for 2000–2015 was estimated to be 38.08 and 104.17 shutdown-years for BWRs and PWRs, respectively, for the purposes of LOOP and LO4160V estimates.

Loss of RHR is not poolable between BWRs and PWRs. For the LORHR rate estimate, the data were analyzed for years 2000-2009. The failure data are taken from the EPRI report, *An Analysis of Loss of Decay Heat Removal and Loss of Inventory Event Trends* (1990-2009) (EPRI, 2010). The exposure term for the LORHR is based on the non-critical time in the monthly operating reports for 2000-2009. The total shutdown-years for PWRs in this period is 65.78. An adjustment to the exposure term was necessary to account for the fact that the RHR system is not continuously in operation during an outage. The records available in the INL database did not show how many hours RHR cooling systems were in use; records only showed the total time spent in shutdown. A review of the reference plant's outages from 2000-2015 was performed to inform a representative fraction of outage hours where RHR cooling is in use. It was found that approximately 80% of outage hours were refueling outage hours and 14% of outage hours were maintenance outages with RHR in use. RHR cooling was noted to be in use during 61 percent of the refueling outage hours and 86 percent of the maintenance outage hours. As such, the exposure time for LORHR at PWRs was calculated, based on the reference plant experience, to be $(0.80 \times 0.61 + 0.14 \times 0.86) \times 65.78 = 40.02$ shutdown-years.

Table 8-1 summarizes the initiating event frequency values (including uncertainties) for LOOP, LO4160V, and LORHR.

Table 8-1 L3PRA LPSD Model Initiating Event Frequency Summary for LOOP, LO4160V and LORHR

Event Type	5 th (per shutdown-year)	MLE ¹ (per shutdown-year)	95 th (per shutdown-year)	α	β
LOOP (pooled)	6.50E-03	0.163	0.276	1.28	7.84
LO4160V (pooled)	1.44E-02	0.340	0.461	1.19	3.50
LORHR (PWR)	8.12E-04	0.367	0.442	0.887	2.42
1. MLE refers to maximum likelihood estimation. See (Atwood, 2003) for discussion on use of MLE in initiating event frequency estimation.					

A review of past operating experience (see 12APPENDIX D and the events that are included in the frequency estimate) was used to characterize the plant conditions leading to the events and the response after the events. The plant conditions and the POS can have an influence on the initiating event frequencies. Unique POS conditions are accounted for in the loss of inventory and overdrain events, discussed in the following sections. However, for other initiating event categories, the information needed to sub-divide into POS-specific categories is not available. For the loss of RHR cooling initiating event category, the following characteristics are noted in reviewing 13 events that are used to estimate the initiating event frequency (EPRI, 2010).

- The plant where the event occurred could not be identified for 4 of 13 loss of RHR cooling events. The event descriptions from (EPRI, 2010) were the only information available to support the analysis.
- All 13 loss of RHR events were recovered by restoring RHR cooling in relatively short time after the event occurrence, with an average time to restore RHR cooling of 15 minutes.
- The available information was not sufficient to distinguish the POS at the time of the initiating event. For most events, the Technical Specification operating mode can be determined from the event description. However, this information cannot be used to identify the POS because there is no data collection on the duration of operating modes, which can vary significantly from plant-to-plant and outage-to-outage. Though the information about the POS conditions is limited, none of the loss of RHR events appear to have happened during mid-loop operations.
- The event causes for loss of RHR cooling include: loss of cooling to the RHR heat exchanger (e.g., CCW issues or isolating flow to the heat exchanger), electrical power supply issues causing valve mispositions in the RHR system, maintenance and testing activities resulting in inadvertent isolation of RHR cooling, and other inadvertent isolations.

The available information was not sufficient to distinguish the status of the RHR trains (e.g., whether one or two RHR trains were operating at the time of the event).

The available information about the loss of RHR cooling events has helped to inform the modeling approach for these events. The initiating events are defined to occur at a functional level, rather than a train level. The specific status and availability of trains is handled in the initiating event response modeling and accounts for the POS conditions. While past studies have provided initiating event frequencies for specific POSs (EPRI, 2001), the approach for this project is to use one initiating event frequency applied to all shutdown POSs. The initiating event operating experience review shows that loss of RHR events are typically recoverable. So, all loss of RHR events consider the possibility of restoring the previously operating RHR train. The human failure events and the RHR system equipment reliability are addressed in the fault tree modeling in the initiating event response.

8.1.2 Frequency Estimates for Losses of Inventory

The loss of inventory operating experience data was reviewed to inform the approach for estimating initiating event frequencies. The data sources for the operating experience review were the EPRI reports 1021176, *An Analysis of Loss of Decay Heat Removal and Loss of*

Inventory Event Trends (1990-2009) (EPRI, 2010) and 1003113, An Analysis of Loss of Decay Heat Removal Trends and Initiating Event Frequencies (1989-2000) (EPRI, 2001).

Loss of inventory events can result from a variety of causes, including pipe or pressure vessel failures, human errors resulting in inadvertent flow diversions, or lifting of pressure relief valves. The most recent available operating experience from 2000-2009 shows that eight loss of inventory events occurred during the observation period. All eight of the events are related to pressure transients causing opening of relief valves during POSs with the RCS intact. The following observations are noted from the operating experience review.

- All recent loss of inventory events are related to lifting of relief valves.
- The next most common type of inventory loss involves inadvertent transfer of inventory (i.e., losses due to human errors without a system breach).
- A small number of events from the older observation period (EPRI, 2001) are due to system breaches. These events involve leaking valves.
- All the reviewed loss of inventory events were able to be isolated.

Outside of the lifting relief valve events, there is very little recent operating experience to inform loss of inventory frequency estimates that would be relevant to the modeled reference plant. Given the limited operating experience data, an alternate approach was used to represent the range of possible loss of inventory failures that could occur. The EPRI approach for pipe rupture frequencies (EPRI, 2013) is used to estimate the failure frequencies of the connected pipe systems that can result in a loss of inventory.

The initiating event category frequency estimates for SD-LOINV-SP, SD-LOINV, and SD-LOINV-LG are developed using the EPRI approach, *Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments* (EPRI, 2013). The initiating event frequency, f , for a given pipe failure scenario is given by the following expression,

$$f = l \times \lambda_{pipe} \times P_{pipe}(R|F) \quad [1]$$

where, l is the length of pipe (in feet) located in the relevant plant area.

λ_{pipe} is the failure rate of the pipe per foot-critical reactor year.

$P_{pipe}(R|F)$ is the conditional rupture probability given pipe failure.

The EPRI report defines failure as any condition in which pipe repair or replacement was performed. Failures can include wall thinning, cracks, pinhole leaks, leaks, and major structural failures. A failure will not necessarily result in a loss of inventory event, but the occurrence of any failure will be associated with the conditional probability of a rupture. A rupture is a substantial failure that results in the initiation of a loss of inventory event if the piping segment is connected to the RCS or an RCS drain path. In this report, the terms rupture and break are used interchangeably to refer to substantial pipe failures that result in loss of inventory events.

The intended use of the EPRI approach is to support internal flooding PRAs for at-power operation. The failure rates were developed in terms of reactor operating years. However, the failure rates are expected to apply to systems that are in use during shutdown operations. The EPRI report (EPRI, 2013) states:

The failure rates were developed in terms of events per reactor operating year and linear foot of piping system. Reporting requirements do not require the reporting of many pipe failures and flood events during plant shutdown modes. For systems such as service water and fire protection, which are normally operating during shutdown modes, it is expected that pipe failure rates would be no different at shutdown.

There are examples of the pipe failures that are included in the at-power internal flooding frequency estimates that were discovered during outages; and therefore, these failures should be considered relevant to shutdown frequency estimates. The lack of comprehensive shutdown event reporting makes it difficult to determine the exact proportion of failures that would be applicable to shutdown. Taking this information into account, using the at-power failure rates and conditional rupture probabilities from the EPRI report was deemed to be the best available approach for estimating loss of inventory frequencies for shutdown.

Total Feet of Piping:

The total feet of system piping are used as input to the exposure term in the frequency estimation process. The plant-specific feet of piping may not be precisely known and is often estimated from generic sources. This contributes to the uncertainty in the frequency estimate. The median feet of piping are estimated from available source documents, and the distribution of piping length is represented by a lognormal distribution with an error factor of 3. The total feet of piping used for this project are derived from the safety injection and recirculation (SIR) systems piping shown in Table 8-2 below. The references that provide information on the feet of system piping combine different plant systems into the SIR piping category. The SIR piping category includes the chemical and volume control system, containment spray system, safety injection system, and RHR system. The reference sources also categorize the systems by pipe size diameter. The same pipe size categories are retained here.

Table 8-2 Safety Injection and Recirculation Length of System Piping

System Category	Nominal Pipe Size Diameter (in.)	Median Pipe Length (feet)
SIR Piping (includes CVCS, SI, CS, and RHR piping inside containment) (Note 1)	2 < NPS ≤ 6	2111
	6 < NPS ≤ 10	7036
	NPS > 10	4925
SIR Piping (includes CVCS, SI, CS, and RHR piping outside containment) (Note 2)	2 < NPS ≤ 6	4020
	6 < NPS ≤ 10	13400
	NPS > 10	9380
SIR Piping total feet (inside and outside containment) (Note 3)	All sizes	40871
Notes: (1) The feet of piping inside containment are estimated from the difference between the total feet (inside and outside containment) and the feet of piping outside containment as reported in the plant-specific internal flooding PRA. The same proportions of feet of piping to total system piping are selected for all pipe size categories. (2) The feet of piping outside containment are reported in the plant-specific internal flooding PRA report. (3) The total feet of piping for Westinghouse plants is estimated in EPRI report TR-111880, "Piping System Failure Rates and Rupture Frequencies for Use In Risk Informed In-Service Inspection Applications," Table A-5 (EPRI, 1999).		

The loss of inventory events at shutdown focus only on the RHR system piping because the RHR piping has an RCS connection that is normally unisolated during shutdown that could result in RCS draining. The total feet of just the RHR system piping is shown in Table 8-3 below.

Table 8-3 Residual Heat Removal Length of System Piping

System Category	Nominal Pipe Size Diameter (in.)	Median Pipe Length (feet)
RHR piping inside containment contributing to LOINV (Note 1)	2 < NPS ≤ 6	528
	6 < NPS ≤ 10	1759
	NPS > 10	1231
RHR piping outside containment contributing to LOINV (Note 2)	2 < NPS ≤ 6	181
	6 < NPS ≤ 10	710
	NPS > 10	315
Total RHR piping contributing to LOINV	All sizes	4724
Notes: (1) The feet of RHR piping inside containment is estimated to be 1/4 of the SIR piping inside containment. All RHR piping inside containment is assumed to contribute to the loss of inventory frequency. (2) The feet of RHR piping outside containment that are relevant to loss of inventory events are limited to the areas within the Auxiliary Building reported in the plant-specific internal flooding PRA report.		

Loss of Inventory Event Categories:

As discussed above, the loss of inventory event frequencies are based on the EPRI approach for estimating flooding frequencies for internal flooding PRAs (EPRI, 2013). The EPRI report develops frequency estimated for a range of flood types, which can result in a range of possible impacts at the plant. Three flood type categories are defined to represent the range of frequencies and impacts:

- Spray events were defined to represent break sizes that produce between 1 gpm and 100 gpm.
- Flood events were defined to represent break sizes that produce between 100 gpm and 2000 gpm.
- Major flood events were defined to represent break sizes that produce greater than 2000 gpm.

A similar approach is used for the loss of inventory event categories. The loss of inventory event categories are defined as:

- SD-LOINV-SP: Spray events – small break sizes resulting in a representative loss of inventory rate of 100 gpm.
- SD-LOINV: Loss of inventory events – moderate break sizes resulting in a representative loss of inventory rate of 1000 gpm.
- SD-LOINV-LG: Large loss of inventory events – large breaks sizes resulting in a representative loss of inventory rate of 3500 gpm. Major loss of inventory events only includes pipe sizes 6 inch diameter and larger.

Table 8-4 provides the RHR piping length distributions used for loss of inventory events.

Table 8-4 Residual Heat Removal System Length for Loss of Inventory Events

System Category	Nominal Pipe Size Diameter (in.)	5 th percentile	Median Pipe Length	95 th percentile
RHR piping	2 < NPS ≤ 6	236	709	2127
RHR piping	6 < NPS ≤ 10	823	2469	7407
RHR piping	NPS > 10	515	1546	4638
Total RHR piping contributing to Spray and LOINV events	NPS > 2	1575	4724	14172
Total RHR piping contributing to Large LOINV events	NPS > 6	1338	4015	12045

Failure Data for Plant-Specific Frequency Update:

The EPRI pipe rupture frequency report (EPRI, 2013) provides generic failure data for different types of plant systems. The data are further categorized in terms of the severity of pipe failure (e.g., wall thinning, pinhole leak, leak, major structural failure) and pipe size. The generic data and failure rates in the report are used to develop prior distributions for the RHR pipe failure rates and conditional rupture probabilities. The prior distributions are updated with plant-specific data. The plant-specific data considered for the frequency analysis cover the period from January 1, 1990, through December 31, 2015. The plant-specific flooding data are shown in Table 8-5. The plant-specific data are taken from the following data sources:

- The plant-specific internal flooding PRA report identifies one RHR pipe failure event that occurred during the period from January 1, 1990, through July 31, 2007.
- Search of plant-specific LERs for events including “leak,” “leakage,” or “RHR” in the title through December 2015.
- INL’s NROD site, which includes plant-specific operating experience records through December 2015 (INL, 2013).

Table 8-5 Plant-Specific Pipe Failure Events

System	Nominal Pipe Size / Diameter (in.)	Non Through Wall	Spray (≤ 100 gpm)	Loss of Inventory (100-2000 gpm)	Large Loss of Inventory (> 2000 gpm)
RHR piping	2 < NPS ≤ 6	0	2 ⁽¹⁾⁽²⁾	0	0
	6 < NPS ≤ 10	0	0	0	0
	NPS > 10	0	0	0	0
<i>Notes:</i> (1) Leak causing RHR pump room to flood with approximately two inches of water. Source: plant-specific internal flooding PRA report (2) RCS pressure boundary leakage detected at welded connections around RHR loop suction valve. Source: plant-specific LER					

Analyst judgment is exercised to determine the appropriate statistical models to use for the initiating event frequencies for losses of inventory. The NRC's "Handbook of Parameter Estimation for Probabilistic Risk Assessment", NUREG/CR-6823 (Atwood, 2003), was referenced for guidance in selecting distributions and performing Bayesian updates. Failure rates are assumed to have a gamma uncertainty distribution. A constrained noninformative prior distribution is used with the prior mean taken from the generic estimates in the EPRI report (EPRI, 2013). The plant-specific data is used to update the failure rate with the number of failures from Table 8-5, the relevant length of piping from Table 8-4, and the unit operating years. The two-unit site accounts for 47.3 reactor critical years for the observation period from 1990 to 2015. A plant availability factor of 0.91 is applied to the observation period to convert to reactor critical years.³¹ Although the frequencies are applied to shutdown risk, the update process uses reactor critical years to be consistent with the units reported for the generic failure rates from the EPRI report. The plant availability factor is only applied for the update process. The final results are scaled to give frequencies in terms of calendar years.

The data used for the plant-specific pipe failure rates are summarized in Table 8-6, Table 8-7, and Table 8-8 for the three RHR pipe size categories.

Table 8-6 RHR Pipe Failure Rate Data for Nominal Pipe Size between 2 and 6 inches

Gamma CNI prior distribution	Alpha Prior	Beta Prior	Reference	Notes
	0.5	3.21E+05	Table 4-8 of (EPRI, 2013)	For CNI prior, alpha prior = 0.5, beta prior = 0.5/mean frequency
Evidence	Failures	Feet - Critical Years		
	2	3.35E+04	Tables 8.1.2-3 and 8.1.2-4	709 ft of RHR piping, 47.3 critical years

Table 8-7 RHR Pipe Failure Rate Data for Nominal Pipe Size between 6 and 10 inches

Gamma CNI prior distribution	Alpha Prior	Beta Prior	Reference	Notes
	0.5	1.06E+06	Table 4-9 of (EPRI, 2013)	For CNI prior, alpha prior = 0.5, beta prior = 0.5/mean frequency
Evidence	Failures	Feet - Critical Years		
	0	1.17E+05	Tables 8.1.2-3 and 8.1.2-4	2469 ft of RHR piping, 47.3 critical years

³¹ The plant availability factor represents an average availability for the observation period (1990-2015) used for the loss inventory initiating event analysis. The plant availability factor used for determining POS and POT durations uses more recent data (i.e., 2009 and later). A plant availability factor of 0.93 was determined for the later observation period. The observation period for loss of inventory events was extended to include earlier years due to the rarity of loss of inventory events.

Table 8-8 RHR Pipe Failure Rate Data for Nominal Pipe Size greater than 10 inches

Gamma CNI prior distribution	Alpha Prior	Beta Prior	Reference	Notes
	0.5	6.39E+05	Table 4-10 of (EPRI, 2013)	For CNI prior, alpha prior = 0.5, beta prior = 0.5/mean frequency
Evidence	Failures	Feet - Critical Years		
	0	2.35E+04	Tables 8.1.2-3 and 8.1.2-4	1546 ft of RHR piping, 47.3 critical years

Conditional Rupture Probability:

Conditional rupture probabilities are assumed to have a beta uncertainty distribution. The data are assumed to be binomially distributed. A beta prior distribution is used with parameters selected based on analyst judgment, and example prior distribution parameters provided in EPRI (EPRI, 2013) are referenced for guidance. The data used to develop the conditional rupture probabilities for spray events and loss of inventory events are provided in Table 8-9 through Table 8-14.

Table 8-9 Conditional Rupture Probability for Spray for RHR Pipe Size between 2 and 6 inches

Beta prior distribution	Alpha Prior	Beta Prior	Reference	Notes
	1	9		Based on judgment of the analyst.
Evidence	Ruptures	Failures		
	17	31	Table 4-1 of (EPRI, 2013)	

Table 8-10 Conditional Rupture Probability for Spray for RHR Pipe Size between 6 and 10 inches

Beta prior distribution	Alpha Prior	Beta Prior	Reference	Notes
	1	9		Based on judgment of the analyst.
Evidence	Ruptures	Failures		
	13	31	Table 4-1 of (EPRI, 2013)	

Table 8-11 Conditional Rupture Probability for Spray for RHR Pipe Size greater than 10 inches

Beta prior distribution	Alpha Prior	Beta Prior	Reference	Notes
	1	9		Based on judgment of the analyst.
Evidence	Ruptures	Failures		
	15	36	Table 4-1 of (EPRI, 2013)	

Table 8-12 Conditional Rupture Probability for LOINV for RHR Pipe Size between 2 and 6 inches

Beta prior distribution	Alpha Prior	Beta Prior	Reference	Notes
	1	99		Based on judgment of the analyst.
Evidence	Ruptures	Failures		
	0	31	Table 4-1 of (EPRI, 2013)	

Table 8-13 Conditional Rupture Probability for Spray for RHR Pipe Size between 2 and 6 inches

Beta prior distribution	Alpha Prior	Beta Prior	Reference	Notes
	1	99		Based on judgment of the analyst.
Evidence	Ruptures	Failures		
	0	31	Table 4-1 of (EPRI, 2013)	

Table 8-14 Conditional Rupture Probability for Spray for RHR Pipe Size between 2 and 6 inches

Beta prior distribution	Alpha Prior	Beta Prior	Reference	Notes
	1	99		Based on judgment of the analyst.
Evidence	Ruptures	Failures		
	0	36	Table 4-1 of (EPRI, 2013)	

There is limited evidence to indicate failures of safety injection and recirculation piping that would result in a loss of inventory event or a large loss of inventory event. The generic data in Table 4-1 of (EPRI, 2013) show that of 98 pipe failures, none of the events involved a leak greater than 100 gpm. In the EPRI approach, the prior distribution is generally assumed to have a smaller conditional rupture probability as the effective break size of the rupture increases. However, this is not always supported by the operating experience data. In some pipe categories, the data indicates a higher number of large failures resulting in a break flow >2000 gpm than failures resulting in break flow 100–2000 gpm. The data relevant to RHR piping shows zero failures for both the large and moderate flood categories. For this project, the conditional rupture probability developed for the moderate flood category (and applied to the LOINV initiating event) is also used for the major flood category (i.e., LOINV-LG initiating event). The same conditional rupture probability is applied to both failure categories due to the limited data and to not overly influence the estimate by prior assumptions.

Loss of Inventory Frequency Results:

A Gibbs sampling process is used to generate the combined initiating event frequency posterior distributions.³² The sampling process produces an empirical posterior distribution. The resulting empirical distribution is expected to resemble a gamma distribution based on the choice of prior distribution for the failure rates. Gamma distributions are routinely used to model initiating event frequency uncertainty distributions. Gamma function parameters are fit to the empirical distribution using a maximum likelihood estimate approach.³³ The fitted gamma distribution parameters are used to specify the mean initiating event frequencies and shape parameters in the LPSP PRA model.

The results of the loss of inventory initiating event frequency analysis are shown in Table 8-15.

Table 8-15 Loss of Inventory Initiating Event Frequencies

Loss of Inventory Event Category	Mean IE Frequency per Shutdown-Year	Shape Parameter or Error Factor	5th Percentile	Median Value	95th Percentile
LOINV-SP	3.0E-03	1.9	4.8E-04	2.5E-03	7.2E-03
LOINV	6.4E-05	0.91	2.4E-06	4.3E-05	2.0E-04
LOINV-LG	1.2E-05	0.62	1.4E-07	6.6E-06	4.3E-05

Loss of Inventory with RCS Intact:

All of the LOINV events identified in (EPRI, 2013) are associated with conditions with the RCS pressurized. These events are incorporated into the initiating event frequency estimate for SD-LOINV-OP.

While EPRI report 1021176 (EPRI, 2010) is used to identify recent SD-LOINV-OP event occurrences, the report does not estimate the industry shutdown hours associated with those events. A similar approach as used for the LORHR frequency, as described in Section 8.1.3 above, is used to estimate the applicable shutdown hours. The total PWR shutdown-years with RHR operation are estimated to be 40.02. The SD-LOINV-OP category is only applicable during operations when the RCS is at pressure. The POS durations defined in Table 2-23 are used to estimate the fraction of shutdown-years when the RCS is at pressure. The total RHR operation hours comprise the POS durations from POS 3 through POS 13. The POSs with the RCS at pressure are POS 3–5A and POS 11–13. Based on this data, 27 percent of the RHR operation hours are associated with RCS at pressure. The PWR shutdown-years for RHR operation at pressure are 10.83 years (94,910 hours), and the PWR shutdown-years for RHR operation with RCS vented are 29.19 years (255,905 hours). Uncertainty is accounted for in the estimates by assuming a lognormal distribution with an error factor of 1.5.

³² The sampling is performed using the OpenBUGS version 3.2.3 software (Lunn, 2000). For each frequency estimate, 10,000 samples were run. Sampling simulations were performed for two separate chains. Trace history plots of the two sampling chains were reviewed for evidence of parameter convergence.

³³ The fit is performed using the R statistical computing environment version 3.6.1 with the MASS function package (R, 2013).

The event data from (EPRI, 2010) were used to update the prior frequency estimates that are reported in (EPRI, 2001). The LOINV initiating event frequencies in (EPRI, 2001) are estimated for two categories:

- small draindowns – RCS leaks or diversions resulting in less than 10,000 gallons lost or diverted
- large draindowns – RCS leaks or diversions resulting in greater than 10,000 gallons lost or diverted

The frequency distribution for small draindowns from (EPRI, 2001) is used as the prior distribution for the SD-LOINV-OP category.

The prior distributions and evidence used to update the LOINV-OP frequency for this project, based on the approach in NUREG/CR-6823, Section 6.2.2.7 (Atwood, 2003), are summarized in Table 8-16.

Table 8-16 Summary of Prior Distributions and Evidence for LOINV-OP Frequency

LOINV Category	Lognormal Prior Distribution		No. Events	Lognormal Shutdown Hours Distribution		References
	Mean Frequency (per shut-down hour)	Error Factor		Mean (shutdown hours)	Error Factor	
SD-LOINV-OP	1.1E-05	9.8	8	94,910	1.5	Table 5-14 of (EPRI, 2001) Table A-2 of (EPRI, 2010)

A Gibbs sampling process (Lunn, 2000) is used to generate the LOINV-OP initiating event frequency posterior distributions. The LOINV-OP initiating event frequency results are summarized in Table 8-17.

Table 8-17 LOINV-OP Initiating Event Frequency Estimate

IE Category	Mean Frequency (per shut-down year)	Error Factor	5th	Median	95th
LOINV-OP	6.07E-01	1.98E+00	2.81E-01	5.57E-01	1.10E+00

Loss of Inventory Displacement Events:

The loss of inventory displacement events (LOINV-DS) represent inadvertent transfers of RCS inventory due to human errors, inaccurate level indication, or a combination of both. The frequency estimate is based on the historical loss of inventory frequencies from (EPRI, 2001) and updated with more recent operating experience. The more recent operating experience from (EPRI, 2010) identifies no events in this initiating event category.

The frequency distribution for small draindowns from (EPRI, 2001) is used as the prior distribution for the LOINV-DS category. The frequency is updated with no additional events for the years 2000-2009. The PWR shutdown-years for RHR operation with RCS vented are estimated to be 255,905 hours, as described in the LOINV-OP analysis above. The prior distributions and evidence used to update the LOINV-DS frequency for this project, based on the approach in NUREG/CR-6823, Section 6.2.2.7 (Atwood, 2003), are summarized in Table 8-18.

Table 8-18 Summary of Prior Distributions and Evidence for LOINV-DS Frequency

LOINV Category	Lognormal Prior Distribution		No. Events	Lognormal Shutdown Hours Distribution		References
	Mean Frequency (per shutdown hour)	Error Factor		Mean (shutdown hours)	Error Factor	
LOINV-DS	1.1E-05	9.8	0	255,905	1.5	Table 5-14 of (EPRI, 2001) Table A-2 of (EPRI, 2010)

A Gibbs sampling process (Lunn, 2000) is used to generate the LOINV-DS initiating event frequency posterior distributions. The LOINV-DS initiating event frequency results are summarized in Table 8-19.

Table 8-19 LOINV-DS Initiating Event Frequency Estimate

IE Category	Mean Frequency (per shutdown year)	Error Factor	5th	Median	95th
LOINV-DS	2.09E-02	4.49E+00	3.08E-03	1.38E-02	6.19E-02

8.1.3 Frequency Estimate for Overdrain Entering Mid-loop

The overdrain event is a demand-based event. Overdrain is only considered when the reactor is drained to enter the mid-loop condition. The initiating event frequency for overdrain is calculated by taking the product of the frequency of entering mid-loop during a refueling outage and the conditional probability of over-draining RCS level when entering mid-loop.

$$\text{Overdrain IE frequency} = (\text{frequency of mid-loop}) \times (\text{conditional probability of overdrain}) \quad [2]$$

Frequency of Entering Mid-loop during Refueling Outage:

The frequency of the reference plant entering mid-loop during a refueling outage is based on a review of plant-specific refueling outage histories for 2009-2014. Eight outages were reviewed that occurred during this period. The number of mid-loop conditions entered varied from one to three per outage. Note that only mid-loops with fuel in the reactor vessel were counted. Entrances to mid-loop while defueled were not counted. The total number of mid-loops that the reference plant performed during 2009-2014 is 13. This plant-specific data was used to update

a generic prior based on the assumption of one early mid-loop entry (prior to refueling) and one late mid-loop entry (after refueling) for each refueling outage with an 18 month refueling cycle. The resulting posterior distributions are used as the frequencies of entering hot and cold mid-loop (see Table 8-20). The overdrain initiating event frequency is given by the product of the frequency of entering mid-loop and the conditional probability of over-draining RCS level when entering mid-loop.

Table 8-20 Plant-Specific Frequency for Entering Mid-Loop Condition

	Gamma Distribution using Constrained Noninformative Prior			Evidence		Posterior Distribution		
Mid-Loop Type	Prior Mean (per year)	α	β	Events	Reactor-Years	α	β	Posterior Mean (per year)
Early mid-loop	0.67	0.5	0.75	7	12	7.5	12.75	0.588
Late mid-loop	0.67	0.5	0.75	6	12	6.5	12.75	0.510

Conditional Probability of Overdrain:

The results for conditional probability of overdrain (shown in Table 8-21) are derived from a review of two overdrain events that occurred in the 1990s. The period reviewed for this calculation is 1990-2009. Review of PWR outage records estimate 614 draindowns to mid-loop during this period. The data are used to estimate an industry-average conditional failure probability of overdraining the RCS level when entering mid-loop conditions, using the approach in Section 6.3.2.2 of Atwood (2003).

Table 8-21 Overdrain Conditional Failure Probability Estimate

Non-Informative Prior Distribution		Evidence		Posterior Distribution				
α	β	Failures	Demands	α	β	5%	Mean	95%
0.5	0.5	2	614	2.5	612.5	9.3E-04	4.07E-03	9.0E-03

The point value estimate of the overdrain initiating event frequency is given by the product of the mean values of the mid-loop frequency and the overdrain conditional failure probability.

$$\text{ODML-EARLY IE frequency} = 0.588 / \text{year} \times 4.07 \times 10^{-3} = 2.4 \times 10^{-3} / \text{year}$$

$$\text{ODML-LATE IE frequency} = 0.510 / \text{year} \times 4.07 \times 10^{-3} = 2.1 \times 10^{-3} / \text{year}$$

8.1.4 Frequency Estimate for Loss of Nuclear Service Cooling Water

The initiating event frequency for loss of nuclear service cooling water is based on a support system initiating event fault tree. The same approach is used for the L3PRA project Level 1 PRA for at-power operation (NRC, 2022). The initiating event fault trees are adjusted for POSs depending on the number of NSCW trains that are in operation at the time of the initiating event. Failure of a single train of NSCW results in an initiating event for POSs with a single RHR train in operation. Failure of both trains of NSCW results in an initiating event for POSs with both

RHR trains in operation. Aside from these adjustments to the POS conditions, the NSCW initiating event modeling is the same as that described in (NRC, 2022).

8.2 Loss of Inventory Outside Containment

The loss of inventory events can result in a break pathway that bypasses the containment boundary, similar to interfacing systems loss-of-coolant accidents (ISLOCAs). During much of the outage, the RHR system is in operation. A break or failure in the system can result in a containment bypass pathway, if not isolated. The effects of the loss of inventory leading to core damage are assessed as part of the LPSD Level 1 PRA in this report. The effects of a containment bypass release pathway are assessed in the Level 2 PRA. In order to carry the analysis forward to the Level 2 PRA, the Level 1 PRA accident sequences and frequencies contributing to losses of inventory outside containment must be identified.

The approach for identifying the sequences and frequencies involves three steps:

1. Identify the initiating event categories with potential losses of inventory outside of containment.
2. Identify the accident sequences where isolation of the loss of inventory is failed.
3. Determine the fraction of the loss of inventory initiating event frequency that would result in losses occurring outside containment, rather than inside containment.

Identify initiating event categories with potential losses of inventory outside of containment:

The loss of inventory initiating event categories that involve system breaks, breaches, and failures related to pressure transients are included as potential losses outside containment. The initiating event categories are:

- SD-LOINV-OP
- SD-LOINV-SP
- SD-LOINV
- SD-LOINV-LG

Identify accident sequences contributing to losses of inventory outside of containment:

If the loss of inventory initiating event is identified and isolated, then the potential for a bypass release pathway is averted. However, if the initial diagnosis and isolation fails, then the sequence is carried forward for inclusion in the Level 2 PRA release category for losses outside containment.

Each loss of inventory event tree is assessed for applicable POSs. Each POS includes an event tree top event that models failure of the initial diagnosis and isolation of the inventory loss. Table 8-22 identifies the top event failures and sequence identifying numbers that are to be assessed for loss of inventory outside containment in the Level 2 PRA.

Table 8-22 Loss of Inventory Outside Containment Accident Sequences

Event Tree Name	POS	Top Event Failure	Sequence Identifiers
1-SD-LOINV-OP	POS 5A	1-SD-LOI-D-POS5A	5-05-23-1 5-05-29-1 5-05-30-1
1-SD-LOINV-SP	POS 5B	1-SD-LOI-D-M5-I	5-06-22-1 5-06-28-1 5-06-29-1
	POS 6	1-SD-LOISP-D-POS6	5-07-22-1 5-07-28-1 5-07-29-1
	POS 9	1-SD-LOI-D-M5-I	5-12-22-1 5-12-28-1 5-12-29-1
	POS 10	1-SD-LOISP-D-POS6	5-13-22-1 5-13-28-1 5-13-29-1
1-SD-LOINV	POS 5B	1-SD-LOI-D-M5	5-06-22-1 5-06-28-1 5-06-29-1
	POS 6	1-SD-LOI-D-POS6	5-07-22-1 5-07-28-1 5-07-29-1
	POS 9	1-SD-LOI-D-M5	5-12-22-1 5-12-28-1 5-12-29-1
	POS 10	1-SD-LOI-D-POS6	5-13-22-1 5-13-28-1 5-13-29-1
1-SD-LOINV-LG	POS 7	1-SD-LOI-D-POS7	5-08-22-1 5-08-28-1 5-08-29-1
	POS 8E	1-SD-LOI-D-POS8E	5-09-22-1 5-09-28-1 5-09-29-1
	POS 8L	1-SD-LOI-D-POS8E	5-11-22-1 5-11-28-1 5-11-29-1

Fraction of loss of inventory events occurring outside containment:

The frequencies associated with the accident sequences identified above include potential failures both inside and outside containment. To estimate the frequency contribution for only the losses outside containment, the fraction of the event frequency occurring outside containment is determined. There are two approaches used for determining the fraction of the event frequency outside containment. For the LOINV-OP initiating event category, the operating experience events are assessed to determine the fraction occurring inside and outside containment. For the other LOINV categories, the fraction is based on the estimated plant-specific length of piping inside and outside containment for the systems contributing to the initiating event frequencies.

For LOINV-OP events, the initiating event frequency is based on historical event occurrences identified in (EPRI, 2010). A total of eight events are identified to develop the initiating event frequency, as shown in Table 8-16. The events were reviewed to identify how many involved failures resulting in losses outside containment. One of the eight events was determined to result in losses outside containment. These events are used to develop the fraction of events outside containment. A beta distribution with a non-informative prior distribution is used to develop a distribution to represent this fraction. The results are summarized in Table 8-23.

Table 8-23 Fraction of Events Outside Containment for LOINV-OP

Prior Distribution			Evidence		Posterior Distribution		
Prior Mean Value	α	β	Events Outside Cont.	Total Events	α	β	Posterior Mean Value
0.5	0.5	0.5	1	8	1.5	7.5	1.67E-01

For the other loss of inventory initiating event categories, the initiating event frequencies are based on the plant-specific length of RHR piping summarized in Table 8-3. The fraction of events causing losses outside of containment is given by taking the ratio of the length of piping outside containment to the total length of piping. The result is the lognormal distribution shown in Table 8-24.

Table 8-24 Fraction of Events Outside Containment for LOINV-SP, LOINV, and LOINV-LG

IE Fraction of Events Outside Containment	Mean Value	Error Factor	Median Value	95th
1-SD-LOINV-SP 1-SD-LOINV 1-SD-LOINV-LG	2.89E-01	2.41	2.56E-01	6.18E-01

With the approach outlined above, the frequency contribution for losses of inventory outside containment can be estimated. The actual frequency contribution is evaluated as part of the Level 2 PRA by transferring the identified sequences to the Level 2 PRA model and gathering the contributions as part of a release category representing a containment bypass release pathway to the auxiliary building through a failure in the RHR system.

8.3 Unavailability for Test and Maintenance

The unavailability of equipment for testing and maintenance is typically pre-planned and scheduled during refueling outages. Unlike normal power operations PRA modeling, where unavailability is based on average hours of unavailability for a component, the average hours of unavailability do not provide an adequate representation of the plant status because testing and maintenance is scheduled and coordinated to occur during specific states of the outage. The past reference plant's refueling outage reports and procedure for monitoring outage risk were reviewed to understand the plant-specific approach to managing testing and maintenance during shutdown.

The key equipment unavailabilities can vary from outage to outage. For this project, the unavailability of key systems and components was based on reference plant's experience with refueling outages from 2009 and later. According to discussions with reference plant staff, in 2009 they began a focused effort to reduce the overall time of their outages, and the evolutions performed have been consistent from outage to outage. Part of the approach to managing maintenance is the use of "protected" equipment when redundant equipment is out of service. The protected equipment denotes that the equipment will not be subjected to maintenance work and support equipment required to maintain operable status will be available. The key assumptions with respect to equipment unavailabilities are summarized below:

- Both trains of safety-related equipment (trains A and B) are protected until Mode 5 is reached.
- From Mode 5 until the reactor is defueled one train is protected and maintenance and testing can be performed on the other train. In this project the protected train is assumed to be train B, and the maintenance train is train A.
- While the reactor is defueled, the protected train is swapped to the opposite train.
- Maintenance and testing on the unprotected train of 120 vac vital instrument power and dc power can occur after entering Mode 5 and prior to entering mid-loop. This includes maintenance on the safety features sequencer system. Maintenance on the sequencer makes automatic actuation of the related train emergency diesel generator unavailable. However, the diesel can be operated locally. This condition is exited prior to entering mid-loop, according to the plant outage risk monitoring procedure and consistent with past outage practices.
- Most maintenance activities on the unprotected train electrical power, valves, and pumps begin after the refueling cavity is flooded.
- After refueling and after ESFAS testing is completed, both safety-related trains are protected. Both trains are protected through reinstallation of the reactor vessel head, mid-loop entry, and reactor start up to power generation.
- No consideration was given to correlated maintenance during shutdown. Correlated maintenance might have an impact on the results, though it is not expected to be significant since it would likely occur on a per train basis.

The implementation of these key assumptions is applied in the PRA model by developing flag sets that are used with the SAPHIRE PRA software. The flag sets are used to identify conditional processing of testing and maintenance basic events. The basic event calculation type settings are applied for different POS conditions. The basic events in the flag sets are either set to be “ignored” (I), meaning that the equipment is protected and will not be in maintenance, or “true” (T), meaning that the equipment is out of service for maintenance.

The flag sets are developed to address the unavailability of key systems and components during the modeled outage. The key safety systems and components that support key safety functions include:

- RCS loops
- RHR
- SI
- CVCS charging
- NSCW
- CCW
- SFPC
- Reserve auxiliary transformers (RATs)
- Standby auxiliary transformer (SAT)
- 1E electrical buses
- Diesel generators
- Containment
- Source range nuclear instrumentation
- RWST
- Fuel handling building (FHB) post-accident (PA) filtration

As the scope of this project is focused on the Level 1 PRA core damage risk associated with the reactor, the following systems/components are not addressed in the Level 1 flag sets: SFPC, containment, source range NIs, and FHB PA filtration. The other key systems and components are addressed in the flag sets. Also, the containment status is noted for each POS, and the impacts of the containment system unavailabilities are addressed in the Level 2 PRA for the reactor at-shutdown. For other modeled systems and components (e.g., non-safety related systems), the nominal at-power testing and maintenance unavailabilities are retained in the model.

A set of four flag sets were developed to model equipment unavailabilities in different POS conditions. The flag sets and their basic event settings are described below and in Table 8-25 through Table 8-28.

- Flag set name *1SD45ALORHR* is applied to POS 4 and POS 5A. This flag set models the impact of sequencer maintenance, which results in unavailability of automatic actuation of the train A emergency diesel generator.
- Flag set name *1SDM5LORHR* is developed for reduced inventory and mid-loop operation in POSs 5B, 6, 9, and 10. This flag set is applied when both safety-related trains are protected and no maintenance on key systems and components is being performed. This flag set is also applied to POSs that are not defined in any of the other flag set categories, and there were no routine POS-specific maintenance conditions identified in reviewing past outage experience.

- Flag set name *1SDPOS7LORHR* is applied to POS 7, where filling of the refueling cavity is underway and train B is protected. RHR train B is in operation for shutdown cooling. Maintenance on train A systems can begin during this POS. Some train A maintenance events (e.g., those related to NSCW train A) are set to “ignore”, but for sequences requiring train A equipment, a human failure event is modeled for failure to restore the equipment for operation.
- Flag set name *1SDPOS8LORHR* is applied to POSs 8E and 8L. This flag set is used to model Train B as the protected train, and Train A is under maintenance. The maintenance includes the normal train A offsite power source from the train A reserve auxiliary transformer. Although the protected train is usually swapped between POS 8E and POS 8L, the same flag set is applied to both POSs. No significant asymmetries between the trains were identified, so as a modeling simplification this flag set is used to model the impact of one train being protected and one train under maintenance.

Table 8-25 Flag Set 1SD45ALORHR Applied to POS 4 and 5A

Basic Event Name	Calc. Type	Basic Event Description	Notes and Justification
Both RHR trains operating. Sequencer train A in maintenance.			
1-ACP-BAC-MA-AA02	I	BUS 1AA02 IN MAINTENANCE	
1-ACP-BAC-MA-AB04	I	480V SWITCHGEAR 1AB04 IN MAINTENANCE	
1-ACP-BAC-MA-AB05	I	480V SWITCHGEAR 1AB05 IN MAINTENANCE	
1-ACP-BAC-MA-AB15	I	480V SWITCHGEAR 1AB15 IN MAINTENANCE	
1-ACP-BAC-MA-BA03	I	4160 V BUS 1BA03 IN MAINTENANCE	
1-ACP-BAC-MA-BB06	I	480V SWITCHGEAR 1BB06 IN MAINTENANCE	
1-ACP-BAC-MA-BB07	I	480V SWITCHGEAR 1BB07 IN MAINTENANCE	
1-ACP-BAC-MA-BB16	I	480V SWITCHGEAR 1BB16 IN MAINTENANCE	
1-ACP-BAC-MA-MCCABA	I	480V MCC 1ABA INMAINTENANCE	
1-ACP-BAC-MA-MCCABD	I	480V MCC 1ABD INMAINTENANCE	
1-ACP-BAC-MA-MCCABE	I	480V MCC 1ABE INMAINTENANCE	
1-ACP-BAC-MA-MCCBBA	I	480V MCC 1BBA INMAINTENANCE	
1-ACP-BAC-MA-MCCBBD	I	480V MCC 1BBD IN MAINTENANCE	
1-ACP-BAC-MA-MCCBBE	I	480V MCC 1BBE INMAINTENANCE	
1-ACP-INV-MA-AD1I11	I	INVERTER 1AD1I11 IN MAINTENANCE	
1-ACP-INV-MA-BD1I12	I	INVERTER 1BD1I12 IN MAINTENANCE	
1-ACP-INV-MA-CD1I5	I	INVERTER 1CD1I5 IN MAINTENANCE	
1-ACP-INV-MA-DD1I6	I	INVERTER 1DD1I6 IN MAINTENANCE	
1-AFW-MDP-MA-P4002	I	MDAFWP B (P4-002) UNAVAILABLE DUE TO T&M	
1-AFW-MDP-MA-P4003	I	MDAFWP A (P4-003) UNAVAILABLE DUE TO T&M	
1-AFW-TDP-MA-P4001	I	TDAFWP (P4-001) UNAVAILABLE DUE TO T&M	
1-CVC-MDP-MA-CCPA	I	CCP-A UNAVAILABLE DUE TO MAINTENANCE	
1-CVC-MDP-MA-CCPB	I	CCP-B UNAVAILABLE DUE TO MAINTENANCE	
1-CVC-MDP-TE-CCPA	I	CCP-A UNAVAILABLE DUE TO TEST	
1-CVC-MDP-TE-CCPB	I	CCP-B UNAVAILABLE DUE TO TEST	

Table 8-25 Flag Set 1SD45ALORHR Applied to POS 4 and 5A (cont.)

Basic Event Name	Calc. Type	Basic Event Description	Notes and Justification
1-DCP-BAT-MA-AD1B_____	I	BATTERY 1AD1B IN MAINTENANCE	
1-DCP-BAT-MA-BD1B_____	I	BATTERY 1BD1B IN MAINTENANCE	
1-DCP-BAT-MA-CD1B_____	I	BATTERY 1CD1B IN MAINTENANCE	
1-DCP-BAT-MA-DD1B_____	I	BATTERY 1DD1B IN MAINTENANCE	
1-DCP-BCH-MA-AD1CA	I	BATTERY CHARGER 1AD1CA IN MAINTENANCE	
1-DCP-BCH-MA-AD1CB	I	BATTERY CHARGER 1AD1CB IN MAINTENANCE	
1-DCP-BCH-MA-BD1CA	I	BATTERY CHARGER 1BD1CA IN MAINTENANCE	
1-DCP-BCH-MA-BD1CB	I	BATTERY CHARGER 1BD1CB IN MAINTENANCE	
1-DCP-BDC-MA-AD1	I	125V DC BUS 1AD1 IN MAINTENANCE	
1-DCP-BDC-MA-BD1	I	125V DC BUS 1BD1 IN MAINTENANCE	
1-DCP-BDC-MA-CD1	I	125V DC BUS 1CD1 IN MAINTENANCE	
1-DCP-BDC-MA-DD1	I	125V DC BUS 1DD1IN MAINTENANCE	
1-EPS-DGN-MA-G4001	T	DG1A IN MAINTENANCE	The impact of sequencer and DC maintenance assumes EDG A is unavailable.
1-EPS-DGN-MA-G4002	I	DG1B IN MAINTENANCE	
1-HE-LOCHS	T	Loss of Condenser heat sink occurs	
1-HE-LORHR	T	LOSS OF RHR during Shutdown	
1-HE-LPSD	T	Plant is in low power or shutdown state	
1-LPI-MDP-MA-RHRA	I	RHR PUMP A IN MAINTENANCE	
1-LPI-MDP-MA-RHRB	I	RHR PUMP B IN MAINTENANCE	
1-SIS-MDP-MA-SIA	I	SI PUMP A IN MAINTENANCE	
1-SIS-MDP-MA-SIB	I	SI PUMP B IN MAINTENANCE	
1-SWS-CTF-MA-_B_1234_	I	ALL FOUR NSCW TRAIN B TOWER FANS UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-_A_1234	I	ALL FOUR NSCW TRAIN A TOWER FANS UNAVAILABLE DUE TO MAINTENANCE	

Table 8-25 Flag Set 1SD45ALORHR Applied to POS 4 and 5A (cont.)

Basic Event Name	Calc. Type	Basic Event Description	Notes and Justification
1-SWS-FAN-MA-_B_1234	I	ALL FOUR NSCW TRAIN B TOWER FANS UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-1_F01	I	NSCW TOWER FAN 1-F01 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-1_F02	I	NSCW TOWER FAN 1-F02 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-1_F03	I	NSCW TOWER FAN 1-F03 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-1_F04	I	NSCW TOWER FAN 1-F04 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2_F01	I	NSCW TOWER FAN 2-F01 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2_F02	I	NSCW TOWER FAN 2-F02 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2_F03	I	NSCW TOWER FAN 2-F03 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2_F04	I	NSCW TOWER FAN 2-F04 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_001	I	NSCW PUMP P4-001 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_00135-3	I	ALL 3 NSCW TRAIN A PUMPS UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_002	I	NSCW PUMP P4-002 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_00246-3	I	ALL 3 NSCW TRAIN B PUMPS UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_003	I	NSCW PUMP P4-003 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_004	I	NSCW PUMP P4-004 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_005	I	NSCW PUMP P4-005 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_006	I	NSCW PUMP P4-006 UNAVAILABLE DUE TO MAINTENANCE	

Table 8-25 Flag Set 1SD45ALORHR Applied to POS 4 and 5A (cont.)

Basic Event Name	Calc. Type	Basic Event Description	Notes and Justification
1-SWS-MOV-MA-1669ACT	I	NSCW TR B SPRAY VALVE HV1669A CLOSED for CT MAINT.	
HE-U1	T	UNIT 1 EVENT HAS OCCURRED	

Table 8-26 Flag Set 1SDM5LORHR Applied to POS 5B, 6, 9, and 10

Basic Event Name	Calc. Type	Basic Event Description	Notes and Justification
Both trains protected. Both RHR trains operating. No significant maintenance.			
1-ACP-BAC-MA-AA02	I	BUS 1AA02 IN MAINTENANCE	
1-ACP-BAC-MA-AB04	I	480V SWITCHGEAR 1AB04 IN MAINTENANCE	
1-ACP-BAC-MA-AB05	I	480V SWITCHGEAR 1AB05 IN MAINTENANCE	
1-ACP-BAC-MA-AB15	I	480V SWITCHGEAR 1AB15 IN MAINTENANCE	
1-ACP-BAC-MA-BA03	I	4160 V BUS 1BA03 IN MAINTENANCE	
1-ACP-BAC-MA-BB06	I	480V SWITCHGEAR 1BB06 IN MAINTENANCE	
1-ACP-BAC-MA-BB07	I	480V SWITCHGEAR 1BB07 IN MAINTENANCE	
1-ACP-BAC-MA-BB16	I	480V SWITCHGEAR 1BB16 IN MAINTENANCE	
1-ACP-BAC-MA-MCCABA	I	480V MCC 1ABA INMAINTENANCE	
1-ACP-BAC-MA-MCCABD	I	480V MCC 1ABD INMAINTENANCE	
1-ACP-BAC-MA-MCCABE	I	480V MCC 1ABE INMAINTENANCE	
1-ACP-BAC-MA-MCCBBA	I	480V MCC 1BBA INMAINTENANCE	
1-ACP-BAC-MA-MCCBBD	I	480V MCC 1BBD IN MAINTENANCE	
1-ACP-BAC-MA-MCCBBE	I	480V MCC 1BBE INMAINTENANCE	
1-ACP-INV-MA-AD1I11	I	INVERTER 1AD1I11 IN MAINTENANCE	
1-ACP-INV-MA-BD1I12	I	INVERTER 1BD1I12 IN MAINTENANCE	
1-ACP-INV-MA-CD1I5	I	INVERTER 1CD1I5 IN MAINTENANCE	
1-ACP-INV-MA-DD1I6	I	INVERTER 1DD1I6 IN MAINTENANCE	
1-AFW-MDP-MA-P4002	I	MDAFWP B (P4-002) UNAVAILABLE DUE TO T&M	
1-AFW-MDP-MA-P4003	I	MDAFWP A (P4-003) UNAVAILABLE DUE TO T&M	
1-AFW-TDP-MA-P4001	I	TDAFWP (P4-001) UNAVAILABLE DUE TO T&M	
1-CVC-MDP-MA-CCPA	I	CCP-A UNAVAILABLE DUE TO MAINTENANCE	
1-CVC-MDP-MA-CCPB	I	CCP-B UNAVAILABLE DUE TO MAINTENANCE	
1-CVC-MDP-TE-CCPA	I	CCP-A UNAVAILABLE DUE TO TEST	
1-CVC-MDP-TE-CCPB	I	CCP-B UNAVAILABLE DUE TO TEST	

Table 8-26 Flag Set 1SDM5LORHR Applied to POS 5B, 6, 9, and 10 (cont.)

Basic Event Name	Calc. Type	Basic Event Description	Notes and Justification
1-DCP-BAT-MA-AD1B	I	BATTERY 1AD1B IN MAINTENANCE	
1-DCP-BAT-MA-BD1B	I	BATTERY 1BD1B IN MAINTENANCE	
1-DCP-BAT-MA-CD1B	I	BATTERY 1CD1B IN MAINTENANCE	
1-DCP-BAT-MA-DD1B	I	BATTERY 1DD1B IN MAINTENANCE	
1-DCP-BCH-MA-AD1CA	I	BATTERY CHARGER 1AD1CA IN MAINTENANCE	
1-DCP-BCH-MA-AD1CB	I	BATTERY CHARGER 1AD1CB IN MAINTENANCE	
1-DCP-BCH-MA-BD1CA	I	BATTERY CHARGER 1BD1CA IN MAINTENANCE	
1-DCP-BCH-MA-BD1CB	I	BATTERY CHARGER 1BD1CB IN MAINTENANCE	
1-DCP-BDC-MA-AD1	I	125V DC BUS 1AD1 IN MAINTENANCE	
1-DCP-BDC-MA-BD1	I	125V DC BUS 1BD1 IN MAINTENANCE	
1-DCP-BDC-MA-CD1	I	125V DC BUS 1CD1 IN MAINTENANCE	
1-DCP-BDC-MA-DD1	I	125V DC BUS 1DD1 IN MAINTENANCE	
1-EPS-DGN-MA-G4001	I	DG1A IN MAINTENANCE	
1-EPS-DGN-MA-G4002	I	DG1B IN MAINTENANCE	
1-HE-LORHR	T	LOSS OF RHR during Shutdown	
1-HE-LPSD	T	Plant is in low power or shutdown state	
1-LPI-MDP-MA-RHRA	I	RHR PUMP A IN MAINTENANCE	
1-LPI-MDP-MA-RHRB	I	RHR PUMP B IN MAINTENANCE	
1-SIS-MDP-MA-SIA	I	SI PUMP A IN MAINTENANCE	
1-SIS-MDP-MA-SIB	I	SI PUMP B IN MAINTENANCE	
1-SWS-CTF-MA-_A_1234_	I	ALL FOUR NSCW TRAIN A TOWER FANS UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-CTF-MA-_B_1234_	I	ALL FOUR NSCW TRAIN B TOWER FANS UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-_A_1234	I	ALL FOUR NSCW TRAIN A TOWER FANS UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-_B_1234	I	ALL FOUR NSCW TRAIN B TOWER FANS UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-1_F01	I	NSCW TOWER FAN 1-F01	

Table 8-26 Flag Set 1SDM5LORHR Applied to POS 5B, 6, 9, and 10 (cont.)

Basic Event Name	Calc. Type	Basic Event Description	Notes and Justification
		UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-1_F02	I	NSCW TOWER FAN 1-F02 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-1_F03	I	NSCW TOWER FAN 1-F03 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-1_F04	I	NSCW TOWER FAN 1-F04 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2_F01	I	NSCW TOWER FAN 2-F01 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2_F02	I	NSCW TOWER FAN 2-F02 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2_F03	I	NSCW TOWER FAN 2-F03 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2_F04	I	NSCW TOWER FAN 2-F04 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_001	I	NSCW PUMP P4-001 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_00135-3	I	ALL 3 NSCW TRAIN A PUMPS UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_002	I	NSCW PUMP P4-002 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_00246-3	I	ALL 3 NSCW TRAIN B PUMPS UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_003	I	NSCW PUMP P4-003 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_004	I	NSCW PUMP P4-004 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_005	I	NSCW PUMP P4-005 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_006	I	NSCW PUMP P4-006 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MOV-MA-1668ACT	I	NSCW TR A RETURN ISOLATION VALVE HV1668A CLOSED FOR CT MAINT.	
1-SWS-MOV-MA-	I	NSCW TR B SPRAY VALVE	

Table 8-26 Flag Set 1SDM5LORHR Applied to POS 5B, 6, 9, and 10 (cont.)

Basic Event Name	Calc. Type	Basic Event Description	Notes and Justification
1669ACT		HV1669A CLOSED for CT MAINT.	
HE-U1	T	UNIT 1 EVENT HAS OCCURRED	

Table 8-27 Flag Set 1SDPOS7LORHR Applied to POS 7

Basic Event Name	Calc. Type	Basic Event Description	Notes and Justification
1-ACP-BAC-MA-BA03	I	4160 V BUS 1BA03 IN MAINTENANCE	
1-ACP-BAC-MA-BB06	I	480V SWITCHGEAR 1BB06 IN MAINTENANCE	
1-ACP-BAC-MA-BB07	I	480V SWITCHGEAR 1BB07 IN MAINTENANCE	
1-ACP-BAC-MA-BB16	I	480V SWITCHGEAR 1BB16 IN MAINTENANCE	
1-ACP-BAC-MA-MCCBBA	I	480V MCC 1BBA INMAINTENANCE	
1-ACP-BAC-MA-MCCBBD	I	480V MCC 1BBD IN MAINTENANCE	
1-ACP-BAC-MA-MCCBBE	I	480V MCC 1BBE INMAINTENANCE	
1-ACP-INV-MA-BD1I12	I	INVERTER 1BD1I12 IN MAINTENANCE	
1-ACP-INV-MA-DD1I6	I	INVERTER 1DD1I6 IN MAINTENANCE	
1-CVC-MDP-MA-CCPB	I	CCP-B UNAVAILABLE DUE TO MAINTENANCE	
1-CVC-MDP-TE-CCPB	I	CCP-B UNAVAILABLE DUE TO TEST	
1-DCP-BAT-MA-BD1B	I	BATTERY 1BD1B IN MAINTENANCE	
1-DCP-BAT-MA-CD1B	I	BATTERY 1CD1B IN MAINTENANCE	Bus CD1 powers RHR train A iso valve HV-8701B - assume available until refuel cavity fill is complete.
1-DCP-BAT-MA-DD1B	I	BATTERY 1DD1B IN MAINTENANCE	
1-DCP-BCH-MA-BD1CA	I	BATTERY CHARGER 1BD1CA IN MAINTENANCE	
1-DCP-BCH-MA-BD1CB	I	BATTERY CHARGER 1BD1CB IN MAINTENANCE	
1-DCP-BDC-MA-BD1	I	125V DC BUS 1BD1 IN MAINTENANCE	
1-DCP-BDC-MA-CD1	I	125V DC BUS 1CD1 IN MAINTENANCE	Bus CD1 powers RHR train A iso valve HV-8701B - assume available until refuel cavity fill is complete.
1-DCP-BDC-MA-DD1	I	125V DC BUS 1DD1IN MAINTENANCE	Bus DD1 powers RHR train B iso valve HV-8702A.
1-EPS-DGN-MA-G4002	I	DG1B IN MAINTENANCE	EDG B protected.
1-HE-LORHR	T	LOSS OF RHR during Shutdown	
1-HE-LPSD	T	Plant is in low power or shutdown state	

Table 8-27 Flag Set 1SDPOS7LORHR Applied to POS 7 (cont.)

Basic Event Name	Calc. Type	Basic Event Description	Notes and Justification
1-HPI-XHE-XR-XVM207	I	OPERATORS FAIL TO RESTORE RWST XVM 207 AFTER T&M	Gravity fill from RWST is in progress. This fail to restore event is not applicable.
1-LPI-MDP-MA-RHRB	I	RHR PUMP B IN MAINTENANCE	
1-SIS-MDP-MA-SIB	I	SI PUMP B IN MAINTENANCE	
1-SWS-CTF-MA-A_1234_	I	ALL FOUR NSCW TRAIN A TOWER FANS UNAVAILABLE DUE TO MAINTENANCE	Impact of potential train A unavailability is modeled by requiring HFE to restore.
1-SWS-CTF-MA-B_1234_	I	ALL FOUR NSCW TRAIN B TOWER FANS UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-A_1234	I	ALL FOUR NSCW TRAIN A TOWER FANS UNAVAILABLE DUE TO MAINTENANCE	Impact of potential train A unavailability is modeled by requiring HFE to restore.
1-SWS-FAN-MA-B_1234	I	ALL FOUR NSCW TRAIN B TOWER FANS UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2_F01	I	NSCW TOWER FAN 2-F01 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2_F02	I	NSCW TOWER FAN 2-F02 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2_F03	I	NSCW TOWER FAN 2-F03 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2_F04	I	NSCW TOWER FAN 2-F04 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F0102_-2	I	NSCW TOWER FAN 2-F01 and F02 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F010203-3	I	NSCW TOWER FAN 2-F01, F02 and F03 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F010204-3	I	NSCW TOWER FAN 2-F01, F02 and F04 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F0103_-2	I	NSCW TOWER FAN 2-F01 and F03 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F010304-3	I	NSCW TOWER FAN 2-F01, F03 and F04 UNAVAILABLE DUE TO MAINTENANCE	

Table 8-27 Flag Set 1SDPOS7LORHR Applied to POS 7 (cont.)

Basic Event Name	Calc. Type	Basic Event Description	Notes and Justification
1-SWS-FAN-MA-2F0104__-2	I	NSCW TOWER FAN 2-F01 and F04 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F0203__-2	I	NSCW TOWER FAN 2-F03 and F03 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F020304-3	I	NSCW TOWER FAN 2-F02, F03 and F04 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F0204__-2	I	NSCW TOWER FAN 2-F03 and F04 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F0304__-2	I	NSCW TOWER FAN 2-F01 and F04 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_002	I	NSCW PUMP P4-002 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_00246-3	I	ALL 3 NSCW TRAIN B PUMPS UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_004	I	NSCW PUMP P4-004 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_006	I	NSCW PUMP P4-006 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MOV-MA-1669ACT	I	NSCW TR B SPRAY VALVE HV1669A CLOSED for CT MAINT.	
HE-U1	T	UNIT 1 EVENT HAS OCCURRED	

Table 8-28 Flag Set 1SDPOS8LORHR Applied to POS 8E and 8L

Basic Event Name	Calc. Type	Basic Event Description	Notes and Justification
1-ACP-BAC-MA-AA02	T	BUS 1AA02 IN MAINTENANCE	Train A in maintenance.
1-ACP-BAC-MA-BA03	I	4160 V BUS 1BA03 IN MAINTENANCE	Train B protected.
1-ACP-BAC-MA-BB06	I	480V SWITCHGEAR 1BB06 IN MAINTENANCE	
1-ACP-BAC-MA-BB07	I	480V SWITCHGEAR 1BB07 IN MAINTENANCE	
1-ACP-BAC-MA-BB16	I	480V SWITCHGEAR 1BB16 IN MAINTENANCE	
1-ACP-BAC-MA-MCCBBA	I	480V MCC 1BBA INMAINTENANCE	
1-ACP-BAC-MA-MCCBBD	I	480V MCC 1BBD IN MAINTENANCE	
1-ACP-BAC-MA-MCCBBE	I	480V MCC 1BBE INMAINTENANCE	
1-ACP-INV-MA-BD1I12	I	INVERTER 1BD1I12 IN MAINTENANCE	
1-ACP-INV-MA-DD1I6	I	INVERTER 1DD1I6 IN MAINTENANCE	
1-ACP-TFW-MA-NXRA__	T	RESERVE AUXILIARY XFMR 1NXRA IN MAINTENANCE	RAT train A in maintenance.
1-CVC-MDP-MA-CCPA	T	CCP-A UNAVAILABLE DUE TO MAINTENANCE	
1-CVC-MDP-MA-CCPB	I	CCP-B UNAVAILABLE DUE TO MAINTENANCE	
1-CVC-MDP-TE-CCPA	T	CCP-A UNAVAILABLE DUE TO TEST	
1-CVC-MDP-TE-CCPB	I	CCP-B UNAVAILABLE DUE TO TEST	
1-DCP-BAT-MA-BD1B__	I	BATTERY 1BD1B IN MAINTENANCE	
1-DCP-BAT-MA-DD1B__	I	BATTERY 1DD1B IN MAINTENANCE	
1-DCP-BCH-MA-BD1CA	I	BATTERY CHARGER 1BD1CA IN MAINTENANCE	
1-DCP-BCH-MA-BD1CB	I	BATTERY CHARGER 1BD1CB IN MAINTENANCE	
1-DCP-BDC-MA-BD1	I	125V DC BUS 1BD1 IN MAINTENANCE	
1-DCP-BDC-MA-DD1	I	125V DC BUS 1DD1IN MAINTENANCE	
1-EPS-DGN-MA-G4001	T	DG1A IN MAINTENANCE	EDG A in maintenance.
1-EPS-DGN-MA-G4002	I	DG1B IN MAINTENANCE	EDG B protected.
1-HE-LORHR	T	LOSS OF RHR during Shutdown	
1-HE-LPSD	T	Plant is in low power or shutdown state	
1-LPI-MDP-MA-RHRB	I	RHR PUMP B IN MAINTENANCE	

Table 8-28 Flag Set 1SDPOS8LORHR Applied to POS 8E and 8L (cont.)

Basic Event Name	Calc. Type	Basic Event Description	Notes and Justification
1-SIS-MDP-MA-SIA	T	SI PUMP A IN MAINTENANCE	
1-SIS-MDP-MA-SIB	I	SI PUMP B IN MAINTENANCE	
1-SWS-CTF-MA-_B_1234_	I	ALL FOUR NSCW TRAIN B TOWER FANS UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-_B_1234	I	ALL FOUR NSCW TRAIN B TOWER FANS UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2_F01	I	NSCW TOWER FAN 2-F01 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2_F02	I	NSCW TOWER FAN 2-F02 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2_F03	I	NSCW TOWER FAN 2-F03 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2_F04	I	NSCW TOWER FAN 2-F04 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F0102__-2	I	NSCW TOWER FAN 2-F01 and F02 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F010203-3	I	NSCW TOWER FAN 2-F01, F02 and F03 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F010204-3	I	NSCW TOWER FAN 2-F01, F02 and F04 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F0103__-2	I	NSCW TOWER FAN 2-F01 and F03 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F010304-3	I	NSCW TOWER FAN 2-F01, F03 and F04 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F0104__-2	I	NSCW TOWER FAN 2-F01 and F04 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F0203__-2	I	NSCW TOWER FAN 2-F03 and F03 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F020304-3	I	NSCW TOWER FAN 2-F02, F03 and F04 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-FAN-MA-2F0204__-2	I	NSCW TOWER FAN 2-F03 and F04 UNAVAILABLE DUE TO MAINTENANCE	

Table 8-28 Flag Set 1SDPOS8LORHR Applied to POS 8E and 8L (cont.)

Basic Event Name	Calc. Type	Basic Event Description	Notes and Justification
1-SWS-FAN-MA-2F0304__-2	I	NSCW TOWER FAN 2-F01 and F04 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_002	I	NSCW PUMP P4-002 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_00246-3	I	ALL 3 NSCW TRAIN B PUMPS UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_004	I	NSCW PUMP P4-004 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MDP-MA-P4_006	I	NSCW PUMP P4-006 UNAVAILABLE DUE TO MAINTENANCE	
1-SWS-MOV-MA-1669ACT	I	NSCW TR B SPRAY VALVE HV1669A CLOSED for CT MAINT.	
HE-U1	T	UNIT 1 EVENT HAS OCCURRED	

8.4 RHR Restoration and Repair Data

Losses of RHR can occur from several different causes. The time required for restoring the RHR depends on the specific cause. Operational data documented in EPRI reports (EPRI-1003113 (EPRI, 2001) and EPRI-1021176 (EPRI, 2010)) is used to identify different causes of RHR failures and estimate time for RHR recovery. Five failure cause categories were identified by reviewing the operational data:

- Loss of RHR due to air or nitrogen ingestion into RHR line
- Loss of RHR due to isolation
- Loss of RHR due to electrical faults
- Loss of RHR due to CCW isolation
- Loss of RHR due to RHR pump failures

The failure categories and associated times to restore RHR cooling are summarized below. The RHR pump failures are considered the most limiting category, that is, these failures tend to take the longest time to restore the RHR function. The repair time for RHR pumps is used as the basis for modeling the non-recovery probabilities in the model.

Restoration time after air or nitrogen is ingested into RHR line:

The restoration times were recorded for three events: 19, 37.8, and 50 minutes. In some cases the time is not recorded, and in other cases RHR function was established by aligning the standby train. The average time to restoration from the three recorded events is estimated to be 35.6 minutes, with a standard deviation of 12.8 minutes.

Restoration time for losses of RHR due to isolation:

There are 23 events where an RHR train was secured and then was returned to service by aligning the valves and starting the pumps. In some cases, the RHR was isolated in order to stop leakage through an open RHR relief valve. The average time to restoration is estimated to be 11.2 minutes, with a standard deviation of 12.2 minutes.

Restoration time for losses of RHR due to electrical faults unrelated to SBO:

There are 28 events where a restoration duration was recorded for a loss of RHR due to an electrical fault unrelated to an SBO. The average time to restoration is estimated to be 6.5 minutes, with a standard deviation of 11.4 minutes.

Restoration time for losses of RHR due to isolation of CCW requiring valve alignment:

There are nine restoration times for losses of RHR due to loss of support systems, especially isolation of CCW, recorded in the database. The restoration time includes the time for both diagnosis and execution. Some events included local manual opening of the valves and had a slightly longer duration. The average time to restoration is estimated to be 16.3 minutes, with a standard deviation of about 10.6 minutes.

In addition to the nine events used for estimating the average restoration time, the database contains two more events which were excluded from the average. One event took 690 minutes (11.5 hours). In this event, the CCW configuration and alignment was so unexpected that it took a long time to figure out what needed to be done. The other event took 270 minutes (4.5 hours). This event involved special alignment and diagnostic testing of a decay heat removal discharge valve.

If the two events described above were not excluded from the calculation, then the average time to restoration would be estimated to be 97.5 minutes, with a standard deviation of about 201 minutes. This difference does not impact the results of the project. Even with the two outlier events included, the average restoration time is still significantly less than the time to repair and restore a failed RHR pump, which is discussed below. The repair time for a failed RHR pump bounds the restoration time for all other system failure mechanisms.

Repair time for RHR pump:

The review of plant-specific shutdown initiating events and EPRI-1003113 (EPRI, 2001) recorded three events involving damage to at least one RHR pump. These were:

- High vibration on one RHR pump
- An RHR pump failed 7 minutes after starting; casing fused to impeller
- Both RHR pumps inboard bearing temperature exceeded the alarm set point and stopped

These three events were recorded as non-recoverable. Only one event involved loss of both RHR pumps (the last one, with the high bearing temperature alarm). The other two events involved the failure of one RHR pump, and the standby pump was not affected and could have

been brought back to service when needed. In reviewing this event, it was found that the one RHR pump motor bearing had to be replaced during a refueling outage. This repair took about 2 days (48 hours).

In a 2012 document entitled "Review of Maintenance and Repair Times for Components in Technological Facilities," INL/EXT-12-27734 (Cadwallader, 2012), Table 5 reports the minimum, mean, and maximum repair times for an RHR train as 2, 77, and 300 hours, respectively. This report contains older data which are publicly available.

The review of RHR recovery operating experience provides insights about the time required to restore RHR cooling after a system failure.

- Most loss of RHR events can be recovered within around 30 minutes or less, as shown by the review of operating experience.
- Restoration of RHR cooling is dependent on successful human actions. HFEs are used to model the recovery
- Failures requiring repair are not recoverable as repair times would likely exceed the time available for restoring RHR cooling.

Based on this review the following modeling assumptions are applied:

- RHR cooling is assumed to be capable of being restored after an initiating event causing a loss of RHR cooling.³⁴
- Human failure events are developed to represent failure of actions to restore RHR cooling. If initial human actions fail, then long-term actions are credited (see 1-RHR-XHE-XL-LTREC and 1-RHR-XHE-XL-LTREC-LT5H) if conditions to restart RHR are established based on procedural direction.
- Restoration of RHR cooling is not credited if RHR equipment failures (e.g., common-cause failure of RHR pumps) occur during the accident sequence mission time. Such failures are expected to require repairs that would extend beyond the mission time.

³⁴ A review of loss of RHR cooling initiating event data supports the assumption that restoration of RHR cooling is possible. The loss of RHR cooling initiating event data are discussed in Section 8.1.1.

9 MODEL QUANTIFICATION

9.1 Quantification Approach

The objective of this section is to describe the LPSD Level 1 PRA results. The results of the modeled shutdown accident scenarios are presented here for a unit 1 refueling outage of the reference plant. The model was developed and quantified using the NRC's SAPHIRE software (Smith, 2011). The shutdown accident scenarios were quantified on October 18, 2021, using model version 419 and SAPHIRE version 8.2.4. The truncation level used for quantification was 10^{-12} per year. The minimal cut set upper bound method was used for quantifying the cut set results.

The scope of the model quantification is limited to shutdown accident scenarios for a refueling outage. Other low power and shutdown states are discussed throughout the report but are not quantified. The quantified results represent the risks of the modeled scenarios, but they do not give a complete representation of all low power and shutdown risks. The scope of the model quantification is reiterated here in terms of the modeled outage types, plant operating states, initiating events, and accident scenarios.

Modeled Plant Outage Types: The modeled plant outage type is a refueling outage (POT-4). Other plant outage types (POT-1, POT-2, and POT-3) are not modeled but are discussed as model uncertainties in Section 10.3. The plant outage types are defined in Section 2.3.

Modeled Plant Operating States: The modeled plant operating states include POSs 3, 4, 5A, 5B, 6, 7, 8E, 8L, 9, 10, 11, 12, and 13. These states represent shutdown operations during Technical Specification Modes 4, 5, and 6. The low power and hot standby states are not modeled but are discussed as model uncertainties in Section 10.3.

Modeled Initiating Events: The modeled initiating events include the six shutdown initiating event categories identified in Table 4-3:

- Loss of RHR cooling
- Loss of inventory
- Overdrain
- Loss of electrical power to a class 1E 4,160V AC bus
- Loss of offsite power
- Loss of nuclear service cooling water

The loss of inventory event category is divided into four subcategories:

- SD-LOINV-SP: Small spray loss of inventory event used with POSs with RCS vented (POSs 5B, 6, 9, and 10)
- SD-LOINV: Loss of inventory used with POSs with RCS vented (POSs 5B, 6, 9, and 10)
- SD-LOINV-OP: Loss of inventory with RCS at pressure (modeled for POS 5A)
- SD-LOINV-LG: Large loss of inventory used with POSs with refueling cavity flooded (POSs 7, 8E, and 8L)

The overdrain initiating event category is divided into three subcategories:

- SD-ODML-EARLY: Overdrain entering mid-loop prior to refueling (POS 6)
- SD-ODML-LATE: Overdrain entering mid-loop after refueling (POS 10)
- SD-LOINV-DS: Loss of inventory due to inadvertent displacement (POSs 5B and 9)

Other shutdown initiating event categories are discussed qualitatively in Section 3 but are not modeled.

Modeled Accident Scenarios: The shutdown accident scenarios include the initiating event, the plant operating state conditions, and the plant response to the event. A shutdown accident scenario prioritization scheme was developed to focus the accident sequence analysis work on the most risk-significant conditions and accidents. The results of the accident scenario selection are shown in Table 4-2. Each modeled scenario is represented by an event tree, as described in Section 4.3. Each scenario can comprise several accident sequences, which involve different combinations of operator errors and/or mitigating system failures resulting in core damage. Each accident sequence represents a unique event tree branch and end state in the model.

The LPSD Level 1 PRA results are provided and discussed in the following sections:

Section 9.2 – Core damage frequency (CDF) results obtained for each of the modeled initiating event categories

Section 9.3 – CDF results by plant operating state

Section 9.4 – CDF results for significant shutdown accident sequences

Section 9.5 – Review of significant cutsets

Section 9.6 – Basic event importances

Section 9.7 – CDF truncation limit and convergence of the results

Section 9.8 – Comparison of results with other LPSD PRA studies for similar plants

Uncertainty analyses associated with the results are provided in Section 10.

9.2 CDF Results by Initiating Event Category

The estimated CDF results organized by initiating event category are shown in Table 9-1 and Figure 9-1.

The highest contributing initiating event category is SD-LOOP, loss of offsite power events, at 68.1 percent of the total shutdown CDF. This is similar to the L3PRA project at-power model results, which show LOOP events are a dominant contributor to the at-power, internal events risk (NRC, 2022). The shutdown LOOP initiating event frequency is larger than the corresponding at-power LOOP initiating event frequency. The relatively high LOOP initiating event frequency, as well as the modeling assumptions about LOOP recovery actions (see Section 4.3.5), lead to the SD-LOOP being a significant contributor to shutdown CDF.

Although the contribution to shutdown CDF from LOOP events is large, the contribution from consequential LOOP events occurring in response to non-LOOP initiating events is small. The CDF contribution from consequential LOOP events is 1.1×10^{-7} per year, about 0.9 percent of the total shutdown CDF. The LPSD model uses the same probability of a consequential LOOP for transient-type initiating events as used in the at-power Level 1 PRA model (NRC, 2022).

The next highest contribution to shutdown CDF is associated with the SD-ODML-EARLY event (overdrain entering mid-loop). This event is only applicable when entering the mid-loop condition in POS 6. The POS conditions involve reduced RCS inventory, and consequently short time available to restore RHR. Human failure events in this scenario have high failure probabilities and are important contributors to the CDF.

The SD-LOINV-OP initiating event contributes to 6.2 percent of the shutdown CDF. The initiating event models the loss of RCS inventory due to an over-pressure condition causing relief valves to open. Although this event is only applicable during POS conditions with the RCS intact, it has a significant contribution to the overall shutdown CDF. The relatively high initiating event frequency, which is based on review of recent operating experience (see Section 8.1.2), and significant human failure events are important contributors to this scenario.

9.3 CDF Results by Plant Operating State

The CDF results include factors accounting for the fraction of time spent in the outage and in each POS. The CDF results are calculated by multiplying the POT duration fraction and POS duration fraction to estimate the average CDF contribution per year. The POT and POS fractions are included in the model cut sets and their impacts are accounted for in the model quantification. The assumptions made about POT and POS durations are summarized in Table 9-2 and Table 9-3. The impact of these fractions is illustrated in the example below.

$$[\text{CDF per year for POS-IE scenario}] = [\text{POT fraction}] \times [\text{POS fraction}] \times [\text{IE frequency}] \times [\text{CCDP}]$$

A specific example is show here:

$$\begin{aligned} [\text{CDF per year for POS 5B LORHR}] = & [\text{POT-4 fraction from Table 9-2}] \times \\ & [\text{POS 5B fraction from Table 9-3}] \times \\ & [\text{IE frequency shown in Table 9-1}] \times \\ & [\text{CCDP for POS 5B LORHR scenario from Table 9-7}] \end{aligned}$$

$$\begin{aligned} [\text{CDF per year for POS 5B LORHR}] &= 5.5 \times 10^{-2} \times 5.5 \times 10^{-2} \times 3.7 \times 10^{-1} \text{ per year} \times 8.8 \times 10^{-5} \\ &= 9.9 \times 10^{-8} \text{ per year} \end{aligned}$$

The scope of the results presented here are limited to only POT-4 refueling outages. The POT-4 fraction is the only one that is used in the model quantification. The other POT duration fractions are included in Table 9-2. The sum of all POT fractions gives the total fraction of annual hours in outage, which is 0.069. The fraction of time at full power operations can then be calculated by subtracting the summed POT fractions from 1.0.

Table 9-1 CDF Results by Initiating Event Category

IE Category	IE Frequency per shutdown-year	CDF (per year) ⁽¹⁾	% Contr. to CDF	Cut Set Count
1-SD-LOOP Loss of Offsite Power	1.6E-01	8.5E-06	68.1%	14011
1-SD-ODML-EARLY ⁽²⁾ Overdrain Mid-loop	2.4E-03	1.2E-06	9.8%	417
1-SD-LOINV-OP Loss of Inventory at pressure	6.1E-01	7.8E-07	6.2%	432
1-SD-LORHR Loss of RHR	3.7E-01	6.7E-07	5.4%	1852
1-SD-LONSCW Loss of Service Water	3.5E-05	6.4E-07	5.1%	164
1-SD-LO4160V Loss of power to 4,160V AC bus	3.4E-01	3.7E-07	3.0%	2842
1-SD-ODML-LATE ⁽²⁾ Overdrain Mid-loop	2.1E-03	2.3E-07	1.8%	241
1-SD-LOINV Loss of Inventory	6.4E-05	4.2E-08	0.3%	69
1-SD-LOINV-DS Loss of Inventory Displacement	2.1E-02	2.0E-08	0.2%	120
1-SD-LOINV-SP Loss of Inventory Spray	3.0E-03	1.8E-08	0.1%	196
1-SD-LOINV-LG Loss of Inventory Large	1.2E-05	1.3E-10	< 0.1%	13
Total =	1.5	1.3E-05	100%	20357
Note (1): The CDF values in this table are reported as “per year,” since they already account for the fraction of the year that the plant is in a particular POS (as further discussed in Section 9.3). As such, the total CDF (1.3×10^{-5}/year) can be added to the at-power internal event CDF (after adjusting for the plant availability factor) to arrive at an estimate of total internal event CDF for at-power and shutdown modes of plant operation. Note (2): IE frequency for the over drain events (1-SD-ODML-EARLY and 1-SD-ODML-LATE) are the product of the frequency of entering mid-loop during the applicable POS per year and the conditional probability of over-draining RCS level when entering mid-loop, for POS 6: $0.59 / \text{year} \times 4.1 \times 10^{-3} = 2.4 \times 10^{-3} / \text{year}$, for POS 10: $0.51 / \text{year} \times 4.1 \times 10^{-3} = 2.1 \times 10^{-3} / \text{year}$.				

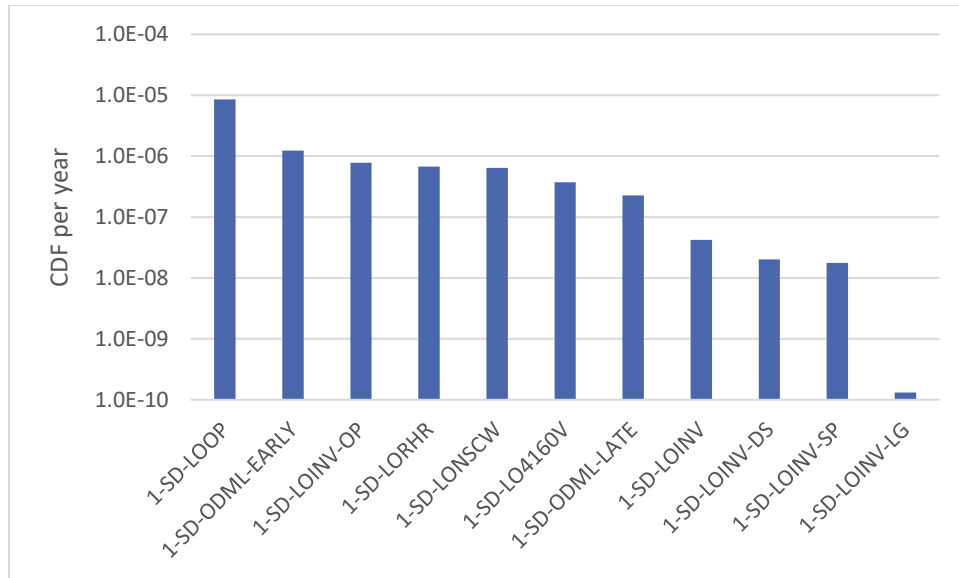


Figure 9-1 CDF Results by Initiating Event Category

Table 9-2 Summary of POT Durations

Plant Outage Type	Defined POT Duration (hours)	Annual Frequency of POT	Fraction of Annual Hours in POT
POT-1	72	1.0	8.2E-03
POT-2	246	0.1	2.8E-03
POT-3	246	0.1	2.8E-03
POT-4 Refueling Outage	726	0.67	5.5E-02
Total fraction of annual hours in outage			6.9E-02
POS 0 At-Power Plant Availability			9.3E-01

Table 9-3 Summary of POS Durations (for POT-4)

Shutdown POS	Defined POS Duration (hours)	Modeled POS Duration (hours)	Fraction of Modeled POS Duration to POT Duration
POS 1 Low power	6	--	not modeled
POS 2 Cooldown with SGs	5	--	not modeled
POS 3 Mode 4 cooldown	3	3	4.13E-03
POS 4 Mode 5 cooldown	4	4	5.51E-03
POS 5A Pressurizer solid	30	30	4.13E-02
POS 5B Draining RCS	40	40	5.51E-02
POS 6 Mid-loop (pre-refuel)	14	14	1.93E-02
POS 7 Filling refuel cavity	45	15 ⁽¹⁾	2.07E-02
POS 8E Refuel cavity flooded (pre-refuel)	68	68	9.37E-02
Defueled	120	--	not modeled
POS 8L Refuel cavity flooded (post-refuel)	52	52	7.16E-02
POS 9 Draining refuel cavity	120	30 ⁽¹⁾	4.13E-02
POS 10 Mid-loop (post-refuel)	25	25	3.44E-02
POS 11 Vacuum fill RCS	24	24	3.31E-02
POS 12 Mode 5 heatup	54	54	7.44E-02
POS 13 Mode 4 heatup	20	20	2.75E-02
POS 14 Startup with SGs	40	--	not modeled
POS 15A Reactor start-up and low power	20	--	not modeled
POS 15B Reactor start-up and low power	36	--	not modeled
Total for POT-4 Refueling Outage	726	379	5.22E-01
Note (1): These POSs cover a variety of conditions, and the modeled POS represents a subset of the POS conditions. Consequentially, the modeled POS duration was reduced to the estimated time that the modeled POS conditions are in place. Refer to Section 4.4.6 (POS 7) and Section 4.4.9 (POS 9) for discussion of the POS modeling assumptions.			

The CDF results for each POS are presented in Table 9-4 and Figure 9-2. The initiating event frequencies calculated per year of shutdown operation are shown in the table. The CDF results are calculated in units per year (i.e., they account for the time spent in the POS). The far-right column in Table 9-4 includes the estimated time to core damage for each POS. The time to core damage calculations are discussed Section 5.3. The calculations are based on a LORHR scenario and assume no operator intervention. These times are provided as a reference point for the variation in the time available for mitigation in different POSs. The specific time to core damage can vary for each accident sequence depending on the successes and failures of mitigating functions modeled in the sequence. For example, the loss of inventory scenarios generally have less time available than the reference time noted in the table. The loss of inventory events impact the RHR cooling function and RCS inventory is being lost. The combined impacts result in less time for mitigation and human actions.

The modeling assumptions regarding the plant configuration and equipment availability have significant influence on the POS CDF results. These assumptions should be considered when comparing the results of different POSs. A summary of key assumptions is provided here.

- During POS 4 and POS 5A, automatic actuation of one EDG is unavailable due to sequencer testing and maintenance.
- During POS 8E and POS 8L, one train of safety-related equipment is assumed unavailable while one reserve auxiliary transformer (RAT) is in maintenance.
- For LOOP events, the restoration of offsite power is limited to a maximum of two hours from the time of the initiating event. Although some POSs have long times available prior to core damage (e.g., POSs 7, 8E, and 8L), the assumptions about power restoration limit the credit that can be taken for the additional available time.
- The LOINV column in Table 9-4 includes all four LOINV initiating event categories as defined in Section 4.3.2. The four categories have different assumptions about the flow rates associated with the losses of inventory. These different conditions should be considered when comparing the results for different POSs.
 - During POS 5A, with RCS pressure at 350 psig, the modeled flow rate is 800 gallons per minute.
 - During POSs 5B, 6, 9, and 10, with RCS vented and at reduced inventory, the modeled flow rate for loss of inventory is 1,000 gallons per minute.
 - During POSs 5B, 6, 9, and 10, with RCS vented and at reduced inventory, the modeled flow rate for spray events is 100 gallons per minute and the initiating event frequency is much higher than the losses with larger break flow rates.
 - During POSs 7, 8E, and 8L, a conservative flow rate is modeled corresponding to the typical RHR pump flow rate of 3,500 gallons per minute.
 - For the POSs where multiple LOINV categories are evaluated, the CDF contributions are summed.

- Establishing forced injection (via safety injection or charging pumps) with a large RCS vent (i.e., “feed and bleed” cooling) is not credited as a means of reaching a safe, stable state. Restoration of RHR cooling or steam generator cooling (if available) is required to reach a safe and stable state.

The contribution of each POS is also influenced by the duration of the POS. As POS durations can vary due to plant-specific and outage-specific outage activities, planning, and management approach, the results should be interpreted considering these POT and POS durations. The assumptions made about POT and POS durations are summarized in Section 2.6. The resulting POT and POS fractions are shown in Table 9-2 and Table 9-3. As mentioned previously, the POT and POS fractions are included in the quantification results that are shown in Table 9-1 and Table 9-4.

From the results in Table 9-4, the most significant POSs in terms of CDF contribution are the two POSs where the refuel cavity is flooded (POS 8E and POS 8L), which contribute approximately 26 percent and 19 percent, respectively. While a flooded refuel cavity affords the operators more time to respond to an initiating event, the risk significance of these two POSs is amplified by their relatively long durations and the maintenance configurations, which limit the equipment available for accident mitigation.

Other significant contributing POSs include POS 5A (pressurizer solid) and POS 6 (mid-loop operations). POS 5A contributes approximately 16 percent to the shutdown CDF. The POS 5A results are related to two key modeling choices. First, the SD-LOINV-OP initiating event frequency is higher for this POS than for other POSs, based on recent operating experience during POSs with the RCS pressurized (see Section 8.1.2 for discussion of initiating event frequencies). Second, the loss of offsite power events are significant contributors during POS 5A due to the assumption that sequencer maintenance prevents the automatic start of the train A emergency diesel generator. POS 6 contributes 13 percent to the shutdown CDF. The dominant contributors to POS 6 include operator failures, which are influenced by the short time available for responding to events while in the mid-loop condition.

The shutdown CDF results in Table 9-4 are reported in units of CDF per year, which are appropriate units for comparison to CDF results from other operating states and hazard categories. The total shutdown CDF of 1.3×10^{-5} per year is comparable to the total at-power CDF for internal events of approximately 5.9×10^{-5} per year (based on 6.4×10^{-5} per reactor critical year, as documented in (NRC, 2022), and scaled by the plant availability factor of 0.93, as shown in Table 9-2 above). Caution should be exercised when comparing risk results from different parts of the L3PRA project. Differences in scope and key assumptions influence the results, and these differences should be considered when comparing results.

Table 9-4 shows that there are several accident scenarios that are not modeled (as denoted by note 9). The selection of shutdown accident scenarios was informed by the prioritization approach discussed in Section 4.2.2. The unmodeled scenarios are not expected to be significant contributors to the shutdown CDF. Potential CDF contributions from the unmodeled scenarios are discussed in a series of sensitivity studies in Section 10.3.

Another important attribute to consider in evaluating the shutdown CDF results is the containment status during plant operating states. There may be large containment openings (e.g., open containment equipment hatch) during many of the shutdown POSs. The initial containment status is an important input to assessing potential radiological releases. Table 9-5

shows the shutdown CDF results with the initial containment status for each POS. The open containment status refers to a POS where the containment equipment hatch and containment personnel airlock are open. A significant fraction of the total shutdown CDF (80 percent) occurs during POSs with the containment initially open. The shutdown CDF results of this Level 1 PRA model are carried forward to analyze release frequencies in the shutdown Level 2 PRA, where actions for closing containment penetrations are modeled.

Table 9-4 CDF Results by Plant Operating State

IE Categories: IE Frequency: (per shutdown- year) Shutdown POS	LORHR	LOINV	ODML / LOINV-DS	LO4160V	LOOP	LONSCW	Total POS CDF (per year)	% Contr. to CDF	Time to CD (hr.)
	3.7E-1	6.1E-1 ⁽¹⁾ 3.0E-3 ⁽²⁾ 6.4E-5 ⁽³⁾ 1.2E-5 ⁽⁴⁾	2.4E-3 ⁽⁵⁾ 2.1E-3 ⁽⁶⁾ 2.1E-2 ⁽⁷⁾	3.4E-1	1.6E-1	3.5E-5 ⁽⁸⁾			
POS 3 Mode 4 cooldown	< 1E-12	(Note 9)	N/A	(Note 9)	2.3E-08	(Note 9)	2.3E-08	0.2%	(Note 10)
POS 4 Mode 5 cooldown	2.4E-12	(Note 9)	N/A	(Note 9)	1.7E-07	(Note 9)	1.7E-07	1.3%	(Note 10)
POS 5A Pressurizer solid	3.9E-11	7.8E-07	N/A	(Note 9)	1.3E-06	(Note 9)	2.0E-06	16.3%	27
POS 5B Draining RCS	9.9E-08	2.2E-08	1.2E-08	2.1E-08	4.9E-07	1.1E-07	7.5E-07	5.9%	5.3
POS 6 Mid-loop (pre-refuel)	1.3E-07	7.7E-09	1.2E-06	1.4E-08	2.2E-07	3.7E-08	1.6E-06	13.0%	3.3
POS 7 Filling refuel cavity	2.7E-08	1.2E-11	N/A	1.7E-08	1.6E-07	(Note 9)	2.0E-07	1.6%	52
POS 8E Refuel cavity flooded (pre-refuel)	1.4E-07	1.1E-10	N/A	1.8E-07	2.7E-06	1.9E-07	3.3E-06	26.0%	90
POS 8L Refuel cavity flooded (post-refuel)	1.0E-07	1.3E-11	N/A	1.0E-07	2.1E-06	1.5E-07	2.4E-06	19.3%	214
POS 9 Draining refuel cavity	5.6E-08	1.7E-08	8.6E-09	1.6E-08	2.8E-07	8.5E-08	4.7E-07	3.7%	15
POS 10 Mid-loop (post-refuel)	7.4E-08	1.4E-08	2.3E-07	2.5E-08	3.1E-07	7.1E-08	7.2E-07	5.8%	11
POS 11 Vacuum fill RCS	4.6E-08	(Note 9)	N/A	(Note 9)	2.3E-07	(Note 9)	2.7E-07	2.2%	(Note 10)
POS 12 Mode 5 heatup	6.4E-12	(Note 9)	N/A	(Note 9)	4.2E-07	(Note 9)	4.2E-07	3.4%	177
POS 13 Mode 4 heatup	1.3E-12	(Note 9)	N/A	(Note 9)	1.6E-07	(Note 9)	1.6E-07	1.2%	(Note 10)
Total:	6.7E-07	8.4E-07	1.5E-06	3.7E-07	8.5E-06	6.4E-07	1.3E-05		

Table 9-4 CDF Results by Plant Operating State (cont.)

IE Categories:	LORHR	LOINV	ODML / LOINV-DS	LO4160V	LOOP	LONSCW	Total POS CDF (per year)	% Contr. to CDF	Time to CD (hr.)
IE Frequency: (per shutdown- year)	3.7E-1	6.1E-1 ⁽¹⁾ 3.0E-3 ⁽²⁾ 6.4E-5 ⁽³⁾ 1.2E-5 ⁽⁴⁾	2.4E-3 ⁽⁵⁾ 2.1E-3 ⁽⁶⁾ 2.1E-2 ⁽⁷⁾	3.4E-1	1.6E-1	3.5E-5 ⁽⁸⁾			
Shutdown POS									
Note (1) Loss of inventory over-pressure event frequency 6.1×10^{-1}/shutdown-year is applied in POS 5A. Note (2) Loss of inventory spray event frequency 3.0×10^{-3}/shutdown-year is applied in POS 5B, 6, 9, and 10. Note (3) Loss of inventory event frequency 6.4×10^{-5}/shutdown-year is applied in POS 5B, 6, 9, and 10. Note (4) Loss of inventory initiating event frequency 1.2×10^{-5}/shutdown-year is applied in POS 7, 8E, and 8L. Note (5) Frequency of mid-loop entry in POS 6 multiplied by conditional probability of overdrain. Note (6) Frequency of mid-loop entry in POS 10 multiplied by conditional probability of overdrain. Note (7) Loss of inventory due to displacement frequency 2.1×10^{-2}/shutdown-year is applied in POS 5B and 9. Note (8) Frequency calculated by fault tree is approximately 3.5×10^{-5}/shutdown-year, but varies by POS. Note (9) This accident scenario was not modeled. Note (10) Time to core damage was not evaluated for this POS (Section 5.3 describes timing assumptions).									

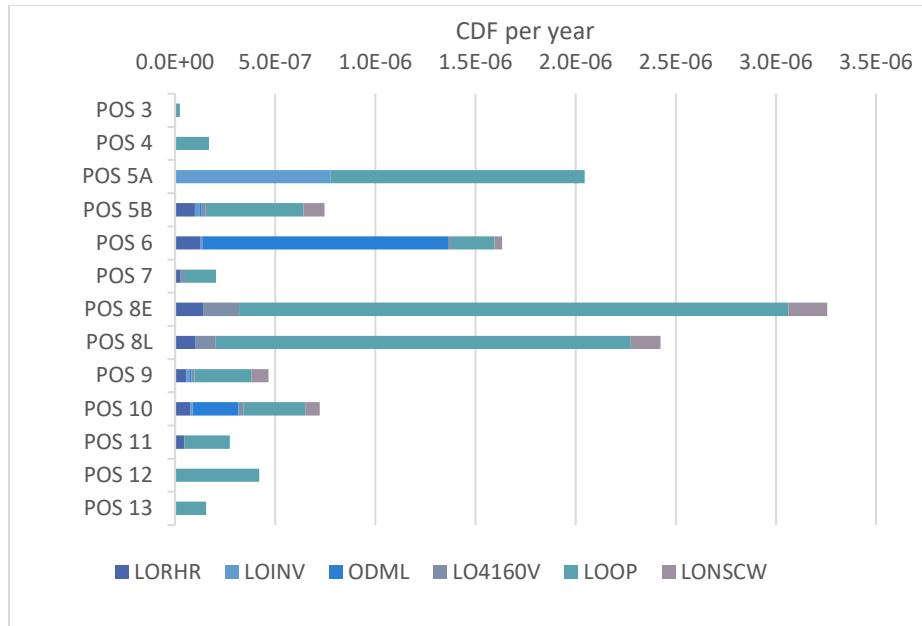


Figure 9-2 CDF Results by Plant Operating State

Table 9-5 CDF by Plant Operating State with Containment Status

Plant Operating State	Plant Operating State Description	CDF (per year)	% Contr. to CDF	Containment Status (Note 1)	CDF Cont. Open (per year)	% Contr. to CDF
POS 3	Mode 4 cooldown	2.3E-08	0.2%	Closed		
POS 4	Mode 5 cooldown	1.7E-07	1.3%	Open	1.7E-07	1.3%
POS 5A	Pressurizer solid	2.0E-06	16.3%	Open	2.0E-06	16.3%
POS 5B	Draining RCS	7.5E-07	5.9%	Open	7.5E-07	5.9%
POS 6	Mid-loop (pre-refuel)	1.6E-06	13.0%	Closed		
POS 7	Filling refuel cavity	2.0E-07	1.6%	Open	2.0E-07	1.6%
POS 8E	Refuel cavity flooded (pre-refuel)	3.3E-06	26.0%	Open	3.3E-06	26.0%
POS 8L	Refuel cavity flooded (post-refuel)	2.4E-06	19.3%	Open	2.4E-06	19.3%
POS 9	Draining refuel cavity	4.7E-07	3.7%	Open	4.7E-07	3.7%
POS 10	Mid-loop (post-refuel)	7.2E-07	5.8%	Closed		
POS 11	Vacuum fill RCS	2.7E-07	2.2%	Open	2.7E-07	2.2%
POS 12	Mode 5 heatup	4.2E-07	3.4%	Open	4.2E-07	3.4%
POS 13	Mode 4 heatup	1.6E-07	1.2%	Closed		
Total:		1.3E-05	100%		1.0E-05	79.8

Note 1. Open containment status indicates the containment equipment hatch and personnel airlock are open at the time of initiating event occurrence.

Table 9-6 presents the POS results independent of the POS duration. The POS and POT fractions were set to 1.0, and the model was re-quantified. The CDF results are reported in units of frequency per year assuming a one year duration for each POS (i.e., one POS-year). These results highlight those POSs that have the greatest likelihood of core damage per unit time. From the results in Table 9-5, POS 5A (pressurizer solid) is the largest contributor per unit time, followed by POS 8E and POS 8L. These POSs are the times during a refueling outage when significant maintenance activities are underway, specifically those involving emergency diesel generators and AC power distribution systems. The results suggest that maintenance configurations are important considerations in a plant's risk assessment (in particular, those maintenance activities that impact the ability to cope with a loss of offsite power). The mid-loop POSs (POS 6 and POS 10) also have a significant contribution to CDF per unit time, but their LOOP contributions are less than other POSs. The practices to protect equipment during mid-loop operations help to reduce the risk.

The conditional core damage probability (CCDP) associated with each POS (for each initiating event) is presented in Table 9-7. The CCDP is calculated by dividing the CDF contribution by the initiating event frequency. The POS and POT fractions are also set to 1.0. The CCDP provides a measure of the likelihood of core damage occurring if an initiating event has occurred during a given POS. The CCDP for all loss of NSCW events is 1.0. This is an expected result because the NSCW system provides the ultimate cooling for decay heat removal. If the system is lost, then decay heat removal cannot be restored. The risk due to loss of NSCW is mitigated by the high redundancy of the system and low initiating event frequency. Operator actions to restore NSCW in single pump operation are included in the initiating event fault tree and their failures are considered as part of the initiating event frequency.

Aside from the loss of NSCW results, the highest CCDPs are associated with the POSs with reduced RCS inventory. This includes the two mid-loop POSs (POS 6 and POS 10) and the lowered inventory levels in POS 5B and POS 9. The reduced inventory in the RCS greatly limits the amount of time available to take action to prevent core damage. In each of these POSs the loss of inventory events have significant CCDPs due to the limited time available for operators to respond to the events. Note that the mid-loop CCDPs for LOINV events are approximately half of the CCDP for POS 5B. This difference is related to the preferred RHR configuration during mid-loop where one RHR pump is secured in standby. This avoids possible damage to both RHR pumps in responding to an event, and later recovery of RHR cooling is more likely.

The results for POS 8E and 8L show CCDP values that may be larger than expected compared to other POSs. The largest POS 8E and 8L CCDP values are associated with the responses to loss of offsite power events. The impact of having the refueling cavity flooded could be expected to reduce the likelihood for events to progress to core damage, but the results show comparable, and in some cases higher, CCDPs than for POSs occurring earlier in the outage. The response to loss of offsite power events is a significant contributor to the CCDP results for these states, and the maintenance configurations limiting the redundancy of equipment to respond to these events is the primary reason for the relatively high CCDPs.

Table 9-6 CDF per Year in POS

IE Categories:	LORHR	LOINV	ODML / LOINV-DS	LO4160V	LOOP	LONSCW	CDF (per year in POS)	Time to CD (hr.)
IE Frequency: (per shutdown-year)	3.7E-1	6.1E-1 ⁽¹⁾ 3.0E-3 ⁽²⁾ 6.4E-5 ⁽³⁾ 1.2E-5 ⁽⁴⁾	2.4E-3 ⁽⁵⁾ 2.1E-3 ⁽⁶⁾ 2.1E-2 ⁽⁷⁾	3.4E-1	1.6E-1	3.5E-5 ⁽⁸⁾		
Shutdown POS								
POS 3 Mode 4 cooldown	3.7E-07	(Note 9)	N/A	(Note 9)	1.0E-04	(Note 9)	1.0E-04	Note (10)
POS 4 Mode 5 cooldown	3.9E-07	(Note 9)	N/A	(Note 9)	5.6E-04	(Note 9)	5.6E-04	Note (10)
POS 5A Pressurizer solid	1.1E-06	3.4E-04	N/A	(Note 9)	5.6E-04	(Note 9)	9.0E-04	27
POS 5B Draining RCS	3.2E-05	7.3E-06	3.8E-06	7.5E-06	1.6E-04	3.5E-05	2.5E-04	5.3
POS 6 Mid-loop (pre-refuel)	1.2E-04	7.2E-06	1.2E-06	1.3E-05	2.1E-04	3.5E-05	3.8E-04	3.3
POS 7 Filling refuel cavity	2.6E-05	1.6E-08	N/A	1.6E-05	1.4E-04	(Note 9)	1.8E-04	52
POS 8E Refuel cavity flooded (pre-refuel)	2.8E-05	2.2E-08	N/A	3.5E-05	5.3E-04	3.7E-05	5.9E-04	90
POS 8L Refuel cavity flooded (post-refuel)	2.7E-05	7.2E-09	N/A	2.6E-05	5.2E-04	3.7E-05	5.8E-04	214
POS 9 Draining refuel cavity	2.5E-05	7.3E-06	3.8E-06	7.5E-06	1.2E-04	3.7E-05	2.1E-04	15
POS 10 Mid-loop (post-refuel)	3.9E-05	7.2E-06	2.3E-07	1.3E-05	1.6E-04	3.7E-05	2.2E-04	11
POS 11 Vacuum fill RCS	2.5E-05	(Note 9)	N/A	(Note 9)	1.2E-04	(Note 9)	1.5E-04	Note (10)
POS 12 Mode 5 heatup	3.7E-07	(Note 9)	N/A	(Note 9)	1.0E-04	(Note 9)	1.0E-04	177
POS 13 Mode 4 heatup	3.7E-07	(Note 9)	N/A	(Note 9)	1.0E-04	(Note 9)	1.0E-04	Note (10)
Note (1) Loss of inventory over-pressure event frequency 6.1×10^{-1} /shutdown-year is applied in POS 5A. Note (2) Loss of inventory spray event frequency 3.0×10^{-3} /shutdown-year is applied in POS 5B, 6, 9, and 10. Note (3) Loss of inventory event frequency 6.4×10^{-5} /shutdown-year is applied in POS 5B, 6, 9, and 10.								

Table 9-6 CDF per Year in POS (cont.)

IE Categories:	LORHR	LOINV	ODML / LOINV-DS	LO4160V	LOOP	LONSCW	CDF (per year in POS)	Time to CD (hr.)
IE Frequency: (per shutdown-year)	3.7E-1	6.1E-1 ⁽¹⁾ 3.0E-3 ⁽²⁾ 6.4E-5 ⁽³⁾ 1.2E-5 ⁽⁴⁾	2.4E-3 ⁽⁵⁾ 2.1E-3 ⁽⁶⁾ 2.1E-2 ⁽⁷⁾	3.4E-1	1.6E-1	3.5E-5 ⁽⁸⁾		
Shutdown POS								
Note (4) Loss of inventory initiating event frequency 1.2×10^{-5}/shutdown-year is applied in POS 7, 8E, and 8L. Note (5) Frequency of mid-loop entry in POS 6 multiplied by conditional probability of overdrain. Note (6) Frequency of mid-loop entry in POS 10 multiplied by conditional probability of overdrain. Note (7) Loss of inventory due to displacement frequency 2.1×10^{-2}/shutdown-year is applied in POS 5B and 9. Note (8) Frequency calculated by fault tree is approximately 3.5×10^{-5}/shutdown-year, but varies by POS. Note (9) This accident scenario was not modeled. Note (10) Time to core damage was not evaluated for this POS (Section 5.3 describes timing assumptions).								

Table 9-7 CCDP for each Initiating Event and POS

IE Categories: IE Frequency: (per shutdown-year) Shutdown POS	LORHR	LOINV	ODML / LOINV-DS	LO4160V	LOOP	Subtotal CCDP for POS	LONSCW	Time to CD (hr.)
	3.7E-1	6.1E-1 ⁽¹⁾ 3.0E-3 ⁽²⁾ 6.4E-5 ⁽³⁾ 1.2E-5 ⁽⁴⁾	2.4E-3 ⁽⁵⁾ 2.1E-3 ⁽⁶⁾ 2.1E-2 ⁽⁷⁾	3.4E-1	1.6E-1		3.5E-5 ⁽⁸⁾	
POS 3 Mode 4 cooldown	1.0E-06	(Note 9)	N/A	(Note 9)	6.3E-04	6.3E-04	(Note 9)	(Note 10)
POS 4 Mode 5 cooldown	1.1E-06	(Note 9)	N/A	(Note 9)	3.4E-03	3.4E-03	(Note 9)	(Note 10)
POS 5A Pressurizer solid	3.1E-06	5.6E-04	N/A	(Note 9)	3.4E-03	4.0E-03	(Note 9)	27
POS 5B Draining RCS	8.8E-05	1.0E-01	1.8E-04	2.2E-05	9.8E-04	1.0E-01	1.0	5.3
POS 6 Mid-loop (pre-refuel)	3.3E-04	4.5E-02	5.1E-04	3.9E-05	1.3E-03	4.7E-02	1.0	3.3
POS 7 Filling refuel cavity	7.0E-05	1.3E-03	N/A	4.7E-05	8.7E-04	2.3E-03	(Note 9)	52
POS 8E Refuel cavity flooded (pre-refuel)	7.5E-05	1.8E-03	N/A	1.0E-04	3.2E-03	5.2E-03	1.0	90
POS 8L Refuel cavity flooded (post-refuel)	7.2E-05	6.0E-04	N/A	7.8E-05	3.2E-03	4.0E-03	1.0	214
POS 9 Draining refuel cavity	6.7E-05	1.0E-01	1.8E-04	2.2E-05	7.6E-04	1.0E-01	1.0	15
POS 10 Mid-loop (post-refuel)	1.1E-04	4.5E-02	1.1E-04	3.9E-05	1.0E-03	4.6E-02	1.0	11
POS 11 Vacuum fill RCS	6.9E-05	(Note 9)	N/A	(Note 9)	7.6E-04	8.3E-04	(Note 9)	(Note 10)
POS 12 Mode 5 heatup	1.0E-06	(Note 9)	N/A	(Note 9)	6.3E-04	6.3E-04	(Note 9)	177
POS 13 Mode 4 heatup	1.0E-06	(Note 9)	N/A	(Note 9)	6.3E-04	6.3E-04	(Note 9)	(Note 10)
Note (1) Loss of inventory over-pressure event frequency 6.1×10^{-1} /shutdown-year is applied in POS 5A. Note (2) Loss of inventory spray event frequency 3.0×10^{-3} /shutdown-year is applied in POS 5B, 6, 9, and 10. Note (3) Loss of inventory event frequency 6.4×10^{-5} /shutdown-year is applied in POS 5B, 6, 9, and 10.								

Table 9-7 CCDP for each Initiating Event and POS (cont.)

IE Categories:	LORHR	LOINV	ODML / LOINV-DS	LO4160V	LOOP	Subtotal CCDP for POS	LONSCW	Time to CD (hr.)
IE Frequency: (per shutdown-year)	3.7E-1	6.1E-1 ⁽¹⁾ 3.0E-3 ⁽²⁾ 6.4E-5 ⁽³⁾	2.4E-3 ⁽⁵⁾ 2.1E-3 ⁽⁶⁾ 2.1E-2 ⁽⁷⁾	3.4E-1	1.6E-1		3.5E-5 ⁽⁸⁾	
Shutdown POS		1.2E-5 ⁽⁴⁾						
Note (4) Loss of inventory initiating event frequency 1.2×10^{-5}/shutdown-year is applied in POS 7, 8E, and 8L. Note (5) Frequency of mid-loop entry in POS 6 multiplied by conditional probability of overdrain. Note (6) Frequency of mid-loop entry in POS 10 multiplied by conditional probability of overdrain. Note (7) Loss of inventory due to displacement frequency 2.1×10^{-2}/shutdown-year is applied in POS 5B and 9. Note (8) Frequency calculated by fault tree is approximately 3.5×10^{-5}/shutdown-year, but varies by POS. Note (9) This accident scenario was not modeled. Note (10) Time to core damage was not evaluated for this POS (Section 5.3 describes timing assumptions).								

9.4 CDF Results by Significant Accident Sequences

The significant accident sequences are identified in Table 9-8. The significant accident sequences are those sequences that, when rank-ordered by decreasing CDF, sum to 95 percent of the total CDF, and any sequences with an individual contribution to CDF of at least 1 percent.

The following are observations about the significant accident sequences contributing to CDF:

- Similar to the initiating event results (see Table 9-1), the largest CDF contributions are associated with accident sequences involving loss of offsite power events.
- The majority of LOOP accident sequences are station blackout sequences. The CDF contribution from SBO sequences is 8.2×10^{-6} per year, which accounts for approximately 65 percent of the total shutdown CDF and 96 percent of the LOOP CDF.
- Many of the LOOP SBO sequences involve successful actions to extend the sequence (either gravity injection from RWST or steam generator cooling with turbine-drive AFW pump), but core damage occurs due to failure to restore power and RHR cooling. These sequences account for approximately 36 percent of the total shutdown CDF. If available, alternate equipment and strategies could be implemented to prevent core damage given that actions to extend the sequences are successful.³⁵
- After the LOOP sequences, the next highest contributing accident sequence is a loss of inventory due to a pressure transient causing the RHR relief valves to open. Operators fail to diagnose and isolate the loss of inventory before makeup is required for continued RHR cooling.
- Accident sequences related to overdrain entering mid-loop also have notable contributions to shutdown CDF.
- For significant accident sequences that do not involve LOOP events, there are many sequences where successful injection with ECCS pumps extends the sequence, but core damage occurs due to failure to restore RHR cooling. These sequences account for approximately 17 percent of the total shutdown CDF. If alternate equipment and strategies for providing decay heat removal were available, these could be implemented to prevent core damage given the additional time available due to successful injection.

³⁵ Note that this study is based on information that reflects the reference plant as it was designed and operated as of 2012. It does not reflect the potential impact of FLEX strategies implemented after 2012. The impacts of FLEX strategies are explored in the Level 3 PRA project overview report on the shutdown Level 1, 2, and 3 PRAs.

Table 9-8 LPSD PRA CDF Results from Significant Accident Sequences

Sequence Identifier	Sequence Description	CDF (per year)	% Contr. to CDF	Cumulative % Contr. to CDF	Cut Set Count
1-SD-LOOP:5-09-3-8-1	Loss of offsite power in POS 8E. Maintenance on one RAT. Failure of opposite train EDG. Failure to restore offsite power. SBO occurs. Failure to align gravity feed. Failure of long-term actions to restore onsite power and RHR cooling.	1.5E-06	11.8	11.8	847
1-SD-LOOP:5-05-3-2-1	Loss of offsite power in POS 5A. Maintenance on one train sequencer. Failure of opposite train EDG. SBO occurs. Successful SG cooling with AFW TDP. Failure to restore offsite or onsite power.	1.2E-06	9.7	21.5	616
1-SD-LOOP:5-09-3-6-1	Loss of offsite power in POS 8E. Maintenance on one RAT. Failure of opposite train EDG. Failure to restore offsite power. SBO occurs. Successful gravity feed. Failure of long-term actions to restore onsite power and RHR cooling.	1.2E-06	9.6	31.1	776
1-SD-LOOP:5-11-3-8-1	Loss of offsite power in POS 8L. Maintenance on one RAT. Failure of opposite train EDG. Failure to restore offsite power. SBO occurs. Failure to align gravity feed. Failure of long-term actions to restore onsite power and RHR cooling.	1.1E-06	8.9	40.0	599
1-SD-LOOP:5-11-3-6-1	Loss of offsite power in POS 8L. Maintenance on one RAT. Failure of opposite train EDG. Failure to restore offsite power. SBO occurs. Successful gravity feed. Failure of long-term actions to restore onsite power and RHR cooling.	9.1E-07	7.3	47.3	557
1-SD-LOINV-OP:5-05-23-1	Loss of inventory event in POS 5A. Failure to diagnose and isolate before makeup is required for RHR cooling. Successful pumped injection. Failure of long-term RHR restoration.	7.4E-07	5.9	53.2	3

Table 9-8 LPSD PRA CDF Results from Significant Accident Sequences (cont.)

Sequence Identifier	Sequence Description	CDF (per year)	% Contr. to CDF	Cumulative % Contr. to CDF	Cut Set Count
1-SD-ODML-EARLY:2-08-1	Overdrain entering POS 6. Successful isolation of overdrain. Failure to restart RHR before RCS boiling. Successful pumped injection. Failure of long-term RHR restoration.	6.0E-07	4.8	58.0	54
1-SD-LOOP:5-15-3-2-1	Loss of offsite power in POS 12. Failure of onsite emergency power. SBO occurs. Successful SG cooling with AFW TDP. Failure to restore offsite or onsite power.	4.1E-07	3.3	61.3	793
1-SD-ODML-EARLY:2-22-1	Overdrain entering POS 6. Failure to diagnose and stop drain before RHR cooling is lost. Successful pumped injection. Failure of long-term RHR restoration.	3.8E-07	3.1	64.3	4
1-SD-LOOP:5-06-3-7-1	Loss of offsite power in POS 5B. Failure of onsite emergency power. Failure to restore offsite power. SBO occurs. Failure to align gravity feed.	2.6E-07	2.1	66.4	664
1-SD-LOOP:5-12-3-7-1	Loss of offsite power in POS 9. Failure of onsite emergency power. Failure to restore offsite power. SBO occurs. Failure to align gravity feed.	2.0E-07	1.6	68.0	584
1-SD-LONSCW:5-09-6-1	Loss of service water in POS 8E. Maintenance on one RAT. Combination of hardware failures and operator actions to restore service water cooling.	1.9E-07	1.5	69.5	21
1-SD-LOOP:5-06-3-6-1	Loss of offsite power in POS 5B. Failure of onsite emergency power. Failure to restore offsite power. SBO occurs. Successful gravity feed. Failure of long-term actions to restore onsite power and RHR cooling.	1.8E-07	1.4	70.9	434

Table 9-8 LPSD PRA CDF Results from Significant Accident Sequences (cont.)

Sequence Identifier	Sequence Description	CDF (per year)	% Contr. to CDF	Cumulative % Contr. to CDF	Cut Set Count
1-SD-LOOP:5-13-3-7-1	Loss of offsite power in POS 10. Failure of onsite emergency power. Failure to restore offsite power. SBO occurs. Failure to align gravity feed.	1.6E-07	1.3	72.2	544
1-SD-LO4160V:5-09-09-1	Loss of power to 4,160V AC bus in POS 8E. Maintenance on one RAT. Combination of hardware failures and operator actions prevent restoring power to bus. Failure of long-term actions to restore onsite power and RHR cooling.	1.6E-07	1.3	73.5	503
1-SD-LOOP:5-04-3-2-1	Loss of offsite power in POS 4. Maintenance on one train sequencer. Failure of opposite train EDG. SBO occurs. Successful SG cooling with AFW TDP. Failure to restore offsite or onsite power.	1.6E-07	1.3	74.8	336
1-SD-ODML-LATE:2-08-1	Overdrain entering POS 10. Successful isolation of overdrain. Failure to restart RHR before RCS boiling. Successful pumped injection. Failure of long-term RHR restoration.	1.6E-07	1.3	76.1	118
1-SD-LOOP:5-14-3-7-1	Loss of offsite power in POS 11. Failure of onsite emergency power. Failure to restore offsite power. SBO occurs. Failure to align gravity feed.	1.6E-07	1.3	77.4	544
1-SD-LOOP:5-16-3-2-1	Loss of offsite power in POS 13. Failure of onsite emergency power. SBO occurs. Successful SG cooling with AFW TDP. Failure to restore offsite or onsite power.	1.5E-07	1.2	78.6	496
1-SD-LONSCW:5-11-6-1	Loss of service water in POS 8L. Maintenance on one RAT. Combination of hardware failures and operator actions to restore service water cooling.	1.5E-07	1.2	79.8	21

Table 9-8 LPSD PRA CDF Results from Significant Accident Sequences (cont.)

Sequence Identifier	Sequence Description	CDF (per year)	% Contr. to CDF	Cumulative % Contr. to CDF	Cut Set Count
1-SD-ODML-EARLY:2-15-1	Overdrain entering POS 6. Successful isolation of overdrain. Failure to restart RHR before RCS boiling. Failure of RCS injection.	1.3E-07	1.1	80.8	83
1-SD-LORHR:5-09-06-1	Loss of RHR cooling in POS 8E. Maintenance on one RAT. Combination of hardware failures and operator actions prevent restoring RHR cooling. Successful pumped injection. Failure of long-term RHR restoration.	1.3E-07	1.0	81.8	178
1-SD-LOOP:5-13-3-6-1	Loss of offsite power in POS 10. Failure of onsite emergency power. Failure to restore offsite power. SBO occurs. Successful gravity feed. Failure of long-term actions to restore onsite power and RHR cooling.	1.1E-07	0.9	82.7	353
1-SD-LORHR:5-07-05-1	Loss of RHR cooling in POS 6. Combination of hardware failures and operator actions prevent restoring RHR cooling. Successful pumped injection. Failure of long-term RHR restoration.	1.0E-07	0.8	83.5	96
1-SD-LOOP:5-07-3-7-1	Loss of offsite power in POS 6. Failure of onsite emergency power. Failure to restore offsite power. SBO occurs. Failure to align gravity feed.	1.0E-07	0.8	84.3	517
1-SD-ODML-EARLY:2-29-1	Overdrain entering POS 6. Failure to diagnose and stop drain before RHR cooling is lost. Failure of RCS injection.	9.7E-08	0.8	85.1	74
1-SD-LORHR:5-11-06-1	Loss of RHR cooling in POS 8L. Maintenance on one RAT. Combination of hardware failures and operator actions prevent restoring RHR cooling. Successful pumped injection. Failure of long-term RHR restoration.	9.4E-08	0.8	85.8	180

Table 9-8 LPSD PRA CDF Results from Significant Accident Sequences (cont.)

Sequence Identifier	Sequence Description	CDF (per year)	% Contr. to CDF	Cumulative % Contr. to CDF	Cut Set Count
1-SD-LO4160V:5-11-09-1	Loss of power to 4,160V AC bus in POS 8L. Maintenance on one RAT. Combination of hardware failures and operator actions prevent restoring power to bus. Failure of long-term actions to restore onsite power and RHR cooling.	9.4E-08	0.8	86.6	343
1-SD-LORHR:5-06-05-1	Loss of RHR cooling in POS 5B. Combination of hardware failures and operator actions prevent restoring RHR cooling. Successful pumped injection. Failure of long-term RHR restoration.	8.6E-08	0.7	87.3	41
1-SD-LONSCW:5-12-05-1	Loss of service water in POS 9. Combination of hardware failures and operator actions to restore service water cooling.	8.5E-08	0.7	87.9	21
1-SD-LONSCW:5-06-05-1	Loss of service water in POS 5B. Combination of hardware failures and operator actions to restore service water cooling.	8.3E-08	0.7	88.6	16
1-SD-LOOP:5-08-3-7-1	Loss of offsite power in POS 7. Failure of onsite emergency power. Failure to restore offsite power. SBO occurs. Failure to align gravity feed.	8.1E-08	0.7	89.3	614
1-SD-LONSCW:5-13-05-1	Loss of service water in POS 10. Combination of hardware failures and operator actions to restore service water cooling.	7.1E-08	0.6	89.8	21
1-SD-LOOP:5-08-3-6-1	Loss of offsite power in POS 7. Failure of onsite emergency power. Failure to restore offsite power. SBO occurs. Successful gravity feed. Failure of long-term actions to restore onsite power and RHR cooling.	6.6E-08	0.5	90.3	564

Table 9-8 LPSD PRA CDF Results from Significant Accident Sequences (cont.)

Sequence Identifier	Sequence Description	CDF (per year)	% Contr. to CDF	Cumulative % Contr. to CDF	Cut Set Count
1-SD-LOOP:5-07-3-6-1	Loss of offsite power in POS 6. Failure of onsite emergency power. Failure to restore offsite power. SBO occurs. Successful gravity feed. Failure of long-term actions to restore onsite power and RHR cooling.	6.2E-08	0.5	90.8	271
1-SD-LOOP:5-12-3-6-1	Loss of offsite power in POS 9. Failure of onsite emergency power. Failure to restore offsite power. SBO occurs. Successful gravity feed. Failure of long-term actions to restore onsite power and RHR cooling.	5.9E-08	0.5	91.3	433
1-SD-LORHR:5-13-05-1	Loss of RHR cooling in POS 10. Combination of hardware failures and operator actions prevent restoring RHR cooling. Successful pumped injection. Failure of long-term RHR restoration.	5.7E-08	0.5	91.8	121
1-SD-LOOP:5-09-1-06-1	Loss of offsite power in POS 8E. Maintenance on one RAT. Successful emergency onsite power. Combination of hardware failures and operator actions prevent restoring RHR cooling. Successful pumped injection. Failure of long-term RHR restoration.	5.6E-08	0.4	92.2	126
1-SD-LOOP:5-05-3-8-1	Loss of offsite power in POS 5A. Maintenance on one train sequencer. Failure of opposite train EDG. SBO occurs. Failure of SG cooling with AFW TDP.	5.5E-08	0.4	92.6	400
1-SD-LORHR:5-12-05-1	Loss of RHR cooling in POS 9. Combination of hardware failures and operator actions prevent restoring RHR cooling. Successful pumped injection. Failure of long-term RHR restoration.	5.0E-08	0.4	93.0	60

Table 9-8 LPSD PRA CDF Results from Significant Accident Sequences (cont.)

Sequence Identifier	Sequence Description	CDF (per year)	% Contr. to CDF	Cumulative % Contr. to CDF	Cut Set Count
1-SD-LOOP:5-14-3-6-1	Loss of offsite power in POS 11. Failure of onsite emergency power. Failure to restore offsite power. SBO occurs. Successful gravity feed. Failure of long-term actions to restore onsite power and RHR cooling.	4.7E-08	0.4	93.4	399
1-SD-LOOP:5-07-1-05-1	Loss of offsite power in POS 6. Successful emergency onsite power. Combination of hardware failures and operator actions prevent restoring RHR cooling. Successful pumped injection. Failure of long-term RHR restoration.	4.4E-08	0.4	93.8	31
1-SD-LOOP:5-11-1-06-1	Loss of offsite power in POS 8L. Maintenance on one RAT. Successful emergency onsite power. Combination of hardware failures and operator actions prevent restoring RHR cooling. Successful pumped injection. Failure of long-term RHR restoration.	4.2E-08	0.3	94.1	129
1-SD-LORHR:5-14-09-1	Loss of RHR cooling in POS 11. Combination of hardware failures and operator actions prevent restoring RHR cooling. Successful pumped injection. Failure of long-term RHR restoration.	4.0E-08	0.3	94.4	60
1-SD-ODML-LATE:2-22-1	Overdrain entering POS 10. Failure to diagnose and stop drain before RHR cooling is lost. Successful pumped injection. Failure of long-term RHR restoration.	3.9E-08	0.3	94.7	3
1-SD-LOOP:5-06-1-05-1	Loss of offsite power in POS 5B. Successful emergency onsite power. Combination of hardware failures and operator actions prevent restoring RHR cooling. Successful pumped injection. Failure of long-term RHR restoration.	3.8E-08	0.3	95.0	25

9.5 Review of CDF Cut Sets

The significant cut sets are those cut sets that, when rank-ordered by decreasing CDF, sum to 95 percent of the total CDF, and any cut sets with an individual contribution to CDF of at least 1 percent. The top 100 cut sets (accounting for 79 percent of the total CDF) were reviewed to ensure they represent reasonable and logical combinations of failures that could result in core damage. An additional sampling of cut sets from each significant sequence were reviewed to ensure that they are reasonable and represent realistic accident sequences. Example cut sets for selected sequences are discussed below, organized by initiating event. The top 700 cut sets, contributing 95 percent of the total CDF, are shown in Appendix B. A sampling of non-significant cut sets was reviewed to ensure that they also represent reasonable and realistic accident sequences.

9.5.1 Cut Sets for Sequence 1-SD-LOOP:5-09-3-8-1

Loss of Offsite Power Sequences Resulting in SBO with One Normal Source of Offsite Power in Maintenance

Example cut sets for sequence 1-SD-LOOP:5-09-3-8-1 are shown in Table 9-9. This sequence represents a shutdown LOOP event during POS 8E with the train A RAT (normal offsite power source for safety-related train A 4,160V AC bus) unavailable for maintenance. The sequence results in a station blackout. Operators fail to gravity-fill the RCS with the limited RWST inventory available during this POS. Core damage is reached due to failures to restore offsite or onsite ac power.

Table 9-9 shows the top cut sets contributing more than 1 percent to the sequence frequency. Other significant cut sets in the sequence follow a similar pattern. The cut sets include equipment failures that prevent train B EDG from powering the safety-related bus. Other failures include failure to restore the EDG, failure to restore offsite power, and unavailability of the alternative offsite power source. The final failure in the cut sets is operator failure to establish gravity feed from the RWST.

Sequence 1-SD-LOOP:5-11-3-8-1 is similar to sequence 1-SD-LOOP:5-09-3-8-1, but occurs during POS 8L, and includes the same types of failure combinations. Sequences 1-SD-LOOP:5-09-3-6-1 and 1-SD-LOOP:5-11-3-6-1 also represent similar LOOP sequences during POS 8E and POS 8L, with the only difference being success of gravity feed. The cut sets for all these sequences are similar. Successful gravity feed can delay the onset of core damage, but core damage is still assumed to occur because recovery of onsite and offsite power fails.

Table 9-9 Cut Sets for Sequence 1-SD-LOOP: 5-09-3-8-1

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
1-SD-LOOP:5-09-3-8-1	1.475E-06	11.77		847	
1	3.906E-07	3.12	26.48	1-IE-SD-LOOP, 1-EPS-DGN-FR-G4002____, 1-EPS-XHE-XL-NR04H, 1-LPI-XHE-XM-GFEED-LOOP, 1-OEP-XHE-XL-NR02H, 1-OEP-XHE-XX-NR02H1, 1-POS-8E-POT-4, 1-POT-4, OEP-VCF-LP-GRWR	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.297E-2			1-EPS-DGN-FR-G4002____	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.500E-1			1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.729E-1			1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	9.370E-2			1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.100E-1			OEP-VCF-LP-GRWR	SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION

Table 9-9 Cut Sets for Sequence 1-SD-LOOP: 5-09-3-8-1 (cont.)

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
2	2.323E-07	1.85	15.75	1-IE-SD-LOOP, 1-ACP-CRB-CC-BA0301__, 1-EPS-XHE-XL-NR04H, 1-LPI-XHE-XM-GFEED-LOOP, 1-OEP-XHE-XL-NR02H, 1-POS-8E-POT-4, 1-POT-4, OEP-VCF-LP-GRWR	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.350E-3			1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.500E-1			1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.370E-2			1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.100E-1			OEP-VCF-LP-GRWR	SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
3	1.625E-07	1.30	11.02	1-IE-SD-LOOP, 1-ACP-CRB-CF-A205301, 1-LPI-XHE-XM-GFEED-LOOP, 1-POS-8E-POT-4, 1-POT-4	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN

Table 9-9 Cut Sets for Sequence 1-SD-LOOP: 5-09-3-8-1 (cont.)

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
	3.498E-4			1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	5.500E-1			1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	9.370E-2			1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
4	1.446E-07	1.15	9.80	1-IE-SD-LOOP, 1-EPS-SEQ-FO-1821U302, 1-EPS-XHE-XL-NR04H, 1-LPI-XHE-XM-GFEED-LOOP, 1-OEP-XHE-XL-NR02H, 1-POS-8E-POT-4, 1-POT-4, OEP-VCF-LP-GRWR	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.330E-3			1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.500E-1			1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.370E-2			1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)

Table 9-9 Cut Sets for Sequence 1-SD-LOOP: 5-09-3-8-1 (cont.)

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.100E-1			OEP-VCF-LP-GRWR	SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
5	1.276E-07	1.02	8.65	1-IE-SD-LOOP, 1-EPS-DGN-FS-G4002____, 1-EPS-XHE-XL-NR04H, 1-LPI-XHE-XM-GFEED-LOOP, 1-OEP-XHE-XL-NR02H, 1-POS-8E-POT-4, 1-POT-4, OEP-VCF-LP-GRWR	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.940E-3			1-EPS-DGN-FS-G4002____	DG1B RANDOMLY FAILS TO START
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.500E-1			1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.370E-2			1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.100E-1			OEP-VCF-LP-GRWR	SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION

Table 9-9 Cut Sets for Sequence 1-SD-LOOP: 5-09-3-8-1 (cont.)

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
6	8.764E-08	0.70	5.94	1-IE-SD-LOOP, 1-EPS-DGN-FR-G4002____, 1-EPS-XHE-XL-NR04H, 1-LPI-XHE-XM-GFEED-LOOP, 1-OA-ALIGNPW-G-H, 1-OEP-XHE-XL-NR02H, 1-OEP-XHE-XX-NR02H1, 1-POS-8E-POT-4, 1-POT-4	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.297E-2			1-EPS-DGN-FR-G4002____	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.500E-1			1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	9.200E-2			1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN Alternate offsite AC - General case
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.729E-1			1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	9.370E-2			1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
7	6.969E-08	0.56	4.73	1-IE-SD-LOOP, 1-EPS-SEQ-CF-FOAB, 1-EPS-XHE-XL-NR04H, 1-LPI-XHE-XM-GFEED-LOOP, 1-POS-8E-POT-4, 1-POT-4	

Table 9-9 Cut Sets for Sequence 1-SD-LOOP: 5-09-3-8-1 (cont.)

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.148E-4			1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.500E-1			1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	9.370E-2			1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
8	5.212E-08	0.42	3.53	1-IE-SD-LOOP, 1-ACP-CRB-CC-BA0301__, 1-EPS-XHE-XL-NR04H, 1-LPI-XHE-XM-GFEED-LOOP, 1-OA-ALIGNPW-G-H, 1-OEP-XHE-XL-NR02H, 1-POS-8E-POT-4, 1-POT-4	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.350E-3			1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.500E-1			1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	9.200E-2			1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN Alternate offsite AC - General case

Table 9-9 Cut Sets for Sequence 1-SD-LOOP: 5-09-3-8-1 (cont.)

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.370E-2			1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
9	3.244E-08	0.26	2.20	1-IE-SD-LOOP, 1-EPS-SEQ-FO-1821U302, 1-EPS-XHE-XL-NR04H, 1-LPI-XHE-XM-GFEED-LOOP, 1-OA-ALIGNPW-G-H, 1-OEP-XHE-XL-NR02H, 1-POS-8E-POT-4, 1-POT-4	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.330E-3			1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.500E-1			1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	9.200E-2			1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN Alternate offsite AC - General case
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.370E-2			1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)

Table 9-9 Cut Sets for Sequence 1-SD-LOOP: 5-09-3-8-1 (cont.)

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
10	2.864E-08	0.23	1.94	1-IE-SD-LOOP, 1-EPS-DGN-FS-G4002____, 1-EPS-XHE-XL-NR04H, 1-LPI-XHE-XM-GFEED-LOOP, 1-OA-ALIGNPW-G-H, 1-OEP-XHE-XL-NR02H, 1-POS-8E-POT-4, 1-POT-4	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.940E-3			1-EPS-DGN-FS-G4002____	DG1B RANDOMLY FAILS TO START
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.500E-1			1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	9.200E-2			1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN Alternate offsite AC - General case
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.370E-2			1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

9.5.2 Cut Sets for Sequence 1-SD-LOOP:5-05-3-2-1

Loss of Offsite Power Sequences Resulting in SBO with One Train Emergency Diesel Generator Automatic Actuation Unavailable due to Maintenance

Example cut sets for sequence 1-SD-LOOP:5-05-3-2-1 are shown in Table 9-10. This sequence represents a shutdown LOOP event during POS 5A with testing and maintenance making automatic actuation of the train A EDG unavailable. The sequence results in a station blackout. Operators successfully establish steam generator cooling with the turbine-driven auxiliary feedwater pump. Failure to restore onsite or offsite AC power and DC power results in core damage.

Table 9-10 shows the top cut sets contributing more than 1 percent to the sequence frequency. Other significant cut sets in the sequence follow a similar pattern. The cut sets include equipment failures that prevent train B EDG from powering the safety-related bus. Other failures include failure to restore the EDG, failure to restore offsite power, and unavailability of the alternative offsite power source.

Table 9-10 Cut Sets for Sequence 1-SD-LOOP: 5-05-3-2-1

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
1-SD-LOOP:5-05-3-2-1	1.215E-06	9.69		616	
1	3.130E-07	2.50	25.77	1-IE-SD-LOOP, 1-EPS-DGN-FR-G4002____, 1-EPS-XHE-XL-NR04H, 1-OEP-XHE-XL-NR02H, 1-OEP-XHE-XX-NR02H1, 1-POS-5A-POT-4, 1-POT-4, OEP-VCF-LP-GRWR	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.297E-2			1-EPS-DGN-FR-G4002____	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.729E-1			1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	4.130E-2			1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.100E-1			OEP-VCF-LP-GRWR	SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
2	1.861E-07	1.48	15.32	1-IE-SD-LOOP, 1-ACP-CRB-CC-BA0301____, 1-EPS-XHE-XL-NR04H, 1-OEP-XHE-XL-NR02H, 1-POS-5A-POT-4, 1-POT-4, OEP-VCF-LP-GRWR	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN

Table 9-10 Cut Sets for Sequence 1-SD-LOOP: 5-05-3-2-1 (cont.)

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
	5.350E-3			1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	4.130E-2			1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.100E-1			OEP-VCF-LP-GRWR	SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
3	1.302E-07	1.04	10.72	1-IE-SD-LOOP, 1-ACP-CRB-CF-A205301, 1-POS-5A-POT-4, 1-POT-4	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.498E-4			1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	4.130E-2			1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
4	1.159E-07	0.92	9.54	1-IE-SD-LOOP, 1-EPS-SEQ-FO-1821U302, 1-EPS-XHE-XL-NR04H, 1-OEP-XHE-XL-NR02H, 1-POS-5A-POT-4, 1-POT-4, OEP-VCF-LP-GRWR	

Table 9-10 Cut Sets for Sequence 1-SD-LOOP: 5-05-3-2-1 (cont.)

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.330E-3			1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	4.130E-2			1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.100E-1			OEP-VCF-LP-GRWR	SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
5	1.023E-07	0.82	8.42	1-IE-SD-LOOP, 1-EPS-DGN-FS-G4002____, 1-EPS-XHE-XL-NR04H, 1-OEP-XHE-XL-NR02H, 1-POS-5A-POT-4, 1-POT-4, OEP-VCF-LP-GRWR	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.940E-3			1-EPS-DGN-FS-G4002____	DG1B RANDOMLY FAILS TO START
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS

Table 9-10 Cut Sets for Sequence 1-SD-LOOP: 5-05-3-2-1 (cont.)

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
	4.130E-2			1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.100E-1			OEP-VCF-LP-GRWR	SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
6	7.023E-08	0.56	5.78	1-IE-SD-LOOP, 1-EPS-DGN-FR-G4002____, 1-EPS-XHE-XL-NR04H, 1-OA-ALIGNPW-G-H, 1-OEP-XHE-XL-NR02H, 1-OEP-XHE-XX-NR02H1, 1-POS-5A-POT-4, 1-POT-4	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.297E-2			1-EPS-DGN-FR-G4002____	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.200E-2			1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN Alternate offsite AC - General case
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.729E-1			1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	4.130E-2			1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table 9-10 Cut Sets for Sequence 1-SD-LOOP: 5-05-3-2-1 (cont.)

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
7	5.585E-08	0.45	4.60	1-IE-SD-LOOP, 1-EPS-SEQ-CF-FOAB, 1-EPS-XHE-XL-NR04H, 1-POS-5A-POT-4, 1-POT-4	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.148E-4			1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	4.130E-2			1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
8	4.177E-08	0.33	3.44	1-IE-SD-LOOP, 1-ACP-CRB-CC-BA0301__, 1-EPS-XHE-XL-NR04H, 1-OA-ALIGNPW-G-H, 1-OEP-XHE-XL-NR02H, 1-POS-5A-POT-4, 1-POT-4	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.350E-3			1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.200E-2			1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN Alternate offsite AC - General case
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS

Table 9-10 Cut Sets for Sequence 1-SD-LOOP: 5-05-3-2-1 (cont.)

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
	4.130E-2			1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
9	2.600E-08	0.21	2.14	1-IE-SD-LOOP, 1-EPS-SEQ-FO-1821U302, 1-EPS-XHE-XL-NR04H, 1-OA-ALIGNPW-G-H, 1-OEP-XHE-XL-NR02H, 1-POS-5A-POT-4, 1-POT-4	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.330E-3			1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.200E-2			1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN Alternate offsite AC - General case
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	4.130E-2			1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
10	2.295E-08	0.18	1.89	1-IE-SD-LOOP, 1-EPS-DGN-FS-G4002___, 1-EPS-XHE-XL-NR04H, 1-OA-ALIGNPW-G-H, 1-OEP-XHE-XL-NR02H, 1-POS-5A-POT-4, 1-POT-4	

Table 9-10 Cut Sets for Sequence 1-SD-LOOP: 5-05-3-2-1 (cont.)

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.940E-3			1-EPS-DGN-FS-G4002____	DG1B RANDOMLY FAILS TO START
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.200E-2			1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN Alternate offsite AC - General case
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	4.130E-2			1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
11	1.392E-08	0.11	1.15	1-IE-SD-LOOP, 1-EPS-TNK-MA-DFOSTKB_, 1-EPS-XHE-XL-NR04H, 1-OEP-XHE-XL-NR02H, 1-POS-5A-POT-4, 1-POT-4, OEP-VCF-LP-GRWR	
	1.630E-1			1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	4.000E-4			1-EPS-TNK-MA-DFOSTKB_	TR B. DIESEL FUEL OIL STORAGE TANK 1-2403-T4-002 IN MAINTENANCE
	6.984E-1			1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS

Table 9-10 Cut Sets for Sequence 1-SD-LOOP: 5-05-3-2-1 (cont.)

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
	3.264E-1			1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	4.130E-2			1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.100E-1			OEP-VCF-LP-GRWR	SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION

9.5.3 Cut Sets for Sequence 1-SD-LOINV-OP:5-05-23-1

Loss of Inventory due to Overpressure Resulting in Lifting of Pressure Relief Valves

Example cut sets for sequence 1-SD-LOINV-OP: 5-05-23-1 are shown in Table 9-11. This sequence represents a loss of inventory due to an RCS pressure excursion resulting in lifting of RHR pressure relief valves. After lifting, an RHR relief valve fails to reclose resulting in a sustained loss of inventory. Operators fail to diagnose and isolate the inventory loss prior to RHR cooling being impacted. Successful pumped injection is established. Actions to restore RHR cooling fail resulting in core damage. The failure to restore RHR cooling can include operator failures to restore RHR cooling, equipment failures, or failure to isolate the inventory loss.

Table 9-11 Cut Sets for Sequence 1-SD-LOINV-OP: 5-05-23-1

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
1-SD-LOINV-OP:5-05-23-1	7.445E-07	5.94		3	
1	3.492E-07	2.79	46.90	1-IE-SD-LOINV-OP, 1-L2-BE-RHRVSTUCK-SD, 1-LOI-XHE-XD-POS5A, 1-POS-5A-POT-4, 1-POT-4, 1-RHR-XHE-XL-LTREC-LT5H	
	6.070E-1			1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	6.694E-2			1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve Does Not Fail Open During CD (SD)
	2.500E-2			1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.130E-2			1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.505E-1			1-RHR-XHE-XL-LTREC-LT5H	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
2	3.387E-07	2.70	45.50	1-IE-SD-LOINV-OP, 1-L2-BE-RHRVSTUCK-SD, 1-LOI-XHE-XD-POS5A, 1-POS-5A-POT-4, 1-POT-4, 1-RHR-XHE-XM-TRIP-MD	
	6.070E-1			1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	6.694E-2			1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve Does Not Fail Open During CD (SD)
	2.500E-2			1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY

Table 9-11 Cut Sets for Sequence 1-SD-LOINV-OP: 5-05-23-1 (cont.)

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
	4.130E-2			1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.460E-1			1-RHR-XHE-XM-TRIP-MD	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS (MODERATE DEPENDENCY)
3	5.661E-08	0.45	7.60	1-IE-SD-LOINV-OP, 1-L2-BE-RHRVSTUCK-SD, 1-LOI-XHE-XD-POS5A, 1-LOI-XHE-XM-ISO2-LT, 1-POS-5A-POT-4, 1-POT-4	
	6.070E-1			1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	6.694E-2			1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve Does Not Fail Open During CD (SD)
	2.500E-2			1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	2.440E-2			1-LOI-XHE-XM-ISO2-LT	FAIL TO ISOLATE LOI AFTER SUCCESSFUL INJECTION
	4.130E-2			1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.530E-2			1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

9.5.4 Cut Sets for Sequence 1-SD-ODML-EARLY:2-08-1

Significant Overdrain Entering Mid-loop Sequence Cut Sets

The cut sets for sequence 1-SD-ODML-EARLY:2-08-1 are shown in Table 9-12. This sequence represents an overdrain of RCS level when draining to enter mid-loop condition during POS 6. Operators successfully diagnose the overdrain and isolate the draining. RHR cooling is unavailable until level is restored greater than one foot above mid-loop centerline. Operators successfully align pumped injection from the RWST to control RCS level and extend the sequence. Operators fail long-term restoration of RHR cooling. The restoration of RHR cooling can fail due to a combination of hardware failures or human failures.

Sequence 1-SD-ODML- EARLY:2-22-1 is similar to sequence 1-SD-ODML-EARLY:2-08-1, but the initial diagnosis and isolation of the draining fails. After successful injection to control RCS level, the long-term restoration of the standby RHR train fails due to a combination of hardware failures or human failures.

Similar sequences are 1-SD-ODML-LATE:2-08-1 and 1-SD-ODML-LATE:2-22-1. These sequences involve overdrain of RCS level when entering mid-loop after refueling in POS 10. Compared to the POS 6 sequences, the decay heat level is lower after refueling, which allows more time for operators to respond to the event. Operator actions in POS 10 are more likely to be successful.

Table 9-12 Cut Sets for Sequence 1-SD-ODML-EARLY: 2-08-1

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
1-SD-ODML- EARLY:2-08-1	6.005E-07	4.79		54	
1	4.637E-07	3.70	77.21	1-IE-SD-ODML-EARLY, 1-POS-6- POT-4-MLDRL, 1-RHR-XHE-XL- LTREC, 1-RHR-XHE-XL-TRN- MLDRL	
	5.880E-1			1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	4.070E-3			1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT- 4)-(demand-based)
	1.550E-2			1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.250E-2			1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN - MLDRL
2	1.167E-07	0.93	19.44	1-IE-SD-ODML-EARLY, 1-LPI- MDP-CF-START, 1-POS-6-POT- 4-MLDRL	
	5.880E-1			1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	4.878E-5			1-LPI-MDP-CF-START	CCF OF RHR PUMPS A & B TO START
	4.070E-3			1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT- 4)-(demand-based)
3	9.429E-09	0.08	1.57	1-IE-SD-ODML-EARLY, 1-LPI- MDP-CF-RUN, 1-POS-6-POT-4- MLDRL	
	5.880E-1			1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling

Table 9-12 Cut Sets for Sequence 1-SD-ODML-EARLY: 2-08-1 (cont.)

Sequence Name / Cut Set #	Prob./Freq. (per year)	% Contrib. to Total CDF	% Contrib. to Seq.	Basic Event Name	Event Description
	3.940E-6			1-LPI-MDP-CF-RUN	CCF OF RHR PUMPS A & B TO RUN
	4.070E-3			1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT- 4)-(demand-based)

9.6 Basic Event Importance Measures

Importance measures provide different ways to determine the contribution of individual basic events to the overall risk. In this project, the following importance measures are calculated:

- Fussell-Vesely (F-V) - an indication of the percentage of the overall risk metric result (CDF) contributed by the cut sets containing the basic event.
- Risk Increase Ratio (RIR) – also referred to as risk achievement worth; an indication of how much the overall risk metric (CDF) would go up if the specific event had probability equal to 1.0, corresponding to totally unreliable equipment or action.
- Risk Reduction Ratio (RRR) – also referred to as risk reduction worth; an indication of how much the overall risk metric (CDF) would be reduced if the specific event probability equaled zero, corresponding to a totally reliable piece of equipment or action.
- Birnbaum - an indication of the sensitivity of the overall risk metric (CDF) with respect to the basic event of concern.

Additional information on how the importance measures are calculated is provided in the SAPHIRE documentation (Smith, 2011). The importance measures for all significant basic events are included in 12APPENDIX B of this report. Significant basic events are defined as those basic events that have a Fussell-Vesely importance greater than 0.005 or a risk-achievement greater than 2 (NRC, 2020b). A subset of these significant basic events is presented in the following tables. Table 9-13 provides all the basic events with a Fussell-Vesely importance measure greater than 0.005. However, specifically excluded from the table are the basic events representing the shutdown initiating event frequencies, as well as the POT and POS duration factors, since information on these events is provided in the tables in Sections 9.2 and 9.3, respectively.

From Table 9-13, the highest Fussell-Vesely importance events involve failures related to restoring onsite or offsite AC power in response to LOOP events. Many of the other high Fussell-Vesely events are related to the modeling of LOOP events. The third highest event is related to the availability of the alternate AC power source during LOOP events, which is assumed unavailable if the offsite grid is impacted. The highest human failure event is failure to establish gravity-driven feed to the RCS during a LOOP event. Other important failure events include equipment failures related to emergency AC power (e.g., emergency diesel generators, RAT supply circuit breakers, and EDG load sequencers) and RHR pump failures. Several other human failures related to response to shutdown initiating events have significant Fussell-Vesely importance measures.

To further evaluate the types of failures that contribute to the overall CDF result, the subsequent tables in this section provide the top (typically ten) basic events in terms of Fussell-Vesely importance for each of the following categories: human failure events, recovery/restoration events, equipment failures, and common-cause failures.

As mentioned above, one category of basic events that have high contributions in terms of Fussell-Vesely importance is human failure events. The top events in this category (which excludes human failure events associated with recovery/restoration events) are shown in Table 9-14. The large contribution to shutdown CDF from human failure events is expected

because the response to most shutdown initiating events requires operator action to mitigate the accident progression, as opposed to full power conditions where there is a greater reliance on automatically actuated safety systems. The most important human failures are those associated with gravity-driven feed during LOOP events. The next most important human failures involve failure to align the alternate AC power source in response to LOOP, failure to diagnose a loss of inventory event during pressurizer solid operation (POS 5A), and failure to recover RHR in response to an overdrain event entering mid-loop.

The next category of important basic events relates to recovering failed systems or restoration actions. The top events in this category are shown in Table 9-15. These basic events represent failures of actions to restore RHR cooling or to restore AC power. The importance of these events reflects the assumptions regarding reaching a safe and stable state (as discussed in Section 5.2.2). As mentioned previously, long-term restoration of RHR cooling is required to reach a safe and stable state for all accident sequences where steam generator cooling is unavailable. For those sequences where offsite power is lost, restoring AC power, either by restoring normal offsite power or emergency diesel generators, is required before RHR cooling can be re-established.

Table 9-16 and Table 9-17 show importance measures for basic events representing independent and common-cause failures of equipment, respectively. Several individual component failures in Table 9-16 have higher Fussell-Vesely measures than the CCF events in Table 9-17. Many of the events involve failures related to the onsite emergency power system (e.g., diesel generators, circuit breakers, and sequencer). For some POSs (e.g., POS 5A, POS 8E, and POS 8L), only one train of emergency power is available. Under these conditions the importance of the single train failure events is heightened.

Table 9-13 Overall Important Basic Events with Fussell-Vesely Measure > 0.005

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-EPS-XHE-XL-NR04H	6252	6.984E-01	4.55E-01	1.20E+00	1.835E+00	8.166E-06	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
1-OEP-XHE-XL-NR02H	9019	3.264E-01	4.33E-01	1.90E+00	1.765E+00	1.665E-05	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
OEP-VCF-LP-GRWR	4071	4.100E-01	3.48E-01	1.50E+00	1.534E+00	1.065E-05	SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
1-LPI-XHE-XM-GFEED-LOOP	4881	5.500E-01	2.84E-01	1.23E+00	1.396E+00	6.463E-06	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
1-EPS-DGN-FR-G4002	2158	3.297E-02	1.85E-01	6.43E+00	1.227E+00	7.039E-05	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
1-OEP-XHE-XX-NR02H1	1532	2.729E-01	1.69E-01	1.45E+00	1.204E+00	7.774E-06	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
1-ACP-CRB-CF-A205301	214	3.498E-04	1.29E-01	3.70E+02	1.149E+00	4.628E-03	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
1-ACP-CRB-CC-BA0301	1528	5.350E-03	1.08E-01	2.11E+01	1.121E+00	2.535E-04	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
1-RHR-XHE-XL-LTREC	311	1.550E-02	9.29E-02	6.90E+00	1.102E+00	7.517E-05	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
1-OA-ALIGNPW-G-H	2281	9.200E-02	7.81E-02	1.77E+00	1.085E+00	1.064E-05	OPERATOR FAILS TO ALIGN Alternate offsite AC - General case
1-LPI-MDP-CF-START	62	4.878E-05	7.20E-02	1.47E+03	1.078E+00	1.836E-02	CCF OF RHR PUMPS A & B TO START
1-EPS-SEQ-FO-1821U302	1309	3.330E-03	6.76E-02	2.12E+01	1.072E+00	2.544E-04	SEQUENCER B FAILS TO OPERATE
1-EPS-SEQ-CF-FOAB	238	2.148E-04	6.50E-02	3.03E+02	1.070E+00	3.788E-03	CCF OF SEQUENCERS TO OPERATE
1-L2-BE-RHRVSTUCK-SD	171	6.694E-02	6.14E-02	1.86E+00	1.065E+00	1.150E-05	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
1-LOI-XHE-XD-POS5A	171	2.500E-02	6.14E-02	3.40E+00	1.065E+00	3.080E-05	OPERATOR FAILS TO DIAGNOSE LOI EARLY
1-EPS-DGN-FS-G4002	1171	2.940E-03	5.78E-02	2.06E+01	1.061E+00	2.465E-04	DG1B RANDOMLY FAILS TO START

Table 9-13 Overall Important Basic Events with Fussell-Vesely Measure > 0.005 (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-RHR-XHE-XL-TRN-MLDRL	84	1.250E-02	4.76E-02	4.76E+00	1.050E+00	4.776E-05	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN -MLDRL
1-IE-SWS-MDP-CR-123456	8	3.031E-05	4.22E-02	1.38E+03	1.044E+00	1.732E-02	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-
1-RHR-XHE-XM-MLDRL-I	85	1.020E-02	3.83E-02	4.72E+00	1.040E+00	4.707E-05	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML
1-RHR-XHE-XL-LTREC-LT5H	432	1.505E-01	3.36E-02	1.19E+00	1.035E+00	2.802E-06	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
1-RHR-XHE-XM-TRIP-MD	13	1.460E-01	2.75E-02	1.16E+00	1.028E+00	2.357E-06	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS (MODERATE DEPENDENCY)
1-OEP-XHE-XL-NR01H	2304	5.196E-01	2.10E-02	1.02E+00	1.021E+00	5.070E-07	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 1 HOUR
1-HPI-XHE-XM-FFFEED-MLDRL	8	7.950E-03	2.04E-02	3.55E+00	1.021E+00	3.220E-05	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-MLDRL
1-EPS-DGN-FR-G4001___	1486	3.297E-02	1.66E-02	1.49E+00	1.017E+00	6.328E-06	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
1-LPI-XHE-XM-GFEED-LD	29	4.520E-01	1.17E-02	1.01E+00	1.012E+00	3.233E-07	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-(LOW DEPENDENCY)-LOI
1-NSCWCT-SPRAY	1858	9.040E-01	1.12E-02	1.00E+00	1.011E+00	1.553E-07	NSCW CTS IN SPRAY MODE (FRACTION OF TIME)
1-RHR-XHE-XL-TRN-POS6	87	1.250E-02	1.10E-02	1.87E+00	1.011E+00	1.101E-05	OPERATORS FAIL TO RECOVER ONE RHR TRAIN
1-LPI-XHE-XM-GFEED-MD	81	5.050E-01	1.06E-02	1.01E+00	1.011E+00	2.639E-07	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED- (MODERATE DEPENDENCY)-MLDRL
1-ACP-CRB-CC-AA0205__	1107	5.350E-03	1.04E-02	2.93E+00	1.010E+00	2.426E-05	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN

Table 9-13 Overall Important Basic Events with Fussell-Vesely Measure > 0.005 (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-OA-OSW-----H	140	2.000E-02	8.83E-03	1.43E+00	1.009E+00	5.532E-06	OPERATORS FAIL TO ESTABLISH SINGLE PUMP NSCW PUMP OPERATION
1-OEP-VCF-LP-CLOPT	2066	5.300E-03	8.41E-03	2.58E+00	1.008E+00	1.989E-05	CONSEQUENTIAL LOSS OF OFFSITE POWER - TRANSIENT
1-SWS-MOV-CC-1669A____	541	3.530E-04	7.00E-03	2.08E+01	1.007E+00	2.487E-04	NSCW CT B SPRAY VALVE HV1669A FAILS TO OPEN ON DEMAND
1-LPI-MDP-CF-RUN	31	3.940E-06	6.84E-03	1.72E+03	1.007E+00	2.152E-02	CCF OF RHR PUMPS A & B TO RUN
1-EPS-DGN-CF-FRUN1	218	3.237E-04	6.82E-03	2.21E+01	1.007E+00	2.639E-04	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN
1-EPS-SEQ-FO-1821U301	939	3.330E-03	6.72E-03	3.01E+00	1.007E+00	2.531E-05	SEQUENCER A FAILS TO OPERATE
1-OEP-XHE-XX-NR01H1	536	2.258E-01	6.61E-03	1.02E+00	1.007E+00	3.672E-07	CONVOLUTION FACTOR FOR 1FTR-OPR (1HR AVAIL)
1-LOI-XHE-XM-ISO2-LT	14	2.440E-02	5.83E-03	1.23E+00	1.006E+00	2.994E-06	FAIL TO ISOLATE LOI AFTER SUCCESSFUL INJECTION
1-LPI-MDP-FS-RHRB	723	1.000E-03	5.79E-03	6.79E+00	1.006E+00	7.261E-05	RHR PUMP B RANDOMLY FAILS TO START
1-AFW-TDP-FR-P4001	633	3.802E-02	5.32E-03	1.14E+00	1.005E+00	1.753E-06	TDAFWP (P4-001) FAILS TO RUN

Table 9-14 Human Failure Event Importance Measures

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-LPI-XHE-XM-GFEED-LOOP	4881	5.500E-01	2.84E-01	1.23E+00	1.396E+00	6.463E-06	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
1-RHR-XHE-XL-LTREC	311	1.550E-02	9.29E-02	6.90E+00	1.102E+00	7.517E-05	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
1-OA-ALIGNPW-G-H	2281	9.200E-02	7.81E-02	1.77E+00	1.085E+00	1.064E-05	OPERATOR FAILS TO ALIGN Alternate offsite AC - General case
1-LOI-XHE-XD-POS5A	171	2.500E-02	6.14E-02	3.40E+00	1.065E+00	3.080E-05	OPERATOR FAILS TO DIAGNOSE LOI EARLY
1-RHR-XHE-XL-TRN-MLDRL	84	1.250E-02	4.76E-02	4.76E+00	1.050E+00	4.776E-05	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN -MLDRL
1-RHR-XHE-XM-MLDRL-I	85	1.020E-02	3.83E-02	4.72E+00	1.040E+00	4.707E-05	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML
1-RHR-XHE-XM-TRIP-MD	13	1.460E-01	2.75E-02	1.16E+00	1.028E+00	2.357E-06	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS (MODERATE DEPENDENCY)
1-HPI-XHE-XM-FFEED-MLDRL	8	7.950E-03	2.04E-02	3.55E+00	1.021E+00	3.220E-05	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-MLDRL
1-LPI-XHE-XM-GFEED-LD	29	4.520E-01	1.17E-02	1.01E+00	1.012E+00	3.233E-07	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-(LOW DEPENDENCY)-LOI
1-RHR-XHE-XL-TRN-POS6	87	1.250E-02	1.10E-02	1.87E+00	1.011E+00	1.101E-05	OPERATORS FAIL TO RECOVER ONE RHR TRAIN

Table 9-15 Repair and Restoration Basic Event Importance Measures

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-EPS-XHE-XL-NR04H	6252	6.984E-01	4.55E-01	1.20E+00	1.835E+00	8.166E-06	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
1-OEP-XHE-XL-NR02H	9019	3.264E-01	4.33E-01	1.90E+00	1.765E+00	1.665E-05	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
1-RHR-XHE-XL-LTREC	311	1.550E-02	9.29E-02	6.90E+00	1.102E+00	7.517E-05	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
1-RHR-XHE-XL-LTREC-LT5H	432	1.505E-01	3.36E-02	1.19E+00	1.035E+00	2.802E-06	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection within less than 5 hours
1-OEP-XHE-XL-NR01H	2304	5.196E-01	2.10E-02	1.02E+00	1.021E+00	5.070E-07	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 1 HOUR

Table 9-16 Equipment Failure Importance Measures

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-EPS-DGN-FR-G4002____	2158	3.297E-02	1.85E-01	6.43E+00	1.227E+00	7.039E-05	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
1-ACP-CRB-CC-BA0301__	1528	5.350E-03	1.08E-01	2.11E+01	1.121E+00	2.535E-04	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
1-EPS-SEQ-FO-1821U302	1309	3.330E-03	6.76E-02	2.12E+01	1.072E+00	2.544E-04	SEQUENCER B FAILS TO OPERATE
1-L2-BE-RHRVSTUCK-SD	171	6.694E-02	6.14E-02	1.86E+00	1.065E+00	1.150E-05	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
1-EPS-DGN-FS-G4002____	1171	2.940E-03	5.78E-02	2.06E+01	1.061E+00	2.465E-04	DG1B RANDOMLY FAILS TO START
1-EPS-DGN-FR-G4001____	1486	3.297E-02	1.66E-02	1.49E+00	1.017E+00	6.328E-06	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
1-ACP-CRB-CC-AA0205__	1107	5.350E-03	1.04E-02	2.93E+00	1.010E+00	2.426E-05	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
1-SWS-MOV-CC-1669A____	541	3.530E-04	7.00E-03	2.08E+01	1.007E+00	2.487E-04	NSCW CT B SPRAY VALVE HV1669A FAILS TO OPEN ON DEMAND
1-EPS-SEQ-FO-1821U301	939	3.330E-03	6.72E-03	3.01E+00	1.007E+00	2.531E-05	SEQUENCER A FAILS TO OPERATE
1-LPI-MDP-FS-RHRB____	723	1.000E-03	5.79E-03	6.79E+00	1.006E+00	7.261E-05	RHR PUMP B RANDOMLY FAILS TO START

Table 9-17 Common Cause Failure Importance Measures

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-ACP-CRB-CF-A205301	214	3.498E-04	1.29E-01	3.70E+02	1.149E+00	4.628E-03	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
1-LPI-MDP-CF-START	62	4.878E-05	7.20E-02	1.47E+03	1.078E+00	1.836E-02	CCF OF RHR PUMPS A & B TO START
1-EPS-SEQ-CF-FOAB	238	2.148E-04	6.50E-02	3.03E+02	1.070E+00	3.788E-03	CCF OF SEQUENCERS TO OPERATE
1-IE-SWS-MDP-CR-123456	8	3.031E-05	4.22E-02	1.38E+03	1.044E+00	1.732E-02	CCF OF NSWS PUMPS TO RUN System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-LPI-MDP-CF-RUN	31	3.940E-06	6.84E-03	1.72E+03	1.007E+00	2.152E-02	CCF OF RHR PUMPS A & B TO RUN
1-EPS-DGN-CF-FRUN1	218	3.237E-04	6.82E-03	2.21E+01	1.007E+00	2.639E-04	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN

9.7 Truncation and Model Convergence

The quantification results for all PRA models are impacted by the truncation limits used during the quantification process. The truncation limit used should demonstrate convergence towards a stable result. ASME/ANS Level 1 PRA standard (ASME/ANS, 2009) supporting requirements QU-B2 and QU-B3 dictate that:

- The quantification truncation level is set at a sufficiently low value that dependencies associated with significant cut sets or accident sequences are not eliminated.
- An iterative process with convergence is considered sufficient when successive reductions in truncation value of one decade result in decreasing changes in CDF, and the final change is less than five percent.

The L3PRA Level 1 LPSP PRA model for internal events was quantified at various truncation levels to ensure model (CDF) convergence and to determine how many cut sets were removed at the various levels. The results of these calculations are provided in Table 9-18. Based on these results, a truncation level of 10^{-10} per year is deemed sufficient to satisfy supporting requirements QU-B2 and QU-B3. However, a truncation level of 10^{-12} per year was ultimately used for consistent with the truncation level used for the L3PRA project Level 1 PRA for at-power internal events (NRC, 2022).

Table 9-18 CDF Truncation Analysis Summary

Truncation (per year)	CDF (per year)	Δ CDF (%)	# of cut sets
1.00E-09	1.18E-05	N/A	639
1.00E-10	1.23E-05	4.0%	2232
1.00E-11	1.25E-05	1.3%	7079
1.00E-12	1.25E-05	0.4%	20357
1.00E-13	1.26E-05	0.2%	55716

9.8 Comparison to Other Studies

There is limited experience with developing low power and shutdown PRAs for U.S. plants, which makes it difficult to identify a basis for comparison. For the few LPSP PRAs that have been developed for U.S. plants, the documentation and results are not available to the authors. One study that was identified for comparison is NUREG/CR-6144, *Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1* (NRC, 1994a).

It should be noted that there are many differences between the L3PRA project and NUREG/CR-6144 that should be taken into consideration while comparing the results. Some of the notable differences are outlined here. First, the plants are different designs, with Surry being a 3-loop PWR and the reference plant being a 4-loop PWR. Also, the two PRA studies have a different scope. NUREG/CR-6144 provides a detailed screening analysis, but the reported results focus on three mid-loop POSs: POS 6 and POS 10 during a refueling outage, and POS 6 during a drained maintenance outage. The quantitative results for the reference plant LPSP PRA cover POS 3 through POS 13 for a refueling outage. The list of modeled initiating events also differs

between the two studies, with NUREG/CR-6144 covering the following initiating event categories:

- Loss of RHR (including loss of RHR due to overdraining)
- Loss of offsite power
- Loss of 4kV AC bus
- Loss of vital bus
- Loss of instrument air
- Loss of CCW
- Loss of emergency switchgear room cooling
- Inadvertent ESFAS actuation
- Boron dilution

The reference plant LPSD PRA covers the following initiating event groups:

- Loss of RHR
- Loss of offsite power
- Loss of power to 4kV AC bus
- Loss of inventory
- Overdraining entering mid-loop
- Loss of service water

Also, NUREG/CR-6144 was completed in 1994, with much of the work completed in the early 1990s. During this time, the U.S. industry was in the process of adjusting to a risk-informed approach to outage management. Some of the key information sources had recently become available around this time:

- Generic Letter 88-17 (NRC, 1988) was issued in 1988, with licensee responses coming in the following years
- NUREG-1410 (NRC, 1990), which provides a summary of the loss of offsite power event during shutdown that occurred at Reference Plant in 1990, was published that same year
- NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management* (NUMARC, 1991) was issued in 1991
- NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States* (NRC, 1993) was published in 1993

In response to these and other information, many changes to the licensees' approach to outage management were implemented around the mid-1990s. It is important to take these differences into consideration when comparing results from these studies that were completed in two different eras.

Some of the highest contributing accident sequences in the reference plant LPSD PRA are losses of inventory during POSs with RCS intact. NUREG/CR-6144 did not quantify results for loss of inventory or LOCAs. All the LOCA sequences considered in the NUREG/CR-6144 screening analysis were assigned to low or medium frequency categories, and none were assigned to the high frequency categories. In NUREG/CR-6144, the frequencies for LOCAs that

are relevant to shutdown POSs were derived from a review of Licensee Event Reports, and the frequencies range from 1×10^{-3} to 5×10^{-3} per shutdown-year (see Section 4.5.4 of (NRC, 1994a)). By comparison, the loss of inventory frequencies for this project have a much broader range from 1×10^{-5} to 6×10^{-1} per shutdown-year. The difference in initiating event frequencies is likely the main reason for the different modeling approaches taken by the two studies.

Another reference point for the loss of inventory results is identified in NUREG-1449. In a survey of LPSD PRA studies, the report identifies a study by Science Applications International Corporation that addresses postulated core melt accidents initiated by LOCAs at the Sequoyah nuclear plant. NUREG-1449 states, “[a] total of 20 cases were analyzed with varying assumptions regarding time of LOCA initiation following shutdown, LOCA size, availability of offsite power, and maintenance status. The postulated core-melt frequency was estimated to be in the range from 7.53×10^{-5} to 8.5×10^{-7} per reactor-year.” For the L3PRA, the total contribution to shutdown CDF from loss of inventory scenarios is 8.6×10^{-7} per year.

One of the key similarities between the two studies is the focus on risk related to mid-loop operation. Through a detailed screening analysis, NUREG/CR-6144 identified mid-loop POSs as the highest contributors to the most severe screening criteria categories. The Reference Plant LPSD PRA shows that about 19 percent of the total estimated shutdown CDF is associated with mid-loop POSs (POS 6 and POS 10). Mid-loop is an important contributor, but there are other POSs with similar CDF contributions. A summary of comparable mid-loop CDF estimates for similar POS and accident scenarios is shown in Table 9-19.

Table 9-19 Comparison of Mid-loop CDF Estimates to NUREG/CR-6144

Initiating Event Category	NUREG/CR-6144 CDF (per year) in POS R6	Reference Plant LPSD PRA CDF (per year) in POS 6	References and Notes
Loss of RHR (excluding overdrain)	2.2E-07	1.3E-07	Results from Table 1 of (Chu, 1995) summing all loss of RHR categories, excluding RHR2A-Over Draining
Overdrain	1.8E-07	1.5E-06	RHR2A-Over Draining from Table 1 of (Chu, 1995)
Loss of offsite power	5.6E-07	2.2E-07	All LOOP categories summed from Table 1 of (Chu, 1995)
Loss of 4 kvac bus	1.4E-07	1.4E-08	Loss of 4 kV bus from Table 1 of (Chu, 1995)

The results for loss of RHR and loss of offsite power appear to be similar for both studies, but there are many modeling differences, including the definitions of initiating event categories, what recovery actions are credited, quantification of human failure events, and configuration of the plant. Another important difference between the studies is the timing and duration of the POS, which factors into the annual CDF calculation. NUREG/CR-6144 bases the fraction of time spent in mid-loop on past experience at Surry. The fraction of time in mid-loop is much greater than what is shown in the reference plant’s recent operating experience. In NUREG/CR-6144, the average annual fraction in mid-loop for POS R6 (mid-loop before refueling) is 1.6 percent. The average annual fraction in POS 6 for this LPSD PRA is 0.1 percent. Though some of the

numbers in Table 9-19 are comparable, the difference in POS duration suggests the conditional risk results from this LPSD PRA are higher than those modeled in the Surry study.

NUREG/CR-6144 also presents CDF results conditional on different conditions that could occur during a POS. A comparison of conditional CDF results is shown in Table 9-20. The average conditional CDF result reported in NUREG/CR-6144 for POS R6 is 8.3×10^{-5} . This result is based on a weighted average of conditional CDF for four different time windows in which mid-loop operations could occur. Each of the time windows is modeled with different plant and operator responses, due to differences in decay heat load, maintenance unavailabilities, and likelihood of plant configurations that can impact mitigating systems. For example, the Surry study generally credits reflux cooling with steam generators as a viable decay heat removal method during mid-loop operations. The reference plant outage history and operating procedures indicate that a large vent will be established prior to entering mid-loop, and this plant configuration prevents any effective cooling with steam generators. Therefore, reflux cooling is never credited during mid-loop in the reference plant LPSD PRA model. NUREG/CR-6144 defines Time Window 1 (13 to 75 hours after plant shutdown) with a high probability of 0.3 that the steam generator loops are isolated, which makes reflux cooling impossible. Consequently, the conditional CDF results for Time Window 1 are much higher than the averaged results. The results for Time Window 1, with reduced credit for reflux cooling, are more appropriate for comparison to the reference plant LPSD PRA results.

Table 9-20 Comparison of Mid-loop Conditional CDF Estimates to NUREG/CR-6144

Source of Results	NUREG/CR-6144 Conditional CDF (per year) in POS R6	Reference Plant LPSD PRA Conditional CDF (per year) in POS 6	References and Notes
NUREG/CR-6144 POS R6, Average	8.3E-05	3.8E-04	Average result for R6 in Table 3 of (Chu, 1995). POS 6 result from Table 9-6 above.
NUREG/CR-6144 POS R6, Window 1	1.0E-03		Window 1 result for R6 in Table 3 of (Chu, 1995). Result with reduced credit for reflux cooling may be appropriate for comparison.

A direct comparison of different LPSD PRA studies can be misleading. Any comparison must be made with consideration of the possible differences in key modeling assumptions. As there is limited experience with development of LPSD PRAs for U.S. plants, there is not a consistent state of practice for many of these modeling assumptions. A list of some LPSD-related model inputs that could be important is provided below.

- Plant design (e.g., 3-loop PWR versus 4-loop PWR)
- Plant approach to outage management
- Design of key systems used for low power and shutdown operation. For example, the RHR system in shutdown cooling mode may have only one hot leg suction line in some plants, while other plants have two.

- The scope of the PRA models. Models may cover different outage types, POSs, initiating events, and hazard categories.
- Plant operating state definitions. Plants may use similar terms and numbering to identify POSs, but due to differences in plant design, operation, and/or modeling approach, the POSs may have very different boundary conditions.
- Plant operating state frequency, timing, and duration. These factors can vary significantly based on the outage type, from plant to plant, and from outage to outage.
- Initiating event category definitions. For example, a loss of RHR initiating event can include many different cause categories, may or may not include support system failures, and may be recoverable or non-recoverable.
- The data used for developing initiating event frequencies and failure rates. The timeframe of the generic and plant-specific data can be important, as outage management practices have changed over time. Also, some data sources may be incomplete for LPSD operating experience. For example, not all LPSD events are documented in Licensee Event Reports.
- Credit for accident mitigating systems and strategies. The approach to crediting steam generators for cooling in different plant configurations can vary. The approach for crediting “feed and bleed” or “fill and spill” cooling can vary. Plants may have different operational approaches, procedural guidance, and supporting technical basis calculations, which can all influence the credit given for accident mitigation.
- Human failure event quantification. The choice of HRA methods and dependency models used to quantify human failure events can have a significant impact on results, given the importance of human failures to LPSD accident sequences.

10 UNCERTAINTY ANALYSIS

10.1 CDF Uncertainty Analysis Approach

This section describes the approach for assessing the uncertainty of the CDF results for the LPSD Level 1 PRA. The uncertainties impacting the CDF results are of the following two types: parameter uncertainty and modeling uncertainty. The two types of uncertainty are treated in different ways.

Parameter uncertainty is any uncertainty that relates to a parameter in the PRA logic model that has a probability value. The uncertainty of each parameter is characterized by a probability distribution. The overall uncertainty is estimated by using a sampling approach to propagate the uncertainty of all the contributing parameters. The results of the parameter uncertainty assessment are described in Section 10.2.

Model uncertainty is any uncertainty that impacts the model results related to modeling assumptions, model scope, or underlying supporting analyses. The model uncertainties are assessed using sensitivity studies. The sensitivity studies consider alternate assumptions and consider the impacts on the model results. The model uncertainty sensitivity studies are described in Section 10.3.

10.2 CDF Parameter Uncertainty

The mean CDF is estimated using a Monte Carlo sampling approach using the uncertainty distributions of the contributing basic events. The state-of-knowledge correlations are addressed by assigning correlation classes to basic events with similar component types and failure modes, as described in the SAPHIRE technical reference manual (Smith, 2011). The CDF mean value and uncertainty results are provided in Table 10-1. The shutdown CDF cumulative distribution function is shown in Figure 10-1, and the probability density is shown in Figure 10-2.

Table 10-1 CDF Parameter Uncertainty Results

Point Estimate CDF	Mean CDF	5 th Percentile	Median	95 th Percentile	Standard Deviation
1.3E-05/yr	1.2E-05/yr	2.7E-06/yr	8.9E-06/yr	2.9E-05/yr	9.3E-06/yr

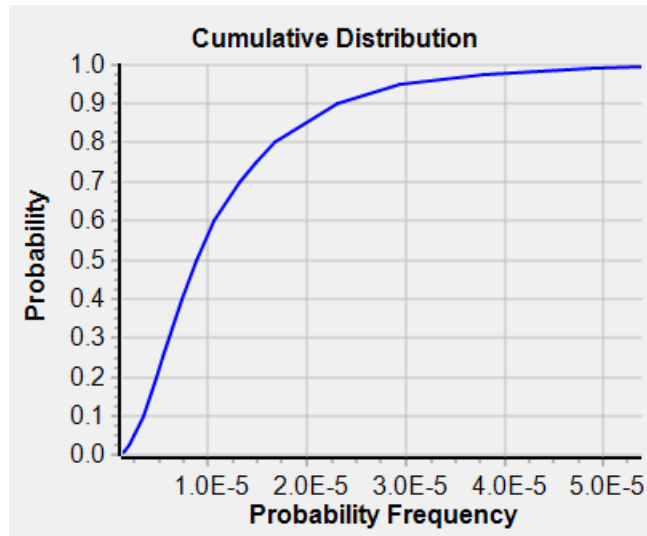


Figure 10-1 Cumulative Distribution Function for Shutdown CDF

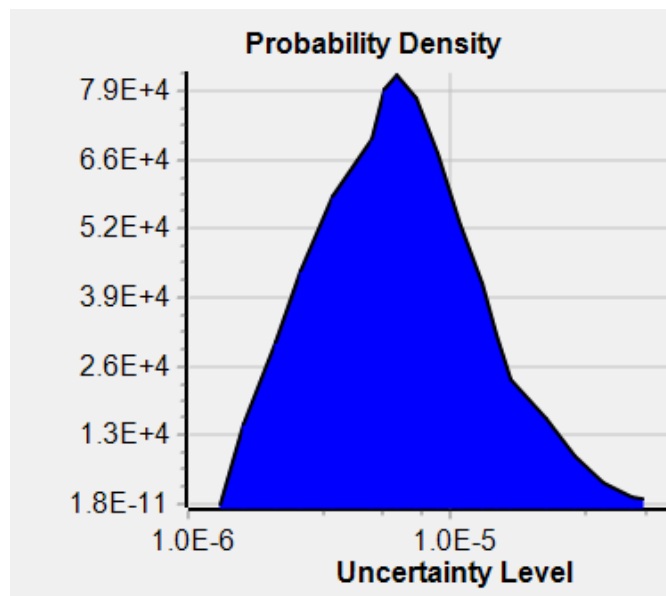


Figure 10-2 Probability Density for Shutdown CDF

10.3 Model Uncertainty and Sensitivity Studies

This section assesses potential model uncertainty and considers the impacts on the model results. Sources of model uncertainty were identified by a systematic review of the model elements and supporting analyses. Refer to Appendix C, for a complete list of modeling uncertainties and a qualitative characterization of their impacts. A set of sensitivity studies were developed to assess the impacts of important model uncertainties. These sensitivity studies were selected based on reviewer feedback from the project Technical Advisory Group (TAG), feedback during ACRS briefings, and consideration of the expected impact on the PRA results.

Table 10-2 shows identified sources of model uncertainty where a sensitivity study has been performed to assess the impacts on the model results. The sensitivity cases are described in Sections 10.3.1 through 10.3.10 and summarized in Section 10.3.11.

10.3.1 Modeling Scope of Shutdown Accident Scenarios

The results described in Section 9 outline the contributions to shutdown CDF from the modeled initiating events and plant operating state conditions for a refueling outage. The modeled POSs include all shutdown states from POS 3 through POS 13, and there are six initiating event categories represented. The contributions from the POS and initiating event combinations are summarized in Table 9-4. However, as shown in the table, there are several POS-initiating event combinations that are not modeled.

This sensitivity study develops estimated CDF contributions for the unmodeled POS-initiating event combinations. The approach is to use the calculated conditional core damage probability for a scenario with similar conditions or conditions that would bound the conditions of the unmodeled scenario. There are limitations with this approach because there are aspects of the POS-specific modeling that are not addressed, for example, decay heat level, sequence timing, human failure events, and plant configurations. Despite the limitations, the numerical results give a sense of the potential contributions from unmodeled scenarios.

These are the key assumptions for estimating the unmodeled shutdown scenarios:

- Loss of inventory during POS 5A is a significant contributor to shutdown CDF. The high initiating event frequency is one of the key factors contributing to the significant CDF result. Similar loss of inventory scenarios could occur during other POSs with intact RCS. The CCDP for loss of inventory in POS 5A is applied to POSs 3 and 4. The post-refueling conditions in POS 11 through 13 are likely over-estimated with this approach. For these POSs, the loss of inventory frequency and CCDP for POS 5B are used as a reference point.
- The loss of NSCW shutdown scenarios are dominated by the common-cause failure of the NSCW pumps. For all POSs, re-establishing single pump operation is possible if any pump is available. The different POS configurations have only a limited impact on the CDF results. The initiating event frequency and CCDP for POS 5B are used as a reference point for all the unmodeled POSs. The limited time available to respond to the event due to lowered inventory level in POS 5B is expected to be more restrictive than the conditions of the unmodeled POSs.
- For the loss of power to 4,160V AC safety-related bus scenarios, the availability of the redundant train bus is the key factor influencing the CDF results. However, for the unmodeled POSs, both trains are assumed to be available. The initiating event frequency and CCDP for POS 5B are used as a reference point for POSs with both 4,160V AC safety-related buses initially available. Again, the limited time available to respond to the event due to lowered inventory level in POS 5B is expected to be more restrictive than the conditions of the unmodeled POSs.

Table 10-2 Sources of Model Uncertainty

Source of Model Uncertainty	Assessment of Model Impact	Sensitivity Study
Modeling scope of shutdown accident scenarios (i.e., modeled initiating events and POSs)	The large number of potential initiating events and POSs can require significant time and resources to address each in a detailed manner. A prioritization scheme (see Section 4.2.2) was developed to focus the modeling on the most significant scenarios. The remaining unmodeled scenarios are expected to have a small contribution to shutdown CDF (e.g., <1% of CDF).	The approach used to prioritize shutdown accident scenarios is expected to capture the risk significant contributors to shutdown CDF. However, the contributions from unmodeled accident scenarios is not known. A sensitivity case, described in Section 10.3.1, was developed to estimate the CDF contributions from unmodeled shutdown accident scenarios.
Low power and hot standby POS risk contribution	The model development focuses on the shutdown POSs. Section 4.2.1 describes potential accident scenarios during low power and hot standby POSs, but these were not included in the base model. The CDF contribution from low power and hot standby POSs is expected to be a small fraction of the total outage CDF.	Section 10.3.2 describes sensitivity cases to estimate the CDF contributions from low power and hot standby POSs during a refueling outage.
Other outage types	The model development for this project focuses on a refueling outage (POT-4). While refueling outages are expected to be the dominant CDF contributors, other types of plant outages can also contribute to low power and shutdown risk.	Section 10.3.3 describes sensitivity cases to estimate the CDF contributions from other outage types (POT-1, POT-2, and POT-3).
Plant outage with significant maintenance activities	Unavailability of risk-significant equipment due to maintenance activities can significantly influence the outage risk results.	Section 10.3.4 describes a sensitivity case to assess the estimated CDF contribution from a cold shutdown maintenance outage (POT-2) with one safety-related emergency diesel generator unavailable for maintenance.
Loss of offsite power initiating event frequencies	The uncertainties associated with how the loss of offsite power initiating event frequencies are applied to shutdown POSs can significantly influence the results.	Section 10.3.5 considers alternate loss of offsite power initiating event frequencies and assesses the impacts on the shutdown CDF.

Table 10-2 Sources of Model Uncertainty (cont.)

Source of Model Uncertainty	Assessment of Model Impact	Sensitivity Study
Likelihood of overdrain entering mid-loop	The conditional probability of overdraining the RCS level when entering a mid-loop condition is difficult to estimate due to limitations of the shutdown initiating event data collection and tracking the number of mid-loop entries at plants. The probability used in the model is based on events occurring in the 1990s as documented in (EPRI, 2010).	While there has not been consistent data collection on overdrain events in recent years, the possible impacts of incorporating more recent data are explored in a series of sensitivity studies in Section 10.3.6.
Time available for recovery of offsite and onsite power	In response to loss of offsite power events, the recovery of offsite and onsite power sources is limited by the plant batteries that are necessary for restoring power. This limits the time available that is credited for power recovery. However, some shutdown scenarios are very slowly developing (e.g., 3-4 days for core damage initiation).	The impact of crediting additional time for offsite and onsite power recovery for slowly developing scenarios is assessed in Section 10.3.7.
Estimation of human error probabilities	The modeling of human failure events (HFEs) and estimating human error probabilities (HEPs) is a key source of modeling uncertainty in the shutdown PRA. Several HFEs show significant contributions to the shutdown CDF results. Therefore, the estimation of HEP values is important. The HEP estimates can vary depending on the HRA method used and modeling assumptions.	A sensitivity study is performed in Section 10.3.8 to assess the impacts of alternate HEP values for significant HFEs.
Impact of steam generator nozzle dam failures.	Nozzle dams are installed during mid-loop operation to isolate steam generators for surveillance and maintenance. Information on nozzle dam failure mechanisms and operating experience are not widely available, but the impacts of nozzle dam failures are potentially significant.	A sensitivity study to assess the impacts of nozzle dam failures is provided in Section 10.3.9.

Table 10-2 Sources of Model Uncertainty (cont.)

Source of Model Uncertainty	Assessment of Model Impact	Sensitivity Study
Equipment unavailability for maintenance and testing	The assumptions regarding equipment unavailabilities for maintenance and testing can have a significant influence on the shutdown PRA results. The specific equipment unavailability, timing and duration of unavailability can vary from outage to outage. As the LPSD is not modeling an actual outage, the equipment unavailabilities are based on assumptions about routinely performed activities and informed by past outage reports from the reference plant.	A sensitivity study to assess the impacts of equipment unavailability assumptions is developed in Section 10.3.10 with variations on the durations of major equipment outages.

The estimated CDF contributions, including the unmodeled POS-initiating event combinations, are shown in Table 10-3. The estimated contributions from the unmodeled POS-initiating event combinations show that the unmodeled scenarios are not expected to have dominant contributions to the shutdown CDF. Table 10-4 shows the rank order results for the POS-initiating event combinations contributing to the top 95 percent of the shutdown CDF. The unmodeled scenarios are all estimated to contribute about 1 percent or less of the shutdown CDF, and only four scenarios contribute to the top 95 percent of CDF. The four scenarios are loss of NSCW during POS 12 and POS 5A and loss of inventory during POS 4 and POS 3. The total estimated shutdown CDF including contributions from modeled and unmodeled scenarios is summarized in Table 10-5.

The potentially significant CDF contribution from loss of NSCW during POS 12 and POS 5A are related to completeness in representing the risk from this initiating event category. In this case, the POS conditions have only a limited impact on the results. The POS 12 and POS 5A contributions are driven by the duration of the POSs. While the loss of NSCW frequency is low, the impacts of the event are significant due to several safety functions that are impacted (e.g., cooling for ECCS pumps, emergency diesel generators, and CCW heat exchangers). If additional cooling capabilities were available, such as cross-tie to the service water from the other unit on site, then the impacts of the event would be reduced, in particular for post-refueling POSs where lower decay heat levels extend the time available for restoration actions. However, no such procedure-supported restoration actions were identified or credited for the modeled loss of NSCW scenario.

The potentially significant CDF contribution from loss of inventory during POS 4 and POS 3 is consistent with the model results because the frequency and impacts are expected to be similar to loss of inventory during POS 5A, which is one of the top contributing scenarios in the shutdown PRA. The primary factor in the significance of this scenario is the high initiating event frequency. As discussed in Section 8.1.2, the available operating experience data suggests that shutdown loss of inventory events are most frequently caused by pressure transients and lifting of relief valves during POSs with the RCS pressurized. Assuming the same initiating event frequency applies to POS 4 and POS 3, and possibly other POSs with RCS at pressure, results

in potentially significant CDF contributions. This demonstrates the importance of understanding the POS conditions that can contribute to increased initiating event frequencies. Risk insights may be missed if a generic initiating event frequency is applied across all POSs without accounting for POS-specific differences. In this project's initial shutdown accident scenario prioritization (described in Section 4.2.2), a generic loss of inventory initiating event frequency was applied, and the significance of loss of inventory during POS 5A was not recognized in that process. Following from this finding, additional research on shutdown initiating event frequencies is recommended. Shutdown PRA can benefit from improved initiating event data collection, characterization, and analysis with accounting for the key shutdown conditions that can contribute to increased initiating event frequencies.

Table 10-3 Estimated CDF Results by Plant Operating State

IE Categories: IE Frequency: (per shutdown- year) Shutdown POS	LORHR	LOINV	ODML / LOINV-DS	LO4160V	LOOP	LONSCW	Total POS CDF (per year)	% Contr. to CDF	Time to CD (hr.)
	3.7E-1	6.1E-1 ⁽¹⁾ 3.0E-3 ⁽²⁾ 6.4E-5 ⁽³⁾ 1.2E-5 ⁽⁴⁾	2.4E-3 ⁽⁵⁾ 2.1E-3 ⁽⁶⁾ 2.1E-2 ⁽⁷⁾	3.4E-1	1.6E-1	3.5E-5 ⁽⁸⁾			
POS 3 Mode 4 cooldown	< 1E-12	7.8E-08 (Note 9)	N/A	1.7E-09 (Note 9)	2.3E-08	7.9E-09 (Note 9)	1.1E-07	0.8%	(Note 10)
POS 4 Mode 5 cooldown	2.4E-12	1.0E-07 (Note 9)	N/A	2.3E-09 (Note 9)	1.7E-07	1.1E-08 (Note 9)	2.9E-07	2.2%	(Note 10)
POS 5A Pressurizer solid	3.9E-11	7.8E-07	N/A	1.7E-08 (Note 9)	1.3E-06	7.9E-08 (Note 9)	2.1E-06	16.2%	27
POS 5B Draining RCS	9.9E-08	2.2E-08	1.2E-08	2.1E-08	4.9E-07	1.1E-07	7.5E-07	5.6%	5.3
POS 6 Mid-loop (pre-refuel)	1.3E-07	7.7E-09	1.2E-06	1.4E-08	2.2E-07	3.7E-08	1.6E-06	12.3%	3.3
POS 7 Filling refuel cavity	2.7E-08	1.2E-11	N/A	1.7E-08	1.6E-07	3.9E-08 (Note 9)	2.4E-07	1.8%	52
POS 8E Refuel cavity flooded (pre-refuel)	1.4E-07	1.1E-10	N/A	1.8E-07	2.7E-06	1.9E-07	3.3E-06	24.6%	90
POS 8L Refuel cavity flooded (post-refuel)	1.0E-07	1.3E-11	N/A	1.0E-07	2.1E-06	1.5E-07	2.4E-06	18.3%	214
POS 9 Draining refuel cavity	5.6E-08	1.7E-08	8.6E-09	1.6E-08	2.8E-07	8.5E-08	4.7E-07	3.5%	15
POS 10 Mid-loop (post- refuel)	7.4E-08	1.4E-08	2.3E-07	2.5E-08	3.1E-07	7.1E-08	7.2E-07	5.4%	11
POS 11 Vacuum fill RCS	4.6E-08	1.3E-08 (Note 9)	N/A	1.4E-08 (Note 9)	2.3E-07	6.3E-08 (Note 9)	3.6E-07	2.7%	(Note 10)
POS 12 Mode 5 heatup	6.4E-12	3.0E-08 (Note 9)	N/A	3.1E-08 (Note 9)	4.2E-07	1.4E-07 (Note 9)	6.2E-07	4.7%	177
POS 13 Mode 4 heatup	1.3E-12	1.1E-08 (Note 9)	N/A	1.1E-08 (Note 9)	1.6E-07	5.3E-08 (Note 9)	2.3E-07	1.7%	(Note 10)
Total:	6.7E-07	1.1E-06	1.5E-06	4.5E-07	8.5E-06	1.0E-06	1.3E-05		

Table 10-3 Estimated CDF Results by Plant Operating State (cont.)

IE Categories:	LORHR	LOINV	ODML / LOINV-DS	LO4160V	LOOP	LONSCW	Total POS CDF (per year)	% Contr. to CDF	Time to CD (hr.)
IE Frequency: (per shutdown- year)	3.7E-1	6.1E-1 ⁽¹⁾ 3.0E-3 ⁽²⁾ 6.4E-5 ⁽³⁾ 1.2E-5 ⁽⁴⁾	2.4E-3 ⁽⁵⁾ 2.1E-3 ⁽⁶⁾ 2.1E-2 ⁽⁷⁾	3.4E-1	1.6E-1	3.5E-5 ⁽⁸⁾			
Shutdown POS									
Note (1) Loss of inventory over-pressure event frequency 6.1×10^{-1}/shutdown-year is applied in POS 5A. Note (2) Loss of inventory spray event frequency 3.0×10^{-3}/shutdown-year is applied in POS 5B, 6, 9, and 10. Note (3) Loss of inventory event frequency 6.4×10^{-5}/shutdown-year is applied in POS 5B, 6, 9, and 10. Note (4) Loss of inventory initiating event frequency 1.2×10^{-5}/shutdown-year is applied in POS 7, 8E, and 8L. Note (5) Frequency of mid-loop entry in POS 6 multiplied by conditional probability of overdrain. Note (6) Frequency of mid-loop entry in POS 10 multiplied by conditional probability of overdrain. Note (7) Loss of inventory due to displacement frequency 2.1×10^{-2}/shutdown-year is applied in POS 5B and 9. Note (8) Frequency calculated by fault tree is approximately 3.5×10^{-5}/shutdown-year, but varies by POS. Note (9) This accident scenario was not modeled, but an estimated CDF contribution is provided based on bounding assumptions about the POS conditions. Note (10) Calculation of time to core damage was not performed for this POS.									

Table 10-4 Estimated CDF Results Significant Scenarios

Initiating Event Category	POS	CDF (per year)	CDF % Contr.	Cumulative CDF %
LOOP	POS 8E	2.7E-06	20.7%	20.7%
LOOP	POS 8L	2.1E-06	15.6%	36.4%
LOOP	POS 5A	1.3E-06	9.6%	46.0%
ODML	POS 6	1.2E-06	9.3%	55.2%
LOINV	POS 5A	7.8E-07	5.9%	61.1%
LOOP	POS 5B	4.9E-07	3.7%	64.8%
LOOP	POS 12	4.2E-07	3.2%	67.9%
LOOP	POS 10	3.1E-07	2.3%	70.3%
LOOP	POS 9	2.8E-07	2.1%	72.4%
ODML	POS 10	2.3E-07	1.7%	74.1%
LOOP	POS 11	2.3E-07	1.7%	75.8%
LOOP	POS 6	2.2E-07	1.7%	77.5%
LONSCW	POS 8E	1.9E-07	1.5%	79.0%
LO4160V	POS 8E	1.8E-07	1.4%	80.3%
LOOP	POS 4	1.7E-07	1.3%	81.6%
LOOP	POS 7	1.6E-07	1.2%	82.8%
LOOP	POS 13	1.6E-07	1.2%	84.0%
LONSCW	POS 8L	1.5E-07	1.1%	85.1%
LONSCW	POS 12	1.4E-07 (Note 1)	1.1%	86.1%
LORHR	POS 8E	1.4E-07	1.1%	87.2%
LORHR	POS 6	1.3E-07	1.0%	88.2%
LONSCW	POS 5B	1.1E-07	0.8%	89.0%
LOINV	POS 4	1.0E-07 (Note 1)	0.8%	89.8%
LORHR	POS 8L	1.0E-07	0.8%	90.5%
LO4160V	POS 8L	1.0E-07	0.8%	91.3%
LORHR	POS 5B	9.9E-08	0.7%	92.0%
LONSCW	POS 9	8.5E-08	0.6%	92.7%
LONSCW	POS 5A	7.9E-08 (Note 1)	0.6%	93.3%
LOINV	POS 3	7.8E-08 (Note 1)	0.6%	93.9%
LORHR	POS 10	7.4E-08	0.6%	94.4%
LONSCW	POS 10	7.1E-08	0.5%	95.0%
Shutdown CDF significant scenarios		1.26E-05	95.0%	95.0%
Total estimated shutdown CDF		1.32E-05	100%	100%
Note (1) This accident scenario was not modeled, but an estimated CDF contribution is provided based on bounding assumptions about the POS conditions.				

Table 10-5 Total Estimated Shutdown CDF

Shutdown CDF Contribution	CDF (per year)	CDF % Contr.
Total shutdown CDF all modeled scenarios	1.3E-05	94.7%
Estimated shutdown CDF unmodeled scenarios	7.1E-07	5.3%
Total estimated shutdown CDF	1.3E-05	100%

10.3.2 Low Power and Hot Standby Plant Operating States Risk Contribution

The sensitivity study in Section 10.3.1 describes the potential CDF contributions from unmodeled shutdown accident scenarios and POSs. However, it does not include risks during low power and hot standby states. This sensitivity study develops estimated CDF contributions for the unmodeled low power and hot standby POSs, including POSs 1, 2, 14, 15A and 15B, during a refueling outage (POT-4).

The approach uses the calculated results from the L3PRA project at-power Level 1 PRA (NRC, 2022) to estimate the contributions from low power and hot standby POSs. While the conditions are expected to be similar to at-power operations some significant differences could be present. Notably, the changing plant conditions and system alignments, as discussed in Section 3.2.2, could contribute to potentially higher initiating event frequencies during low power and hot standby when compared to at-power.

The following key assumptions are applied to the sensitivity case to estimate risk contributions from the low power and hot standby POSs.

- The CDF contributions for low power and hot standby scenarios are estimated by multiplying the initiating event frequency for the accident type by the CCDP from the at-power Level 1 PRA (NRC, 2022) and scaling by the POS and POT duration factors as described in Section 9.3.
- The at-power initiating event frequencies are adjusted to account for plant conditions during low power POSs that can lead to higher event frequencies. Section 4.2.1 provides an assessment of low power initiating events. Though there is considerable uncertainty in the simplified evaluation of low power event rate, it does demonstrate that the low power initiating event rate is potentially higher than the corresponding full power initiating event rate. To account for this potential higher frequency, the full power initiating event frequencies are scaled by a factor of 10 for the following initiating event categories: reactor trip, other transients, main turbine trip, loss of main feedwater, and loss of condenser heat sink.
- The loss of offsite power initiating event frequency during plant shutdown operations is approximately six times higher than the frequency during full power operations. The shutdown initiating event frequency is applied to the hot standby POSs (POS 2 and POS 14). The CCDP values for POS 2 and POS 14 are based on the modeled loss of offsite power scenarios for POS 3 and POS 13, respectively.

- During the transition from power to shutdown operation, the safety injection actuation signals are inhibited, accumulators are isolated, and safety injection pumps are prevented from automatically starting. The safety injection functions are later restored during reactor startup. These transition activities could result in an inadvertent safety injection. The initiating event frequency for inadvertent safety injection is increased by a factor of 10 during the transition POSs, POS 2 and POS 14.
- All other at-power initiating events are included with the nominal initiating event frequencies that are used in the at-power PRA portion of the L3PRA project (NRC, 2022).
- Some at-power initiating events are no longer applicable after the reactor is shut down at the beginning of POS 2. These initiating events are: reactor trip, other transients, main turbine trip, loss of main feedwater, and loss of condenser heat sink. When reactor startup begins in POS 15A, reactor trip and other transients are again applicable. The other initiating events are applicable in POS 15B when the plant transitions from AFW to MFW and the main turbine begins to roll.
- Two additional initiating event categories are introduced for hot standby POSs.
 - Loss of auxiliary feedwater (1-LP-LOAFW) – The initiating event frequency is estimated from the failure-to-run rate for an AFW motor-driven pump (2.2×10^{-2} per year). The CCDP is based on assessing the L3PRA project at-power Level 1 PRA for a reactor trip event with one train of AFW unavailable (CCDP is 9.4×10^{-5}).
 - Loss of secondary side heat removal through atmospheric relief valves (1-LP-LOARV) – The initiating event frequency is based on the ARV failure to close after demand probability (2.95×10^{-3} on demand). The CCDP is based on the at-power Level 1 PRA CCDP for a secondary side break upstream of MSIVs.

There are limitations with this approach because there are aspects of the POS-specific modeling that are not addressed, for example, decay heat level, sequence timing, human failure events, and unique aspects of the plant configurations. While the decay heat and sequence timing are not significantly different for the POSs occurring prior to shutdown, the POSs occurring after shutdown are likely over-estimated using this approach. Despite the limitations, the numerical results give a sense of the potential contributions from unmodeled scenarios.

Table 10-6 shows the CDF contributions from the initiating event categories. The highest contributing initiating event category is loss of offsite power during reactor shutdown. The loss of offsite power initiating event frequency during shutdown is the primary reason for the significant contribution of this scenario. The estimated CCDP for loss of offsite power during reactor shutdown is lower than the CCDP for at-power operation, but the initiating event frequency is approximately six times higher than the frequency for at-power. The loss of offsite power frequency estimate for this project is described in Section 8.1.1. The loss of offsite power frequencies for at-power and shutdown are estimated on a recurring basis, and those results are published (NRC, 2021a) and used for other applications beyond this project. The results consistently show a higher loss of offsite power frequency during shutdown compared to at-power. The higher frequency during shutdown is acknowledged as an important source of modeling uncertainty contributing to the shutdown results. The loss of offsite power frequency during shutdown is discussed further in Section 10.3.5.

The second highest contributing initiating event scenario for the low power and hot standby POSs is other transients. The contribution is influenced by the assumption that the initiating event frequency is a factor of 10 higher during low power POSs. The contributions from reactor trip, main turbine trip, loss of main feedwater, and loss of condenser heat sink are similarly elevated. These contributions highlight the importance of developing a consistent framework for operating experience collection and initiating event frequency estimates for low power and hot standby operations. The assessment of low power initiating event operating experience in Section 4.2.1 suggests that the initiating event frequencies could be even greater than a factor of 10 higher than the full power estimates. If low power event occurrences are not assessed separately from full power event occurrences, then the frequency estimates could be significantly underestimating the rate of events that occur over relatively short exposure periods during reactor shutdown and startup.

Other significant contributors to the CDF for low power and hot standby POSs include the at-power loss of offsite power events and loss of nuclear service cooling water. These results are consistent with the L3PRA project at-power Level 1 PRA, where those scenarios are also dominant contributors to CDF.

The CDF contributions from individual low power and hot standby POSs are shown at the bottom of Table 10-6. The POSs at the end of the outage period (POSs 14, 15A, and 15B) account for nearly 90 percent of the low power and hot standby CDF. The large contribution is due to the much longer exposure time during the later POSs compared to the beginning of the outage.

The total estimated CDF contribution from the low power and hot standby POSs during a refueling outage is shown in Table 10-7. The total estimated CDF contribution from low power and hot standby POSs is about six percent of the total CDF for a refueling outage. Note that the CDF estimates are based on the average annual risk assuming an 18-month refueling outage cycle. If CDF is estimated on a year-to-year basis, then a calendar year that includes a refueling outage would have a factor of 1.5 higher contribution.

Table 10-6 Low Power and Hot Standby POS CDF (per Year) during Refueling Outage

Initiating Event Category	POS 1	POS 2	POS 14	POS 15A	POS 15B	Total	%
LOOP during reactor shutdown (Note 1)	N/A	3.9E-08	3.1E-07	N/A	N/A	3.5E-07	39.0%
Other Transients	1.1E-08 (Note 2)	N/A	N/A	3.6E-08 (Note 2)	6.4E-08 (Note 2)	1.1E-07	12.3%
Grid-Related LOOP (at-power)	7.8E-09	N/A	N/A	2.6E-08	4.7E-08	8.0E-08	8.9%
Loss of NSCW	3.7E-09	3.1E-09	2.5E-08	1.2E-08	2.2E-08	6.6E-08	7.4%
Switchyard-Centered LOOP (at-power)	4.4E-09	N/A	N/A	1.5E-08	2.6E-08	4.6E-08	5.1%
Reactor Trip	4.1E-09 (Note 2)	N/A	N/A	1.4E-08 (Note 2)	2.5E-08 (Note 2)	4.3E-08	4.8%
Weather-Related LOOP (at-power)	3.8E-09	N/A	N/A	1.3E-08	2.3E-08	4.0E-08	4.4%
Turbine Trip	4.5E-09 (Note 2)	N/A	N/A	N/A	2.7E-08 (Note 2)	3.2E-08	3.5%
Medium LOCA	9.9E-10	8.3E-10	6.6E-09	3.3E-09	6.0E-09	1.8E-08	2.0%
Loss of Main Feedwater	2.2E-09 (Note 2)	N/A	N/A	N/A	1.3E-08 (Note 2)	1.5E-08	1.7%
Loss of Condenser Heat Sink	2.0E-09 (Note 2)	N/A	N/A	N/A	1.2E-08 (Note 2)	1.4E-08	1.6%
Secondary-Side Break downstream MSIVs or upstream MFIVs	6.8E-10	5.6E-10	4.5E-09	2.3E-09	4.1E-09	1.2E-08	1.3%
Loss of 4.16kV AC Bus A	6.1E-10	5.1E-10	4.0E-09	2.0E-09	3.6E-09	1.1E-08	1.2%
Loss of Auxiliary Feedwater	N/A	7.9E-10	6.3E-09	3.1E-09	N/A	1.0E-08	1.1%
Plant-Centered LOOP (at-power)	8.1E-10	N/A	N/A	2.7E-09	4.9E-09	8.4E-09	0.9%
Loss of RCP Seal Injection	4.4E-10	3.7E-10	2.9E-09	1.5E-09	2.6E-09	7.9E-09	0.9%
Loss of 4.16kV AC Bus B	3.7E-10	3.1E-10	2.5E-09	1.2E-09	2.2E-09	6.6E-09	0.7%
Loss of 125V DC Bus BD1	3.7E-10	3.0E-10	2.4E-09	1.2E-09	2.2E-09	6.5E-09	0.7%
Inadvertent Safety Injection Actuation	6.2E-11	5.2E-10 (Note 2)	4.1E-09 (Note 2)	2.1E-10	3.7E-10	5.3E-09	0.6%
Loss of Atmospheric Relief Valves	N/A	3.3E-10	2.6E-09	1.3E-09	N/A	4.2E-09	0.5%
Loss of 125V DC Bus AD1	1.5E-10	1.3E-10	1.0E-09	5.1E-10	9.1E-10	2.7E-09	0.3%
Loss of Aux. Component Cooling Water	1.0E-10	8.7E-11	7.0E-10	3.5E-10	6.3E-10	1.9E-09	0.2%
Small LOCA	1.0E-10	8.5E-11	6.8E-10	3.4E-10	6.1E-10	1.8E-09	0.2%
Interfacing-system LOCA in RHR hot leg suction	9.6E-11	8.0E-11	6.4E-10	3.2E-10	5.7E-10	1.7E-09	0.2%

Table 10-6 Low Power and Hot Standby POS CDF (per Year) during Refueling Outage (cont.)

Initiating Event Category	POS 1	POS 2	POS 14	POS 15A	POS 15B	Total	%
Steam generator tube rupture	5.3E-11	4.4E-11	3.5E-10	1.8E-10	3.2E-10	9.4E-10	0.1%
Secondary-Side Break upstream MSIVs or downstream MFIVs	4.5E-11	3.8E-11	3.0E-10	1.5E-10	2.7E-10	8.0E-10	0.1%
Excessive LOCA (Reactor Vessel Rupture)	4.2E-11	3.5E-11	2.8E-10	1.4E-10	2.5E-10	7.6E-10	0.1%
Loss of 120V AC Panels A and B	4.1E-11	3.4E-11	2.7E-10	1.4E-10	2.4E-10	7.3E-10	0.1%
ISLOCA from RHR Cold Leg Injection Line A	1.8E-11	1.5E-11	1.2E-10	5.9E-11	1.1E-10	3.2E-10	0.0%
ISLOCA from RHR Cold Leg Injection Line B	1.8E-11	1.5E-11	1.2E-10	5.9E-11	1.1E-10	3.2E-10	0.0%
Large LOCA	1.5E-11	1.3E-11	1.0E-10	5.0E-11	9.1E-11	2.7E-10	0.0%
ISLOCA from RCP Stage 1 Seal Leak Off	1.4E-11	1.2E-11	9.6E-11	4.8E-11	8.6E-11	2.6E-10	0.0%
Loss of Instrument Air	1.0E-11	8.7E-12	7.0E-11	3.5E-11	6.3E-11	1.9E-10	0.0%
Loss of 120V AC Panels A and C	3.2E-14	2.7E-14	2.1E-13	1.1E-13	1.9E-13	5.7E-13	0.0%
Loss of 120V AC Panels A and D	1.8E-14	1.5E-14	1.2E-13	6.1E-14	1.1E-13	3.3E-13	0.0%
Loss of 120V AC Panels B and C	1.8E-14	1.5E-14	1.2E-13	6.1E-14	1.1E-13	3.3E-13	0.0%
Loss of 120V AC Panels B and D	1.8E-14	1.5E-14	1.2E-13	6.1E-14	1.1E-13	3.3E-13	0.0%
Loss of 120V AC Panels C and D	1.8E-14	1.5E-14	1.2E-13	6.1E-14	1.1E-13	3.3E-13	0.0%
ISLOCA from RCP Thermal Barrier Heat Exchanger	1.5E-14	1.2E-14	1.0E-13	5.0E-14	9.0E-14	2.7E-13	0.0%
Total	4.8E-08	4.7E-08	3.8E-07	1.4E-07	2.9E-07	9.0E-07	100%
POS % Contribution	5.4%	5.2%	42.0%	15.2%	32.3%		
Note (1) The initiating event frequency for loss of offsite power during shutdown is applied to hot standby POSs, POS 2 and POS 14. Note (2) The initiating event frequency is increased by a factor of 10 to account for the uncertainty in initiating event frequency during low power and hot standby accident scenarios.							

Table 10-7 Total Estimated Refueling Outage CDF

Low Power / Shutdown CDF Contribution	CDF (per year)	CDF % Contr.
Total shutdown CDF all modeled scenarios	1.3E-05	88.6%
Estimated shutdown CDF unmodeled scenarios	7.1E-07	5.0%
Estimated low power and hot standby CDF unmodeled scenarios	9.0E-07	6.4%
Total estimated refueling outage CDF	1.4E-05	100%

10.3.3 Other Outage Types Risk Contribution

The PRA modeling for this project is focused on estimating the risk of low power and shutdown operations during a planned refueling outage. Due to the relatively long duration and unique plant configurations, refueling outages are expected to include the most significant contributors to low power and shutdown risk. However, other types of outages can pose challenges for low power and shutdown operations.

This section describes a series of sensitivity calculations to estimate the CDF contributions from unmodeled outage types. The approach is similar to the sensitivity studies presented in Section 10.3.1 and 10.3.2, above. Assumptions are made about initiating event frequencies and conditional core damage probabilities by using results from accident scenarios with similar conditions or conditions that would bound the conditions of the unmodeled scenarios. As with the previous sensitivity studies, there are limitations with this approach because there are aspects of the POT and POS-specific modeling that are not addressed, for example, detailed analyses of decay heat levels, sequence timing, human failure events, and unique aspects of the plant configurations. Despite the limitations, the numerical results give a sense of the potential contributions from unmodeled accident scenarios during other outage types.

The key assumptions and results for the following outage types are discussed:

- Non-drained maintenance outage without the use of RHR cooling (POT-1)
- Non-drained maintenance outage with the use of RHR cooling (POT-2)
- Drained maintenance outage with the use of RHR cooling (POT-3).

Non-drained maintenance outage without the use of RHR cooling (POT-1):

The POT-1 maintenance outage considers an emergent reactor shutdown, entrance into Technical Specification Mode 3 Hot Standby operations, and restoration to full power operation. The following key assumptions are applied to the sensitivity case to estimate risk contributions from a POT-1 outage:

- The outage is assumed to initiate on an unplanned reactor trip. POS 1 (low power and reactor shutdown) is not entered, and the outage begins in POS 2 (reactor shutdown, cooldown with steam generators).

- The outage is entered after reactor shutdown and short duration maintenance activities that do not require RCS cooldown are completed before restoring full power operations. The modeled outage assumes that there is no risk-significant equipment out of service.
- The frequency of the outage is assumed to occur once per year, and the duration is assumed to be 72 hours. See Section 2.3.1 for a review of plant-specific operating experience of POT frequencies and durations. The POT frequency and duration modeling assumptions are summarized in Table 9-2.
- The assumptions about initiating event categories and frequencies during low power, hot standby, and shutdown POSs have a significant influence on the estimated CDF contributions. Similar assumptions as discussed in Section 10.3.2 are applied to POT-1 and are summarized here:
 - The full power initiating event frequencies are scaled by a factor of 10 for the following initiating event categories: reactor trip, other transients, main turbine trip, loss of main feedwater, and loss of condenser heat sink.
 - The shutdown loss of offsite power initiating event frequency is applied to hot standby POS 2.
 - Some at-power initiating events are no longer applicable after the reactor is shut down and are applicable again when reactor startup begins.
 - Initiating events for loss of auxiliary feedwater and loss of secondary side heat removal through atmospheric relief valves are included for applicable POSs.

Table 10-8 shows the CDF contributions from the initiating event categories for POT-1. The list of significant initiating events is similar to the low power and hot standby POSs from Section 10.3.2 for a refueling outage. The highest contributing initiating event category is loss of offsite power during reactor shutdown. The second highest contributing initiating event scenario is other transients. Both results are highly influenced by assumptions about the initiating event frequencies. Other initiating events with significant contributions include reactor trip, main turbine trip, loss of main feedwater, and loss of condenser heat sink, which all are assumed to have elevated initiating event frequencies.

Other significant contributors to the CDF for POT-1 POSs include the at-power loss of offsite power events and loss of nuclear service cooling water. These results are consistent with the L3PRA project at-power Level 1 PRA, where those scenarios are also dominant contributors to CDF.

Non-drained maintenance outage with the use of RHR cooling (POT-2):

The POT-2 maintenance outage considers a controlled shutdown to Technical Specification Mode 5 cold shutdown with decay heat removal provided by the RHR system. The following key assumptions are applied to the sensitivity case to estimate risk contributions from a POT-2 outage:

- The outage is entered for maintenance activities that require entrance to Mode 5 cold shutdown; examples include investigations of adverse equipment trends (e.g., main

turbine or pump vibrations, main condenser leaks, steam leaks, reactor coolant leaks, cooling water leaks), elective repairs or plant improvements, and shutdowns that are required by the Technical Specifications. However, the modeled outage assumes that there is no risk-significant equipment out of service that would impact the plant response to an initiating event.

- The frequency of the outage is assumed to be 0.1 outage per year, and the duration is assumed to be 10 days. The frequency and duration of such an outage is highly dependent on the plant-specific conditions that motivate the outage. The frequency and duration can vary significantly from plant-to-plant. See Section 2.3.1 for a review of plant-specific operating experience of POT frequencies and durations. The POT frequency and duration modeling assumptions are summarized in Table 9-2.
- As this outage type includes entry into shutdown POSs (POSs 3–13), all applicable shutdown initiating event categories are included in the sensitivity study.
- The assumptions about initiating event categories and frequencies during low power, hot standby, and shutdown POSs have a significant influence on the estimated CDF contributions. Similar assumptions as discussed in Section 10.3.2 are applied to POT-2 and are summarized here:
 - The full power initiating event frequencies are scaled by a factor of 10 for the following initiating event categories: reactor trip, other transients, main turbine trip, loss of main feedwater, and loss of condenser heat sink.
 - The shutdown loss of offsite power initiating event frequency is applied to hot standby POS 2 and POS 14.
 - Some at-power initiating events are no longer applicable after the reactor is shut down and are applicable again when reactor startup begins.
 - Initiating events for loss of auxiliary feedwater and loss of secondary side heat removal through atmospheric relief valves are included for applicable POSs.

Table 10-9 shows the CDF contributions from the initiating event categories for POT-2. It is noteworthy that the shutdown initiating events and POSs dominate the CDF contributions. Approximately 78 percent of the CDF is associated with the shutdown POSs (POSs 3–13). The highest contributing initiating event category is loss of inventory during shutdown. The CDF contribution from loss of inventory during POS 4 is consistent with previously discussed model results where losses of inventory with RCS intact (i.e., not vented) were significant contributors. The primary factor in the significance of this scenario is the high initiating event frequency. The long duration of POS 4 (assumed to be 68 hours) also contributes to the result. The outage timing assumes that the plant is held in this state (i.e., cold shutdown with RHR cooling) while the outage maintenance activities are underway.

The second highest contributing initiating event scenario is loss of offsite power during reactor shutdown, which is also a significant contributor to POT-1 and the shutdown POSs for POT-4. Other significant contributors include other transients and reactor trip, which are assumed to have elevated initiating event frequencies during the transition from full power to shutdown. The

at-power loss of offsite power events and loss of nuclear service cooling water are also significant contributors.

Drained maintenance outage with the use of RHR cooling (POT-3):

The POT-3 maintenance outage considers a controlled shutdown to Technical Specification Mode 5 cold shutdown with decay heat removal provided by the RHR system and draining the RCS to a reduced inventory level. This plant configuration is entered if maintenance requires low RCS level without unloading the core; examples include addressing RCS leakage, reactor coolant pump seal maintenance, and steam generator tube maintenance. The following key assumptions are applied to the sensitivity case to estimate risk contributions from a POT-3 outage:

- The outage is entered for maintenance activities that require lowering the RCS inventory level. The RCS may be reduced to near the reactor vessel flange or further reduced to the mid-loop level depending on the type of work needed to be performed. It is assumed that mid-loop operation will be avoided unless it is unavoidable. The estimated CDF for POT-3 assumes that mid-loop is not entered. As seen in the refueling outage (POT-4) CDF results, mid-loop operations (POS 6) have a significant CDF contribution (see Table 9-4). An alternative version of the POT-3 outage with mid-loop entry is provided to demonstrate the potential significant CDF contribution associated with mid-loop entry.
- As with the other POT assessments, the event frequencies and CCDF values for similar at-power and refueling outage accident scenarios are used to estimate CDF values. However, an additional assumption is made for the POSs with reduced inventory (POS 5B and POS 6) in POT-3. Due to the shorter duration of the outage and the cooldown POSs, the reduced inventory states may be reached at a shorter time after reactor shutdown. This would result in higher decay heat levels for those POSs. Also, the analyzed decay heat assumes a controlled coast down at the end of cycle for a refueling outage. An outage occurring earlier in the cycle period and without a controlled coast down would have somewhat higher decay heat during the first few days of the outage. To account for the potentially higher decay heat level, the human error probabilities are increased by a factor of 10. This is applied to the human failure events in POS 5B and POS 6 that involve the initial, time-sensitive operator response to diagnose the event and restore RHR cooling. The increased human error probabilities are applied to shutdown accident scenarios, loss of offsite power, loss of RHR, loss of 4.16kV AC bus, and loss of NSCW. The loss of inventory human failure events are not adjusted because the timing is driven by the inventory loss, rather than the decay heat level.
- The frequency of the outage is assumed to be 0.1 outage per year, and the duration is assumed to be 10 days. The frequency and duration of such an outage is highly dependent on the plant-specific conditions that motivate the outage. The frequency and duration can vary significantly from plant-to-plant. See Section 2.3.1 for a review of plant-specific operating experience of POT frequencies and durations. The POT frequency and duration modeling assumptions are summarized in Table 9-2.
- As this outage type includes entry into shutdown POSs (POSs 3–13), all applicable shutdown initiating event categories are included in the sensitivity study.

- The assumptions about initiating event categories and frequencies during low power, hot standby, and shutdown POSs have a significant influence on the estimated CDF contributions. Similar assumptions as discussed in Section 10.3.2 are applied to POT-3 and are summarized here:
 - The full power initiating event frequencies are scaled by a factor of 10 for the following initiating event categories: reactor trip, other transients, main turbine trip, loss of main feedwater, and loss of condenser heat sink.
 - The shutdown loss of offsite power initiating event frequency is applied to hot standby POS 2 and POS 14.
 - Some at-power initiating events are no longer applicable after the reactor is shut down and are applicable again when reactor startup begins.
 - Initiating events for loss of auxiliary feedwater and loss of secondary side heat removal through atmospheric relief valves are included for applicable POSs.

Table 10-10 and Table 10-11 show the CDF contributions from the initiating event categories for POT-3. Table 10-10 shows the estimated CDF contributions from shutdown POSs, and Table 10-11 shows the estimated CDF contributions from low power and hot standby POSs. The shutdown initiating event scenarios and POSs are the dominant contributors to the CDF estimates. The shutdown POSs (POSs 3–13) contribute approximately 80 percent of the estimated CDF for the outage. The highest contributing plant operating state is POS 5B, cold shutdown with reduced RCS inventory. The highest contributing initiating event category is loss of offsite power during shutdown. The other shutdown initiating event categories also contribute significantly to the estimated CDF. At-power accident scenarios for other transients, loss of offsite power events, and loss of nuclear service cooling water are also significant contributors.

The CDF contributions from shutdown POSs are even more significant if a mid-loop entry is included in a POT-3 outage. Table 10-12 and Table 10-13 show the CDF contributions for POT-3 with mid-loop entry. Table 10-12 shows the estimated CDF contributions from shutdown POSs, and Table 10-13 shows the estimated CDF contributions from low power and hot standby POSs. The same assumptions are applied as in the case shown in Table 10-10. The only differences are that the duration of POS 5B (40 hours) is split between POS 5B and POS 6 (20 hours in each state) and the overdrain entering mid-loop scenario is included in POS 6. One entry into the mid-loop state is assumed with the frequency of the outage assumed to be 0.1 outage per year.

The inclusion of mid-loop entry and POS 6 greatly increases the POT-3 CDF estimate from 6×10^{-7} per year to 1.1×10^{-6} per year. The POS 6 CDF contributes more than 50 percent of the outage risk. The highest contributing initiating event category is the loss of offsite power during shutdown. The other shutdown initiating event categories also contribute significantly to the estimated CDF.

Table 10-8 CDF (per Year) during Non-Drained Maintenance Outage without the Use of RHR Cooling (POT-1)

Initiating Event	POS 2	POS 15A	POS 15B	Total	%
LOOP during reactor shutdown (Note 1)	1.9E-07	N/A	N/A	1.9E-07	21.7%
Other Transients	N/A	5.4E-08 (Note 2)	9.7E-08 (Note 2)	1.5E-07	17.4%
Grid-Relate LOOP (at-power)	N/A	3.9E-08	7.0E-08	1.1E-07	12.6%
Loss of NSCW	1.5E-08	1.9E-08	3.3E-08	6.7E-08	7.7%
Switchyard-Centered LOOP (at-power)	N/A	2.2E-08	4.0E-08	6.2E-08	7.1%
Reactor Trip	N/A	2.1E-08 (Note 2)	3.7E-08 (Note 2)	5.8E-08	6.7%
Weather-Relate LOOP (at-power)	N/A	1.9E-08	3.4E-08	5.4E-08	6.2%
Turbine Trip	N/A	N/A	4.1E-08 (Note 2)	4.1E-08	4.7%
Loss of Main Feedwater	N/A	N/A	2.0E-08 (Note 2)	2.0E-08	2.3%
Loss of Condenser Heat Sink	N/A	N/A	1.8E-08 (Note 2)	1.8E-08	2.1%
Medium LOCA	4.0E-09	5.0E-09	8.9E-09	1.8E-08	2.1%
Secondary-Side Break downstream MSIVs or upstream MFIVs	2.7E-09	3.4E-09	6.1E-09	1.2E-08	1.4%
Plant-Centered LOOP (at-power)	N/A	4.1E-09	7.3E-09	1.1E-08	1.3%
Loss of 4.16kV AC Bus A	2.4E-09	3.0E-09	5.5E-09	1.1E-08	1.3%
Loss of Auxiliary Feedwater	3.8E-09	4.7E-09	N/A	8.5E-09	1.0%
Loss of RCP Seal Injection	1.8E-09	2.2E-09	4.0E-09	7.9E-09	0.9%
Loss of 4.16kV AC Bus B	1.5E-09	1.9E-09	3.4E-09	6.7E-09	0.8%
Loss of 125V DC Bus BD1	1.5E-09	1.8E-09	3.3E-09	6.6E-09	0.8%
Loss of Atmospheric Relief Valves	1.6E-09	2.0E-09	N/A	3.5E-09	0.4%
Loss of 125V DC Bus AD1	6.1E-10	7.6E-10	1.4E-09	2.7E-09	0.3%
Loss of Aux. Component Cooling Water	4.2E-10	5.2E-10	9.4E-10	1.9E-09	0.2%
Small LOCA	4.1E-10	5.1E-10	9.1E-10	1.8E-09	0.2%
Interfacing-system LOCA in RHR hot leg suction	3.8E-10	4.8E-10	8.6E-10	1.7E-09	0.2%
Inadvertent Safety Injection Actuation	2.5E-10	3.1E-10	5.6E-10	1.1E-09	0.1%

Table 10-8 CDF (per Year) during Non-Drained Maintenance Outage without the Use of RHR Cooling (POT-1) (cont.)

Initiating Event	POS 2	POS 15A	POS 15B	Total	%
Steam generator tube rupture	2.1E-10	2.6E-10	4.7E-10	9.5E-10	0.1%
Secondary-Side Break upstream MSIVs or downstream MFIVs	1.8E-10	2.3E-10	4.1E-10	8.1E-10	0.1%
Excessive LOCA (Reactor Vessel Rupture)	1.7E-10	2.1E-10	3.8E-10	7.6E-10	0.1%
Loss of 120V AC Panels A and B	1.6E-10	2.0E-10	3.7E-10	7.3E-10	0.1%
ISLOCA from RHR Cold Leg Injection Line A	7.1E-11	8.9E-11	1.6E-10	3.2E-10	0.0%
ISLOCA from RHR Cold Leg Injection Line B	7.1E-11	8.9E-11	1.6E-10	3.2E-10	0.0%
Large LOCA	6.0E-11	7.6E-11	1.4E-10	2.7E-10	0.0%
ISLOCA from RCP Stage 1 Seal Leak Off	5.7E-11	7.2E-11	1.3E-10	2.6E-10	0.0%
Loss of Instrument Air	4.2E-11	5.2E-11	9.4E-11	1.9E-10	0.0%
Loss of 120V AC Panels A and C	1.3E-13	1.6E-13	2.9E-13	5.7E-13	0.0%
Loss of 120V AC Panels A and D	7.3E-14	9.1E-14	1.6E-13	3.3E-13	0.0%
Loss of 120V AC Panels B and C	7.3E-14	9.1E-14	1.6E-13	3.3E-13	0.0%
Loss of 120V AC Panels B and D	7.3E-14	9.1E-14	1.6E-13	3.3E-13	0.0%
Loss of 120V AC Panels C and D	7.3E-14	9.1E-14	1.6E-13	3.3E-13	0.0%
ISLOCA from RCP Thermal Barrier Heat Exchanger	6.0E-14	7.5E-14	1.3E-13	2.7E-13	0.0%
Total	2.2E-07	2.1E-07	4.4E-07	8.7E-07	100%
POS % Contribution	26.0%	23.7%	50.4%		
Note (1) The initiating event frequency for loss of offsite power during shutdown is applied to hot standby POSs, POS 2 and POS 14. Note (2) The initiating event frequency is increased by a factor of 10 to account for the uncertainty in initiating event frequency during low power and hot standby accident scenario.					

Table 10-9 CDF (per Year) during Non-Drained Maintenance Outage without the Use of RHR Cooling (POT-2)

Initiating Event	POS 1	POS 2	POS 3	POS 4	POS 12	POS 13	POS 14	POS 15A	POS 15B	Total	%
Loss of inventory during reactor shutdown	N/A	N/A	1.2E-08	2.6E-07	4.0E-09	1.7E-09	N/A	N/A	N/A	2.8E-07	46.5%
LOOP during reactor shutdown (Note 1)	N/A	5.9E-09	3.5E-09	8.0E-08	5.6E-08	2.3E-08	4.7E-08	N/A	N/A	2.2E-07	35.7%
Other Transients	1.6E-09 (Note 2)	N/A	N/A	N/A	N/A	N/A	N/A	5.4E-09 (Note 2)	9.7E-09 (Note 2)	1.7E-08	2.8%
Loss of NSCW during reactor shutdown	N/A	N/A	2.9E-10	6.5E-09	4.6E-09	1.9E-09	N/A	N/A	N/A	1.3E-08	2.2%
Grid-Relate LOOP (at-power)	1.2E-09	N/A	N/A	N/A	N/A	N/A	N/A	3.9E-09	7.0E-09	1.2E-08	2.0%
Loss of 4.16kV AC Bus during reactor shutdown	N/A	N/A	2.6E-10	5.8E-09	4.1E-09	1.7E-09	N/A	N/A	N/A	1.2E-08	2.0%
Loss of NSCW (at-power)	5.6E-10	4.7E-10	N/A	N/A	N/A	N/A	3.7E-09	1.9E-09	3.3E-09	1.0E-08	1.6%
Switchyard-Centered LOOP (at-power)	6.6E-10	N/A	N/A	N/A	N/A	N/A	N/A	2.2E-09	4.0E-09	6.8E-09	1.1%
Reactor Trip	6.2E-10 (Note 2)	N/A	N/A	N/A	N/A	N/A	N/A	2.1E-09 (Note 2)	3.7E-09 (Note 2)	6.4E-09	1.1%
Weather-Relate LOOP (at-power)	5.7E-10	N/A	N/A	N/A	N/A	N/A	N/A	1.9E-09	3.4E-09	5.9E-09	1.0%
Turbine Trip	6.8E-10 (Note 2)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	4.1E-09 (Note 2)	4.8E-09	0.8%
Medium LOCA	1.5E-10	1.2E-10	N/A	N/A	N/A	N/A	9.9E-10	5.0E-10	8.9E-10	2.7E-09	0.4%
Loss of Main Feedwater	3.3E-10 (Note 2)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	2.0E-09 (Note 2)	2.3E-09	0.4%
Loss of Condenser Heat Sink	3.0E-10 (Note 2)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	1.8E-09 (Note 2)	2.1E-09	0.4%
Secondary-Side Break downstream MSIVs or upstream MFIVs	1.0E-10	8.4E-11	N/A	N/A	N/A	N/A	6.8E-10	3.4E-10	6.1E-10	1.8E-09	0.3%
Loss of 4.16kV AC Bus A	9.1E-11	7.6E-11	N/A	N/A	N/A	N/A	6.1E-10	3.0E-10	5.5E-10	1.6E-09	0.3%

Table 10-9 CDF (per Year) during Non-Drained Maintenance Outage without the Use of RHR Cooling (POT-2) (cont.)

Initiating Event	POS 1	POS 2	POS 3	POS 4	POS 12	POS 13	POS 14	POS 15A	POS 15B	Total	%
Loss of Auxiliary Feedwater	N/A	1.2E-10	N/A	N/A	N/A	N/A	9.4E-10	4.7E-10	N/A	1.5E-09	0.3%
Plant-Centered LOOP (at-power)	1.2E-10	N/A	N/A	N/A	N/A	N/A	N/A	4.1E-10	7.3E-10	1.3E-09	0.2%
Loss of RCP Seal Injection	6.6E-11	5.5E-11	N/A	N/A	N/A	N/A	4.4E-10	2.2E-10	4.0E-10	1.2E-09	0.2%
Loss of 4.16kV AC Bus B	5.6E-11	4.7E-11	N/A	N/A	N/A	N/A	3.7E-10	1.9E-10	3.4E-10	1.0E-09	0.2%
Loss of 125V DC Bus BD1	5.5E-11	4.6E-11	N/A	N/A	N/A	N/A	3.7E-10	1.8E-10	3.3E-10	9.8E-10	0.2%
Loss of RHR cooling during reactor shutdown	N/A	N/A	1.4E-11	3.1E-10	2.2E-10	9.2E-11	N/A	N/A	N/A	6.4E-10	0.1%
Loss of Atmospheric Relief Valves	N/A	4.9E-11	N/A	N/A	N/A	N/A	3.9E-10	2.0E-10	N/A	6.3E-10	0.1%
Loss of 125V DC Bus AD1	2.3E-11	1.9E-11	N/A	N/A	N/A	N/A	1.5E-10	7.6E-11	1.4E-10	4.1E-10	0.1%
Loss of Aux. Component Cooling Water	1.6E-11	1.3E-11	N/A	N/A	N/A	N/A	1.0E-10	5.2E-11	9.4E-11	2.8E-10	0.0%
Small LOCA	1.5E-11	1.3E-11	N/A	N/A	N/A	N/A	1.0E-10	5.1E-11	9.1E-11	2.7E-10	0.0%
Interfacing-system LOCA in RHR hot leg suction	1.4E-11	1.2E-11	N/A	N/A	N/A	N/A	9.6E-11	4.8E-11	8.6E-11	2.6E-10	0.0%
Inadvertent Safety Injection Actuation	9.3E-12	7.8E-12	N/A	N/A	N/A	N/A	6.2E-11	3.1E-11	5.6E-11	1.7E-10	0.0%
Steam generator tube rupture	7.9E-12	6.6E-12	N/A	N/A	N/A	N/A	5.3E-11	2.6E-11	4.7E-11	1.4E-10	0.0%
Secondary-Side Break upstream MSIVs or downstream MFIVs	6.8E-12	5.6E-12	N/A	N/A	N/A	N/A	4.5E-11	2.3E-11	4.1E-11	1.2E-10	0.0%
Excessive LOCA (Reactor Vessel Rupture)	6.4E-12	5.3E-12	N/A	N/A	N/A	N/A	4.2E-11	2.1E-11	3.8E-11	1.1E-10	0.0%

Table 10-9 CDF (per Year) during Non-Drained Maintenance Outage without the Use of RHR Cooling (POT-2) (cont.)

Initiating Event	POS 1	POS 2	POS 3	POS 4	POS 12	POS 13	POS 14	POS 15A	POS 15B	Total	%
Loss of 120V AC Panels A and B	6.1E-12	5.1E-12	N/A	N/A	N/A	N/A	4.1E-11	2.0E-11	3.7E-11	1.1E-10	0.0%
ISLOCA from RHR Cold Leg Injection Line A	2.7E-12	2.2E-12	N/A	N/A	N/A	N/A	1.8E-11	8.9E-12	1.6E-11	4.8E-11	0.0%
ISLOCA from RHR Cold Leg Injection Line B	2.7E-12	2.2E-12	N/A	N/A	N/A	N/A	1.8E-11	8.9E-12	1.6E-11	4.8E-11	0.0%
Large LOCA	2.3E-12	1.9E-12	N/A	N/A	N/A	N/A	1.5E-11	7.6E-12	1.4E-11	4.0E-11	0.0%
ISLOCA from RCP Stage 1 Seal Leak Off	2.2E-12	1.8E-12	N/A	N/A	N/A	N/A	1.4E-11	7.2E-12	1.3E-11	3.8E-11	0.0%
Loss of Instrument Air	1.6E-12	1.3E-12	N/A	N/A	N/A	N/A	1.0E-11	5.2E-12	9.4E-12	2.8E-11	0.0%
Loss of 120V AC Panels A and C	4.8E-15	4.0E-15	N/A	N/A	N/A	N/A	3.2E-14	1.6E-14	2.9E-14	8.5E-14	0.0%
Loss of 120V AC Panels A and D	2.7E-15	2.3E-15	N/A	N/A	N/A	N/A	1.8E-14	9.1E-15	1.6E-14	4.9E-14	0.0%
Loss of 120V AC Panels B and C	2.7E-15	2.3E-15	N/A	N/A	N/A	N/A	1.8E-14	9.1E-15	1.6E-14	4.9E-14	0.0%
Loss of 120V AC Panels B and D	2.7E-15	2.3E-15	N/A	N/A	N/A	N/A	1.8E-14	9.1E-15	1.6E-14	4.9E-14	0.0%
Loss of 120V AC Panels C and D	2.7E-15	2.3E-15	N/A	N/A	N/A	N/A	1.8E-14	9.1E-15	1.6E-14	4.9E-14	0.0%
ISLOCA from RCP Thermal Barrier Heat Exchanger	2.2E-15	1.9E-15	N/A	N/A	N/A	N/A	1.5E-14	7.5E-15	1.3E-14	4.0E-14	0.0%
Total	7.3E-09	7.2E-09	1.6E-08	3.6E-07	6.9E-08	2.9E-08	5.6E-08	2.1E-08	4.4E-08	6.0E-07	100%
POS % Contribution	1.2%	1.2%	2.6%	58.9%	11.4%	4.8%	9.3%	3.4%	7.2%		
Note (1) The initiating event frequency for loss of offsite power during shutdown is applied to hot standby POSs, POS 2 and POS 14. Note (2) The initiating event frequency is increased by a factor of 10 to account for the uncertainty in initiating event frequency during low power and hot standby accident scenario.											

Table 10-10 CDF (per Year) for Shutdown POSs during Drained Maintenance Outage (POT-3)

Initiating Event	POS 3	POS 4	POS 5A	POS 5B	POS 11	POS 12	POS 13	Total	%
LOOP during reactor shutdown	3.5E-09	4.7E-09	2.3E-08	9.7E-08 (Note 1)	3.6E-08	5.6E-08	2.3E-08	2.4E-07	40.6%
Loss of inventory during reactor shutdown	1.2E-08	1.6E-08	7.8E-08	3.3E-09	1.7E-09	4.0E-09	1.7E-09	1.2E-07	19.1%
Loss of RHR cooling during reactor shutdown	1.4E-11	1.8E-11	9.2E-11	5.5E-08 (Note 1)	7.4E-09	2.2E-10	9.2E-11	6.3E-08	10.5%
Loss of NSCW during reactor shutdown	2.8E-10	3.7E-10	1.9E-09	3.2E-08 (Note 1)	8.0E-09	4.5E-09	1.9E-09	4.9E-08	8.1%
Loss of 4.16kV AC Bus during reactor shutdown	2.6E-10	3.4E-10	1.7E-09	6.1E-09 (Note 1)	1.7E-09	4.1E-09	1.7E-09	1.6E-08	2.6%
Total Shutdown POSs	1.6E-08	2.1E-08	1.0E-07	1.9E-07	5.5E-08	6.9E-08	2.9E-08	4.9E-07	80.9%
POS % Contribution	2.6%	3.5%	17.4%	32.1%	9.2%	11.5%	4.8%		80.9%
Total All POSs								6.0E-07	100%
Note (1) Human error probabilities are increased by a factor of ten to account for potentially higher decay heat levels in reduced inventory POSs during POT-3 compared to the modeled refueling outage POT-4.									

Table 10-11 CDF (per Year) for Low Power and Hot Standby POSs during Drained Maintenance Outage (POT-3)

Initiating Event	POS 1	POS 2	POS 14	POS 15A	POS 15B	Total	%
LOOP during reactor shutdown (Note 1)	N/A	5.9E-09	4.7E-08	N/A	N/A	5.3E-08	8.7%
Other Transients	1.6E-09 (Note 2)	N/A	N/A	5.4E-09 (Note 2)	5.4E-09 (Note 2)	1.2E-08	2.0%
Grid-Relate LOOP (at-power)	1.2E-09	N/A	N/A	3.9E-09	3.9E-09	8.9E-09	1.5%
Loss of NSCW (at-power)	5.6E-10	4.7E-10	3.7E-09	1.9E-09	1.9E-09	8.5E-09	1.4%
Switchyard-Centered LOOP (at-power)	6.6E-10	N/A	N/A	2.2E-09	2.2E-09	5.1E-09	0.8%
Reactor Trip	6.2E-10 (Note 2)	N/A	N/A	2.1E-09 (Note 2)	2.1E-09 (Note 2)	4.8E-09	0.8%
Weather-Relate LOOP (at-power)	5.7E-10	N/A	N/A	1.9E-09	1.9E-09	4.4E-09	0.7%
Turbine Trip	6.8E-10 (Note 2)	N/A	N/A	N/A	2.3E-09 (Note 2)	3.0E-09	0.5%
Medium LOCA	1.5E-10	1.2E-10	9.9E-10	5.0E-10	5.0E-10	2.3E-09	0.4%
Secondary-Side Break downstream MSIVs or upstream MFIVs	1.0E-10	8.4E-11	6.8E-10	3.4E-10	3.4E-10	1.5E-09	0.3%
Loss of Auxiliary Feedwater	N/A	1.2E-10	9.4E-10	4.7E-10	N/A	1.5E-09	0.3%
Loss of Main Feedwater	3.3E-10 (Note 2)	N/A	N/A	N/A	1.1E-09 (Note 2)	1.4E-09	0.2%
Loss of 4.16kV AC Bus A	9.1E-11	7.6E-11	6.1E-10	3.0E-10	3.0E-10	1.4E-09	0.2%
Loss of Condenser Heat Sink	3.0E-10 (Note 2)	N/A	N/A	N/A	1.0E-09 (Note 2)	1.3E-09	0.2%
Loss of RCP Seal Injection	6.6E-11	5.5E-11	4.4E-10	2.2E-10	2.2E-10	1.0E-09	0.2%
Plant-Centered LOOP (at-power)	1.2E-10	N/A	N/A	4.1E-10	4.1E-10	9.3E-10	0.2%
Loss of 4.16kV AC Bus B	5.6E-11	4.7E-11	3.7E-10	1.9E-10	1.9E-10	8.5E-10	0.1%
Loss of 125V DC Bus BD1	5.5E-11	4.6E-11	3.7E-10	1.8E-10	1.8E-10	8.3E-10	0.1%
Loss of Atmospheric Relief Valves	N/A	4.9E-11	3.9E-10	2.0E-10	N/A	6.3E-10	0.1%
Loss of 125V DC Bus AD1	2.3E-11	1.9E-11	1.5E-10	7.6E-11	7.6E-11	3.5E-10	0.1%
Loss of Aux. Component Cooling Water	1.6E-11	1.3E-11	1.0E-10	5.2E-11	5.2E-11	2.4E-10	0.0%
Small LOCA	1.5E-11	1.3E-11	1.0E-10	5.1E-11	5.1E-11	2.3E-10	0.0%
Interfacing-system LOCA in RHR hot leg suction	1.4E-11	1.2E-11	9.6E-11	4.8E-11	4.8E-11	2.2E-10	0.0%
Inadvertent Safety Injection Actuation	9.3E-12	7.8E-12	6.2E-11	3.1E-11	3.1E-11	1.4E-10	0.0%

Table 10-11 CDF (per Year) for Low Power and Hot Standby POSs during Drained Maintenance Outage (POT-3) (cont.)

Initiating Event	POS 1	POS 2	POS 14	POS 15A	POS 15B	Total	%
Steam generator tube rupture	7.9E-12	6.6E-12	5.3E-11	2.6E-11	2.6E-11	1.2E-10	0.0%
Secondary-Side Break upstream MSIVs or downstream MFIVs	6.8E-12	5.6E-12	4.5E-11	2.3E-11	2.3E-11	1.0E-10	0.0%
Excessive LOCA (Reactor Vessel Rupture)	6.4E-12	5.3E-12	4.2E-11	2.1E-11	2.1E-11	9.7E-11	0.0%
Loss of 120V AC Panels A and B	6.1E-12	5.1E-12	4.1E-11	2.0E-11	2.0E-11	9.3E-11	0.0%
ISLOCA from RHR Cold Leg Injection Line A	2.7E-12	2.2E-12	1.8E-11	8.9E-12	8.9E-12	4.0E-11	0.0%
ISLOCA from RHR Cold Leg Injection Line B	2.7E-12	2.2E-12	1.8E-11	8.9E-12	8.9E-12	4.0E-11	0.0%
Large LOCA	2.3E-12	1.9E-12	1.5E-11	7.6E-12	7.6E-12	3.4E-11	0.0%
ISLOCA from RCP Stage 1 Seal Leak Off	2.2E-12	1.8E-12	1.4E-11	7.2E-12	7.2E-12	3.3E-11	0.0%
Loss of Instrument Air	1.6E-12	1.3E-12	1.0E-11	5.2E-12	5.2E-12	2.4E-11	0.0%
Loss of 120V AC Panels A and C	4.8E-15	4.0E-15	3.2E-14	1.6E-14	1.6E-14	7.3E-14	0.0%
Loss of 120V AC Panels A and D	2.7E-15	2.3E-15	1.8E-14	9.1E-15	9.1E-15	4.2E-14	0.0%
Loss of 120V AC Panels B and C	2.7E-15	2.3E-15	1.8E-14	9.1E-15	9.1E-15	4.2E-14	0.0%
Loss of 120V AC Panels B and D	2.7E-15	2.3E-15	1.8E-14	9.1E-15	9.1E-15	4.2E-14	0.0%
Loss of 120V AC Panels C and D	2.7E-15	2.3E-15	1.8E-14	9.1E-15	9.1E-15	4.2E-14	0.0%
ISLOCA from RCP Thermal Barrier Heat Exchanger	2.2E-15	1.9E-15	1.5E-14	7.5E-15	7.5E-15	3.4E-14	0.0%
Total Low Power and Hot Standby POSs	7.3E-09	7.0E-09	5.6E-08	2.1E-08	2.4E-08	1.2E-07	19.1%
POS % Contribution	1.2%	1.2%	9.3%	3.4%	4.0%		19.1%
Total All POSs						6.0E-07	100%
Note (1) The initiating event frequency for loss of offsite power during shutdown is applied to hot standby POSs, POS 2 and POS 14.							
Note (2) The initiating event frequency is increased by a factor of 10 to account for the uncertainty in initiating event frequency during low power and hot standby accident scenario.							

Table 10-12 CDF (per Year) for Shutdown POSs during Drained Maintenance Outage (POT-3) with Mid-Loop Entry

Initiating Event	POS 3	POS 4	POS5A	POS 5B	POS 6	POS 11	POS 12	POS 13	Total	%
LOOP during reactor shutdown	3.5E-09	4.7E-09	2.3E-08	4.8E-08 (Note 1)	1.3E-07 (Note 1)	3.6E-08	5.6E-08	2.3E-08	3.3E-07	29.8%
Loss of RHR cooling during reactor shutdown	1.4E-11	1.8E-11	9.2E-11	2.8E-08 (Note 1)	2.3E-07 (Note 1)	7.4E-09	2.2E-10	9.2E-11	2.6E-07	23.6%
Overdrain entering mid-loop	N/A	N/A	N/A	N/A	2.1E-07	N/A	N/A	N/A	2.1E-07	18.7%
Loss of inventory during reactor shutdown	1.2E-08	1.6E-08	7.8E-08	1.7E-09	1.6E-09	1.7E-09	4.0E-09	1.7E-09	1.2E-07	10.4%
Loss of NSCW during reactor shutdown	2.8E-10	3.7E-10	1.9E-09	1.6E-08 (Note 1)	1.6E-08 (Note 1)	8.0E-09	4.5E-09	1.9E-09	4.9E-08	4.4%
Loss of 4.16kV AC Bus during reactor shutdown	2.6E-10	3.4E-10	1.7E-09	3.0E-09 (Note 1)	1.7E-08 (Note 1)	1.7E-09	4.1E-09	1.7E-09	3.0E-08	2.7%
Total Shutdown POSs	1.6E-08	2.1E-08	1.0E-07	9.7E-08	6.0E-07	5.5E-08	6.9E-08	2.9E-08	9.9E-07	89.6%
POS % Contribution	1.4%	1.9%	9.4%	8.7%	54.3%	5.0%	6.2%	2.6%		89.6%
Total All POSs									1.1E-06	100%
Note (1) Human error probabilities are increased by a factor of ten to account for potentially higher decay heat levels in reduced inventory POSs during POT-3 compared to the modeled refueling outage POT-4.										

Table 10-13 CDF (per Year) for Low Power and Hot Standby POSs during Drained Maintenance Outage (POT-3) with Mid-Loop Entry

Initiating Event	POS 1	POS 2	POS 14	POS 15A	POS 15B	Total	%
LOOP during reactor shutdown (Note 1)	N/A	5.9E-09	4.7E-08	N/A	N/A	5.3E-08	4.8%
Other Transients	1.6E-09 (Note 2)	N/A	N/A	5.4E-09 (Note 2)	5.4E-09 (Note 2)	1.2E-08	1.1%
Grid-Relate LOOP (at-power)	1.2E-09	N/A	N/A	3.9E-09	3.9E-09	8.9E-09	0.8%
Loss of NSCW (at-power)	5.6E-10	4.7E-10	3.7E-09	1.9E-09	1.9E-09	8.5E-09	0.8%
Switchyard-Centered LOOP (at-power)	6.6E-10	N/A	N/A	2.2E-09	2.2E-09	5.1E-09	0.5%
Reactor Trip	6.2E-10 (Note 2)	N/A	N/A	2.1E-09 (Note 2)	2.1E-09 (Note 2)	4.8E-09	0.4%
Weather-Relate LOOP (at-power)	5.7E-10	N/A	N/A	1.9E-09	1.9E-09	4.4E-09	0.4%
Turbine Trip	6.8E-10 (Note 2)	N/A	N/A	N/A	2.3E-09 (Note 2)	3.0E-09	0.3%
Medium LOCA	1.5E-10	1.2E-10	9.9E-10	5.0E-10	5.0E-10	2.3E-09	0.2%
Secondary-Side Break downstream MSIVs or upstream MFIVs	1.0E-10	8.4E-11	6.8E-10	3.4E-10	3.4E-10	1.5E-09	0.1%
Loss of Auxiliary Feedwater	N/A	1.2E-10	9.4E-10	4.7E-10	N/A	1.5E-09	0.1%
Loss of Main Feedwater	3.3E-10 (Note 2)	N/A	N/A	N/A	1.1E-09 (Note 2)	1.4E-09	0.1%
Loss of 4.16kV AC Bus A	9.1E-11	7.6E-11	6.1E-10	3.0E-10	3.0E-10	1.4E-09	0.1%
Loss of Condenser Heat Sink	3.0E-10 (Note 2)	N/A	N/A	N/A	1.0E-09 (Note 2)	1.3E-09	0.1%
Loss of RCP Seal Injection	6.6E-11	5.5E-11	4.4E-10	2.2E-10	2.2E-10	1.0E-09	0.1%
Plant-Centered LOOP (at-power)	1.2E-10	N/A	N/A	4.1E-10	4.1E-10	9.3E-10	0.1%
Loss of 4.16kV AC Bus B	5.6E-11	4.7E-11	3.7E-10	1.9E-10	1.9E-10	8.5E-10	0.1%
Loss of 125V DC Bus BD1	5.5E-11	4.6E-11	3.7E-10	1.8E-10	1.8E-10	8.3E-10	0.1%
Loss of Atmospheric Relief Valves	N/A	4.9E-11	3.9E-10	2.0E-10	N/A	6.3E-10	0.1%
Loss of 125V DC Bus AD1	2.3E-11	1.9E-11	1.5E-10	7.6E-11	7.6E-11	3.5E-10	0.0%
Loss of Aux. Component Cooling Water	1.6E-11	1.3E-11	1.0E-10	5.2E-11	5.2E-11	2.4E-10	0.0%
Small LOCA	1.5E-11	1.3E-11	1.0E-10	5.1E-11	5.1E-11	2.3E-10	0.0%

Table 10-13 CDF (per Year) for Low Power and Hot Standby POSs during Drained Maintenance Outage (POT-3) with Mid-Loop Entry (cont.)

Initiating Event	POS 1	POS 2	POS 14	POS 15A	POS 15B	Total	%
Interfacing-system LOCA in RHR hot leg suction	1.4E-11	1.2E-11	9.6E-11	4.8E-11	4.8E-11	2.2E-10	0.0%
Inadvertent Safety Injection Actuation	9.3E-12	7.8E-12	6.2E-11	3.1E-11	3.1E-11	1.4E-10	0.0%
Steam generator tube rupture	7.9E-12	6.6E-12	5.3E-11	2.6E-11	2.6E-11	1.2E-10	0.0%
Secondary-Side Break upstream MSIVs or downstream MFIVs	6.8E-12	5.6E-12	4.5E-11	2.3E-11	2.3E-11	1.0E-10	0.0%
Excessive LOCA (Reactor Vessel Rupture)	6.4E-12	5.3E-12	4.2E-11	2.1E-11	2.1E-11	9.7E-11	0.0%
Loss of 120V AC Panels A and B	6.1E-12	5.1E-12	4.1E-11	2.0E-11	2.0E-11	9.3E-11	0.0%
ISLOCA from RHR Cold Leg Injection Line A	2.7E-12	2.2E-12	1.8E-11	8.9E-12	8.9E-12	4.0E-11	0.0%
ISLOCA from RHR Cold Leg Injection Line B	2.7E-12	2.2E-12	1.8E-11	8.9E-12	8.9E-12	4.0E-11	0.0%
Large LOCA	2.3E-12	1.9E-12	1.5E-11	7.6E-12	7.6E-12	3.4E-11	0.0%
ISLOCA from RCP Stage 1 Seal Leak Off	2.2E-12	1.8E-12	1.4E-11	7.2E-12	7.2E-12	3.3E-11	0.0%
Loss of Instrument Air	1.6E-12	1.3E-12	1.0E-11	5.2E-12	5.2E-12	2.4E-11	0.0%
Loss of 120V AC Panels A and C	4.8E-15	4.0E-15	3.2E-14	1.6E-14	1.6E-14	7.3E-14	0.0%
Loss of 120V AC Panels A and D	2.7E-15	2.3E-15	1.8E-14	9.1E-15	9.1E-15	4.2E-14	0.0%
Loss of 120V AC Panels B and C	2.7E-15	2.3E-15	1.8E-14	9.1E-15	9.1E-15	4.2E-14	0.0%
Loss of 120V AC Panels B and D	2.7E-15	2.3E-15	1.8E-14	9.1E-15	9.1E-15	4.2E-14	0.0%
Loss of 120V AC Panels C and D	2.7E-15	2.3E-15	1.8E-14	9.1E-15	9.1E-15	4.2E-14	0.0%
ISLOCA from RCP Thermal Barrier Heat Exchanger	2.2E-15	1.9E-15	1.5E-14	7.5E-15	7.5E-15	3.4E-14	0.0%
Total Low Power and Hot Standby POSs	7.3E-09	7.0E-09	5.6E-08	2.1E-08	2.4E-08	1.2E-07	10.4%
POS % Contribution	0.7%	0.6%	5.1%	1.8%	2.2%		10.4%
Total All POSs						1.1E-06	100%
Note (1) The initiating event frequency for loss of offsite power during shutdown is applied to hot standby POSs, POS 2 and POS 14.							
Note (2) The initiating event frequency is increased by a factor of 10 to account for the uncertainty in initiating event frequency during low power and hot standby accident scenario.							

10.3.4 Alternative Maintenance Outage with One Emergency Diesel Generator Unavailable

The unavailability of risk-significant equipment has a significant impact on the outage risk. The PRA model assumptions about maintenance unavailabilities should be formulated with realistic plant configurations and informed by past operating experience. However, detailed industry-wide plant outage configuration information is not readily available, and plant configurations can vary widely from outage to outage.

This sensitivity study assesses a possible maintenance outage with one safety-related emergency diesel generator unavailable for part of the outage duration. The assessed outage is a non-drained maintenance outage with RHR cooling (POT-2). The modeling assumptions are identical to the POT-2 outage assessed in Section 10.3.3 with the additional assumption of the train A emergency diesel generator unavailable for the first part of the outage from POS 1 through POS 4. The diesel generator is assumed to be restored for the heatup and restart portion of the outage, POS 12 through POS 15B. The at-power and shutdown modeled scenarios are re-evaluated with the train A emergency diesel generator unavailable, and the revised CCDF values are applied to the POS 1 through POS 4 estimated CDF values.

Table 10-14 shows the CDF contributions from the initiating event categories for the alternate POT-2 outage with one emergency diesel generator unavailable. The most notable impact is the increased contribution of the shutdown and at-power loss of offsite power scenarios when compared to the nominal POT-2 outage (see Table 10-9). The CDF contributions from transient initiating events (e.g., other transients, turbine trip, reactor trip, and loss of main feedwater) are increased due to the consequential losses of offsite power modeled in those scenarios. The shutdown loss of power to a 4.16kV AC bus scenario also contributes significantly to the CDF estimate. The overall POT-2 outage CDF increases significantly from 6×10^{-7} per year to 1.3×10^{-6} per year.

Table 10-14 CDF (per Year) during Maintenance Outage (POT-2) with One Emergency Diesel Generator Unavailable

Initiating Event	POS 1	POS 2	POS 3	POS 4	POS 12	POS 13	POS 14	POS 15A	POS 15B	Total	%
LOOP during reactor shutdown (Note 1)	N/A	3.2E-08	1.9E-08	4.3E-07	5.6E-08	2.3E-08	4.7E-08	N/A	N/A	6.1E-07	46.4%
Loss of inventory during reactor shutdown	N/A	N/A	1.4E-08	3.3E-07	4.0E-09	1.7E-09	N/A	N/A	N/A	3.5E-07	26.4%
Other Transients	7.1E-08 (Note 2)	N/A	N/A	N/A	N/A	N/A	N/A	5.4E-09 (Note 2)	9.7E-09 (Note 2)	8.6E-08	6.6%
Loss of 4.16kV AC Bus during reactor shutdown	N/A	N/A	3.3E-09	7.4E-08	4.1E-09	1.7E-09	N/A	N/A	N/A	8.3E-08	6.3%
Turbine Trip	3.0E-08 (Note 2)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	4.1E-09 (Note 2)	3.4E-08	2.6%
Reactor Trip	2.8E-08 (Note 2)	N/A	N/A	N/A	N/A	N/A	N/A	2.1E-09 (Note 2)	3.7E-09 (Note 2)	3.4E-08	2.6%
Grid-Relate LOOP (at-power)	1.2E-08	N/A	N/A	N/A	N/A	N/A	N/A	3.9E-09	7.0E-09	2.2E-08	1.7%
Loss of Main Feedwater	1.2E-08 (Note 2)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	2.0E-09 (Note 2)	1.4E-08	1.1%
Loss of NSCW during reactor shutdown	N/A	N/A	2.8E-10	6.3E-09	4.5E-09	1.9E-09	N/A	N/A	N/A	1.3E-08	1.0%
Loss of Condenser Heat Sink	1.1E-08 (Note 2)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	1.8E-09 (Note 2)	1.3E-08	1.0%
Weather-Relate LOOP (at-power)	6.1E-09	N/A	N/A	N/A	N/A	N/A	N/A	1.9E-09	3.4E-09	1.1E-08	0.9%
Switchyard-Centered LOOP (at-power)	3.1E-09	N/A	N/A	N/A	N/A	N/A	N/A	2.2E-09	4.0E-09	9.2E-09	0.7%
Loss of NSCW (at-power)	5.6E-10	4.7E-10	N/A	N/A	N/A	N/A	3.7E-09	1.9E-09	3.3E-09	1.0E-08	0.8%
Loss of RCP Seal Injection	3.2E-09	2.7E-09	N/A	N/A	N/A	N/A	4.4E-10	2.2E-10	4.0E-10	7.0E-09	0.5%
Secondary-Side Break downstream MSIVs or upstream MFIVs	8.6E-10	7.1E-10	N/A	N/A	N/A	N/A	6.8E-10	3.4E-10	6.1E-10	3.2E-09	0.2%
Medium LOCA	2.4E-10	2.0E-10	N/A	N/A	N/A	N/A	9.9E-10	5.0E-10	8.9E-10	2.8E-09	0.2%

Table 10-14 CDF (per Year) during Maintenance Outage (POT-2) with One Emergency Diesel Generator Unavailable (cont.)

Initiating Event	POS 1	POS 2	POS 3	POS 4	POS 12	POS 13	POS 14	POS 15A	POS 15B	Total	%
Loss of 125V DC Bus BD1	7.6E-10	6.3E-10	N/A	N/A	N/A	N/A	3.7E-10	1.8E-10	3.3E-10	2.3E-09	0.2%
Loss of Auxiliary Feedwater	n/a	4.8E-10	N/A	N/A	N/A	N/A	9.4E-10	4.7E-10	n/a	1.9E-09	0.1%
Plant-Centered LOOP (at-power)	5.6E-10	n/a	N/A	N/A	N/A	N/A	n/a	4.1E-10	7.3E-10	1.7E-09	0.1%
Loss of 4.16kV AC Bus B	4.3E-10	3.6E-10	N/A	N/A	N/A	N/A	3.7E-10	1.9E-10	3.4E-10	1.7E-09	0.1%
Loss of 4.16kV AC Bus A	9.1E-11	7.6E-11	N/A	N/A	N/A	N/A	6.1E-10	3.0E-10	5.5E-10	1.6E-09	0.1%
Loss of Atmospheric Relief Valves	n/a	4.9E-10	N/A	N/A	N/A	N/A	3.9E-10	2.0E-10	n/a	1.1E-09	0.1%
Inadvertent Safety Injection Actuation	3.1E-10	2.6E-10	N/A	N/A	N/A	N/A	6.2E-11	3.1E-11	5.6E-11	7.2E-10	0.1%
Loss of RHR cooling during reactor shutdown	N/A	N/A	1.4E-11	3.1E-10	2.2E-10	9.2E-11	N/A	N/A	N/A	6.4E-10	0.0%
Steam generator tube rupture	1.8E-10	1.5E-10	N/A	N/A	N/A	N/A	5.3E-11	2.6E-11	4.7E-11	4.6E-10	0.0%
Loss of 125V DC Bus AD1	2.3E-11	1.9E-11	N/A	N/A	N/A	N/A	1.5E-10	7.6E-11	1.4E-10	4.1E-10	0.0%
Loss of Aux. Component Cooling Water	1.6E-11	1.3E-11	N/A	N/A	N/A	N/A	1.0E-10	5.2E-11	9.4E-11	2.8E-10	0.0%
Small LOCA	1.5E-11	1.3E-11	N/A	N/A	N/A	N/A	1.0E-10	5.1E-11	9.1E-11	2.7E-10	0.0%
Interfacing-system LOCA in RHR hot leg suction	1.4E-11	1.2E-11	N/A	N/A	N/A	N/A	9.6E-11	4.8E-11	8.6E-11	2.6E-10	0.0%
Secondary-Side Break upstream MSIVs or downstream MFIVs	6.8E-11	5.6E-11	N/A	N/A	N/A	N/A	4.5E-11	2.3E-11	4.1E-11	2.3E-10	0.0%
Excessive LOCA (Reactor Vessel Rupture)	6.4E-12	5.3E-12	N/A	N/A	N/A	N/A	4.2E-11	2.1E-11	3.8E-11	1.1E-10	0.0%

Table 10-14 CDF (per Year) during Maintenance Outage (POT-2) with One Emergency Diesel Generator Unavailable (cont.)

Initiating Event	POS 1	POS 2	POS 3	POS 4	POS 12	POS 13	POS 14	POS 15A	POS 15B	Total	%
Loss of 120V AC Panels A and B	6.1E-12	5.1E-12	N/A	N/A	N/A	N/A	4.1E-11	2.0E-11	3.7E-11	1.1E-10	0.0%
ISLOCA from RHR Cold Leg Injection Line A	2.7E-12	2.2E-12	N/A	N/A	N/A	N/A	1.8E-11	8.9E-12	1.6E-11	4.8E-11	0.0%
ISLOCA from RHR Cold Leg Injection Line B	2.7E-12	2.2E-12	N/A	N/A	N/A	N/A	1.8E-11	8.9E-12	1.6E-11	4.8E-11	0.0%
Large LOCA	2.3E-12	1.9E-12	N/A	N/A	N/A	N/A	1.5E-11	7.6E-12	1.4E-11	4.0E-11	0.0%
ISLOCA from RCP Stage 1 Seal Leak Off	2.2E-12	1.8E-12	N/A	N/A	N/A	N/A	1.4E-11	7.2E-12	1.3E-11	3.8E-11	0.0%
Loss of Instrument Air	1.6E-12	1.3E-12	N/A	N/A	N/A	N/A	1.0E-11	5.2E-12	9.4E-12	2.8E-11	0.0%
Loss of 120V AC Panels A and C	4.8E-15	4.0E-15	N/A	N/A	N/A	N/A	3.2E-14	1.6E-14	2.9E-14	8.5E-14	0.0%
Loss of 120V AC Panels A and D	2.7E-15	2.3E-15	N/A	N/A	N/A	N/A	1.8E-14	9.1E-15	1.6E-14	4.9E-14	0.0%
Loss of 120V AC Panels B and C	2.7E-15	2.3E-15	N/A	N/A	N/A	N/A	1.8E-14	9.1E-15	1.6E-14	4.9E-14	0.0%
Loss of 120V AC Panels B and D	2.7E-15	2.3E-15	N/A	N/A	N/A	N/A	1.8E-14	9.1E-15	1.6E-14	4.9E-14	0.0%
Loss of 120V AC Panels C and D	2.7E-15	2.3E-15	N/A	N/A	N/A	N/A	1.8E-14	9.1E-15	1.6E-14	4.9E-14	0.0%
ISLOCA from RCP Thermal Barrier Heat Exchanger	2.2E-15	1.9E-15	N/A	N/A	N/A	N/A	1.5E-14	7.5E-15	1.3E-14	4.0E-14	0.0%
Total	1.8E-07	3.8E-08	3.7E-08	8.4E-07	6.9E-08	2.9E-08	5.6E-08	2.1E-08	4.4E-08	1.3E-06	100%
POS % Contribution	13.8%	2.9%	2.8%	63.8%	5.3%	2.2%	4.3%	1.6%	3.3%		
Note (1) The initiating event frequency for loss of offsite power during shutdown is applied to hot standby POSs, POS 2 and POS 14. Note (2) The initiating event frequency is increased by a factor of 10 to account for the uncertainty in initiating event frequency during low power and hot standby accident scenario.											

10.3.5 Loss of Offsite Power Initiating Event Frequency Sensitivity Study

Loss of offsite power is the highest contributing initiating event to shutdown CDF. The operating experience data show that the loss of offsite power frequency is higher for shutdown operations than critical operations, but the specific causes for this difference are not well understood. This project uses a generic shutdown loss of offsite power frequency across all shutdown POSs, but there may be POS-specific and plant-specific conditions that influence the likelihood of an event occurrence. The uncertainties in how the loss of offsite power initiating event frequencies are applied to shutdown states can significantly influence the results.

The loss of offsite power initiating event frequency for shutdown is based on a review of non-critical operating experience for the years 2000 through 2015 (see Section 8.1.1 for further details). The loss of offsite power initiating event frequency for shutdown is approximately six times higher than the frequency for critical operations. While the operating experience shows this difference between the shutdown and at-power frequencies, there are modeling assumptions associated with how the events are analyzed. Unique conditions related to a specific plant, outage, and POS can influence the applicability of operating experience and the frequency estimates that would apply during specific conditions. However, there generally is not adequate data collection on the details of outage conditions or durations of outages and POSs to support more refined frequency estimates, and caution should be exercised in disregarding operating experience as not applicable to a specific analysis. The discussion here provides an assessment of the reasons why the shutdown frequency is higher than the at-power frequency and considers the impacts of a range of different loss of offsite power frequencies that could be applicable to a refueling outage.

Table 10-15 shows a comparison of the loss of offsite power category frequencies for shutdown and at-power. These are the same loss of offsite power category frequencies that are routinely updated by the NRC reactor operating experience assessment program (NRC, 2021a). In the L3PRA project, the shutdown frequencies are not separated into categories due to the small number of events and uncertainties in the exposure period data. Separating the data into four frequency estimates was deemed to exacerbate the uncertainty of the estimates, so only a single grouped loss of offsite power frequency is used for the shutdown PRA. However, estimated frequencies for each category are shown here for comparison.

The shutdown loss of offsite power frequency estimates are larger than the at-power frequencies for all sub-categories. The higher frequencies may be explained by plant activities during outages; for example, transferring electrical power sources and maintenance in the plant switchyard. These activities are expected to be associated with the plant-centered and switchyard-centered categories, and both categories show a significant increase compared to their at-power counterparts. The weather-related loss of offsite power frequency is also significantly higher during shutdown, but the reasons for the increase are less clear. There may be seasonal impacts related to the timing of outages. There are also cases where plants initiate an outage in anticipation of a forecasted weather event, which then results in a loss of offsite power while the plant is shutdown. For the grid-related losses of offsite power, there are no identified reasons for why shutdown states would have a higher frequency. The difference in frequency may be due to the expected statistical variation.

A series of sensitivity studies were developed to assess the impact of different loss of offsite power frequency assumptions. The base CDF results using the shutdown-specific frequency are included for comparison. The CDF using the at-power loss of offsite power frequency is

assessed. Also, special consideration is given to the POSs with the refueling cavity flooded (i.e., POSs 8E and 8L). The operating experience suggests that many of the plant-centered and switchyard-centered events are associated with maintenance activities that occur during these POSs. Two additional sensitivity cases are included to assess the impact of higher loss of offsite power frequencies during these POSs. The first case applies the shutdown frequency to only POSs 8E and 8L. The at-power frequency is applied to all other shutdown POSs. The second case increases the shutdown loss of offsite power frequency by a factor of three during POSs 8E and 8L, based on the assumption that the shutdown initiating events are concentrated during this portion of the outage, which accounts for about one third of the outage time.

The CDF results for initiating event categories for the four cases described above are shown in Table 10-16. The CDF contributions to loss of offsite power for the shutdown POSs are shown in Table 10-17. As expected, using the at-power frequency shows a much lower contribution from losses of offsite power events and a significant decrease in the overall shutdown CDF (see Case A in Table 10-16 and Table 10-17). Case B shows the shutdown frequency applied only during POSs 8E and 8L. While the POS 8E and 8L CDF contributions are significantly increased (accounting for almost 89 percent of the loss of offsite power CDF), the overall shutdown CDF is slightly lower than the base case. Finally, Case C shows the results with the shutdown frequency increased by a factor of three during POSs 8E and 8L. Again, the POS 8E and 8L CDF contributions account for most of the loss of offsite power CDF, and the overall shutdown CDF is greater than the base case result.

Table 10-15 Comparison of Shutdown and At-Power Loss of Offsite Power Frequencies

LOOP Category	Shutdown Estimated Frequency (per year)	At-Power Frequency (per reactor-critical-year)	At-Power Frequency (per year; 0.93 plant availability)	Ratio of Shutdown to At-Power Frequency
Grid-Related LOOP	2.1E-02	1.2E-02	1.1E-02	1.8
Plant-Centered LOOP	2.1E-02	1.9E-03	1.8E-03	11.7
Switchyard-Centered LOOP	7.0E-02	1.0E-02	9.7E-03	7.2
Weather-Related LOOP	4.2E-02	3.9E-03	3.6E-03	11.6
Total	1.6E-01	2.9E-02	2.7E-02	6.1

Table 10-16 Loss of Offsite Power Frequency Sensitivity Cases CDF Results by Initiating Event Category

Initiating Event Category	Base Case CDF Results		Case A: At-Power LOOP Frequency		Case B: Shutdown LOOP Frequency POS 8E & 8L Only		Case C: Shutdown LOOP Frequency ×3 POS 8E & 8L Only	
	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF
Loss of Offsite Power	8.5E-06	68.1%	1.4E-06	25.7%	5.4E-06	57.6%	1.5E-05	79.0%
Overdrain Mid-loop (before refueling)	1.2E-06	9.8%	1.2E-06	22.8%	1.2E-06	13.0%	1.2E-06	6.4%
Loss of Inventory at pressure	7.8E-07	6.2%	7.8E-07	14.4%	7.8E-07	8.2%	7.8E-07	4.1%
Loss of RHR	6.7E-07	5.4%	6.7E-07	12.5%	6.7E-07	7.1%	6.7E-07	3.5%
Loss of Service Water	6.4E-07	5.1%	6.4E-07	11.9%	6.4E-07	6.8%	6.4E-07	3.4%
Loss of power to 4,160V AC bus	3.7E-07	3.0%	3.7E-07	7.0%	3.7E-07	4.0%	3.7E-07	2.0%
Overdrain Mid-loop (after refueling)	2.3E-07	1.8%	2.3E-07	4.2%	2.3E-07	2.4%	2.3E-07	1.2%
Loss of Inventory	4.2E-08	0.3%	4.2E-08	0.8%	4.2E-08	0.5%	4.2E-08	0.2%
Loss of Inventory Displacement	2.0E-08	0.2%	2.0E-08	0.4%	2.0E-08	0.2%	2.0E-08	0.1%
Loss of Inventory Spray	1.8E-08	0.1%	1.8E-08	0.3%	1.8E-08	0.2%	1.8E-08	0.1%
Loss of Inventory Large	1.3E-10	< 0.1%	1.3E-10	< 0.1%	1.3E-10	< 0.1%	1.3E-10	< 0.1%
Total	1.3E-05	100%	5.4E-06	100%	9.4E-06	100%	1.9E-05	100%

Table 10-17 Loss of Offsite Power Frequency Sensitivity Cases CDF Results by Shutdown Plant Operating State

Shutdown POS	Base Case CDF Results		Case A: At-Power LOOP Frequency		Case B: Shutdown LOOP Frequency POS 8E & 8L Only		Case C: Shutdown LOOP Frequency ×3 POS 8E & 8L Only	
	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF
POS 3 Mode 4 cooldown	2.3E-08	0.3%	3.7E-09	0.3%	3.7E-09	0.1%	3.7E-09	< 0.1%
POS 4 Mode 5 cooldown	1.7E-07	2.0%	2.7E-08	2.0%	2.7E-08	0.5%	2.7E-08	0.2%
POS 5A Pressurizer solid	1.3E-06	14.9%	2.1E-07	14.9%	2.1E-07	3.8%	2.1E-07	1.4%
POS 5B Draining RCS	4.9E-07	5.7%	7.9E-08	5.7%	7.9E-08	1.5%	7.9E-08	0.5%
POS 6 Mid- loop (pre- refuel)	2.2E-07	2.6%	3.5E-08	2.6%	3.5E-08	0.7%	3.5E-08	0.2%
POS 7 Filling refuel cavity	1.6E-07	1.9%	2.6E-08	1.8%	2.6E-08	0.5%	2.6E-08	0.2%
POS 8E Refuel cavity flooded (pre-refuel)	2.7E-06	32.1%	4.5E-07	32.2%	2.7E-06	50.6%	8.2E-06	54.7%
POS 8L Refuel cavity flooded (post-refuel)	2.1E-06	24.3%	3.4E-07	24.3%	2.1E-06	38.2%	6.2E-06	41.3%
POS 9 Draining refuel cavity	2.8E-07	3.3%	4.6E-08	3.3%	4.6E-08	0.8%	4.6E-08	0.3%
POS 10 Mid- loop (post- refuel)	3.1E-07	3.6%	5.0E-08	3.6%	5.0E-08	0.9%	5.0E-08	0.3%
POS 11 Vacuum fill RCS	2.3E-07	2.7%	3.7E-08	2.6%	3.7E-08	0.7%	3.7E-08	0.2%
POS 12 Mode 5 heatup	4.2E-07	4.9%	6.8E-08	4.9%	6.8E-08	1.3%	6.8E-08	0.5%
POS 13 Mode 4 heatup	1.6E-07	1.8%	2.5E-08	1.8%	2.5E-08	0.5%	2.5E-08	0.2%
Total:	8.5E-06	100.0%	1.4E-06	100.0%	5.4E-06	100.0%	1.5E-05	100.0%

10.3.6 Likelihood of Overdrain Entering Mid-loop

The conditional probability of an overdrain event occurring when entering the mid-loop condition is based on an industry-wide operating experience review, as described in Section 8.1.3. However, the operating experience data may not be representative of more recent operating practices in the industry. The overdrain event occurrences are documented in (EPRI, 2010), and the events occurred in the 1990s. The collection of industry-wide mid-loop entries is based on a review of PWR outage records from 1990 through 2009. More recent data on overdrain events and mid-loop entries has not been collected.

The assumptions driving the overdrain probability estimate are significant to the shutdown CDF results. The overdrain events contribute over 10% of the refueling outage shutdown CDF (see Table 9-1). Incorporating more recent operating experience may show that overdrain occurrences are less likely and reduce their significance to shutdown CDF. However, because there has not been comprehensive shutdown initiating event data collection since the publication of (EPRI, 2010), there may be additional overdrain events or precursor events that are not properly accounted. The sensitivity cases discussed here show a range of possible overdrain probabilities with assumptions about more recent mid-loop data.

The first sensitivity case considers the inclusion of additional overdrain events or precursor events that would increase the probability estimate for overdrain occurrence. The data from (EPRI, 2010) show two overdrain events that resulted in loss of decay heat removal. A third event is documented that involves inadvertent draining several inches below the top of the hot leg, but decay heat removal was not lost. Overdrain events such as this should be considered for inclusion in the conditional probability estimate. If operators had not terminated the draining, then loss of cooling and pump damage could have occurred. The operators' response and recovery from the overdrain is credited and modeled in the PRA, so including this event would be consistent with the event scenario modeling assumptions. An alternate overdrain probability estimate including this event is shown in Table 10-18.

In looking at more recent operating experience, challenges have been identified in finding completeness in shutdown initiating events. It is possible that additional overdrain events have occurred, but there are no documented records of such events in recent years. The inclusion of more recent operating experience assuming no additional overdrain events would tend to reduce the estimated overdrain probability. Additional operating years from 2010 to 2015 (consistent with the years assessed for other shutdown events assessed in Section 8.1) are included in this sensitivity. The exact number of industry-wide mid-loop entries is not known, but it is estimated below.

During the six-year period from 2010 to 2015, 67 PWR units were in operation. Each unit is assumed to have four refueling outages during the period and two mid-loop entries per outage. One additional unit operated for approximately half of the six-year period before permanently shutting down. That unit is assumed to have two refueling outages with two mid-loop entries per outage. This yields a total of 540 mid-loop entries. This may be an overestimate given that plants recognize the risk significance of mid-loop operations and would avoid entering mid-loop unless necessary. However, plants have also focused on reducing outage durations, which may make mid-loop entry a more common practice in recent years. The overdrain estimate in Section 8.1.3 shows an estimate of 614 mid-loop entries over the 20-year period from 1990 through 2009. This suggests a much lower rate of mid-loop entry. If this rate were continued from 2010 through 2015, then 184 mid-loop entries would be expected. This range in the

possible mid-loop entries is reflected in the conditional probability estimates in Table 10-18. The prior distribution for these estimates is based on the 1990-2009 estimate from Section 8.1.3.

One more possible overdrain probability estimate is made using a non-informative prior distribution. This estimate reflects the possibility that the overdrain events from the 1990s are no longer representative of current operations, and therefore, should not be included in the probability estimate. This approach results in the lowest estimate of overdrain probability.

The shutdown CDF results considering the alternate estimates of overdrain probability are shown in Table 10-19. The impact of the different overdrain probabilities results in a total shutdown CDF ranging from 1.1×10^{-5} to 1.3×10^{-5} per year. The CDF contribution from overdrain events (including both pre-refueling mid-loop and post-refueling mid-loop) varies from about 3 percent to 15 percent of the shutdown CDF. In all cases, overdrain entering mid-loop remains a significant initiating event contributing to the shutdown CDF. Given this importance, it is recommended that data collection on mid-loop entries and overdrain events are improved to support estimates of overdrain probabilities that are representative of recent operating experience.

Table 10-18 Sensitivity Study of Overdrain Conditional Probability Estimates

Prior Distribution Parameters		Evidence		Posterior Beta Distribution			Sensitivity Description
α	β	Overdrain Events	Mid-Loop Entries	α	β	Mean Value	
0.5	0.5	2	614	2.5	612.5	4.1E-03	Base case (see Section 8.1.3)
0.5	0.5	3	614	3.5	611.5	5.7E-03	Additional overdrain event included
2.5	612.5	0	184	2.5	795.5	3.1E-03	2010-2015 experience with low mid-loop entries
2.5	612.5	0	540	2.5	1152.5	2.2E-03	2010-2015 experience with high mid-loop entries
0.5	0.5	0	540	0.5	540.5	9.2E-04	2010-2015 experience with non-informative prior

Table 10-19 Sensitivity Study of Overdrain Conditional Probability Shutdown CDF Results

Initiating Event Category	Base Case Overdrain Prob. 4.1E-03		Sensitivity A Overdrain Prob. 5.7E-03		Sensitivity B Overdrain Prob 3.1E-03		Sensitivity C Overdrain Prob 2.2E-03		Sensitivity D Overdrain Prob 9.2E-04	
	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF
Loss of Offsite Power	8.5E-06	68.1%	8.5E-06	65.1%	8.5E-06	70.1%	8.5E-06	72.0%	8.5E-06	74.8%
Overdrain Mid-loop (before refueling)	1.2E-06	9.8%	1.7E-06	13.1%	9.3E-07	7.7%	6.6E-07	5.6%	2.8E-07	2.4%
Loss of Inventory at pressure	7.8E-07	6.2%	7.8E-07	5.9%	7.8E-07	6.4%	7.8E-07	6.5%	7.8E-07	6.8%
Loss of RHR	6.7E-07	5.4%	6.7E-07	5.1%	6.7E-07	5.5%	6.7E-07	5.7%	6.7E-07	5.9%
Loss of Service Water	6.4E-07	5.1%	6.4E-07	4.9%	6.4E-07	5.2%	6.4E-07	5.4%	6.4E-07	5.6%
Loss of power to 4,160V AC bus	3.7E-07	3.0%	3.7E-07	2.9%	3.7E-07	3.1%	3.7E-07	3.2%	3.7E-07	3.3%
Overdrain Mid-loop (after refueling)	2.3E-07	1.8%	3.2E-07	2.4%	1.7E-07	1.4%	1.2E-07	1.0%	5.1E-08	0.5%
Loss of Inventory	4.2E-08	0.3%	4.2E-08	0.3%	4.2E-08	0.3%	4.2E-08	0.4%	4.2E-08	0.4%
Loss of Inventory Displacement	2.0E-08	0.2%	2.0E-08	0.2%	2.0E-08	0.2%	2.0E-08	0.2%	2.0E-08	0.2%
Loss of Inventory Spray	1.8E-08	0.1%	1.8E-08	0.1%	1.8E-08	0.1%	1.8E-08	0.1%	1.8E-08	0.2%
Loss of Inventory Large	1.3E-10	< 0.1%	1.3E-10	< 0.1%	1.3E-10	< 0.1%	1.3E-10	< 0.1%	1.3E-10	< 0.1%
Total	1.3E-05	100%	1.3E-05	100%	1.2E-05	100%	1.2E-05	100%	1.1E-05	100%

10.3.7 Time Available for Offsite and Onsite Power Recovery

In response to loss of offsite power events, the recovery of offsite and onsite power sources is limited by the plant batteries that are necessary for restoring power. This limits the time available that is credited for power recovery. However, some shutdown scenarios are very slowly developing, which would potentially allow additional time for recovery. The recovery of offsite power is limited to two hours after the loss of offsite power. The time available for onsite diesel generator recovery is limited to four hours after the loss of offsite power. The assumptions regarding power recovery are described in Section 4.3.5, and the timing assumptions are consistent with the modeling approach used in the L3PRA project Level 1 PRA for at-power operations (NRC, 2022).

The time available for recovery is based on plant-specific battery depletion times, plant procedures, and the DC power requirements for aligning AC power sources. However, for some shutdown accident scenarios (e.g., conditions with refueling cavity flooded), the time available for potential recovery actions is expansive. While there are no procedure guidelines for long-term power recovery,³⁶ it is expected that plant operators would continue attempts to restore power and prevent core damage. The accident simulations performed for the shutdown Level 2 PRA have shown that for accidents during POS 8E with extended loss of offsite power, core damage initiates three to four days after the start of the event. The impact of additional time for power recovery is assessed through a sensitivity study on the probabilities for offsite and onsite power recovery.

The offsite power recovery probabilities are based on a statistical analysis of loss of offsite power event durations (see Loss of Offsite Power section of (NRC, 2021a)). The base modeling assumption uses the non-recovery probability for two hours after loss of power, which is 3.3×10^{-1} . The onsite diesel generator recovery probability is based on an assessment of the time to restore diesel generators after unplanned outages of the generators (NRC, 2005c). The 4-hour non-recovery probability for onsite emergency diesel generators used in the base model is 7.0×10^{-1} .

Considering the much longer times potentially available for power recovery during periods with the refueling cavity flooded, alternate non-recovery probabilities are applied to the model. The alternate non-recovery probabilities are based on 24 hours being available for recovery, which is consistent with the duration assumed for the shutdown PRA mission time. The 24-hour non-recovery probabilities for offsite and onsite power recovery are 2.8×10^{-2} and 2.9×10^{-1} , respectively.

The alternate non-recovery probabilities are applied to events occurring during POS 8E and 8L (refueling cavity flooded) and POS 7 (refueling cavity partially flooded). The impacts of the extended time for power recovery on the CDF results are shown in Table 10-20. The CDF contribution from losses of offsite power is reduced 45 percent, from 8.5×10^{-6} per year to 4.7×10^{-6} per year (overall shutdown CDF is reduced by approximately 31 percent, from 1.3×10^{-5} per year to 8.7×10^{-6} per year). Even with the substantial reduction in loss of offsite power CDF, it remains the dominant contributor to shutdown CDF.

³⁶ Note that this study is based on information that reflects the reference plant as it was designed and operated as of 2012. It does not reflect the potential impact of FLEX strategies implemented after 2012. The impacts of FLEX strategies are explored in the Level 3 PRA project overview report on the shutdown Level 1, 2, and 3 PRAs.

The loss of offsite power CDF contributions by individual POS are shown in Table 10-21. Even though the three POSs impacted by this sensitivity study (i.e., POSs 7, 8E, and 8L) all show reduced CDF contributions, POS 8E and POS 8L still make significant contributions to the overall loss of offsite power CDF. There are some failures that do not benefit from the extended recovery time because recovery or repair of the equipment failures are not credited (e.g., common-cause failure of the offsite power supply circuit breakers to the RATs).

Table 10-20 AC Power Recovery Sensitivity Case CDF Results by Initiating Event Category

Initiating Event Category	Base Case CDF Results		Sensitivity with Extended Power Recovery Time	
	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF
Loss of Offsite Power	8.5E-06	68.1%	4.7E-06	54.1%
Overdrain Mid-loop (before refueling)	1.2E-06	9.8%	1.2E-06	14.1%
Loss of Inventory at pressure	7.8E-07	6.2%	7.8E-07	8.9%
Loss of RHR	6.7E-07	5.4%	6.7E-07	7.7%
Loss of Service Water	6.4E-07	5.1%	6.4E-07	7.3%
Loss of power to 4,160V AC bus	3.7E-07	3.0%	3.7E-07	4.3%
Overdrain Mid-loop (after refueling)	2.3E-07	1.8%	2.3E-07	2.6%
Loss of Inventory	4.2E-08	0.3%	4.2E-08	0.5%
Loss of Inventory Displacement	2.0E-08	0.2%	2.0E-08	0.2%
Loss of Inventory Spray	1.8E-08	0.1%	1.8E-08	0.2%
Loss of Inventory Large	1.3E-10	< 0.1%	1.3E-10	< 0.1%
Total	1.3E-05	100%	8.7E-06	100%

Table 10-21 AC Power Recovery Sensitivity Case CDF Results by Shutdown Plant Operating State

Shutdown POS Contribution to Loss of Offsite Power CDF	Base Case CDF Results		Sensitivity with Extended Power Recovery Time	
	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF
POS 3 Mode 4 cooldown	2.3E-08	0.3%	2.3E-08	0.5%
POS 4 Mode 5 cooldown	1.7E-07	2.0%	1.7E-07	3.6%
POS 5A Pressurizer solid	1.3E-06	14.9%	1.3E-06	27.0%
POS 5B Draining RCS	4.9E-07	5.7%	4.9E-07	10.3%
POS 6 Mid-loop (pre-refuel)	2.2E-07	2.6%	2.2E-07	4.6%
POS 7 Filling refuel cavity	1.6E-07	1.9%	1.3E-07	2.7%
POS 8E Refuel cavity flooded (pre-refuel)	2.7E-06	32.1%	5.8E-07	12.4%
POS 8L Refuel cavity flooded (post-refuel)	2.1E-06	24.3%	4.4E-07	9.3%
POS 9 Draining refuel cavity	2.8E-07	3.3%	2.8E-07	6.0%
POS 10 Mid-loop (post-refuel)	3.1E-07	3.6%	3.1E-07	6.6%
POS 11 Vacuum fill RCS	2.3E-07	2.7%	2.3E-07	4.8%
POS 12 Mode 5 heatup	4.2E-07	4.9%	4.2E-07	8.9%
POS 13 Mode 4 heatup	1.6E-07	1.8%	1.6E-07	3.3%
Total:	8.5E-06	100.0%	4.7E-06	100.0%

10.3.8 Estimation of Human Error Probabilities

The modeling of human failure events and estimating human error probabilities is a key source of modeling uncertainty in the shutdown PRA. During shutdown operations with many key systems are inhibited or unavailable, there is a greater reliance on operator actions to diagnose and respond to initiating events. Several HFEs show significant contributions to the shutdown CDF results. Therefore, the estimation of their HEP values is important to the overall results. Evaluations of HRA methods have shown there can be variability in HEP estimates depending on the method used and/or the analyst applying the method (NRC, 2016).

To assess the impacts of variability of HEP estimates, the significant HFEs contributing to the shutdown CDF were identified and reassessed with alternate HEP values. The significant HFEs were selected by choosing all HFEs with Fussell-Vesely importance values greater than 0.005. The significant HFEs are shown in Table 10-22. Descriptions of the HFEs can be found in Table 7-1. Alternate HEP values are based on the HEP uncertainty distributions with the 5th percentile representing a lower bound value and the 95th percentile representing an upper bound value. All HEPs are modeled using a lognormal uncertainty distribution.

The initiating event CDF results using the lower bound HEP values are shown in Table 10-23 and compared against the base case results. Using the lower bound HEP values results in approximately 24 percent reduction in overall CDF. The largest CDF decreases are seen in the loss of inventory with RCS pressurized and overdrain entering mid-loop before refueling. The CDF reduction by POS is shown in Table 10-24. The largest CDF reductions are associated with mid-loop operations before refueling (POS 6) and pressurizer solid operation (POS 5A).

The initiating event CDF results using the upper bound HEP values are shown in Table 10-25 and compared against the base case results. Using the upper bound HEP values roughly doubles the overall CDF. Substantial CDF increases are seen in the overdrain entering mid-loop events and loss of inventory with RCS pressurized. Loss of inventory spray events and loss of RHR events also show large CDF increases with the upper bound HEP values. The CDF increase by POS is shown in Table 10-26. The largest CDF increases are associated with mid-loop operations before refueling (POS 6) and pressurizer solid operation (POS 5A).

Table 10-22 Significant Human Failure Events by Fussell-Vesely Importance

Human Failure Event Name	Fussell-Vesely	HEP Value	Error Factor	5 th Percentile	Median Value	95 th Percentile
1-LPI-XHE-XM-GFEED-LOOP	2.8E-01	5.5E-01	1.8	2.9E-01	5.2E-01	9.3E-01
1-OA-ALIGNPW-G-H	7.8E-02	9.2E-02	5.0	1.1E-02	5.7E-02	2.9E-01
1-LOI-XHE-XD-POS5A	6.1E-02	2.5E-02	5.0	3.1E-03	1.5E-02	7.7E-02
1-RHR-XHE-XL-TRN-MLDRL	4.8E-02	1.3E-02	5.0	1.5E-03	7.7E-03	3.9E-02
1-RHR-XHE-XM-MLDRL-I	3.8E-02	1.0E-02	5.0	1.3E-03	6.3E-03	3.2E-02
1-RHR-XHE-XM-TRIP-MD	2.7E-02	1.5E-01	5.0	2.9E-02	9.1E-02	4.5E-01
1-HPI-XHE-XM-FFEED-MLDRL	2.0E-02	8.0E-03	5.0	9.9E-04	4.9E-03	2.5E-02
1-LPI-XHE-XM-GFEED-LD	1.2E-02	4.5E-01	2.4	1.9E-01	3.9E-01	9.4E-01
1-RHR-XHE-XL-TRN-POS6	1.1E-02	1.3E-02	5.0	1.5E-03	7.7E-03	3.9E-02
1-LPI-XHE-XM-GFEED-MD	1.1E-02	5.1E-01	2.0	2.5E-01	4.6E-01	9.2E-01
1-OA-OSW-----H	8.8E-03	2.0E-02	5.0	2.5E-03	1.2E-02	6.2E-02
1-LOI-XHE-XM-ISO2-LT	5.8E-03	2.4E-02	5.0	3.0E-03	1.5E-02	7.6E-02

Table 10-23 Lower Bound HEP Sensitivity Case CDF Results by Initiating Event Category

Initiating Event Category	Base Case CDF Results		Sensitivity with Lower Bound HEP Values	
	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF
Loss of Offsite Power	8.5E-06	68.1%	7.5E-06	78.1%
Overdrain Mid-loop (before refueling)	1.2E-06	9.8%	2.6E-07	2.7%
Loss of Inventory at pressure	7.8E-07	6.2%	6.0E-08	0.6%
Loss of RHR	6.7E-07	5.4%	5.8E-07	6.1%
Loss of Service Water	6.4E-07	5.1%	5.4E-07	5.7%
Loss of power to 4,160V AC bus	3.7E-07	3.0%	3.7E-07	3.9%
Overdrain Mid-loop (after refueling)	2.3E-07	1.8%	2.1E-07	2.2%
Loss of Inventory	4.2E-08	0.3%	3.7E-08	0.4%
Loss of Inventory Displacement	2.0E-08	0.2%	1.7E-08	0.2%
Loss of Inventory Spray	1.8E-08	0.1%	1.1E-08	0.1%
Loss of Inventory Large	1.3E-10	< 0.1%	1.2E-10	< 0.1%
Total	1.3E-05	100%	9.6E-06	100%

Table 10-24 Lower Bound HEP Sensitivity Case CDF Results by Shutdown Plant Operating State

Shutdown POS Contribution CDF	Base Case CDF Results		Sensitivity with Lower Bound HEP Values	
	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF
POS 3 Mode 4 cooldown	2.3E-08	0.2%	2.3E-08	0.2%
POS 4 Mode 5 cooldown	1.7E-07	1.3%	1.5E-07	1.5%
POS 5A Pressurizer solid	2.0E-06	16.3%	1.2E-06	12.1%
POS 5B Draining RCS	7.5E-07	5.9%	7.1E-07	7.4%
POS 6 Mid-loop (pre-refuel)	1.6E-06	13.0%	5.2E-07	5.4%
POS 7 Filling refuel cavity	2.0E-07	1.6%	2.0E-07	2.1%
POS 8E Refuel cavity flooded (pre-refuel)	3.3E-06	26.0%	2.9E-06	30.0%
POS 8L Refuel cavity flooded (post-refuel)	2.4E-06	19.3%	2.1E-06	22.3%
POS 9 Draining refuel cavity	4.7E-07	3.7%	3.8E-07	3.9%
POS 10 Mid-loop (post-refuel)	7.2E-07	5.8%	6.7E-07	7.0%
POS 11 Vacuum fill RCS	2.7E-07	2.2%	2.1E-07	2.2%
POS 12 Mode 5 heatup	4.2E-07	3.4%	4.1E-07	4.3%
POS 13 Mode 4 heatup	1.6E-07	1.2%	1.5E-07	1.6%
Total:	1.3E-05	100%	9.6E-06	100%

Table 10-25 Upper Bound HEP Sensitivity Case CDF Results by Initiating Event Category

Initiating Event Category	Base Case CDF Results		Sensitivity with Upper Bound HEP Values	
	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF
Loss of Offsite Power	8.5E-06	68.1%	1.1E-05	42.9%
Overdrain Mid-loop (before refueling)	1.2E-06	9.8%	6.9E-06	26.7%
Loss of Inventory at pressure	7.8E-07	6.2%	4.9E-06	19.2%
Loss of RHR	6.7E-07	5.4%	1.2E-06	4.6%
Loss of Service Water	6.4E-07	5.1%	8.7E-07	3.4%
Loss of power to 4,160V AC bus	3.7E-07	3.0%	3.7E-07	1.5%
Overdrain Mid-loop (after refueling)	2.3E-07	1.8%	3.3E-07	1.3%
Loss of Inventory	4.2E-08	0.3%	5.6E-08	0.2%
Loss of Inventory Displacement	2.0E-08	0.2%	2.7E-08	0.1%
Loss of Inventory Spray	1.8E-08	0.1%	3.3E-08	0.1%
Loss of Inventory Large	1.3E-10	< 0.1%	1.6E-10	0.0%
Total	1.3E-05	100%	2.6E-05	100%

Table 10-26 Upper Bound HEP Sensitivity Case CDF Results by Shutdown Plant Operating State

Shutdown POS Contribution to CDF	Base Case CDF Results		Sensitivity with Upper Bound HEP Values	
	CDF	% Contr. to CDF	CDF	% Contr. to CDF
POS 3 Mode 4 cooldown	2.3E-08	0.2%	2.5E-08	0.1%
POS 4 Mode 5 cooldown	1.7E-07	1.3%	2.2E-07	0.9%
POS 5A Pressurizer solid	2.0E-06	16.3%	6.6E-06	25.7%
POS 5B Draining RCS	7.5E-07	5.9%	8.1E-07	3.2%
POS 6 Midloop (pre-refuel)	1.6E-06	13.0%	8.1E-06	31.2%
POS 7 Filling refuel cavity	2.0E-07	1.6%	2.2E-07	0.9%
POS 8E Refuel cavity flooded (pre-refuel)	3.3E-06	26.0%	4.2E-06	16.3%
POS 8L Refuel cavity flooded (post-refuel)	2.4E-06	19.3%	3.1E-06	12.2%
POS 9 Draining refuel cavity	4.7E-07	3.7%	5.9E-07	2.3%
POS 10 Midloop (post-refuel)	7.2E-07	5.8%	9.1E-07	3.5%
POS 11 Vacuum fill RCS	2.7E-07	2.2%	3.5E-07	1.3%
POS 12 Mode 5 heat-up	4.2E-07	3.4%	4.5E-07	1.7%
POS 13 Mode 4 heat-up	1.6E-07	1.2%	1.7E-07	0.6%
Total:	1.3E-05	100%	2.6E-05	100%

10.3.9 Impact of Steam Generator Nozzle Dam Failure

As part of the mid-loop operations, described in Section 2.5, nozzle dams are installed to isolate the RCS from the steam generators. The nozzle dams are then removed during mid-loop after refueling. The nozzle dams maintain the RCS integrity during the intervening plant operating states. Failures of the nozzle dams are not modeled in the shutdown PRA. The likelihood of a nozzle dam failure occurring is a source of model uncertainty. There is a lack of widely available operating experience data on nozzle dam failures and uncertainty about the design details and failure criteria. However, a nozzle dam failure could result in a large opening in the RCS and loss of inventory.

The potential for nozzle dam failure is discussed in Section 5.3.5. A potential scenario is discussed in which RCS pressure increases related to pumped injection with ECCS pumps can challenge the nozzle dam integrity. Pumped injection is relied upon in many of the modeled accident sequences to provide alternate core cooling when RHR cooling is lost. The significant accident sequences were reviewed to identify those sequences with successful pumped injection for POSs with the nozzle dams installed (see Section 9.4 for significant accident sequences). Those sequences were then adjusted to include a nozzle dam failure and a failure to recover the nozzle dam integrity. The failure probability of the nozzle dams under pumped injection conditions is unknown, so failure probabilities were assumed to assess the impact on the shutdown CDF. Two sensitivity cases were developed assuming a probability of nozzle dam failure after pumped injection of 0.5 and 0.1, respectively. If a nozzle dam failure does occur, then it may be possible that the dam could be reinstalled or replaced to restore RCS integrity. However, the inventory loss and imminent boiling would make any possible recovery actions challenging. Therefore, no recovery from the nozzle dam failure is credited in these sensitivity studies.

The CDF results by POS are shown in Table 10-27. The impacted POSs are POS 6 when nozzle dams are installed, POS 10 when nozzle dams are removed, and all POSs occurring in between. The impacts on the shutdown CDF results are substantial. Assuming a non-recoverable failure probability of 0.5 would increase the shutdown CDF to 4.8×10^{-5} per year, and a non-recoverable failure probability of 0.1 would result in a shutdown CDF of 2.0×10^{-5} per year. The potential increases in CDF are primarily associated with the mid-loop POSs, POS 6 and POS 10. In both cases, about 87 percent of the CDF increase is associated with POS 6 and about 9 percent of the CDF increase is associated with POS 10. It should be recognized that for POS 10 and the other post-refueling POSs, the likelihood of nozzle dam failure is much less because injection flow from a single ECCS pump is sufficient to maintain RCS temperature below 200 °F. The peak RCS pressure from single pump operation is not expected to challenge the nozzle dam integrity. Still, even if the post-refueling CDF contributions are ignored, the potential CDF contributions from pre-refueling mid-loop nozzle dam failures are significant.

Table 10-27 Nozzle Dam Failure Sensitivity CDF Results by Shutdown Plant Operating State

Shutdown POS Contribution to CDF	Base Case CDF Results		Sensitivity 0.5 Nozzle Dam Failure Prob.		Sensitivity 0.1 Nozzle Dam Failure Prob.	
	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF
POS 3 Mode 4 cooldown	2.3E-08	0.2%	2.3E-08	< 0.1%	2.3E-08	0.1%
POS 4 Mode 5 cooldown	1.7E-07	1.3%	1.7E-07	0.4%	1.7E-07	0.9%
POS 5A Pressurizer solid	2.0E-06	16.3%	2.0E-06	4.3%	2.0E-06	10.5%
POS 5B Draining RCS	7.5E-07	5.9%	7.5E-07	1.6%	7.5E-07	3.8%
POS 6 Mid-loop (pre-refuel)	1.6E-06	13.0%	3.2E-05	67.9%	7.8E-06	39.7%
POS 7 Filling refuel cavity	2.0E-07	1.6%	2.0E-07	0.4%	2.0E-07	1.0%
POS 8E Refuel cavity flooded (pre-refuel)	3.3E-06	26.0%	3.3E-06	7.0%	3.3E-06	16.7%
POS 8L Refuel cavity flooded (post-refuel)	2.4E-06	19.3%	3.5E-06	7.4%	2.6E-06	13.5%
POS 9 Draining refuel cavity	4.7E-07	3.7%	5.2E-07	1.1%	4.8E-07	2.4%
POS 10 Mid-loop (post-refuel)	7.2E-07	5.8%	3.9E-06	8.2%	1.4E-06	6.9%
POS 11 Vacuum fill RCS	2.7E-07	2.2%	2.7E-07	0.6%	2.7E-07	1.4%
POS 12 Mode 5 heatup	4.2E-07	3.4%	4.2E-07	0.9%	4.2E-07	2.2%
POS 13 Mode 4 heatup	1.6E-07	1.2%	1.6E-07	0.3%	1.6E-07	0.8%
Total:	1.3E-05	100%	4.8E-05	100%	2.0E-05	100%

10.3.10 Equipment Unavailability for Maintenance and Testing

Equipment unavailability for maintenance and testing activities has a significant impact on the shutdown CDF results and is a source of model uncertainty. The status of equipment, including a detailed analysis of the timing and duration of unavailabilities, is an important input into a shutdown PRA. The outage modeled for the L3PRA project is not based on an actual historical outage, but it is informed by past outages at the reference plant. Assumptions were made about the unavailability of major equipment and electrical support systems. These assumptions are summarized in Section 8.3.

The primary periods of equipment unavailability are during POS 4 and POS 5A, and from POS 8E to POS 8L when the refueling cavity is flooded. Sensitivity cases were developed for each of these periods to show the impacts of reducing or extending the duration of the maintenance activities that occur during these POSs.

The activities during POSs 4 and 5A include maintenance on vital AC and DC power systems and the safety features sequencer system. The maintenance and testing activities are performed prior to entering reduced inventory and mid-loop operations, where equipment is protected, and work controls limit the maintenance activities being performed. The first sensitivity case assumes the maintenance activities during POSs 4 and 5A are not performed at

this time in the outage or defer to another time. The second case considers an extended maintenance period due to unexpected delays or emergent issues, which extends the duration of the POSs. Both POS 4 and POS 5A are assumed to be twice as long as the originally modeled outage. The CDF results by POS are shown in Table 10-28. The no maintenance case significantly reduces the contribution of POS 5A to the overall shutdown CDF with the reduction primarily associated with reduced contribution from loss of offsite power events (overall CDF is reduced by approximately 10 percent). The extended maintenance case shows a significant increase to the POS 5A contribution (overall CDF is increased by approximately 17 percent). Under this assumption POS 5A has the largest CDF contribution of any POS.

The sensitivity cases for POS 8E and POS 8L are based on adjustments to the POS durations. The period with refueling cavity flooded is the primary window for maintenance activities. However, the exposure time could be reduced or extended depending on the timing of when the activities are performed and when the fuel is offloaded and reloaded into the reactor. The first sensitivity case considers a reduced maintenance period with the duration of POS 8E and POS 8L reduced by a factor of two, and the second sensitivity case considers an extended maintenance period with the duration twice as long. The CDF results by POS for the POS 8E and 8L sensitivity cases are shown in Table 10-29. As expected, the duration of the maintenance period has a significant influence on the CDF contribution from POS 8E and POS 8L with the combined CDF contribution ranging from 29 percent to 62 percent of the overall outage CDF. In the reduced maintenance case, overall CDF is reduced by approximately 23 percent; in the extended maintenance case, overall CDF is increased by approximately 45 percent.

Table 10-28 POS 4 and POS 5A Maintenance Sensitivity CDF Results by Shutdown Plant Operating State

Shutdown POS Contribution to CDF	Base Case CDF Results		Sensitivity A: No Maintenance		Sensitivity B: Maintenance Duration Doubled	
	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF
POS 3 Mode 4 cooldown	2.3E-08	0.2%	2.3E-08	0.2%	2.3E-08	0.2%
POS 4 Mode 5 cooldown	1.7E-07	1.3%	3.1E-08	0.3%	3.4E-07	2.3%
POS 5A Pressurizer solid	2.0E-06	16.3%	1.0E-06	8.9%	4.1E-06	27.8%
POS 5B Draining RCS	7.5E-07	5.9%	7.2E-07	6.4%	7.2E-07	4.9%
POS 6 Mid-loop (pre-refuel)	1.6E-06	13.0%	1.6E-06	14.4%	1.6E-06	11.1%
POS 7 Filling refuel cavity	2.0E-07	1.6%	2.0E-07	1.8%	2.0E-07	1.4%
POS 8E Refuel cavity flooded (pre-refuel)	3.3E-06	26.0%	3.3E-06	28.8%	3.3E-06	22.2%
POS 8L Refuel cavity flooded (post-refuel)	2.4E-06	19.3%	2.4E-06	21.4%	2.4E-06	16.5%
POS 9 Draining refuel cavity	4.7E-07	3.7%	4.5E-07	4.0%	4.5E-07	3.1%
POS 10 Mid-loop (post-refuel)	7.2E-07	5.8%	7.1E-07	6.3%	7.1E-07	4.8%
POS 11 Vacuum fill RCS	2.7E-07	2.2%	2.7E-07	2.4%	2.7E-07	1.9%
POS 12 Mode 5 heatup	4.2E-07	3.4%	4.2E-07	3.7%	4.2E-07	2.9%
POS 13 Mode 4 heatup	1.6E-07	1.2%	1.6E-07	1.4%	1.6E-07	1.1%
Total:	1.3E-05	100%	1.1E-05	100%	1.5E-05	100%

Table 10-29 POS 8E and POS 8L Maintenance Sensitivity CDF Results by Shutdown Plant Operating State

Shutdown POS Contribution to CDF	Base Case CDF Results		Sensitivity C: Maintenance Duration Halved		Sensitivity D: Maintenance Duration Doubled	
	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF	CDF (per year)	% Contr. to CDF
POS 3 Mode 4 cooldown	2.3E-08	0.2%	2.3E-08	0.2%	2.3E-08	0.1%
POS 4 Mode 5 cooldown	1.7E-07	1.3%	1.7E-07	1.7%	1.7E-07	0.9%
POS 5A Pressurizer solid	2.0E-06	16.3%	2.0E-06	21.1%	2.0E-06	11.2%
POS 5B Draining RCS	7.5E-07	5.9%	7.5E-07	7.7%	7.5E-07	4.1%
POS 6 Mid-loop (pre-refuel)	1.6E-06	13.0%	1.6E-06	16.8%	1.6E-06	9.0%
POS 7 Filling refuel cavity	2.0E-07	1.6%	2.0E-07	2.1%	2.0E-07	1.1%
POS 8E Refuel cavity flooded (pre-refuel)	3.3E-06	26.0%	1.6E-06	16.8%	6.5E-06	35.8%
POS 8L Refuel cavity flooded (post-refuel)	2.4E-06	19.3%	1.2E-06	12.5%	4.8E-06	26.6%
POS 9 Draining refuel cavity	4.7E-07	3.7%	4.7E-07	4.8%	4.7E-07	2.6%
POS 10 Mid-loop (post-refuel)	7.2E-07	5.8%	7.2E-07	7.4%	7.2E-07	4.0%
POS 11 Vacuum fill RCS	2.7E-07	2.2%	2.7E-07	2.8%	2.7E-07	1.5%
POS 12 Mode 5 heatup	4.2E-07	3.4%	4.2E-07	4.3%	4.2E-07	2.3%
POS 13 Mode 4 heatup	1.6E-07	1.2%	1.6E-07	1.6%	1.6E-07	0.9%
Total:	1.3E-05	100%	9.7E-06	100%	1.8E-05	100%

10.3.11 Summary of Model Uncertainty and Sensitivity Studies

The sources of model uncertainty were assessed to understand the potential impacts on CDF results from varying model assumptions and approaches. A complete review of sources of model uncertainty is contained in Appendix C which includes a qualitative characterization of each source of model uncertainty. The sensitivity studies in the preceding sections were developed to assess the impacts of important model uncertainties that were selected based on reviewer feedback from the project Technical Advisory Group (TAG), feedback during ACRS briefings, and consideration of the expected impact on the PRA results. The sensitivity study results are summarized in Table 10-30. The CDF results of the sensitivity studies are also shown in Figure 10-3. Figure 10-4 shows the percent change in CDF for each sensitivity study compared to the base CDF result.

Table 10-30 Summary of Model Uncertainty Sensitivity Studies

Sensitivity Study	CDF (per year)	CDF % Change from Base Case	Sensitivity Study Description
Base case	1.3E-05		CDF result from the base model for POT-4 and shutdown POSs (Section 9)
1	1.3E-05	6%	Estimated CDF contributions from unmodeled POT-4 scenarios (Section 10.3.1)
2	1.4E-05	13%	Estimated CDF contributions from POT-4 low power and hot standby POSs (Section 10.3.2)
3A	1.6E-05	29%	Estimated CDF contributions from all plant outage types (Section 10.3.3)
3B	1.7E-05	33%	Alternative assessment of POT-3 drained maintenance outage with mid-loop entry (Section 10.3.3)
4	1.7E-05	35%	Alternative assessment of POT-2 hot standby outage with diesel generator unavailable for maintenance (Section 10.3.4)
5A	5.4E-06	-57%	Re-evaluation of loss of offsite power using frequencies for at-power operations (Section 10.3.5)
5B	9.4E-06	-25%	Re-evaluation of loss of offsite power applying shutdown frequencies to only POSs 8E and 8L (Section 10.3.5)
5C	1.9E-05	52%	Re-evaluation of loss of offsite power applying increased shutdown frequencies to only POSs 8E and 8L (Section 10.3.5)
6A	1.3E-05	5%	Re-evaluation of mid-loop overdrain probability considering additional operating experience event (Section 10.3.6)
6B	1.2E-05	-3%	Re-evaluation of mid-loop overdrain probability considering recent operating years with low estimate of mid-loop entries (Section 10.3.6)
6C	1.2E-05	-5%	Re-evaluation of mid-loop overdrain probability considering recent operating years with high estimate of mid-loop entries (Section 10.3.6)
6D	1.1E-05	-9%	Re-evaluation of mid-loop overdrain probability considering recent operating years with non-informative prior distribution (Section 10.3.6)
7	8.7E-06	-31%	Consider extended time for possible SBO recovery actions during POSs with refueling cavity flooded (Section 10.3.7)
8A	9.6E-06	-24%	Significant HFEs quantified with lower bound HEP estimates (Section 10.3.8)
8B	2.6E-05	106%	Significant HFEs quantified with upper bound HEP estimates (Section 10.3.8)
9A	4.8E-05	280%	Consideration of possible steam generator nozzle dam failure with 0.5 failure probability (Section 10.3.9)
9B	2.0E-05	56%	Consideration of possible steam generator nozzle dam failure with 0.1 failure probability (Section 10.3.9)
10A	1.1E-05	-10%	Re-evaluation of POSs 4 and 5A assuming no significant maintenance activities (Section 10.3.10)
10B	1.5E-05	17%	Re-evaluation of POSs 4 and 5A assuming extended maintenance duration (Section 10.3.10)
10C	9.7E-06	-23%	Re-evaluation of POSs 8E and 8L assuming reduced maintenance duration (Section 10.3.10)

Table 10-30 Summary of Model Uncertainty Sensitivity Studies (cont.)

Sensitivity Study	CDF (per year)	CDF % Change from Base Case	Sensitivity Study Description
10D	1.8E-05	45%	Re-evaluation of POSs 8E and 8L assuming extended maintenance duration (Section 10.3.10)

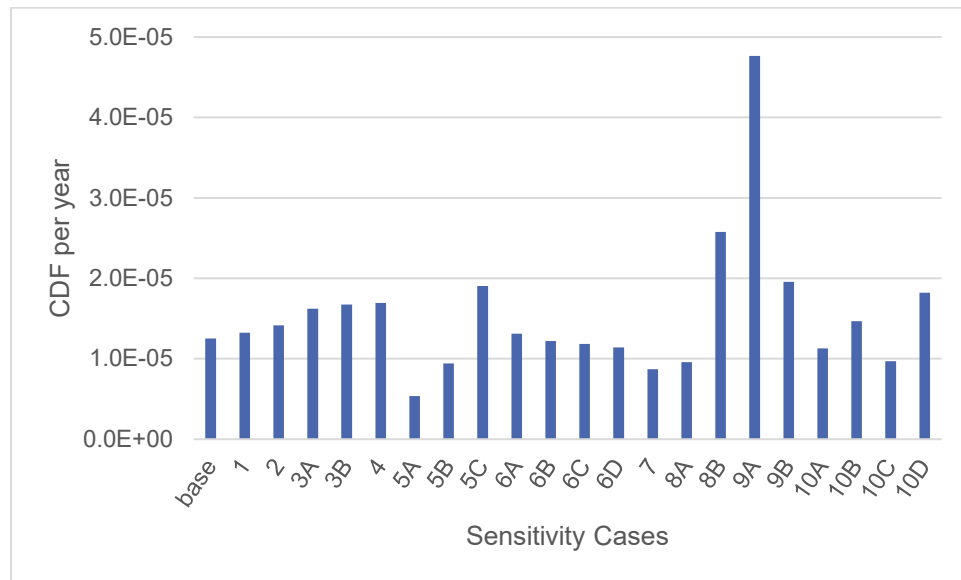


Figure 10-3 CDF results of Sensitivity Studies

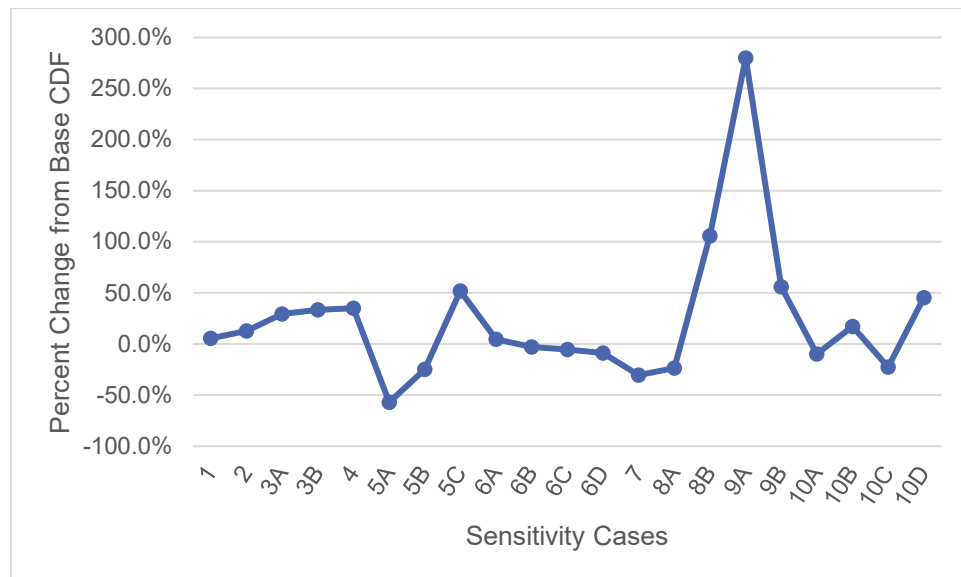


Figure 10-4 Sensitivity Studies Percent Change in CDF Compared to Base Case Result

Reviewing the sensitivity study results shows the largest impact on shutdown CDF is related to the potential for nozzle dam failure. Assuming a failure probability of 0.5 for sequences with pumped injection results in an increase in CDF of nearly a factor of three (case 9A). An assumed failure probability of 0.1 also has a significant impact on the CDF results with an increase of 56 percent (case 9B). No definitive references for the nozzle dam failure criteria were identified for this project, and this is considered a significant model uncertainty. Additional study on the possible mechanisms for nozzle dam failures is suggested as a topic for future work.

The next highest change in CDF is the sensitivity study related to uncertainty in human error probabilities. Increasing the HEPs to their upper bound (95th percentile) estimates for significant HFEs results in a CDF value that is twice the base CDF (case 8B). Human failure events are important contributors to several shutdown accident sequences, so it is expected that varying the HEP values would have a significant impact on the CDF results.

Other sensitivity studies that show significant increase in the CDF include alternate loss of offsite power initiating event frequencies for POSs 8E and 8L (case 5C) and extending the maintenance duration during POSs 8E and 8L (case 10D). The loss of offsite power CDF for POSs 8E and 8L are both significant contributors to the overall CDF, and the assumptions about the initiating event frequency and maintenance unavailabilities are important model uncertainties.

The sensitivity study showing the largest decrease in CDF is also related to loss of offsite power frequencies. Using the initiating event frequency for at-power operation results in a 57 percent reduction in CDF (case 5A). The next largest decrease in CDF is related to assumptions about recovering from loss of offsite power events during POSs with the refueling cavity partially or fully flooded (case 7). Although the accident scenarios progress slowly under these conditions, the time available for restoring power is limited by the plant batteries required to realign offsite power to the safety-related buses. However, if actions could be taken to extend the time available for power recovery, then this could significantly reduce CDF.³⁷ Other significant reductions in shutdown CDF are related to alternate assumptions on the loss of offsite power frequencies (case 5B) and using lower bound (5th percentile) HEP values for significant human failure events (case 8A).

The base case PRA model focuses only on shutdown POSs and a refueling outage (i.e., POT-4). Even within the refueling outage, not all POSs and accident scenarios are modeled. A prioritization scheme was developed to focus on the scenarios expected to be most risk-significant. Some initial prioritization is practically needed when considering the large number of possible plant configurations and accident scenarios that can be experienced in the different outage types. Sensitivity cases 1 through 4 describe the approaches for estimating CDF contributions from the unmodeled shutdown accident scenarios, low power and hot standby POSs, and other outage types. The estimated CDF contributions are summarized in Table 10-31.

The estimated refueling outage CDF accounts for 87 percent of the total estimated CDF for all low power and shutdown outages. The modeled shutdown accident scenarios account for about

³⁷ Note that this study is based on information that reflects the reference plant as it was designed and operated as of 2012. It does not reflect the potential impact of FLEX strategies implemented after 2012. The impacts of FLEX strategies are explored in the Level 3 PRA project overview report on the shutdown Level 1, 2, and 3 PRAs.

95 percent of the shutdown risk for refueling outages (see Table 10-5). The estimated CDF values suggest that there may be a small number of potentially significant scenarios that were not modeled, but most of the shutdown risk is represented by the modeled scenario results discussed in Section 9. While the model results focus on the shutdown risk from refueling outages, the estimated CDF values for all outage types and operating states provide measures of the complete low power and shutdown risk. Also, note that the CDF estimates are based on simplifying assumptions and may be bounding estimates compared to more realistic modeling results. In considering the completeness of the model scope, it is worthwhile to note that completeness is an important aspect of the results, but there are other modeling assumptions that can have larger impacts on the CDF results, as demonstrated by the sensitivity studies.

Table 10-31 Estimated CDF Contribution from All Plant Outage Types

Low Power and Shutdown CDF Contribution	CDF (per year)	CDF % Contr.
Non-drained maintenance outage without RHR (POT-1) CDF	8.7E-07	5.3%
Non-drained maintenance outage with RHR (POT-2) CDF	6.0E-07	3.7%
Drained maintenance outage with RHR (POT-3) CDF	6.0E-07	3.7%
Refueling outage (POT-4) CDF	1.4E-05	87.2%
Total shutdown CDF all modeled scenarios	1.3E-05	77.3%
Estimated shutdown CDF unmodeled scenarios	7.1E-07	4.4%
Estimated low power / hot standby CDF unmodeled scenarios	9.0E-07	5.5%
Total Low Power and Shutdown CDF	1.6E-05	100.0%

11 SUMMARY OF CONCLUSIONS

11.1 Summary of Model Scope

This section summarizes the results and conclusions from the low power and shutdown analysis. The LPSD PRA model was developed to assess the risk associated with the shutdown plant operating states of a refueling outage. A systematic approach to identify plant outage types, operating states, and low power and shutdown initiating events was performed. Six categories of shutdown initiating events were identified for quantitative modeling: loss of offsite power, loss of RHR cooling, loss of inventory, loss of power to a 4,160V AC safety-related bus, overdraining entering mid-loop, and loss of nuclear service cooling water. Other outage types, plant operating states, and initiating event categories were assessed qualitatively.

The total shutdown CDF is estimated to be 1.3×10^{-5} per year for the modeled refueling outage. The CDF values are presented as frequencies on a per year basis, which provides for a comparison to the CDF results from other parts of the L3PRA project (e.g., CDF for at-power operation). However, refueling outages are planned and occur at known times. The CDF contribution is concentrated during the time of scheduled outage. The average shutdown CDF can either overestimate or underestimate the CDF depending on if the period being assessed includes a scheduled outage or not.

A series of sensitivity studies (see Sections 10.3.1 through 10.3.3) provide estimates of the CDF contributions from the outage types, operating states and initiating event scenarios that were not included in the quantitative model. Three types of outage types were assessed: non-drained maintenance outage without RHR cooling, non-drained maintenance outage with RHR cooling, and drained maintenance outage with RHR cooling. Each of the outage types are estimated to contribute approximately 4–5 percent of the total low power and shutdown CDF. For refueling outages, the sensitivity study results show that unmodeled low power and hot standby POSs contribute approximately 5 percent of the total low power and shutdown CDF. The unmodeled initiating event scenarios for a refueling outage are estimated to contribute 4 percent of the total low power and shutdown CDF. The sensitivity studies estimate that collectively these unmodeled portions account for approximately 23 percent of the total low power and shutdown CDF. (See Table 10-31 for a summary of estimates for unmodeled low power and shutdown CDF contributions.) While these estimates are based on simplified and bounding assumptions, they do provide a sense of the relative contributions to overall low power and shutdown risk. The sensitivity studies also provide a guide for considering development of additional outage models for future work.

The quantified PRA model represents a refueling outage that is informed by past operating experience on outage frequency, duration, maintenance activities, and plant states. As discussed above, other outage types are not included in the model. The quantified PRA results include the significant contributors to refueling outage shutdown operations. The plant response for all shutdown plant operating states (POSs 3 through 13) are modeled.

Insights from the quantified model results are discussed in Section 11.2. Section 11.3 summarizes key modeling aspects that are important to carry forward to the Level 2 PRA. Section 11.4 provides discussion of potential areas for future work related to low power and shutdown PRA.

11.2 Summary of Key Results

The highest contributing initiating event category to the shutdown CDF is loss of offsite power. The significant contribution from losses of offsite power is similar to the results from the L3PRA project at-power, internal events PRA (NRC, 2022). The loss of offsite power initiating event frequency for shutdown operation is higher than the corresponding frequency for at-power operation, which contributes to the overall significance of the CDF result. The modeling assumptions regarding the recovery of offsite power also influence the CDF results. The time available for recovery is based on plant-specific battery depletion times, plant procedures, and the DC power requirements for aligning AC power sources. If plant procedures, equipment, and strategies were available to extend the coping time for station blackout sequences, then the CDF could be significantly reduced. The implementation of FLEX mitigating strategies is expected to have a positive impact in this respect; however, the FLEX strategies are not included within the scope of this model.³⁸

The next highest contributing initiating event category is overdrain entering mid-loop prior to refueling. Mid-loop operation is generally recognized as a high-risk evolution in qualitative risk assessments. The short time available until RCS boiling occurs upon loss of cooling and the reliance on operator actions to diagnose and respond to the event in a short timeframe are challenges that contribute to the risk significance of overdrain scenarios. Human failure events are dominant contributors to the overdrain and other mid-loop accident scenarios, as demonstrated by the importance measure results (see Section 9.6).

The loss of inventory with RCS at pressure is also a significant initiating event category. The initiating event models the loss of RCS inventory due to an over-pressure condition causing relief valves to open. Although this event is only applicable during POS conditions with the RCS intact, it has a significant contribution to the overall shutdown CDF. The relatively high initiating event frequency and significant human failure events are important contributors to this scenario.

The shutdown CDF was also parsed into contribution by plant operating state (see Section 9.3). It is noteworthy that a significant portion of the shutdown CDF (80 percent) occurs during POSs when the containment equipment hatch is initially open at the time of the initiating event occurrence. While actions can be taken to close containment (these actions are considered in the Level 2 PRA), the initial containment status is an important attribute to consider in assessing the potential for radiological releases.

The highest contributing plant operating states in terms of CDF are POS 8E and POS 8L, which represent operation with the refueling cavity flooded. While the accident scenarios in these states progress slowly due to the large water inventory available in the refueling cavity, the unavailability of equipment for maintenance is an important factor influencing the risk in these states. In POSs 8E and 8L, one reserve auxiliary transformer and the associated 4,160V AC safety related bus (and its associated emergency diesel generator) are assumed out of service for maintenance. These unavailabilities have a significant impact on the CDF results and, in particular, on the loss of offsite power results. Given the slow developing nature of the accident

³⁸ Note that this study is based on information that reflects the reference plant as it was designed and operated as of 2012. It does not reflect the potential impact of FLEX strategies implemented after 2012. The impacts of FLEX strategies are explored in the Level 3 PRA project overview report on the shutdown Level 1, 2, and 3 PRAs.

scenarios, improved coping strategies to recover from losses of offsite power could significantly reduce the CDF contributions.

The next highest contributing plant operating state is POS 5A, operation with pressurizer solid. The loss of inventory with RCS at pressure is a significant contributor to this POS. The assumptions about maintenance unavailability also influence the POS 5A results. In POS 5A the automatic actuation of one EDG is assumed unavailable due to maintenance and testing on the safety features sequencer.

Mid-loop operations prior to refueling, POS 6, are a significant contributor to the shutdown CDF. The short time available before RCS boiling occurs is an important factor contributing to the POS 6 CDF result. While operators may be able to delay the onset of core damage by establishing injection with ECCS pumps, the injection inventory would be depleted after several hours and restoration of RHR cooling would be required to reach a safe and stable state. The CDF contributions from mid-loop operations can be reduced if mid-loop entry is delayed until later in the outage or avoiding mid-loop entry until fuel is removed from the reactor. However, delaying or avoiding mid-loop could extend the overall outage time depending on what activities are planned for the outage.

The loss of RHR results demonstrate the importance of having a redundant system for providing decay heat removal, as the POSs with steam generator heat removal available (POSs 3, 4, 5A, 12, and 13) have much lower contributions to CDF than the POSs that rely solely on RHR for heat removal. The results also show that the highest loss of RHR contributors are those with a short time available for responding to the event. These are the POSs with reduced inventory (POS 5B and POS 9) and mid-loop conditions (POS 6 and POS 10).

The loss of power to the 4,160V AC bus events have the highest shutdown initiating event frequency in comparison to other initiating event categories. However, the plant impacts are not as severe as the other initiating event categories. Two key assumptions are made about these events: (1) the loss of power to 4,160V AC bus event only impacts one safety-related train, and (2) the impacted bus can be re-energized by starting and aligning the associated EDG. These assumptions are consistent with the recent operating experience for these events, as reported in (EPRI, 2010).

Several different loss of inventory events are considered in the shutdown CDF model. While the losses of inventory with RCS at pressure are significant contributors to CDF, the other loss of inventory categories do not contribute significantly. One of the primary reasons for the relative importance is the difference in initiating event frequencies. The losses of inventory related to piping failures have much lower initiating event frequencies. A review of the recent operating experience (EPRI, 2010) and (EPRI, 2001) shows that there have not been any recent occurrences of significant inventory losses other than those related to lifting relief valves. However, operating experience should be continually monitored to ensure there are no adverse trends in loss of inventory occurrences. The CCDP results show that these can be important risk contributors, but their contributions are limited by the relatively low initiating event frequencies.

The total shutdown CDF of 1.3×10^{-5} per year is less than the total at-power CDF for internal events of approximately 6.0×10^{-5} per year (based on 6.4×10^{-5} per reactor critical year, as documented in (NRC, 2022), and scaled by the plant availability factor of 0.93, as shown in Table 9-2). While the shutdown CDF is nominally less than the at-power CDF when considered on an average per year basis, it is important to remember that the shutdown risk is concentrated

during a much shorter time period (about 20 days assuming a 30-day refueling outage every 18 months) compared to full power operation (93 percent of a calendar year or about 340 days). When considering the relative durations of shutdown and at-power operation, the core damage **probability** is more than three times higher during the modeled shutdown refueling outage than at-power operation. Reporting the average CDF per year also obscures the variation of outage time from year to year. If considering a year where a refueling outage occurs, then the shutdown CDF can be 1.5 times higher than the average CDF per year. Similarly, the plant availability for full power operation can vary from about 90 percent to more than 98 percent for a year with or without a refueling outage, respectively. These considerations are presented to provide some perspective on the comparisons of shutdown and full power CDF and considering average versus conditional risk measures. However, caution should be exercised when comparing risk results from different parts of the L3PRA project. Differences in scope and key assumptions influence the results, and these differences should be considered when comparing results.

11.3 Comments on Level 2 Analysis

The shutdown Level 1 PRA model is to be used as input to the shutdown Level 2 PRA leading to estimation of fission product release category frequencies. In this section, some characteristics of the Level 1 PRA model that can be of interest for the Level 2 PRA are summarized:

- The containment status is an important consideration in the Level 2 PRA. There may be containment openings during many of the shutdown POSs. Refer to the results and discussion in Section 9.3 and Table 9-5, which shows the shutdown CDF results with the initial containment status for each POS.
- Many factors related to the reactor status and containment systems can influence the potential for radiological releases. These include the RCS status (e.g., vented or head removed) and availability and status of containment sprays, containment cooling units, containment purge ventilation system, and containment sump. Refer to Section 4.4 for the assumptions regarding the status of the reactor and containment systems for each modeled POS.
- Interfacing systems loss-of-coolant accidents (ISLOCAs) represent containment bypass scenarios and are typically important contributors in the Level 2 PRA. The ISLOCA scenarios during shutdown can be different than those modeled for full power operation. The term loss of inventory outside containment is used to describe the containment bypass scenarios modeled in the shutdown PRA. Section 8.2 describes the modeling of losses of inventory outside containment and includes the estimates of the fraction of events occurring outside containment for inventory losses during POSs with the RCS pressurized (Table 8-23) and POSs with the RCS vented (Table 8-24).
- Shutdown accident scenarios for post-refueling POSs may develop slowly due to the reduced decay heat level after removal and replacement of a portion of the spent core. Approximately 36 percent of the shutdown CDF is attributed to post-refueling POSs. While these scenarios meet the surrogate criterion for core damage in the Level 1 PRA, the Level 2 PRA accident progress analysis may show that core damage can be arrested or averted for some post-refueling scenarios. The time available for mitigating actions may reduce challenges to containment making releases less likely for some post-refueling scenarios.

- The mission times used in the shutdown Level 1 PRA are limited to 24 hours. If the Level 2 PRA requires longer time periods to fully develop severe accident sequences, then the effect of potential longer mission times should either be modeled or at least assessed.

11.4 Candidates for Future Work

The L3PRA provides a comprehensive assessment of low power and shutdown risks; however, there are several areas that could benefit from additional study. These areas are offered as potential improvements that can be pursued in future revisions or follow-on work to this project. A complete list of candidate issues for future work is provided in Appendix A. The discussion below highlights some of the primary areas where model improvements could be made.

- Initiating event categories and frequency estimates: Operating experience data for many shutdown initiating events are not routinely collected and analyzed. Additional study and establishing an on-going shutdown initiating event data collection and analysis program could improve initiating event frequency estimates. In particular, loss of offsite power frequency estimates have a significant impact on the CDF results.
- Model scope: The qualitative analysis in this project considers the complete range of outage types and plant operating states. However, the quantitative model focuses on a refueling outage and shutdown POSs. Although other outage types and POSs are addressed in sensitivity studies, additional modeling of other outage types and POSs is an area for improvement.
- Steam generator nozzle dam failures: As shown in a sensitivity study (see Section 10.3.9), nozzle dam failure can significantly impact the PRA results. However, there is a lack of widely available operating experience data on nozzle dam failures and uncertainty about the design details and failure criteria.
- Human failure events and human error probability estimates: As part of this project, the HFEs underwent several reviews and reevaluations. Yet considering the important influence these events can have on the CDF results, additional study and evaluation with alternate HRA methods could help to benefit future low power and shutdown PRAs.

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APPENDIX A CANDIDATES FOR FUTURE WORK

The Level 3 Probabilistic Risk Assessment (PRA) project (L3PRA project) provides a comprehensive assessment of low power and shutdown risks; however, there are several areas that could benefit from additional study. These areas are offered as potential improvements that can be pursued in future revisions or follow-on work to this study. The discussion below highlights some of the key areas where model improvements are expected to have significant impacts on the estimated core damage frequency (CDF) results. Table A-1 shows a more complete list of areas that could be the focus of future work.

Initiating Event Frequencies: Operating experience data for many shutdown initiating events are not routinely collected and analyzed. The primary source of shutdown initiating event data is (EPRI, 2010). This report includes events through year 2009. The report does not include data on the shutdown exposure hours relevant to those events. Additional study is needed to develop the shutdown initiating event data to support frequencies that reflect recent shutdown operating experience. In cases where operating experience reveals that initiating event occurrences are rare or have not occurred, then alternative methods for estimating initiating event frequencies should be considered. Some methods to consider might include expert elicitation or human reliability analysis (HRA)-based methods for human-induced initiators. Low power initiating event frequencies could similarly benefit from additional study. Operating experience suggests that some initiating event frequencies can be significantly different during low power operation compared to at-power. Additional effort is needed to develop consistent methods for defining low power initiating event categories and estimating low power exposure hours. Some specific improvements to shutdown initiating events are described below.

- **Loss of offsite power (LOOP) initiating event frequency:** The low power and shutdown (LPSD) PRA for the L3PRA project groups four shutdown LOOP sub-categories (plant-centered, switchyard-centered, grid-related, and weather-related) into a single initiating event category. The initiating event frequency is 1.6×10^{-1} per shutdown-year, compared to the grouped LOOP frequency of approximately 3×10^{-2} per reactor critical year for the at-power model. The LOOP rates are higher in each of the four sub-categories for shutdown events compared to power operation. The differences in LOOP frequencies could be further investigated to identify which events should be applied to frequency estimates for different plant operating states (POSS) and outage types. The LOOP frequency estimates have a significant influence on the shutdown CDF results, as shown in the sensitivity studies in Section 10.3.5.
- **Loss of residual heat removal (RHR) cooling initiating event frequency:** A total of 13 loss of RHR events were identified for pressurized-water reactor (PWR) operating experience from 2000-2009. These events were used for estimating the loss of RHR frequency for this study. The primary source of operating experience is (EPRI, 2010). Some of these losses of RHR events could be confirmed by other data sources, such as the data collection and analysis at Idaho National Laboratory and NRC licensee event reports. However, not all events could be confirmed by other sources, and in some cases the plant where the event occurred could not be identified. In addition, the types of the failures and the conditions in which they occur can be important considerations in the accident sequence modeling. For example, is the failure recoverable or not? Did the failure of the cooling function involve an RHR system failure or a support system failure? These are some of the questions that could benefit from improved data collection and

analysis. In addition, the RHR operating conditions (e.g., operating at reduced inventory or one train versus two trains in operation) can influence the initiating event frequency. Lastly, there is no comprehensive data on the operating hours in these different conditions.

- **Loss of inventory initiating event frequency:** There is limited loss of inventory initiating event data available since the year 2000. The last published loss of inventory initiating event frequencies (EPRI, 2001) are based on data prior to 2000. The losses are categorized into small draindowns (less than 10,000 gallons lost) and large draindowns (greater than 10,000 gallons lost). The more recent operating experience identified in (EPRI, 2010) shows that there have been no recent large draindowns. Also, the types of failures resulting in losses of inventory are different from past experience. Nearly all the inventory losses from year 2000 and later are related to pressure transients and lifting of RHR relief valves. The loss of inventory initiating events could be improved by redefining the types of failures that can occur, including multiple leak locations, leak rates and inadvertent draining due to operator errors, and considering the plant-specific conditions that can influence the frequencies.

Initiating Event Categories: The LPSD PRA model includes six shutdown initiating event categories. The initiating event identification and operating experience reviews highlight the many ways initiating events can occur. The initiating event categories are defined to subsume several sub-categories, which simplifies the modeling approach. In some cases, the model could be improved by separating the sub-categories into unique initiating events. The LORHR initiating event is defined to include RHR system failures, loss of cooling to the RHR heat exchangers due to component cooling water failures, and inadvertent isolation. The initiating event frequencies and plant response could be improved by tailoring them to each LORHR sub-category. The loss of inventory initiating event also could be improved by dividing into different categories of pipe break sizes, locations, and other sources of inventory loss.

Similar issues exist in defining the distinctions between at-power, low power, hot standby, and shutdown initiating event categories. As discussed in Section 4.2.1, low power and hot standby initiating event frequencies could be significantly higher than corresponding at-power frequencies. However, improvements in the categorization of initiating events and accounting for exposure times of plant operating states are needed to support proper frequency estimates. An established state-of-practice exists for at-power initiating events. However, there is less experience with developing low power, hot standby, and shutdown initiating events. The EPRI reports on shutdown operating experience (EPRI, 2001) and (EPRI, 2010) provide useful reference points for categorizing shutdown initiating events, but these reports do not reflect the most recent experience with shutdown events. There may be a need to redefine initiating event categories, or in some cases further sub-categorize them, to support more detailed modeling of initiating event causes and plant conditions.

Human Failure Events and Human Error Probability Estimates: Many of the dominant contributing events to shutdown CDF are human failure events (HFEs). Important HFEs include failures related to responding and recovering from LOOPs, failures to restore RHR cooling, and failures related to initiating events occurring during mid-loop operations with short times available for operator response. Any assessment of HFEs and human error probability (HEP) estimates can vary from analyst to analyst depending on the assumptions and interpretation of plant information, procedures, training, timing evaluation, and other factors. As part of this study, the HFEs underwent several reviews and reevaluations. Yet considering the important influence

these events can have on the CDF results, additional study and evaluation with alternate HRA methods could help to benefit future low power and shutdown PRAs.

Long Term Recovery Actions: Additional study of recovery actions for slowly developing shutdown accident scenarios (e.g., conditions with refueling cavity flooded) could provide additional risk insights and inform best practices. The time available for recovering from LOOPs is a significant modeling uncertainty. The time available for recovery is based on plant-specific battery depletion times, plant procedures, and the DC power requirements for aligning AC power sources. Improved procedural guidelines, strategies, and equipment could significantly reduce the CDF contribution from LOOPs, as demonstrated in the sensitivity studies in Section 10.3.7. The implementation of FLEX mitigating strategies is expected to improve the plant's capability to address extended losses of AC power. The impacts of the FLEX strategies are explored in the Level 3 PRA project overview report on the shutdown Level 1, 2, and 3 PRAs.

Model Scope: The LPSD model is developed to assess the risks of a refueling outage during shutdown POSs. The model does not include all combinations of POSs and initiating event categories. A prioritization scheme was used to focus the model development on those accident scenarios deemed most important to risk. Additional scenarios applicable to a refueling outage could be added in future revisions to the model. Additional outage types could also be modeled. These unmodeled areas are addressed through sensitivity studies (Sections 10.3.1 through 10.3.3). Although the POSs entered during other outage types are similar to those entered during a refueling outage, a more detailed modeling approach would support the unique conditions or unavailabilities that impact the plant response.

Potential for Steam Generator Nozzle Dam Failure: The use of steam generator nozzle dams is an important part of outage operations to support access for surveillance and maintenance activities. Installation of the nozzle dams can introduce a potential weakness in the RCS pressure boundary. Possible failure of the nozzle dams should be addressed in the PRA. As shown in a sensitivity study (see Section 10.3.9), nozzle dam failure can have significant impacts on the PRA results. However, there is a lack of widely available operating experience data on nozzle dam failures and uncertainty about the design details and failure criteria. Possible failure scenarios for nozzle dams are an area that could benefit from additional study.

Equipment Unavailability for Maintenance and Testing: Equipment unavailability for maintenance and testing activities has a significant impact on the shutdown CDF results and is a source of model uncertainty. The status of equipment, including a detailed analysis of the timing and duration of unavailabilities, is an important input into a shutdown PRA. Assumptions were made about the unavailability of major equipment and electrical support systems, as informed by reviews of past outages at the reference plant. Improved data collection on outage-specific test and maintenance durations would improve the realism of the modeling.

Considerations for Establishing a Safe and Stable State: For this study, the criterion for reaching a safe, stable state during shutdown is re-establishing RHR cooling or steam generator cooling (for POSs where steam generators are available). Alternative methods of establishing cooling may be possible by injecting coolant from the RWST using charging or safety injection pumps or by gravity-driven injection. However, this is not credited as a means of establishing a safe, stable state because the RWST inventory will be exhausted. A series of calculations were completed to estimate the timing of injection before usable inventory is exhausted. While the injection can be sustained for several hours, the inventory is expected to be exhausted well

within the typical PRA mission time of 24 hours. While ECCS injection is not credited for establishing a safe, stable state, it is credited for extending the time available for restoration of RHR. This can be important for delaying core damage and allowing time for restoration of RHR cooling. Additional guidance, procedure directions, and training could improve the strategies for ECCS injection and ensure there is adequate time for the necessary actions for restoring a safe and stable state.

Table A-1 Candidates for Future Work

Topic Area	Proposed Area for Future Work
Plant evolutions and outage types	Complete a quantitative model for outage types other than refueling outages. In some cases where POS conditions are not significantly different than those in a refueling outage, it may be sufficient to add the POS durations to the existing refueling outage models. However, many unplanned forced outages can involve equipment failures and unavailabilities that would change the modeled plant response. Additional modeling to represent these outages and associated POSs would improve the completeness and fidelity of the LPSD PRA. Also, note that NUREG/CR-7265 (NRC, 2020a) ranked POSs during drained maintenance outages (i.e., POT-3) as high priorities in addressing LPSD risk.
Low Power and Hot Standby POSs	Complete a quantitative model for low power and hot standby POSs. A screening process could be developed to identify which POSs and initiators are expected to have significant contribution to outage risk.
Low power initiating events	Include low power and hot standby initiating event scenario models for those that are not quantitatively screened out.
Low power and shutdown initiating event frequency data	Improve the data collection and analysis for low power and shutdown initiating events and related operating hours for those event categories. This would help to better define appropriate initiating event categories and improve frequency estimates.
Shutdown loss of offsite power event frequency	This study groups four shutdown LOOP sub-categories (plant-centered, switchyard-centered, grid-related, and weather-related) into a single initiating event category. The initiating event frequency is 1.6×10^{-1} per shutdown year, compared to the grouped LOOP frequency of approximately 3×10^{-2} per reactor critical year for the at-power model. The LOOP rates are higher in each of the four sub-categories for shutdown events compared to power operation. The differences in LOOP frequencies could be further investigated to identify which events should be applied to frequency estimates for different POSs and outage types.
Loss of RHR initiating event frequency	A total of 13 loss of RHR events were identified for PWR operating experience from 2000-2009. These events were used for estimating the loss of RHR frequency for this study. The primary source of operating experience is EPRI report 1021176. Some of these loss of RHR events could be confirmed by other data sources, such as the data collection and analysis at Idaho National Laboratory and NRC Licensee Event Reports. However, not all events could be confirmed by other sources, and in some cases even the plant where the event occurred could not be identified. In addition, the types of the failures and the conditions in which they occur can be important considerations in the accident sequence modeling. For example, is the failure recoverable or not? Did the failure of the cooling function involve an RHR system failure or a support system failure? These are some of the questions that could benefit from improved data collection and analysis. In addition, the RHR operating conditions (e.g., operating at reduced inventory or one train versus two trains in operation) can influence the initiating event frequency. Lastly, there is no comprehensive data on the operating hours in these different conditions.
Loss of inventory initiating event frequency	There is limited loss of inventory initiating event data available since the year 2000. The last published loss of inventory initiating event frequencies (in EPRI report 1003113) are based on data prior to 2000. The losses are categorized into small draindowns (less than 10,000 gallons lost) and large draindowns (greater than 10,000 gallons lost). The more recent operating experience identified in EPRI report 1021176 shows that there have been no recent large draindowns. Also, the types of failures resulting in losses of

Table A-1 Candidates for Future Work (cont.)

Topic Area	Proposed Area for Future Work
	inventory are different from past experience. Nearly all the inventory losses from year 2000 and later are related to pressure transients and lifting of RHR relief valves. The loss of inventory initiating events could be improved by redefining the types of failures that can occur, including multiple leak locations and leak rates, and considering the plant-specific conditions that can influence the frequencies. Additional study could also be focused on human error-induced events that result in inadvertent draining of the RCS. Although recent operating experience does not show these types of errors to be common occurrences, such events have occurred in the past (for example, inadvertent draining RCS at Wolf Creek, licensee event report 482/1994-013) and could be potentially significant contributors to LPSD CDF.
Shutdown IE-POS combinations	Some IE-POS combinations for the modeled shutdown POSs are excluded from the model based on a qualitative approach to scenario prioritization. Modeling all IE-POS combinations, unless they meet the screening criteria defined in the trial use LPSD PRA standard supporting requirement LIE-C6 (ASME/ANS, 2014) or other appropriate quantitative screening criteria, would improve the completeness and fidelity of the LPSD PRA.
Other shutdown initiating events	Addressing other shutdown initiating events, including heavy load drops, reactivity control events, and fuel handling events, would improve the completeness and fidelity of the LPSD PRA. Qualitative analysis or quantitative screening could be used to demonstrate whether these other initiating events are expected to be significant contributors or can be screened out.
Consider impacts of internal fires, internal flooding, and other hazards	As stated in Section 1.2, the L3PRA study only considers internal events for low power and shutdown operation. The resources and level of information needed to support modeling of fire, flooding, seismic and other hazards was deemed to be beyond the scope of this study. However, as these hazard groups could be potentially important for low power and shutdown operations, additional study, either qualitative or quantitative, should be considered for future work.
LOINV scenario selection	The loss of inventory initiating event is defined very broadly to include a variety of breaks, human error-induced flow diversions, and opening of relief valves. The modeled LOINV scenarios all assume a break in the operating RHR loop. Considering other break locations and release pathways would improve the completeness and fidelity of the LPSD PRA. These other breaks could either be modeled or it could be demonstrated that they are bounded or not risk significant.
Loss of inventory outside containment and ISLOCA for shutdown operations	Consider development of a consensus method for identifying, screening, and modeling loss of inventory events that can lead to bypasses of the containment boundary. Modeling approaches are better established for ISLOCA scenarios in modeling for at-power operations, but the types of events and plant configurations that can lead to bypasses at shutdown can be much different than those that are considered for at-power operation.
Likelihood and impacts of nozzle dam failures	A series of sensitivity studies considers the possibility of a nozzle dam failure due to RCS pressure increase related to pumped injection for alternate core cooling when RHR cooling is lost. The potential impacts on the shutdown CDF results are significant. No definitive reference for the nozzle dam failure criteria were identified to support this study, and this is a significant model uncertainty.

Table A-1 Candidates for Future Work (cont.)

Topic Area	Proposed Area for Future Work
Post-refueling mid-loop modeling and containment status	The POS 10 (post-refueling mid-loop) modeling could have significant differences from its pre-refueling counterpart. Lower decay heat levels post-refueling allow longer time before boiling and more time for operator actions. The longer time to boil also can have an impact on containment status. Consistent with typical qualitative risk monitoring practices, the administrative controls on the containment status is informed by comparison of the time to boil with the time required to close containment (usually limited by closure of the equipment hatch). Recent outage reports suggest that the reference plant's practice is to close the equipment hatch during fueled mid-loop operations, both pre-refueling and post-refueling. However, a plant may choose to keep the equipment hatch open if the time to boil is significantly long enough compared to the time required to close containment, and this could be an acceptable condition according to the applicable risk monitoring procedures. Potential for entering a fueled mid-loop condition with the equipment hatch open should be further explored, as this is an important contributor to the Level 1 PRA results and an important consideration for the Level 2 PRA modeling.
Criteria for safe and stable state	For this study, the criterion for reaching a safe, stable state during shutdown is re-establishing RHR cooling or steam generator cooling (for POSs where steam generators are available). Alternative methods of establishing cooling may be possible by injecting coolant from the RWST using charging or safety injection pumps or by gravity-driven injection. Credit for reaching a safe, stable state by so-called "feed and bleed" or "fill and spill" could be considered if detailed guidance on the strategy were available, including plant procedures and supporting calculations.
Plant transition between states and POS operating hours	The POS durations are defined to completely cover a representative refueling outage. The durations are defined in Section 2. However, in some cases (POS 7 and POS 9) only a fraction of the POS is modeled because only a fraction is spent in the representative conditions. These POSs represent significant transitions that are occurring within the plant (e.g., raising RCS level from reduced inventory to refueling cavity flooded). This modeling simplification leaves some portions of the POS un-modeled because only a fraction of the total POS duration is included. This leads to an underestimation of risk. This could be improved by further sub-dividing the POSs and covering the entire duration of the outage in the model.
Test and maintenance durations	Improved data collection on outage-specific test and maintenance durations would improve the realism of the modeling. For many of the key systems and components, the duration may be shorter than the assumed duration, as the equipment is currently assumed unavailable for the entire POS.
Pre-initiator HFEs	Only a small number of pre-initiator HFEs are modeled and these are based on the at-power analysis. The LPSD PRA could be improved by performing a more thorough evaluation of LPSD pre-initiator HFEs and the dependency of post-initiator HFEs on the pre-initiator HFEs.
Consequential LOOP	The modeling of consequential LOOP events was incorporated from the at-power model. The likelihood of consequential LOOP for shutdown initiators is expected to be much different than that for at-power transients and LOCAs. The consequential LOOP modeling could be removed from the shutdown model or adjusted with more appropriate conditional probabilities. It should be noted that consequential LOOP events are not significant contributors to the current model results.
LOOP accident sequence modeling	LOOP accident sequence modeling is simplified by using the same model logic as LORHR for the sequences where emergency power or normal

Table A-1 Candidates for Future Work (cont.)

Topic Area	Proposed Area for Future Work
	offsite power has been restored. The modeling simplifications for LOOP sequences with power recovery are not expected to have a significant impact on the results. The plant and operator responses are similar. However, the specific RHR restoration actions can be different. Reflecting these differences with LOOP-specific HFEs would improve the realism of the model. Other considerations could be differences in sequence timing, staff resources (as operators are responding to LOOP and restoring RHR), and equipment dependencies.
HFEs applied to multiple POSs and IEs	Several HFEs applied to multiple POSs and IEs as shown in Table 7-1. This is a modeling simplification because the HFEs for these different accident conditions are similar. Applying a systematic approach to identify the accident scenario conditions that are significantly different as to suggest development of other unique HFEs would improve the completeness and fidelity of the LPSD PRA.
Human failure events with less than 30 minutes available	Additional evaluation of key HFEs associated with short times available for operator actions could help improve the realism of the associated accident sequences. A more detailed timing evaluation could improve the HEP estimates for those HFEs. This could require additional task analyses, interviews with operators and plant staff, observation of simulator exercises, and additional plant walkdowns. Note that several of the important HFEs in the LPSD PRA are associated with mid-loop sequences with limited time available to perform operator actions.
Applying operating experience to human failure event evaluation and dependency	A review of LPSD operating experience (APPENDIX D and prior studies on initiating events (EPRI, 2001) and (EPRI, 2010) provide much information on how failures have occurred and what restoration and recovery actions were taken. In this study, the operating experience was used to glean general qualitative insights that informed the modeling. Future work could be pursued to incorporate the operating experience in a more structured way to provide data for shaping HFEs, HEP estimates, and HFE dependency.
Consider alternative HRA methods	The SPAR-H method was selected for use in the LPSD study along with a small number of HFEs that were evaluated and adopted from the reference plant's PRA. Given the significance of important HFEs to the overall results of the LPSD PRA, additional study of the HFEs may be beneficial to characterize the possible range and uncertainty of HEP values. For example, the IDHEAS-G method (NRC, 2021b) has been developed by NRC. While this method was not available for use in the LPSD PRA, future work could consider the impact of using IDHEAS-G or other HRA methods. HRA of actions taken during shutdown was also an area identified for additional study by the expert panel that contributed to NUREG/CR-7265.
Credit for EDG recovery	EDG recovery has been credited in the LPSD Level 1 model; however, it is not applied to all failure modes that are modeled in the emergency power fault tree (EPS). Recovery is not applied to failures of RAT breakers. Also, sequencer failures are not recoverable because the data used to model EDG recovery do not include sequencer failures. The LPSD model credits EDG recovery only in POSs 5A, 8E, and 8L. In each of these POSs there are maintenance activities that limit the EDG redundancy. The conditions for restoring at least one EDG are favorable during these POSs due to the availability of time and resources to perform the restoration. Additional study of EDG recovery could provide bases for expanding credit for other types of failure recovery if supported by operating experience, plant procedures, and sufficient time to perform the actions.

Table A-1 Candidates for Future Work (cont.)

Topic Area	Proposed Area for Future Work
Modeling error in application of test and maintenance assumptions	As described in Section 8.3, a set of assumptions were developed regarding the unavailability of equipment due to testing and maintenance. The unavailability assumptions are applied in the model using “flag files” to adjust the basic event settings in the models. Due to a modeling error, the flag files were not correctly applied for a small number of modeling elements. The impacted event trees include 1-SD-LOINV-OP, 1-SD-LONSCW, and 1-SD-LOINV-LG. The test and maintenance assumptions can have significant impacts on results, particularly the LOOP scenarios. However, the impacts on these event tree results are limited. The flag file rules were fixed, and the results were re-generated to assess the impact. For the 1-SD-LONSCW, and 1-SD-LOINV-LG event trees, the change in CDF is negligible. For the 1-SD-LOINV-OP event tree, the CDF was reduced by approximately 2 percent. The reason for the decrease is the removal of test and maintenance basic events that are not performed at the same time as the assumed maintenance activities. The modeling error was discovered after documentation of the model version number 402 model results, and it was deemed not significant enough to re-generate the results documentation. Also, the overall uncertainty related to testing and maintenance unavailabilities during shutdown is expected to overwhelm the small impacts of this modeling error.
Comparison to other low power and shutdown PRAs	Section 9.8 provides a comparison to NRC’s evaluation of LPSD severe accidents at Surry in NUREG/CR-6144 (NRC, 1994a). The L3PRA project and future studies could benefit from additional comparisons to other LPSD PRAs. While the number of LPSD PRAs is limited for U.S. operating plants, other countries have regulatory frameworks that require LPSD PRAs for all plants. Comparisons against LPSD PRAs for foreign plants is an area to consider for future work.

APPENDIX B OUTPUT TABLES

This appendix provides output tables containing (1) the top cutsets contributing 95 percent of total shutdown CDF (Table B-1 in Section B.1) and (2) basic event importance measures (Table B-2 in Section B.2).

B.1 Cut Sets Contributing to CDF

The top 700 cut sets contributing 95 percent of the overall CDF for the LPSD PRA are provided in Table B-1. The cut set results were generated using model version number 419.

Table B-1 Top Cut Sets Contributing 95% of Total CDF

#	Prob/Freq	Total %	Cut Set
Total	1.190E-5	100	Displaying 700 Cut Sets. (20357 Original)
1	4.637E-7	3.90	1-IE-SD-ODML-EARLY,1-POS-6-POT-4-MLDRL,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XL-TRN-MLDRL
2	3.906E-7	3.28	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002___,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
3	3.784E-7	3.18	1-IE-SD-ODML-EARLY,1-POS-6-POT-4-MLDRL,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XM-MLDRL-I
4	3.492E-7	2.93	1-IE-SD-LOINV-OP,1-L2-BE-RHRVSTUCK-SD,1-LOI-XHE-XD-POS5A,1-POS-5A-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC-LT5H
5	3.387E-7	2.85	1-IE-SD-LOINV-OP,1-L2-BE-RHRVSTUCK-SD,1-LOI-XHE-XD-POS5A,1-POS-5A-POT-4,1-POT-4,1-RHR-XHE-XM-TRIP-MD
6	3.196E-7	2.69	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002___,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
7	3.130E-7	2.63	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002___,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
8	2.985E-7	2.51	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002___,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR
9	2.442E-7	2.05	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002___,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
10	2.346E-7	1.97	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-POS-12-POT-4,1-POT-4
11	2.323E-7	1.95	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301___,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
12	1.900E-7	1.60	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301___,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
13	1.861E-7	1.56	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301___,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
14	1.775E-7	1.49	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301___,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
15	1.625E-7	1.37	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-LPI-XHE-XM-GFEED-LOOP,1-POS-8E-POT-4,1-POT-4
16	1.571E-7	1.32	1-IE-LONSCW,1-IE-SWS-MDP-CR-123456,1-POS-8E-POT-4,1-POT-4
17	1.452E-7	1.22	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
18	1.446E-7	1.21	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
19	1.330E-7	1.12	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO
20	1.302E-7	1.09	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-POS-5A-POT-4,1-POT-4
21	1.276E-7	1.07	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002__,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
22	1.242E-7	1.04	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-LPI-XHE-XM-GFEED-LOOP,1-POS-8L-POT-4,1-POT-4
23	1.201E-7	1.01	1-IE-SD-ODML-EARLY,1-HPI-XHE-XM-FFEED-MLDRL,1-LPI-XHE-XM-GFEED-MD,1-POS-6-POT-4-MLDRL,1-RHR-XHE-XL-TRN-MLDRL
24	1.200E-7	1.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-123456,1-POS-8L-POT-4,1-POT-4
25	1.183E-7	0.99	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
26	1.167E-7	0.98	1-IE-SD-ODML-EARLY,1-LPI-MDP-CF-START,1-POS-6-POT-4-MLDRL
27	1.159E-7	0.97	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
28	1.105E-7	0.93	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR
29	1.044E-7	0.88	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
30	1.023E-7	0.86	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
31	1.016E-7	0.85	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO
32	1.013E-7	0.85	1-IE-SD-ODML-LATE,1-LPI-MDP-CF-START,1-POS-10-POT-4-D

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
33	1.006E-7	0.85	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-EPS-XHE-XL-NR04H,1-POS-12-POT-4,1-POT-4
34	9.753E-8	0.82	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR
35	9.556E-8	0.80	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-LPI-XHE-XM-GFEED-LOOP,1-POS-5B-POT-4,1-POT-4
36	9.276E-8	0.78	1-IE-SD-LORHR,1-LPI-MDP-CF-START,1-POS-8E-POT-4,1-POT-4
37	9.038E-8	0.76	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
38	8.772E-8	0.74	1-IE-SD-ODML-EARLY,1-HPI-XHE-XM-FFEED-MLDRL,1-LPI-XHE-XM-GFEED-LD,1-POS-6-POT-4-MLDRL,1-RHR-XHE-XM-MLDRL-I
39	8.764E-8	0.74	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8E-POT-4,1-POT-4
40	8.672E-8	0.73	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-POS-13-POT-4,1-POT-4
41	8.594E-8	0.72	1-IE-SD-LO4160C,1-LPI-MDP-CF-START,1-POS-8E-POT-4,1-POT-4
42	7.980E-8	0.67	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
43	7.819E-8	0.66	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-POS-5B-POT-4,1-POT-4,/1-SD-FEED-G-SBO
44	7.589E-8	0.64	1-IE-SD-LORHR,1-POS-6-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XL-TRN-POS6
45	7.250E-8	0.61	1-IE-LONSCW,1-IE-SWS-MDP-CR-123456,1-POS-5B-POT-4,1-POT-4,/1-SD-NSCW-D-P5B
46	7.171E-8	0.60	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO
47	7.163E-8	0.60	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-LPI-XHE-XM-GFEED-LOOP,1-POS-9-POT-4,1-POT-4
48	7.088E-8	0.60	1-IE-SD-LORHR,1-LPI-MDP-CF-START,1-POS-8L-POT-4,1-POT-4
49	7.023E-8	0.59	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-5A-POT-4,1-POT-4
50	6.969E-8	0.59	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-POS-8E-POT-4,1-POT-4

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
51	6.923E-8	0.58	1-IE-LONSCW,1-IE-SWS-MDP-CR-123456,1-POS-9-POT-4,1-POT-4
52	6.697E-8	0.56	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002___,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8L-POT-4,1-POT-4
53	6.567E-8	0.55	1-IE-SD-LO4160C,1-LPI-MDP-CF-START,1-POS-8L-POT-4,1-POT-4
54	5.966E-8	0.50	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-LPI-XHE-XM-GFEED-LOOP,1-POS-10-POT-4,1-POT-4
55	5.867E-8	0.49	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-LPI-XHE-XM-GFEED-LOOP,1-POS-5B-POT-4,1-POT-4
56	5.766E-8	0.48	1-IE-LONSCW,1-IE-SWS-MDP-CR-123456,1-POS-10-POT-4,1-POT-4
57	5.741E-8	0.48	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-LPI-XHE-XM-GFEED-LOOP,1-POS-11-POT-4,1-POT-4
58	5.702E-8	0.48	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-EPS-XHE-XL-NR04H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO
59	5.661E-8	0.48	1-IE-SD-LOINV-OP,1-L2-BE-RHRVSTUCK-SD,1-LOI-XHE-XD-POS5A,1-LOI-XHE-XM-ISO2-LT,1-POS-5A-POT-4,1-POT-4
60	5.585E-8	0.47	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-EPS-XHE-XL-NR04H,1-POS-5A-POT-4,1-POT-4
61	5.479E-8	0.46	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002___,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO
62	5.455E-8	0.46	1-IE-SD-LORHR,1-LPI-MDP-CF-START,1-POS-5B-POT-4,1-POT-4
63	5.325E-8	0.45	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-POS-8L-POT-4,1-POT-4
64	5.212E-8	0.44	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301___,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4
65	4.881E-8	0.41	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-POS-10-POT-4,1-POT-4,/1-SD-FEED-G-SBO
66	4.800E-8	0.40	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-POS-5B-POT-4,1-POT-4,/1-SD-FEED-G-SBO
67	4.398E-8	0.37	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-LPI-XHE-XM-GFEED-LOOP,1-POS-9-POT-4,1-POT-4
68	4.357E-8	0.37	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-EPS-XHE-XL-NR04H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO
69	4.264E-8	0.36	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301___,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
70	4.177E-8	0.35	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
71	4.176E-8	0.35	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-4-POT-4,1-POT-4,OEP-VCF-LP-GRWR
72	4.120E-8	0.35	1-IE-SD-LOOP,1-LPI-MDP-CF-START,1-POS-8E-POT-4,1-POT-4
73	4.089E-8	0.34	1-IE-SD-LORHR,1-LPI-MDP-CF-START,1-POS-9-POT-4,1-POT-4
74	4.022E-8	0.34	1-IE-SD-ODML-LATE,1-POS-10-POT-4-D,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XL-TRN-ML-P10
75	3.983E-8	0.33	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4
76	3.861E-8	0.32	1-IE-SD-ODML-LATE,1-POS-10-POT-4-D,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XM-POS10-I
77	3.719E-8	0.31	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-EPS-XHE-XL-NR04H,1-POS-13-POT-4,1-POT-4
78	3.663E-8	0.31	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-LPI-XHE-XM-GFEED-LOOP,1-POS-10-POT-4,1-POT-4
79	3.590E-8	0.30	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-LPI-XHE-XM-GFEED-LOOP,1-POS-7-POT-4,1-POT-4
80	3.525E-8	0.30	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-LPI-XHE-XM-GFEED-LOOP,1-POS-11-POT-4,1-POT-4
81	3.406E-8	0.29	1-IE-SD-LORHR,1-LPI-MDP-CF-START,1-POS-10-POT-4,1-POT-4
82	3.371E-8	0.28	1-IE-SD-LOOP,1-POS-6-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XL-TRN-POS6
83	3.347E-8	0.28	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-LPI-XHE-XM-GFEED-LOOP,1-POS-6-POT-4,1-POT-4
84	3.277E-8	0.28	1-IE-SD-LORHR,1-LPI-MDP-CF-START,1-POS-11-POT-4,1-POT-4
85	3.258E-8	0.27	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO
86	3.244E-8	0.27	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4
87	3.148E-8	0.26	1-IE-SD-LOOP,1-LPI-MDP-CF-START,1-POS-8L-POT-4,1-POT-4
88	2.997E-8	0.25	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-POS-10-POT-4,1-POT-4,/1-SD-FEED-G-SBO
89	2.937E-8	0.25	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-POS-7-POT-4,1-POT-4,/1-SD-FEED-G-SBO

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
90	2.864E-8	0.24	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4
91	2.739E-8	0.23	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-POS-6-POT-4,1-POT-4,/1-SD-FEED-G-SBO
92	2.654E-8	0.22	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO
93	2.600E-8	0.22	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
94	2.483E-8	0.21	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-4-POT-4,1-POT-4,OEP-VCF-LP-GRWR
95	2.479E-8	0.21	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4
96	2.423E-8	0.20	1-IE-SD-LOOP,1-LPI-MDP-CF-START,1-POS-5B-POT-4,1-POT-4
97	2.403E-8	0.20	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-POS-9-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
98	2.343E-8	0.20	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002____,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO
99	2.295E-8	0.19	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002____,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
100	2.204E-8	0.19	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-LPI-XHE-XM-GFEED-LOOP,1-POS-7-POT-4,1-POT-4
101	2.189E-8	0.18	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4
102	2.167E-8	0.18	1-IE-SD-LORHR,1-POS-5B-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XL-TRN-3H
103	2.055E-8	0.17	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-LPI-XHE-XM-GFEED-LOOP,1-POS-6-POT-4,1-POT-4
104	2.049E-8	0.17	1-IE-SD-LORHR,1-LPI-MDP-CF-START,1-POS-7-POT-4,1-POT-4
105	2.028E-8	0.17	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO
106	1.986E-8	0.17	1-IE-LONSCW,1-IE-SWS-MDP-CR-123456,1-LPI-XHE-XM-GFEED-NSCW,1-POS-5B-POT-4,1-POT-4

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
107	1.926E-8	0.16	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-POS-11-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
108	1.911E-8	0.16	1-IE-SD-LORHR,1-LPI-MDP-CF-START,1-POS-6-POT-4,1-POT-4
109	1.867E-8	0.16	1-IE-LONSCW,1-IE-SWS-MDP-CR-123456,1-POS-6-POT-4,1-POT-4,/1-SD-NSCW-D-ML
110	1.816E-8	0.15	1-IE-SD-LOOP,1-LPI-MDP-CF-START,1-POS-9-POT-4,1-POT-4
111	1.803E-8	0.15	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-POS-7-POT-4,1-POT-4,/1-SD-FEED-G-SBO
112	1.801E-8	0.15	1-IE-SD-LOINV,1-POS-5B-POT-4,1-POT-4,1-RHR-XHE-XM-LOI-POS5B,1-RHR-XHE-XM-TRIP-HD
113	1.791E-8	0.15	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002____,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO
114	1.759E-8	0.15	1-IE-SD-LORHR,1-HPI-XHE-XM-FFEED-MLDRL,1-LPI-XHE-XM-GFEED-LD,1-POS-6-POT-4,1-POT-4,1-RHR-XHE-XL-TRN-POS6
115	1.748E-8	0.15	1-IE-SD-LO4160C,1-ACP-XHE-XL-ALTAC-6H,1-EPS-DGN-FR-G4002____,1-POS-8E-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC-LT5H
116	1.738E-8	0.15	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-POS-4-POT-4,1-POT-4
117	1.681E-8	0.14	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-POS-6-POT-4,1-POT-4,/1-SD-FEED-G-SBO
118	1.577E-8	0.13	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FR-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H2,1-POS-5B-POT-4,1-POT-4
119	1.546E-8	0.13	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-4-POT-4,1-POT-4,OEP-VCF-LP-GRWR
120	1.513E-8	0.13	1-IE-SD-LOOP,1-LPI-MDP-CF-START,1-POS-10-POT-4,1-POT-4
121	1.455E-8	0.12	1-IE-SD-LOOP,1-LPI-MDP-CF-START,1-POS-11-POT-4,1-POT-4
122	1.392E-8	0.12	1-IE-SD-LOOP,1-EPS-TNK-MA-DFOSTKB_,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
123	1.385E-8	0.12	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,1-SWS-MOV-CC-1669A____,OEP-VCF-LP-GRWR
124	1.368E-8	0.12	1-IE-LONSCW,1-IE-SWS-MDP-CR-123456,1-LPI-XHE-XM-GFEED-MLDRL,1-POS-6-POT-4,1-POT-4

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
125	1.365E-8	0.11	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-4-POT-4,1-POT-4,OEP-VCF-LP-GRWR
126	1.364E-8	0.11	1-IE-SD-LOOP,1-AFW-TDP-FR-P4001____,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR02H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
127	1.353E-8	0.11	1-IE-SD-LORHR,1-POS-10-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XL-TRN-POS10
128	1.353E-8	0.11	1-IE-SD-LORHR,1-POS-10-POT-4,1-POT-4,1-RHR-XHE-XD-ML,1-RHR-XHE-XL-LTREC
129	1.350E-8	0.11	1-IE-SD-LOINV,1-POS-9-POT-4,1-POT-4,1-RHR-XHE-XM-LOI-POS5B,1-RHR-XHE-XM-TRIP-HD
130	1.302E-8	0.11	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-POS-3-POT-4,1-POT-4
131	1.253E-8	0.11	1-IE-SD-LO4160C,1-POS-10-POT-4,1-POT-4,1-RHR-XHE-XD-ML,1-RHR-XHE-XL-LTREC
132	1.182E-8	0.10	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FR-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H2,1-POS-9-POT-4,1-POT-4
133	1.133E-8	0.10	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,1-SWS-MOV-CC-1669A____,OEP-VCF-LP-GRWR
134	1.120E-8	0.09	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FR-G4002____,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H2,1-POS-5B-POT-4,1-POT-4,/1-SD-FEED-G-SBO
135	1.110E-8	0.09	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,1-SWS-MOV-CC-1669A____,OEP-VCF-LP-GRWR
136	1.059E-8	0.09	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,1-SWS-MOV-CC-1669A____,OEP-VCF-LP-GRWR
137	1.038E-8	0.09	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H0,1-POS-5B-POT-4,1-POT-4
138	1.030E-8	0.09	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-EPS-XHE-XL-NR04H,1-POS-9-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
139	9.843E-9	0.08	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FR-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H2,1-POS-10-POT-4,1-POT-4

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
140	9.625E-9	0.08	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H2,1-POS-12-POT-4,1-POT-4,OEP-VCF-LP-GRWR
141	9.623E-9	0.08	1-IE-SD-LOOP,1-POS-5B-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XL-TRN-3H
142	9.471E-9	0.08	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FR-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H2,1-POS-11-POT-4,1-POT-4
143	9.429E-9	0.08	1-IE-SD-ODML-EARLY,1-LPI-MDP-CF-RUN,1-POS-6-POT-4-MLDRL
144	9.370E-9	0.08	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-4-POT-4,1-POT-4
145	9.324E-9	0.08	1-IE-SD-LOOP,1-ACP-INV-FC-BD1I12____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
146	9.324E-9	0.08	1-IE-SD-ODML-LATE,1-HPI-XHE-XM-FFEED-MLDRL,1-LPI-XHE-XM-GFEED-LD,1-POS-10-POT-4-D,1-RHR-XHE-XL-TRN-ML-P10
147	9.102E-9	0.08	1-IE-SD-LOOP,1-LPI-MDP-CF-START,1-POS-7-POT-4,1-POT-4
148	8.951E-9	0.08	1-IE-SD-ODML-LATE,1-HPI-XHE-XM-FFEED-MLDRL,1-LPI-XHE-XM-GFEED-LD,1-POS-10-POT-4-D,1-RHR-XHE-XM-POS10-I
149	8.741E-9	0.07	1-IE-SD-LO4160C,1-EPS-DGN-FR-G4002____,1-LPI-MDP-FS-RHRA____,1-POS-8E-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC-LT5H
150	8.661E-9	0.07	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,1-SWS-MOV-CC-1669A____,OEP-VCF-LP-GRWR
151	8.486E-9	0.07	1-IE-SD-LOOP,1-LPI-MDP-CF-START,1-POS-6-POT-4,1-POT-4
152	8.258E-9	0.07	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-EPS-XHE-XL-NR04H,1-POS-11-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
153	8.178E-9	0.07	1-IE-SD-ODML-LATE,1-LPI-MDP-CF-RUN,1-POS-10-POT-4-D
154	8.127E-9	0.07	1-IE-SD-LOINV-OP,1-HPI-XHE-XM-FFEED-LOI5A,1-L2-BE-RHRVSTUCK-SD,1-LOI-XHE-XD-POS5A,1-LPI-XHE-XM-GFEED-LD,1-POS-5A-POT-4,1-POT-4
155	8.118E-9	0.07	1-IE-SD-LOOP,1-ACP-SSD-MA-1821U302,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
			XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
156	8.111E-9	0.07	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-AFW-TDP-FR-P4001__,1-EPS-XHE-XL-NR02H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
157	7.819E-9	0.07	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205__,1-ACP-CRB-CC-BA0301__,1-LPI-XHE-XM-GFEED-LOOP,1-POS-5B-POT-4,1-POT-4
158	7.814E-9	0.07	1-IE-SD-LOOP,1-HPI-XHE-XM-FFEED-MLDRL,1-LPI-XHE-XM-GFEED-LD,1-POS-6-POT-4,1-POT-4,1-RHR-XHE-XL-TRN-POS6
159	7.777E-9	0.07	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H0,1-POS-9-POT-4,1-POT-4
160	7.629E-9	0.06	1-IE-SD-LOOP,1-ACP-INV-FC-BD1I12__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
161	7.589E-9	0.06	1-IE-SD-LORHR,1-POS-6-POT-4,1-POT-4,1-RHR-XHE-XD-ML,1-RHR-XHE-XL-LTREC
162	7.493E-9	0.06	1-IE-SD-LORHR,1-LPI-MDP-CF-RUN,1-POS-8E-POT-4,1-POT-4
163	7.480E-9	0.06	1-IE-SD-LOOP,1-ACP-BAC-MA-MCCBBB__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
164	7.480E-9	0.06	1-IE-SD-LOOP,1-ACP-BAC-MA-MCCBBF__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
165	7.472E-9	0.06	1-IE-SD-LOOP,1-ACP-INV-FC-BD1I12__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
166	7.451E-9	0.06	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-EPS-XHE-XL-NR04H,1-POS-4-POT-4,1-POT-4
167	7.155E-9	0.06	1-IE-SD-LORHR,1-POS-8E-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC-LT5H,1-RHR-XHE-XL-TRB-POS8E
168	7.125E-9	0.06	1-IE-SD-LOOP,1-ACP-INV-FC-BD1I12__,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR
169	7.031E-9	0.06	1-IE-SD-LO4160C,1-POS-6-POT-4,1-POT-4,1-RHR-XHE-XD-ML,1-RHR-XHE-XL-LTREC
170	6.993E-9	0.06	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001__,1-EPS-DGN-FR-G4002__,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H2,1-POS-10-POT-4,1-POT-4,1-SD-FEED-G-SBO
171	6.941E-9	0.06	1-IE-SD-LO4160C,1-LPI-MDP-CF-RUN,1-POS-8E-POT-4,1-POT-4

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
172	6.834E-9	0.06	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS1__,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
173	6.834E-9	0.06	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS4__,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
174	6.762E-9	0.06	1-IE-SD-LOOP,1-ACP-BAC-MA-BYB1____,1-EPS-XHE-XL-NR04H,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
175	6.642E-9	0.06	1-IE-SD-LOOP,1-ACP-SSD-MA-1821U302,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
176	6.629E-9	0.06	1-IE-SD-LO4160C,1-POS-8E-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC-LT5H,1-RHR-XHE-XL-TRB-POS8E
177	6.506E-9	0.05	1-IE-SD-LOOP,1-ACP-SSD-MA-1821U302,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
178	6.478E-9	0.05	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H0,1-POS-10-POT-4,1-POT-4
179	6.444E-9	0.05	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H0,1-POS-5B-POT-4,1-POT-4,/1-SD-FEED-G-SBO
180	6.233E-9	0.05	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H0,1-POS-11-POT-4,1-POT-4
181	6.204E-9	0.05	1-IE-SD-LOOP,1-ACP-SSD-MA-1821U302,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR
182	6.130E-9	0.05	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FR-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR30M,1-OEP-XHE-XX-NR30M2,1-POS-6-POT-4,1-POT-4
183	6.008E-9	0.05	1-IE-SD-LOOP,1-POS-10-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XL-TRN-POS10
184	6.008E-9	0.05	1-IE-SD-LOOP,1-POS-10-POT-4,1-POT-4,1-RHR-XHE-XD-ML,1-RHR-XHE-XL-LTREC
185	5.873E-9	0.05	1-IE-LONSCW,1-IE-SWS-MDP-CR-24,1-OA-OSW-----H,1-POS-8E-POT-4,1-POT-4
186	5.860E-9	0.05	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205__,1-ACP-CRB-CC-BA0301__,1-LPI-XHE-XM-GFEED-LOOP,1-POS-9-POT-4,1-POT-4
187	5.830E-9	0.05	1-IE-SD-LOOP,1-ACP-INV-FC-BD1I12__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
188	5.725E-9	0.05	1-IE-SD-LORHR,1-LPI-MDP-CF-RUN,1-POS-8L-POT-4,1-POT-4
189	5.653E-9	0.05	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205___,1-EPS-DGN-FR-G4002___,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-5B-POT-4,1-POT-4
190	5.653E-9	0.05	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301___,1-EPS-DGN-FR-G4001___,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-5B-POT-4,1-POT-4
191	5.592E-9	0.05	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS4___,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
192	5.592E-9	0.05	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS1___,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
193	5.585E-9	0.05	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-EPS-XHE-XL-NR04H,1-POS-3-POT-4,1-POT-4
194	5.572E-9	0.05	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301___,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-4-POT-4,1-POT-4
195	5.537E-9	0.05	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H0,1-POS-12-POT-4,1-POT-4,OEP-VCF-LP-GRWR
196	5.497E-9	0.05	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205___,1-ACP-CRB-CC-BA0301___,1-EPS-XHE-XL-NR04H,1-POS-12-POT-4,1-POT-4,OEP-VCF-LP-GRWR
197	5.477E-9	0.05	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS4___,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
198	5.477E-9	0.05	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS1___,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
199	5.476E-9	0.05	1-IE-SD-LOOP,1-AFW-TDP-FR-P4001___,1-EPS-SEQ-CF-FOAB,1-POS-12-POT-4,1-POT-4
200	5.392E-9	0.05	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-OA-ALIGNPW-G-H,1-POS-9-POT-4,1-POT-4,/1-SD-FEED-G-SBO
201	5.304E-9	0.04	1-IE-SD-LO4160C,1-LPI-MDP-CF-RUN,1-POS-8L-POT-4,1-POT-4
202	5.223E-9	0.04	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS1___,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR
203	5.223E-9	0.04	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS4___,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
			XL-NR02H,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR
204	5.216E-9	0.04	1-IE-SD-LOOP,1-EPS-DGN-CF-FSUN1,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-5B-POT-4,1-POT-4
205	5.096E-9	0.04	1-IE-SD-LOOP,1-ACP-CRB-OO-ANA0401_,1-EPS-DGN-FR-G4002_,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8E-POT-4,1-POT-4
206	5.076E-9	0.04	1-IE-SD-LOOP,1-ACP-SSD-MA-1821U302,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
207	5.049E-9	0.04	1-IE-SD-LOOP,1-AFW-TDP-FR-P4001_,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR02H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
208	5.008E-9	0.04	1-IE-SD-LOOP,1-EPS-MDP-FS-XFERPPS_,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-5B-POT-4,1-POT-4
209	4.881E-9	0.04	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205_,1-ACP-CRB-CC-BA0301_,1-LPI-XHE-XM-GFEED-LOOP,1-POS-10-POT-4,1-POT-4
210	4.867E-9	0.04	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301_,1-EPS-SEQ-FO-1821U301,1-LPI-XHE-XM-GFEED-LOOP,1-POS-5B-POT-4,1-POT-4
211	4.867E-9	0.04	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205_,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-POS-5B-POT-4,1-POT-4
212	4.697E-9	0.04	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205_,1-ACP-CRB-CC-BA0301_,1-LPI-XHE-XM-GFEED-LOOP,1-POS-11-POT-4,1-POT-4
213	4.488E-9	0.04	1-IE-LONSCW,1-IE-SWS-MDP-CR-24,1-OA-OSW-----H,1-POS-8L-POT-4,1-POT-4
214	4.457E-9	0.04	1-IE-SD-LOOP,1-AFW-TDP-FR-P4001_,1-EPS-DGN-FS-G4002_,1-EPS-XHE-XL-NR02H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
215	4.437E-9	0.04	1-IE-SD-ODML-EARLY,1-ACP-CRB-CF-A205301,1-OEP-VCF-LP-CLOPT,1-POS-6-POT-4-MLDRL
216	4.406E-9	0.04	1-IE-SD-LORHR,1-LPI-MDP-CF-RUN,1-POS-5B-POT-4,1-POT-4
217	4.399E-9	0.04	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR30M,1-OEP-XHE-XX-NR30M0,1-POS-6-POT-4,1-POT-4
218	4.370E-9	0.04	1-IE-SD-LO4160C,1-EPS-DGN-FR-G4002_,1-POS-8E-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC-LT5H,1-RHR-XHE-XL-TRA-POS8E

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
219	4.321E-9	0.04	1-IE-SD-LOOP,1-ACP-CRB-CF-A205301,1-OA-ALIGNPW-G-H,1-POS-11-POT-4,1-POT-4,/1-SD-FEED-G-SBO
220	4.273E-9	0.04	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS1__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
221	4.273E-9	0.04	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS4__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
222	4.237E-9	0.04	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205__,1-EPS-DGN-FR-G4002__,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-9-POT-4,1-POT-4
223	4.237E-9	0.04	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-EPS-DGN-FR-G4001__,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-9-POT-4,1-POT-4
224	4.177E-9	0.04	1-IE-SD-LOINV-SP,1-LOI-XHE-XM-ISO2-LT,1-POS-10-POT-4,1-POT-4,1-RHR-XHE-XM-LOISP-POS6
225	4.170E-9	0.04	1-IE-SD-LOOP,1-ACP-CRB-OO-ANA0401__,1-EPS-DGN-FR-G4002__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO
226	4.096E-9	0.03	1-IE-SD-LOINV-DS,1-POS-5B-POT-4,1-POT-4,1-RHR-XHE-XM-M5LOI-I,1-RHR-XHE-XM-TRIP-LD
227	4.084E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-OO-ANA0401__,1-EPS-DGN-FR-G4002__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-5A-POT-4,1-POT-4
228	4.082E-9	0.03	1-IE-SD-LO4160C,1-LPI-MDP-CF-RUN,1-POS-5B-POT-4,1-POT-4
229	4.023E-9	0.03	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H0,1-POS-10-POT-4,1-POT-4,/1-SD-FEED-G-SBO
230	3.923E-9	0.03	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001__,1-EPS-DGN-FR-G4002__,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H2,1-POS-6-POT-4,1-POT-4,/1-SD-FEED-G-SBO
231	3.910E-9	0.03	1-IE-SD-LOOP,1-EPS-DGN-CF-FSUN1,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-9-POT-4,1-POT-4
232	3.894E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-OO-ANA0401__,1-EPS-DGN-FR-G4002__,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8L-POT-4,1-POT-4
233	3.849E-9	0.03	1-IE-SD-ODML-LATE,1-ACP-CRB-CF-A205301,1-OEP-VCF-LP-CLOPT,1-POS-10-POT-4-D

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
234	3.835E-9	0.03	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H0,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
235	3.803E-9	0.03	1-IE-SD-LORHR,1-ACP-XHE-XL-ALTAC-6H,1-LPI-MDP-FS-RHRB____,1-POS-8E-POT-4,1-POT-4
236	3.803E-9	0.03	1-IE-SD-LORHR,1-DCP-BCH-MA-AD1CA____,1-LPI-MDP-FS-RHRB____,1-POS-8E-POT-4,1-POT-4
237	3.754E-9	0.03	1-IE-SD-LOOP,1-EPS-MDP-FS-XFERPPS____CC,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-9-POT-4,1-POT-4
238	3.651E-9	0.03	1-IE-SD-ODML-EARLY,1-DCP-BCH-FC-AAABBABB-CC,1-POS-6-POT-4-MLDRL
239	3.648E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-EPS-SEQ-FO-1821U301,1-LPI-XHE-XM-GFEED-LOOP,1-POS-9-POT-4,1-POT-4
240	3.648E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205____,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-POS-9-POT-4,1-POT-4
241	3.564E-9	0.03	1-IE-SD-ODML-EARLY,1-LPI-MDP-FS-RHRA____,1-POS-6-POT-4-MLDRL,1-RHR-XHE-XM-MLDRL-I,1-RHR-XHE-XM-TRIP-MD
242	3.557E-9	0.03	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H2,1-POS-13-POT-4,1-POT-4,OEP-VCF-LP-GRWR
243	3.529E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205____,1-EPS-DGN-FR-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-10-POT-4,1-POT-4
244	3.529E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-EPS-DGN-FR-G4001____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-10-POT-4,1-POT-4
245	3.523E-9	0.03	1-IE-SD-LO4160C,1-ACP-XHE-XL-ALTAC-6H,1-LPI-MDP-FS-RHRB____,1-POS-8E-POT-4,1-POT-4
246	3.519E-9	0.03	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-5B-POT-4,1-POT-4
247	3.519E-9	0.03	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-SEQ-FO-1821U301,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-5B-POT-4,1-POT-4
248	3.511E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-EPS-DGN-FR-G4001____,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-5B-POT-4,1-POT-4,/1-SD-FEED-G-SBO

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
249	3.511E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205__,1-EPS-DGN-FR-G4002__,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-5B-POT-4,1-POT-4,/1-SD-FEED-G-SBO
250	3.468E-9	0.03	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-4-POT-4,1-POT-4
251	3.421E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-EPS-SEQ-FO-1821U301,1-EPS-XHE-XL-NR04H,1-POS-12-POT-4,1-POT-4,OEP-VCF-LP-GRWR
252	3.421E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205__,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-POS-12-POT-4,1-POT-4,OEP-VCF-LP-GRWR
253	3.396E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205__,1-EPS-DGN-FR-G4002__,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-11-POT-4,1-POT-4
254	3.396E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-EPS-DGN-FR-G4001__,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-11-POT-4,1-POT-4
255	3.371E-9	0.03	1-IE-SD-LOOP,1-POS-6-POT-4,1-POT-4,1-RHR-XHE-XD-ML,1-RHR-XHE-XL-LTREC
256	3.328E-9	0.03	1-IE-SD-LOOP,1-LPI-MDP-CF-RUN,1-POS-8E-POT-4,1-POT-4
257	3.303E-9	0.03	1-IE-SD-LORHR,1-LPI-MDP-CF-RUN,1-POS-9-POT-4,1-POT-4
258	3.268E-9	0.03	1-IE-SD-LOINV-OP,1-EPS-SEQ-FO-1821U301,1-L2-BE-RHRVSTUCK-SD,1-LOI-XHE-XD-POS5A,1-LPI-XHE-XM-GFEED-LOI5A,1-OA-NSCWFAN---H,1-POS-5A-POT-4,1-POT-4
259	3.268E-9	0.03	1-IE-SD-LOINV-OP,1-EPS-SEQ-FO-1821U302,1-L2-BE-RHRVSTUCK-SD,1-LOI-XHE-XD-POS5A,1-LPI-XHE-XM-GFEED-LOI5A,1-OA-NSCWFAN---H,1-POS-5A-POT-4,1-POT-4
260	3.257E-9	0.03	1-IE-SD-LOOP,1-EPS-DGN-CF-FSUN1,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-10-POT-4,1-POT-4
261	3.240E-9	0.03	1-IE-SD-LOOP,1-DCP-FUS-OP-BD104__,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
262	3.186E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-OO-ANA0401_,1-EPS-DGN-FR-G4002__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO
263	3.178E-9	0.03	1-IE-SD-LOOP,1-POS-8E-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC-LT5H,1-RHR-XHE-XL-TRB-POS8E

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
264	3.166E-9	0.03	1-IE-SD-ODML-LATE,1-DCP-BCH-FC-AAABBABB-CC,1-POS-10-POT-4-D
265	3.138E-9	0.03	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H0,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
266	3.136E-9	0.03	1-IE-SD-LORHR,1-HPI-XHE-XM-FFEED-MLDRL,1-LPI-XHE-XM-GFEED-LD,1-POS-10-POT-4,1-POT-4,1-RHR-XHE-XL-TRN-POS10
267	3.134E-9	0.03	1-IE-SD-LOOP,1-EPS-DGN-CF-FSUN1,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-11-POT-4,1-POT-4
268	3.126E-9	0.03	1-IE-SD-LOOP,1-EPS-MDP-FS-XFERPPS_ -CC,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-10-POT-4,1-POT-4
269	3.123E-9	0.03	1-IE-SD-LOOP,1-EPS-TNK-MA-DFOSTKB_,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
270	3.109E-9	0.03	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-NSCWCT-SPRAY,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,1-SWS-MOV-CC-1669A__
271	3.106E-9	0.03	1-IE-SD-LOINV-DS,1-LPI-MDP-CF-START,1-POS-5B-POT-4,1-POT-4
272	3.106E-9	0.03	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001__,1-EPS-DGN-FS-G4002__,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-5B-POT-4,1-POT-4
273	3.106E-9	0.03	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002__,1-EPS-DGN-FS-G4001__,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-5B-POT-4,1-POT-4
274	3.074E-9	0.03	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H0,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
275	3.070E-9	0.03	1-IE-SD-LOINV-DS,1-POS-9-POT-4,1-POT-4,1-RHR-XHE-XM-M5LOI-I,1-RHR-XHE-XM-TRIP-LD
276	3.062E-9	0.03	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002__,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-4-POT-4,1-POT-4
277	3.061E-9	0.03	1-IE-SD-LOOP,1-AFW-TDP-FR-P4001__,1-EPS-DGN-FR-G4002__,1-EPS-XHE-XL-NR02H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-5A-POT-4,1-POT-4
278	3.060E-9	0.03	1-IE-SD-LO4160C,1-LPI-MDP-CF-RUN,1-POS-9-POT-4,1-POT-4

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
279	3.038E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-EPS-SEQ-FO-1821U301,1-LPI-XHE-XM-GFEED-LOOP,1-POS-10-POT-4,1-POT-4
280	3.038E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205__,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-POS-10-POT-4,1-POT-4
281	3.031E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-ACP-CRB-OO-ANA0401__,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4
282	3.029E-9	0.03	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U301,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-POS-5B-POT-4,1-POT-4
283	3.017E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-EPS-DGN-FR-G4001__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-12-POT-4,1-POT-4,OEP-VCF-LP-GRWR
284	3.017E-9	0.03	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205__,1-EPS-DGN-FR-G4002__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-12-POT-4,1-POT-4,OEP-VCF-LP-GRWR
285	3.008E-9	0.03	1-IE-SD-LOOP,1-EPS-MDP-FS-XFERPPS_-CC,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-11-POT-4,1-POT-4
286	2.971E-9	0.02	1-IE-SD-LOINV,1-LOI-XHE-XM-ISO2-LT,1-POS-10-POT-4,1-POT-4,1-RHR-XHE-XM-LOI-POS6
287	2.969E-9	0.02	1-IE-SD-LOOP,1-ACP-BAC-FC-BA03____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
288	2.969E-9	0.02	1-IE-SD-LOOP,1-ACP-BAC-FC-BB16____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
289	2.964E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002__,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-VCF-LP-CLOPT,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
290	2.931E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H0,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR
291	2.923E-9	0.02	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-EPS-SEQ-FO-1821U301,1-LPI-XHE-XM-GFEED-LOOP,1-POS-11-POT-4,1-POT-4
292	2.923E-9	0.02	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205__,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-POS-11-POT-4,1-POT-4

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
293	2.906E-9	0.02	1-IE-SD-LORHR,1-ACP-XHE-XL-ALTAC-6H,1-LPI-MDP-FS-RHRB____,1-POS-8L-POT-4,1-POT-4
294	2.906E-9	0.02	1-IE-SD-LORHR,1-DCP-BCH-MA-AD1CA____,1-LPI-MDP-FS-RHRB____,1-POS-8L-POT-4,1-POT-4
295	2.901E-9	0.02	1-IE-SD-LORHR,1-DCP-BCH-FC-AAABBABB-CC,1-POS-8E-POT-4,1-POT-4
296	2.837E-9	0.02	1-IE-SD-LO4160C,1-ACP-CRB-CC-BA0301____,1-ACP-XHE-XL-ALTAC-6H,1-POS-8E-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC-LT5H
297	2.759E-9	0.02	1-IE-SD-LO4160C,1-ACP-BAC-MA-AA02____,1-EPS-DGN-FR-G4002____,1-POS-7-POT-4,1-POT-4
298	2.759E-9	0.02	1-IE-SD-LO4160C,1-ACP-BAC-MA-AB15____,1-EPS-DGN-FR-G4002____,1-POS-7-POT-4,1-POT-4
299	2.751E-9	0.02	1-IE-SD-LORHR,1-LPI-MDP-CF-RUN,1-POS-10-POT-4,1-POT-4
300	2.739E-9	0.02	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205____,1-ACP-CRB-CC-BA0301____,1-LPI-XHE-XM-GFEED-LOOP,1-POS-6-POT-4,1-POT-4
301	2.731E-9	0.02	1-IE-SD-LO4160C,1-EPS-DGN-FR-G4002____,1-NSCWCT-NEED-SWAP,1-NSCWCT-BYPASS,1-OA-NSCWCT-MV-H,1-POS-8E-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC-LT5H
302	2.724E-9	0.02	1-IE-SD-ODML-EARLY,1-EPS-SEQ-CF-FOAB,1-OEP-VCF-LP-CLOPT,1-POS-6-POT-4-MLDRL
303	2.692E-9	0.02	1-IE-SD-LO4160C,1-ACP-XHE-XL-ALTAC-6H,1-LPI-MDP-FS-RHRB____,1-POS-8L-POT-4,1-POT-4
304	2.692E-9	0.02	1-IE-SD-LO4160C,1-DCP-BCH-MA-AD1CA____,1-LPI-MDP-FS-RHRB____,1-POS-8L-POT-4,1-POT-4
305	2.687E-9	0.02	1-IE-SD-LO4160C,1-DCP-BCH-FC-AAABBABB-CC,1-POS-8E-POT-4,1-POT-4
306	2.669E-9	0.02	1-IE-SD-LOINV-OP,1-DCP-BAT-MA-AD1B____,1-L2-BE-RHRVSTUCK-SD,1-LOI-XHE-XD-POS5A,1-LPI-XHE-XM-GFEED-LOI5A,1-OA-NSCWFAN---H,1-POS-5A-POT-4,1-POT-4
307	2.669E-9	0.02	1-IE-SD-LOINV-OP,1-DCP-BAT-MA-BD1B____,1-L2-BE-RHRVSTUCK-SD,1-LOI-XHE-XD-POS5A,1-LPI-XHE-XM-GFEED-LOI5A,1-OA-NSCWFAN---H,1-POS-5A-POT-4,1-POT-4
308	2.654E-9	0.02	1-IE-SD-LOINV-SP,1-POS-10-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XM-LOISP-POS6
309	2.651E-9	0.02	1-IE-SD-LOOP,1-DCP-FUS-OP-BD104____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
310	2.647E-9	0.02	1-IE-SD-LORHR,1-LPI-MDP-CF-RUN,1-POS-11-POT-4,1-POT-4

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
311	2.639E-9	0.02	1-IE-SD-ODML-EARLY,1-EPS-DGN-FR-G4001____,1-LPI-XHE-XM-GFEED-MD,1-OEP-VCF-LP-CLOPT,1-POS-6-POT-4-MLDRL,1-RHR-XHE-XL-TRN-MLDRL
312	2.639E-9	0.02	1-IE-SD-ODML-EARLY,1-EPS-DGN-FR-G4002____,1-LPI-XHE-XM-GFEED-MD,1-OEP-VCF-LP-CLOPT,1-POS-6-POT-4-MLDRL,1-RHR-XHE-XL-TRN-MLDRL
313	2.637E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-9-POT-4,1-POT-4
314	2.637E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-SEQ-FO-1821U301,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-9-POT-4,1-POT-4
315	2.597E-9	0.02	1-IE-SD-LOOP,1-DCP-FUS-OP-BD104____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
316	2.591E-9	0.02	1-IE-SD-LOOP,1-DCP-BAT-MA-ND1B____,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8E-POT-4,1-POT-4
317	2.588E-9	0.02	1-IE-LONSCW,1-IE-SWS-MDP-CR-24,1-OA-OSW-----H,1-POS-9-POT-4,1-POT-4
318	2.548E-9	0.02	1-IE-SD-LO4160C,1-LPI-MDP-CF-RUN,1-POS-10-POT-4,1-POT-4
319	2.543E-9	0.02	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-NSCWCT-SPRAY,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,1-SD-FEED-G-SBO,1-SWS-MOV-CC-1669A____
320	2.543E-9	0.02	1-IE-SD-LOOP,1-LPI-MDP-CF-RUN,1-POS-8L-POT-4,1-POT-4
321	2.491E-9	0.02	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-NSCWCT-SPRAY,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,1-SWS-MOV-CC-1669A____
322	2.480E-9	0.02	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-ACP-CRB-OO-ANA0401____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,1-SD-FEED-G-SBO
323	2.479E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-CF-FSUN1,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR30M,1-POS-6-POT-4,1-POT-4
324	2.476E-9	0.02	1-IE-SD-LOOP,1-DCP-FUS-OP-BD104____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR
325	2.434E-9	0.02	1-IE-SD-LOOP,1-AFW-TDP-FR-P4001____,1-EPS-SEQ-CF-FOAB,1-EPS-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
326	2.429E-9	0.02	1-IE-SD-LOOP,1-ACP-BAC-FC-BB16____,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
327	2.429E-9	0.02	1-IE-SD-LOOP,1-ACP-BAC-FC-BA03____,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
328	2.429E-9	0.02	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-ACP-CRB-OO-ANA0401____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
329	2.425E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-OEP-VCF-LP-CLOPT,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
330	2.404E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H2,1-POS-9-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
331	2.398E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H0,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
332	2.397E-9	0.02	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-EPS-DGN-FR-G4001____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR30M,1-OEP-XHE-XX-NR30M1,1-POS-6-POT-4,1-POT-4
333	2.397E-9	0.02	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205____,1-EPS-DGN-FR-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR30M,1-OEP-XHE-XX-NR30M1,1-POS-6-POT-4,1-POT-4
334	2.393E-9	0.02	1-IE-SD-ODML-EARLY,1-LPI-MDP-FS-RHRA____,1-LPI-MDP-FS-RHRB____,1-POS-6-POT-4-MLDRL
335	2.379E-9	0.02	1-IE-SD-LOOP,1-EPS-MDP-FS-XFERPPS____CC,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR30M,1-POS-6-POT-4,1-POT-4
336	2.376E-9	0.02	1-IE-SD-LORHR,1-HPI-XHE-XM-FFEED-6H,1-POS-5B-POT-4,1-POT-4,1-RHR-XHE-XL-TRN-3H
337	2.375E-9	0.02	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-NSCWCT-SPRAY,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,1-SWS-MOV-CC-1669A____
338	2.363E-9	0.02	1-IE-SD-ODML-LATE,1-EPS-SEQ-CF-FOAB,1-OEP-VCF-LP-CLOPT,1-POS-10-POT-4-D
339	2.344E-9	0.02	1-IE-SD-LOINV-SP,1-LOI-XHE-XM-ISO2-LT,1-POS-6-POT-4,1-POT-4,1-RHR-XHE-XM-LOISP-POS6
340	2.328E-9	0.02	1-IE-SD-LOINV-DS,1-LPI-MDP-CF-START,1-POS-9-POT-4,1-POT-4

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
341	2.328E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FS-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-9-POT-4,1-POT-4
342	2.328E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-DGN-FS-G4001____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-9-POT-4,1-POT-4
343	2.316E-9	0.02	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-ACP-CRB-OO-ANA0401____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4
344	2.312E-9	0.02	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-POS-9-POT-4,1-POT-4,/1-SD-FEED-G-SBO
345	2.303E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-CF-FSUN1,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-12-POT-4,1-POT-4,OEP-VCF-LP-GRWR
346	2.270E-9	0.02	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U301,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-POS-9-POT-4,1-POT-4
347	2.257E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H0,1-POS-6-POT-4,1-POT-4,/1-SD-FEED-G-SBO
348	2.233E-9	0.02	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205____,1-EPS-DGN-FS-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-5B-POT-4,1-POT-4
349	2.233E-9	0.02	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-EPS-DGN-FS-G4001____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-5B-POT-4,1-POT-4
350	2.217E-9	0.02	1-IE-SD-LORHR,1-DCP-BCH-FC-AAABBABB-CC,1-POS-8L-POT-4,1-POT-4
351	2.211E-9	0.02	1-IE-SD-LOOP,1-EPS-MDP-FS-XFERPPS____CC,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-12-POT-4,1-POT-4,OEP-VCF-LP-GRWR
352	2.197E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-10-POT-4,1-POT-4
353	2.197E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-SEQ-FO-1821U301,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-10-POT-4,1-POT-4
354	2.192E-9	0.02	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-EPS-DGN-FR-G4001____,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-10-POT-4,1-POT-4,/1-SD-FEED-G-SBO

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
355	2.192E-9	0.02	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205____,1-EPS-DGN-FR-G4002____,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-10-POT-4,1-POT-4,/1-SD-FEED-G-SBO
356	2.185E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-SEQ-FO-1821U301,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-5B-POT-4,1-POT-4,/1-SD-FEED-G-SBO
357	2.185E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-SEQ-FO-1821U302,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-5B-POT-4,1-POT-4,/1-SD-FEED-G-SBO
358	2.167E-9	0.02	1-IE-SD-LORHR,1-POS-5B-POT-4,1-POT-4,1-RHR-XHE-XD-3H,1-RHR-XHE-XL-LTREC
359	2.160E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H2,1-POS-12-POT-4,1-POT-4
360	2.156E-9	0.02	1-IE-LONSCW,1-IE-SWS-MDP-CR-24,1-OA-OSW-----H,1-POS-10-POT-4,1-POT-4
361	2.152E-9	0.02	1-IE-LONSCW,1-IE-SWS-MDP-CR-246,1-OA-OSW-----H,1-POS-8E-POT-4,1-POT-4
362	2.152E-9	0.02	1-IE-LONSCW,1-IE-SWS-MDP-CR-245,1-OA-OSW-----H,1-POS-8E-POT-4,1-POT-4
363	2.152E-9	0.02	1-IE-LONSCW,1-IE-SWS-MDP-CR-124,1-OA-OSW-----H,1-POS-8E-POT-4,1-POT-4
364	2.152E-9	0.02	1-IE-LONSCW,1-IE-SWS-MDP-CR-234,1-OA-OSW-----H,1-POS-8E-POT-4,1-POT-4
365	2.130E-9	0.02	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U301,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-POS-12-POT-4,1-POT-4,OEP-VCF-LP-GRWR
366	2.128E-9	0.02	1-IE-SD-LOOP,1-AFW-TDP-FS-P4001____,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR02H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
367	2.120E-9	0.02	1-IE-SD-LOOP,1-DCP-BAT-MA-ND1B____,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO
368	2.114E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-11-POT-4,1-POT-4
369	2.114E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-SEQ-FO-1821U301,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-11-POT-4,1-POT-4

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
370	2.109E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FR-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H2,1-POS-7-POT-4,1-POT-4,OEP-VCF-LP-GRWR
371	2.092E-9	0.02	1-IE-SD-LOOP,1-ACP-INV-FC-BD1I12____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4
372	2.077E-9	0.02	1-IE-SD-LOOP,1-DCP-BAT-MA-ND1B____,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-5A-POT-4,1-POT-4
373	2.076E-9	0.02	1-IE-SD-ODML-LATE,1-LPI-MDP-FS-RHRA____,1-LPI-MDP-FS-RHRB____,1-POS-10-POT-4-D
374	2.073E-9	0.02	1-IE-SD-LORHR,1-ACP-CRB-CF-A205301,1-OEP-VCF-LP-CLOPT,1-POS-5B-POT-4,1-POT-4
375	2.073E-9	0.02	1-IE-SD-LOOP,1-ACP-BAC-FC-MCCBBF____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
376	2.073E-9	0.02	1-IE-SD-LOOP,1-ACP-BAC-FC-MCCBBB____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
377	2.073E-9	0.02	1-IE-SD-LOOP,1-ACP-BAC-FC-BB07____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
378	2.054E-9	0.02	1-IE-SD-LO4160C,1-DCP-BCH-FC-AAABBABB-CC,1-POS-8L-POT-4,1-POT-4
379	2.051E-9	0.02	1-IE-SD-LORHR,1-DCP-BCH-FC-AA____BABB-CC,1-POS-8E-POT-4,1-POT-4
380	2.047E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H0,1-POS-13-POT-4,1-POT-4,OEP-VCF-LP-GRWR
381	2.032E-9	0.02	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205____,1-ACP-CRB-CC-BA0301____,1-EPS-XHE-XL-NR04H,1-POS-13-POT-4,1-POT-4,OEP-VCF-LP-GRWR
382	2.026E-9	0.02	1-IE-SD-LOOP,1-DCP-FUS-OP-BD104____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
383	2.024E-9	0.02	1-IE-SD-LOOP,1-AFW-TDP-FR-P4001____,1-EPS-SEQ-CF-FOAB,1-POS-13-POT-4,1-POT-4
384	2.007E-9	0.02	1-IE-SD-LO4160C,1-POS-5B-POT-4,1-POT-4,1-RHR-XHE-XD-3H,1-RHR-XHE-XL-LTREC

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
385	1.980E-9	0.02	1-IE-SD-LOOP,1-DCP-BAT-MA-ND1B____,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8L-POT-4,1-POT-4
386	1.957E-9	0.02	1-IE-SD-LOOP,1-LPI-MDP-CF-RUN,1-POS-5B-POT-4,1-POT-4
387	1.944E-9	0.02	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-NSCWCT-SPRAY,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,1-SWS-MOV-CC-1669A____
388	1.939E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FS-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-10-POT-4,1-POT-4
389	1.939E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-DGN-FS-G4001____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-10-POT-4,1-POT-4
390	1.929E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FS-G4002____,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-5B-POT-4,1-POT-4,/1-SD-FEED-G-SBO
391	1.929E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-DGN-FS-G4001____,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-5B-POT-4,1-POT-4,/1-SD-FEED-G-SBO
392	1.927E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H2,1-POS-11-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
393	1.921E-9	0.02	1-IE-SD-LO4160C,1-ACP-CRB-CF-A205301,1-OEP-VCF-LP-CLOPT,1-POS-5B-POT-4,1-POT-4
394	1.902E-9	0.02	1-IE-SD-LORHR,1-LPI-MDP-FS-RHRA____,1-LPI-MDP-FS-RHRB____,1-POS-8E-POT-4,1-POT-4
395	1.902E-9	0.02	1-IE-SD-LORHR,1-POS-8E-POT-4,1-POT-4,1-SD-HEP-JOINT-93
396	1.900E-9	0.02	1-IE-SD-LO4160C,1-DCP-BCH-FC-AA__BABB-CC,1-POS-8E-POT-4,1-POT-4
397	1.895E-9	0.02	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-ACP-CRB-OO-ANA0401____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO
398	1.891E-9	0.02	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U301,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-POS-10-POT-4,1-POT-4
399	1.887E-9	0.02	1-IE-SD-LOINV,1-POS-10-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XM-LOI-POS6

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
400	1.886E-9	0.02	1-IE-SD-LOOP,1-ACP-CRB-OO-ANA0401_,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4
401	1.878E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-12-POT-4,1-POT-4,OEP-VCF-LP-GRWR
402	1.878E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-SEQ-FO-1821U301,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-12-POT-4,1-POT-4,OEP-VCF-LP-GRWR
403	1.874E-9	0.02	1-IE-SD-LOOP,1-ACP-BAC-FC-BYB1____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
404	1.872E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-CF-FSUN1,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5B-POT-4,1-POT-4,/1-SD-FEED-G-SBO
405	1.866E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FS-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-11-POT-4,1-POT-4
406	1.866E-9	0.02	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-DGN-FS-G4001____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-OEP-XHE-XX-NR01H1,1-POS-11-POT-4,1-POT-4
407	1.865E-9	0.02	1-IE-SD-LOINV-DS,1-LOI-XHE-XM-ISO2-LT,1-POS-5B-POT-4,1-POT-4,1-RHR-XHE-XM-M5LOI-I
408	1.857E-9	0.02	1-IE-SD-LOOP,1-EPS-TNK-MA-DFOSTKB_,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-4-POT-4,1-POT-4,OEP-VCF-LP-GRWR
409	1.853E-9	0.02	1-IE-SD-LOOP,1-EPS-SEQ-CF-FOAB,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-POS-11-POT-4,1-POT-4,/1-SD-FEED-G-SBO
410	1.822E-9	0.02	1-IE-SD-LOOP,1-ACP-SSD-MA-1821U302,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4
411	1.820E-9	0.02	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-AFW-TDP-FR-P4001____,1-EPS-XHE-XL-NR02H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
412	1.820E-9	0.02	1-IE-SD-LOOP,1-AFW-TDP-FR-P4001____,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR02H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-4-POT-4,1-POT-4,OEP-VCF-LP-GRWR

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
413	1.820E-9	0.02	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U301,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-POS-11-POT-4,1-POT-4
414	1.804E-9	0.02	1-IE-SD-ODML-EARLY,1-EPS-DGN-FR-G4001____,1-LPI-XHE-XM-GFEED-MLDRL,1-OEP-VCF-LP-CLOPT,1-POS-6-POT-4-MLDRL,1-RHR-XHE-XM-MLDRL-I
415	1.804E-9	0.02	1-IE-SD-ODML-EARLY,1-EPS-DGN-FR-G4002____,1-LPI-XHE-XM-GFEED-MLDRL,1-OEP-VCF-LP-CLOPT,1-POS-6-POT-4-MLDRL,1-RHR-XHE-XM-MLDRL-I
416	1.766E-9	0.01	1-IE-SD-LO4160C,1-ACP-XHE-XL-ALTAC-6H,1-EPS-SEQ-FO-1821U302,1-POS-8E-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC-LT5H
417	1.762E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-VCF-LP-CLOPT,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
418	1.762E-9	0.01	1-IE-SD-LO4160C,1-LPI-MDP-FS-RHRA____,1-LPI-MDP-FS-RHRB____,1-POS-8E-POT-4,1-POT-4
419	1.762E-9	0.01	1-IE-SD-LO4160C,1-POS-8E-POT-4,1-POT-4,1-SD-HEP-JOINT-93
420	1.725E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FR-G4002____,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H2,1-POS-7-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
421	1.713E-9	0.01	1-IE-SD-ODML-EARLY,1-CCW-HTX-CF-TRNAB,1-POS-6-POT-4-MLDRL
422	1.712E-9	0.01	1-IE-SD-LOOP,1-ACP-INV-FC-BD1I12____,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO
423	1.706E-9	0.01	1-IE-SD-LORHR,1-DCP-BCH-FC-AAABBABB-CC,1-POS-5B-POT-4,1-POT-4
424	1.705E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-EPS-SEQ-FO-1821U301,1-LPI-XHE-XM-GFEED-LOOP,1-POS-6-POT-4,1-POT-4
425	1.705E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205____,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-POS-6-POT-4,1-POT-4
426	1.696E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-MCCBBF____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
427	1.696E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-MCCBBB____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
428	1.696E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-BB07____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
429	1.689E-9	0.01	1-IE-SD-LOOP,1-ACP-XHE-XL-ALTAC-6H,1-LPI-MDP-FS-RHRB____,1-POS-8E-POT-4,1-POT-4

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
430	1.689E-9	0.01	1-IE-SD-LOOP,1-DCP-BCH-MA-AD1CA____,1-LPI-MDP-FS-RHRB____,1-POS-8E-POT-4,1-POT-4
431	1.678E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-MA-MCCBBF____,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
432	1.678E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-MA-MCCBBB____,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
433	1.677E-9	0.01	1-IE-SD-LOOP,1-ACP-INV-FC-BD1I12____,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
434	1.673E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205____,1-EPS-DGN-FS-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-9-POT-4,1-POT-4
435	1.673E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-EPS-DGN-FS-G4001____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-9-POT-4,1-POT-4
436	1.667E-9	0.01	1-IE-SD-LOINV,1-LOI-XHE-XM-ISO2-LT,1-POS-6-POT-4,1-POT-4,1-RHR-XHE-XM-LOI-POS6
437	1.665E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-OO-ANA0401____,1-EPS-DGN-FS-G4002____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4
438	1.662E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-BB07____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
439	1.662E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-BB16____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
440	1.662E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-BA03____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
441	1.662E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-MCCBBB____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
442	1.662E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-MCCBBF____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
443	1.658E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FS-G4002____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-12-POT-4,1-POT-4,OEP-VCF-LP-GRWR
444	1.658E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-DGN-FS-G4001____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-12-POT-4,1-POT-4,OEP-VCF-LP-GRWR

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
445	1.655E-9	0.01	1-IE-SD-LORHR,1-LPI-MDP-CF-RUN,1-POS-7-POT-4,1-POT-4
446	1.645E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-246,1-OA-OSW----- H,1-POS-8L-POT-4,1-POT-4
447	1.645E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-245,1-OA-OSW----- H,1-POS-8L-POT-4,1-POT-4
448	1.645E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-124,1-OA-OSW----- H,1-POS-8L-POT-4,1-POT-4
449	1.645E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-234,1-OA-OSW----- H,1-POS-8L-POT-4,1-POT-4
450	1.631E-9	0.01	1-IE-SD-LO4160C,1-ACP-BAC-FC-AA02____,1-EPS-DGN-FR-G4002____,1-POS-5B-POT-4,1-POT-4
451	1.631E-9	0.01	1-IE-SD-LO4160C,1-ACP-BAC-FC-AB15____,1-EPS-DGN-FR-G4002____,1-POS-5B-POT-4,1-POT-4
452	1.624E-9	0.01	1-IE-SD-LORHR,1-POS-9-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XL-TRN-POS9
453	1.624E-9	0.01	1-IE-SD-LORHR,1-POS-9-POT-4,1-POT-4,1-RHR-XHE-XD-3H,1-RHR-XHE-XL-LTREC
454	1.620E-9	0.01	1-IE-SD-LOOP,1-DCP-BAT-MA-ND1B____,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8L-POT-4,1-POT-4,1-SD-FEED-G-SBO
455	1.618E-9	0.01	1-IE-SD-ODML-EARLY,1-LPI-CKV-CC-009010____-CC,1-POS-6-POT-4-MLDRL
456	1.599E-9	0.01	1-IE-SD-LOOP,1-ACP-INV-FC-BD1I12____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4
457	1.595E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-CF-FSUN1,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
458	1.584E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-MCCBBF____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR
459	1.584E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-MCCBBB____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR
460	1.584E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-BA03____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR
461	1.584E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-BB07____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
462	1.584E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-BB16____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR
463	1.567E-9	0.01	1-IE-SD-LORHR,1-DCP-BCH-FC-AA__BABBB-CC,1-POS-8L-POT-4,1-POT-4
464	1.559E-9	0.01	1-IE-SD-LO4160C,1-ACP-XHE-XL-ALTAC-6H,1-EPS-DGN-FS-G4002____,1-POS-8E-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC-LT5H
465	1.557E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-DGN-MA-G4001____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-7-POT-4,1-POT-4,OEP-VCF-LP-GRWR
466	1.554E-9	0.01	1-IE-SD-LORHR,1-ACP-CRB-CF-A205301,1-OEP-VCF-LP-CLOPT,1-POS-9-POT-4,1-POT-4
467	1.543E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-OO-ANA0401____,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO
468	1.543E-9	0.01	1-IE-SD-LORHR,1-LPI-MDP-CF-RUN,1-POS-6-POT-4,1-POT-4
469	1.541E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-DCP-BAT-MA-ND1B____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4
470	1.534E-9	0.01	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS4____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4
471	1.534E-9	0.01	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS1____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4
472	1.534E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-BYB1____,1-EPS-XHE-XL-NR04H,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
473	1.533E-9	0.01	1-IE-SD-LO4160C,1-LPI-MDP-CF-RUN,1-POS-7-POT-4,1-POT-4
474	1.532E-9	0.01	1-IE-SD-LOOP,1-EPS-MDP-FS-XFERPPS_-CC,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
475	1.523E-9	0.01	1-IE-SD-LOOP,1-LPI-XHE-XM-GFEED-LOOP,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR01H,1-POS-5B-POT-4,1-POT-4,1-SWS-MOV-CF-1668A69A
476	1.517E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-MA-BYB1____,1-EPS-XHE-XL-NR04H,1-NSCWCT-SPRAY,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
477	1.512E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-OO-ANA0401_,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
478	1.511E-9	0.01	1-IE-SD-ODML-EARLY,1-HPI-XHE-XR-XVM207,1-LPI-XHE-XM-GFEED-MD,1-POS-6-POT-4-MLDRL,1-RHR-XHE-XL-TRN-MLDRL
479	1.505E-9	0.01	1-IE-SD-LO4160C,1-POS-9-POT-4,1-POT-4,1-RHR-XHE-XD-3H,1-RHR-XHE-XL-LTREC
480	1.502E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-BYB1____,1-EPS-XHE-XL-NR04H,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
481	1.492E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-SEQ-FO-1821U301,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR30M,1-OEP-XHE-XX-NR30M1,1-POS-6-POT-4,1-POT-4
482	1.492E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR30M,1-OEP-XHE-XX-NR30M1,1-POS-6-POT-4,1-POT-4
483	1.490E-9	0.01	1-IE-SD-LOOP,1-ACP-SSD-MA-1821U302,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO
484	1.490E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-1236,1-OA-OSW-----H,1-POS-8E-POT-4,1-POT-4
485	1.490E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-1346,1-OA-OSW-----H,1-POS-8E-POT-4,1-POT-4
486	1.490E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-1234,1-OA-OSW-----H,1-POS-8E-POT-4,1-POT-4
487	1.490E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-2345,1-OA-OSW-----H,1-POS-8E-POT-4,1-POT-4
488	1.490E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-1245,1-OA-OSW-----H,1-POS-8E-POT-4,1-POT-4
489	1.490E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-2356,1-OA-OSW-----H,1-POS-8E-POT-4,1-POT-4
490	1.490E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-1256,1-OA-OSW-----H,1-POS-8E-POT-4,1-POT-4
491	1.490E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-1456,1-OA-OSW-----H,1-POS-8E-POT-4,1-POT-4
492	1.490E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-3456,1-OA-OSW-----H,1-POS-8E-POT-4,1-POT-4
493	1.489E-9	0.01	1-IE-SD-LOINV-SP,1-POS-6-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XM-LOISP-POS6
494	1.486E-9	0.01	1-IE-SD-ODML-LATE,1-CCW-HTX-CF-TRNAB,1-POS-10-POT-4-D
495	1.481E-9	0.01	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR02H,1-POS-4-POT-4,1-POT-4,1-SWS-MOV-CC-1669A____,OEP-VCF-LP-GRWR

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
496	1.474E-9	0.01	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-NSCWCT-BYPASS,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,1-SWS-MOV-CC-1669A___,OEP-VCF-LP-GRWR
497	1.467E-9	0.01	1-IE-SD-LOOP,1-LPI-MDP-CF-RUN,1-POS-9-POT-4,1-POT-4
498	1.460E-9	0.01	1-IE-SD-LOOP,1-ACP-SSD-MA-1821U302,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
499	1.458E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205___,1-ACP-CRB-CC-BA0301___,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5B-POT-4,1-POT-4,/1-SD-FEED-G-SBO
500	1.453E-9	0.01	1-IE-SD-LORHR,1-LPI-MDP-FS-RHRA___,1-LPI-MDP-FS-RHRB___,1-POS-8L-POT-4,1-POT-4
501	1.453E-9	0.01	1-IE-SD-LORHR,1-POS-8L-POT-4,1-POT-4,1-SD-HEP-JOINT-95
502	1.452E-9	0.01	1-IE-SD-LO4160C,1-DCP-BCH-FC-AA___BABB-CC,1-POS-8L-POT-4,1-POT-4
503	1.442E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301___,1-OEP-VCF-LP-CLOPT,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
504	1.441E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-OO-ANA0401___,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4
505	1.440E-9	0.01	1-IE-SD-LO4160C,1-ACP-CRB-CF-A205301,1-OEP-VCF-LP-CLOPT,1-POS-9-POT-4,1-POT-4
506	1.432E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-BYB1___,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR
507	1.430E-9	0.01	1-IE-SD-LO4160C,1-LPI-MDP-CF-RUN,1-POS-6-POT-4,1-POT-4
508	1.419E-9	0.01	1-IE-SD-LO4160C,1-ACP-CRB-CC-BA0301___,1-LPI-MDP-FS-RHRA___,1-POS-8E-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC-LT5H
509	1.403E-9	0.01	1-IE-SD-ODML-LATE,1-LPI-CKV-CC-009010___CC,1-POS-10-POT-4-D
510	1.399E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-12346,1-OA-OSW----H,1-POS-8E-POT-4,1-POT-4
511	1.399E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-12345,1-OA-OSW----H,1-POS-8E-POT-4,1-POT-4
512	1.399E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-12356,1-OA-OSW----H,1-POS-8E-POT-4,1-POT-4
513	1.399E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-23456,1-OA-OSW----H,1-POS-8E-POT-4,1-POT-4

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
514	1.399E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-13456,1-OA-OSW---- ---H,1-POS-8E-POT-4,1-POT-4
515	1.399E-9	0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-12456,1-OA-OSW---- ---H,1-POS-8E-POT-4,1-POT-4
516	1.398E-9	0.01	1-IE-SD-LOINV-DS,1-LOI-XHE-XM-ISO2-LT,1-POS-9- POT-4,1-POT-4,1-RHR-XHE-XM-M5LOI-I
517	1.394E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205___,1-EPS-DGN- FS-G4002___,1-LPI-XHE-XM-GFEED-LOOP,1-OEP- XHE-XL-NR01H,1-POS-10-POT-4,1-POT-4
518	1.394E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301___,1-EPS-DGN- FS-G4001___,1-LPI-XHE-XM-GFEED-LOOP,1-OEP- XHE-XL-NR01H,1-POS-10-POT-4,1-POT-4
519	1.393E-9	0.01	1-IE-SD-LOOP,1-HPI-XHE-XM-FFEED-MLDRL,1-LPI- XHE-XM-GFEED-LD,1-POS-10-POT-4,1-POT-4,1-RHR- XHE-XL-TRN-POS10
520	1.392E-9	0.01	1-IE-SD-LOOP,1-ACP-SSD-MA-1821U302,1-EPS-XHE- XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA- ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT- 4,1-POT-4
521	1.390E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002___,1-EPS-SEQ- FO-1821U301,1-LPI-XHE-XM-GFEED-LOOP,1-OEP- XHE-XL-NR01H,1-POS-5B-POT-4,1-POT-4
522	1.390E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-FS-G4001___,1-EPS-SEQ- FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-OEP- XHE-XL-NR01H,1-POS-5B-POT-4,1-POT-4
523	1.383E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-EPS-XHE-XL- NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX- NR02H0,1-POS-9-POT-4,1-POT-4,/1-SD-FEED-G- SBO,OEP-VCF-LP-GRWR
524	1.376E-9	0.01	1-IE-SD-LO4160C,1-DCP-BCH-MA-AD1CA___,1-EPS- DGN-FR-G4002___,1-POS-8L-POT-4,1-POT-4,1-RHR- XHE-XL-LTREC
525	1.376E-9	0.01	1-IE-SD-LO4160C,1-ACP-XHE-XL-ALTAC-6H,1-EPS- DGN-FR-G4002___,1-POS-8L-POT-4,1-POT-4,1-RHR- XHE-XL-LTREC
526	1.364E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002___,1-EPS-SEQ- FO-1821U301,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX- NR02H1,1-POS-10-POT-4,1-POT-4,/1-SD-FEED-G-SBO
527	1.364E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001___,1-EPS-SEQ- FO-1821U302,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX- NR02H1,1-POS-10-POT-4,1-POT-4,/1-SD-FEED-G-SBO
528	1.363E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-OO-ANA0401___,1-EPS-DGN- FS-G4002___,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL- NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO
529	1.361E-9	0.01	1-IE-SD-LORHR,1-CCW-HTX-CF-TRNAB,1-POS-8E- POT-4,1-POT-4

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
530	1.346E-9	0.01	1-IE-SD-LO4160C,1-LPI-MDP-FS-RHRA____,1-LPI-MDP-FS-RHRB____,1-POS-8L-POT-4,1-POT-4
531	1.346E-9	0.01	1-IE-SD-LO4160C,1-POS-8L-POT-4,1-POT-4,1-SD-HEP-JOINT-95
532	1.341E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205____,1-EPS-DGN-FS-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-11-POT-4,1-POT-4
533	1.341E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-EPS-DGN-FS-G4001____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-11-POT-4,1-POT-4
534	1.335E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-OO-ANA0401____,1-EPS-DGN-FS-G4002____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
535	1.317E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-DGN-FS-G4001____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR30M,1-OEP-XHE-XX-NR30M1,1-POS-6-POT-4,1-POT-4
536	1.317E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001____,1-EPS-DGN-FS-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR30M,1-OEP-XHE-XX-NR30M1,1-POS-6-POT-4,1-POT-4
537	1.308E-9	0.01	1-IE-SD-LOOP,1-ACP-INV-FC-BD1I12____,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO
538	1.305E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-CF-FSUN1,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
539	1.302E-9	0.01	1-IE-SD-LORHR,1-POS-11-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XL-TRN-POS9
540	1.302E-9	0.01	1-IE-SD-LORHR,1-POS-11-POT-4,1-POT-4,1-RHR-XHE-XD-3H,1-RHR-XHE-XL-LTREC
541	1.296E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-MCCBBB____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
542	1.296E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-MCCBBF____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
543	1.296E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-BB07____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
544	1.296E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-BB16____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
545	1.296E-9	0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-BA03____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
546	1.294E-9	0.01	1-IE-SD-LORHR,1-ACP-CRB-CF-A205301,1-OEP-VCF-LP-CLOPT,1-POS-10-POT-4,1-POT-4
547	1.291E-9	0.01	1-IE-SD-LOOP,1-ACP-XHE-XL-ALTAC-6H,1-LPI-MDP-FS-RHRB____,1-POS-8L-POT-4,1-POT-4
548	1.291E-9	0.01	1-IE-SD-LOOP,1-DCP-BCH-MA-AD1CA____,1-LPI-MDP-FS-RHRB____,1-POS-8L-POT-4,1-POT-4
549	1.288E-9	0.01	1-IE-SD-LOOP,1-DCP-BCH-FC-AAABBABB-CC,1-POS-8E-POT-4,1-POT-4
550	1.285E-9	0.01	1-IE-SD-LORHR,1-LPI-CKV-CC-009010__-CC,1-POS-8E-POT-4,1-POT-4
551	1.279E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-CF-FSUN1,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
552	1.279E-9	0.01	1-IE-SD-LORHR,1-DCP-BCH-FC-AAABBABB-CC,1-POS-9-POT-4,1-POT-4
553	1.274E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002____,1-EPS-DGN-MA-G4001____,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-7-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
554	1.273E-9	0.01	1-IE-SD-LORHR,1-EPS-SEQ-CF-FOAB,1-OEP-VCF-LP-CLOPT,1-POS-5B-POT-4,1-POT-4
555	1.273E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-OO-ANA0401__,1-EPS-DGN-FS-G4002____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4
556	1.265E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-AFW-TDP-FS-P4001____,1-EPS-XHE-XL-NR02H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
557	1.265E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-EPS-SEQ-FO-1821U301,1-EPS-XHE-XL-NR04H,1-POS-13-POT-4,1-POT-4,OEP-VCF-LP-GRWR
558	1.265E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205____,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-POS-13-POT-4,1-POT-4,OEP-VCF-LP-GRWR
559	1.261E-9	0.01	1-IE-SD-LO4160C,1-CCW-HTX-CF-TRNAB,1-POS-8E-POT-4,1-POT-4
560	1.261E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-DCP-BAT-MA-ND1B____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO
561	1.255E-9	0.01	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS1____,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO
562	1.255E-9	0.01	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS4____,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
563	1.253E-9	0.01	1-IE-SD-LOOP,1-EPS-MDP-FS-XFERPPS_-CC,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
564	1.246E-9	0.01	1-IE-SD-LORHR,1-ACP-CRB-CF-A205301,1-OEP-VCF-LP-CLOPT,1-POS-11-POT-4,1-POT-4
565	1.242E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H0,1-POS-12-POT-4,1-POT-4
566	1.235E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-DCP-BAT-MA-ND1B____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
567	1.233E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205__,1-ACP-CRB-CC-BA0301__,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-POS-12-POT-4,1-POT-4
568	1.230E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205__,1-EPS-DGN-FR-G4002____,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-6-POT-4,1-POT-4,/1-SD-FEED-G-SBO
569	1.230E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-EPS-DGN-FR-G4001____,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-6-POT-4,1-POT-4,/1-SD-FEED-G-SBO
570	1.229E-9	0.01	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS1__,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
571	1.229E-9	0.01	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS4__,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
572	1.227E-9	0.01	1-IE-SD-LOOP,1-EPS-MDP-FS-XFERPPS_-CC,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,OEP-VCF-LP-GRWR
573	1.227E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-FS-G4001____,1-EPS-DGN-FS-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-5B-POT-4,1-POT-4
574	1.223E-9	0.01	1-IE-SD-LO4160C,1-ACP-BAC-FC-AA02____,1-EPS-DGN-FR-G4002____,1-POS-9-POT-4,1-POT-4
575	1.223E-9	0.01	1-IE-SD-LO4160C,1-ACP-BAC-FC-AB15____,1-EPS-DGN-FR-G4002____,1-POS-9-POT-4,1-POT-4
576	1.222E-9	0.01	1-IE-SD-LOOP,1-LPI-MDP-CF-RUN,1-POS-10-POT-4,1-POT-4
577	1.219E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-CF-FSUN1,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR
578	1.213E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H0,1-POS-7-POT-4,1-POT-4,OEP-VCF-LP-GRWR

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
579	1.206E-9	0.01	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-NSCWCT-BYPASS,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,1-SWS-MOV-CC-1669A___,OEP-VCF-LP-GRWR
580	1.204E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-FR-G4002___,1-EPS-DGN-FS-G4001___,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-10-POT-4,1-POT-4,/1-SD-FEED-G-SBO
581	1.204E-9	0.01	1-IE-SD-LOOP,1-EPS-DGN-FR-G4001___,1-EPS-DGN-FS-G4002___,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-10-POT-4,1-POT-4,/1-SD-FEED-G-SBO
582	1.204E-9	0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205___,1-ACP-CRB-CC-BA0301___,1-LPI-XHE-XM-GFEED-LOOP,1-POS-7-POT-4,1-POT-4,OEP-VCF-LP-GRWR
583	1.199E-9	0.01	1-IE-SD-LO4160C,1-ACP-CRB-CF-A205301,1-OEP-VCF-LP-CLOPT,1-POS-10-POT-4,1-POT-4
584	1.191E-9	0.01	1-IE-SD-LO4160C,1-LPI-CKV-CC-009010___CC,1-POS-8E-POT-4,1-POT-4
585	1.185E-9	< 0.01	1-IE-SD-LOINV-DS,1-POS-5B-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XM-M5LOI-I
586	1.181E-9	< 0.01	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-NSCWCT-BYPASS,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4,1-SWS-MOV-CC-1669A___,OEP-VCF-LP-GRWR
587	1.179E-9	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-OO-ANA0401___,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO
588	1.179E-9	< 0.01	1-IE-SD-LO4160C,1-EPS-SEQ-CF-FOAB,1-OEP-VCF-LP-CLOPT,1-POS-5B-POT-4,1-POT-4
589	1.177E-9	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301___,1-DCP-BAT-MA-ND1B___,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4
590	1.176E-9	< 0.01	1-IE-SD-LOOP,1-LPI-MDP-CF-RUN,1-POS-11-POT-4,1-POT-4
591	1.172E-9	< 0.01	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS4___,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4
592	1.172E-9	< 0.01	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS1___,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4
593	1.172E-9	< 0.01	1-IE-SD-LOOP,1-ACP-BAC-FC-BYB1___,1-EPS-XHE-XL-NR04H,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
594	1.170E-9	< 0.01	1-IE-SD-LOOP,1-EPS-MDP-FS-XFERPPS_-CC,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,OEP-VCF-LP-GRWR
595	1.169E-9	< 0.01	1-IE-SD-LOOP,1-EPS-DGN-CF-FSUN1,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-10-POT-4,1-POT-4,/1-SD-FEED-G-SBO
596	1.142E-9	< 0.01	1-IE-SD-LOOP,1-LPI-XHE-XM-GFEED-LOOP,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR01H,1-POS-9-POT-4,1-POT-4,1-SWS-MOV-CF-1668A69A
597	1.139E-9	< 0.01	1-IE-SD-LOOP,1-ACP-SSD-MA-1821U302,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO
598	1.139E-9	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-1236,1-OA-OSW-----H,1-POS-8L-POT-4,1-POT-4
599	1.139E-9	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-1346,1-OA-OSW-----H,1-POS-8L-POT-4,1-POT-4
600	1.139E-9	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-1234,1-OA-OSW-----H,1-POS-8L-POT-4,1-POT-4
601	1.139E-9	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-2345,1-OA-OSW-----H,1-POS-8L-POT-4,1-POT-4
602	1.139E-9	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-1245,1-OA-OSW-----H,1-POS-8L-POT-4,1-POT-4
603	1.139E-9	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-2356,1-OA-OSW-----H,1-POS-8L-POT-4,1-POT-4
604	1.139E-9	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-1256,1-OA-OSW-----H,1-POS-8L-POT-4,1-POT-4
605	1.139E-9	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-1456,1-OA-OSW-----H,1-POS-8L-POT-4,1-POT-4
606	1.139E-9	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-3456,1-OA-OSW-----H,1-POS-8L-POT-4,1-POT-4
607	1.133E-9	< 0.01	1-IE-SD-LOOP,1-AFW-TDP-FR-P4001____,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR02H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
608	1.127E-9	< 0.01	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-NSCWCT-BYPASS,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,1-SWS-MOV-CC-1669A____,OEP-VCF-LP-GRWR
609	1.118E-9	< 0.01	1-IE-SD-LORHR,1-POS-5B-POT-4,1-POT-4,1-SD-HEP-JOINT-62
610	1.118E-9	< 0.01	1-IE-SD-LORHR,1-LPI-MDP-FS-RHRA____,1-LPI-MDP-FS-RHRB____,1-POS-5B-POT-4,1-POT-4
611	1.115E-9	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-EPS-DGN-FR-G4001____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-13-POT-4,1-POT-4,OEP-VCF-LP-GRWR

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
612	1.115E-9	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205__,1-EPS-DGN-FR-G4002__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-13-POT-4,1-POT-4,OEP-VCF-LP-GRWR
613	1.109E-9	< 0.01	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H0,1-POS-11-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
614	1.097E-9	< 0.01	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-VCF-LP-CLOPT,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
615	1.087E-9	< 0.01	1-IE-SD-LOINV-OP,1-ACP-CRB-CF-A205301,1-LPI-XHE-XM-GFEED-LOI5A,1-OEP-VCF-LP-CLOPT,1-POS-5A-POT-4,1-POT-4
616	1.082E-9	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-AFW-TDP-FR-P4001__,1-EPS-XHE-XL-NR02H,1-OEP-XHE-XL-NR02H,1-POS-4-POT-4,1-POT-4,OEP-VCF-LP-GRWR
617	1.069E-9	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-12346,1-OA-OSW----H,1-POS-8L-POT-4,1-POT-4
618	1.069E-9	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-12345,1-OA-OSW----H,1-POS-8L-POT-4,1-POT-4
619	1.069E-9	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-12356,1-OA-OSW----H,1-POS-8L-POT-4,1-POT-4
620	1.069E-9	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-23456,1-OA-OSW----H,1-POS-8L-POT-4,1-POT-4
621	1.069E-9	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-13456,1-OA-OSW----H,1-POS-8L-POT-4,1-POT-4
622	1.069E-9	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-12456,1-OA-OSW----H,1-POS-8L-POT-4,1-POT-4
623	1.065E-9	< 0.01	1-IE-SD-LORHR,1-DCP-BCH-FC-AAABBABB-CC,1-POS-10-POT-4,1-POT-4
624	1.061E-9	< 0.01	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U301,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-POS-6-POT-4,1-POT-4
625	1.061E-9	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205__,1-EPS-DGN-FS-G4002__,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR30M,1-POS-6-POT-4,1-POT-4
626	1.061E-9	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-EPS-DGN-FS-G4001__,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR30M,1-POS-6-POT-4,1-POT-4
627	1.059E-9	< 0.01	1-IE-SD-LOINV,1-POS-6-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XM-LOI-POS6
628	1.055E-9	< 0.01	1-IE-SD-LOOP,1-HPI-XHE-XM-FFEED-6H,1-POS-5B-POT-4,1-POT-4,1-RHR-XHE-XL-TRN-3H

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
629	1.042E-9	< 0.01	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002____,1-EPS-SEQ-FO-1821U301,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-9-POT-4,1-POT-4
630	1.042E-9	< 0.01	1-IE-SD-LOOP,1-EPS-DGN-FS-G4001____,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-9-POT-4,1-POT-4
631	1.041E-9	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-OO-ANA0401____,1-EPS-DGN-FS-G4002____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,1-SD-FEED-G-SBO
632	1.040E-9	< 0.01	1-IE-SD-LORHR,1-CCW-HTX-CF-TRNAB,1-POS-8L-POT-4,1-POT-4
633	1.036E-9	< 0.01	1-IE-SD-LO4160C,1-POS-5B-POT-4,1-POT-4,1-SD-HEP-JOINT-62
634	1.033E-9	< 0.01	1-IE-SD-ODML-EARLY,1-HPI-XHE-XR-XVM207,1-LPI-XHE-XM-GFEED-MLDRL,1-POS-6-POT-4-MLDRL,1-RHR-XHE-XM-MLDRL-I
635	1.030E-9	< 0.01	1-IE-SD-LOOP,1-EPS-MDP-FR-XFERPPS____CC,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-5B-POT-4,1-POT-4
636	1.025E-9	< 0.01	1-IE-SD-LORHR,1-DCP-BCH-FC-AAABBABB-CC,1-POS-11-POT-4,1-POT-4
637	1.018E-9	< 0.01	1-IE-SD-LO4160C,1-ACP-BAC-FC-AA02____,1-EPS-DGN-FR-G4002____,1-POS-10-POT-4,1-POT-4
638	1.018E-9	< 0.01	1-IE-SD-LO4160C,1-ACP-BAC-FC-AB15____,1-EPS-DGN-FR-G4002____,1-POS-10-POT-4,1-POT-4
639	1.000E-9	< 0.01	1-IE-SD-LOOP,1-AFW-TDP-FR-P4001____,1-EPS-DGN-FS-G4002____,1-EPS-XHE-XL-NR02H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-5A-POT-4,1-POT-4
640	9.980E-10	< 0.01	1-IE-SD-LOOP,1-ACP-BAC-MA-MCCBBF____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-4-POT-4,1-POT-4,OEP-VCF-LP-GRWR
641	9.980E-10	< 0.01	1-IE-SD-LOOP,1-ACP-BAC-MA-MCCBBB____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-4-POT-4,1-POT-4,OEP-VCF-LP-GRWR
642	9.975E-10	< 0.01	1-IE-SD-LOOP,1-EPS-DGN-CF-FSUN1,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
643	9.969E-10	< 0.01	1-IE-SD-LOOP,1-ACP-INV-FC-BD1I12____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-4-POT-4,1-POT-4,OEP-VCF-LP-GRWR
644	9.926E-10	< 0.01	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H0,1-POS-7-POT-4,1-POT-4,1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
645	9.858E-10	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-EPS-DGN-FS-G4001____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-12-POT-4,1-POT-4,OEP-VCF-LP-GRWR

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
646	9.858E-10	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205__,1-EPS-DGN-FS-G4002__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-12-POT-4,1-POT-4,OEP-VCF-LP-GRWR
647	9.853E-10	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205__,1-ACP-CRB-CC-BA0301__,1-POS-7-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
648	9.845E-10	< 0.01	1-IE-SD-LOOP,1-DCP-BCH-FC-AAABBABB-CC,1-POS-8L-POT-4,1-POT-4
649	9.823E-10	< 0.01	1-IE-SD-LORHR,1-LPI-CKV-CC-009010__-CC,1-POS-8L-POT-4,1-POT-4
650	9.733E-10	< 0.01	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,1-SWS-RLY-FC-AX3_69AB,OEP-VCF-LP-GRWR
651	9.733E-10	< 0.01	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,1-SWS-RLY-FC-162_1X69,OEP-VCF-LP-GRWR
652	9.685E-10	< 0.01	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002__,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-VCF-LP-CLOPT,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
653	9.655E-10	< 0.01	1-IE-SD-LO4160C,1-EPS-DGN-FR-G4002__,1-POS-7-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC-LT5H,1-RHR-XHE-XL-STBY-POS7
654	9.635E-10	< 0.01	1-IE-SD-LO4160C,1-CCW-HTX-CF-TRNAB,1-POS-8L-POT-4,1-POT-4
655	9.634E-10	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301__,1-DCP-BAT-MA-ND1B__,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO
656	9.623E-10	< 0.01	1-IE-SD-LOOP,1-POS-5B-POT-4,1-POT-4,1-RHR-XHE-XD-3H,1-RHR-XHE-XL-LTREC
657	9.591E-10	< 0.01	1-IE-SD-LOOP,1-DCP-BAT-MA-ND1B__,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4
658	9.588E-10	< 0.01	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS1__,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO
659	9.588E-10	< 0.01	1-IE-SD-LOOP,1-ACP-DCP-FC-1B_PS4__,1-EPS-XHE-XL-NR04H,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO
660	9.576E-10	< 0.01	1-IE-SD-LOOP,1-EPS-MDP-FS-XFERPPS_-CC,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
661	9.542E-10	< 0.01	1-IE-SD-LORHR,1-EPS-SEQ-CF-FOAB,1-OEP-VCF-LP-CLOPT,1-POS-9-POT-4,1-POT-4

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
662	9.508E-10	< 0.01	1-IE-SD-LORHR,1-LPI-MDP-FS-RHRB____,1-POS-8E-POT-4,1-POT-4,1-RHR-XHE-XL-TRA-POS8E
663	9.508E-10	< 0.01	1-IE-SD-LOOP,1-LPI-XHE-XM-GFEED-LOOP,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR01H,1-POS-10-POT-4,1-POT-4,1-SWS-MOV-CF-1668A69A
664	9.498E-10	< 0.01	1-IE-SD-LOOP,1-ACP-TFW-MA-SAT____,1-EPS-DGN-FR-G4002____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H1,1-POS-8E-POT-4,1-POT-4
665	9.488E-10	< 0.01	1-IE-SD-LOOP,1-ACP-TFW-FC-BB16X____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,OEP-VCF-LP-GRWR
666	9.487E-10	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-246,1-OA-OSW-----H,1-POS-9-POT-4,1-POT-4
667	9.487E-10	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-245,1-OA-OSW-----H,1-POS-9-POT-4,1-POT-4
668	9.487E-10	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-124,1-OA-OSW-----H,1-POS-9-POT-4,1-POT-4
669	9.487E-10	< 0.01	1-IE-LONSCW,1-IE-SWS-MDP-CR-234,1-OA-OSW-----H,1-POS-9-POT-4,1-POT-4
670	9.257E-10	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-EPS-DGN-MA-G4001____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-7-POT-4,1-POT-4,OEP-VCF-LP-GRWR
671	9.217E-10	< 0.01	1-IE-SD-LOOP,1-EPS-XHE-XL-NR04H,1-NSCWCT-BYPASS,1-OEP-XHE-XL-NR02H,1-POS-8L-POT-4,1-POT-4,/1-SD-FEED-G-SBO,1-SWS-MOV-CC-1669A____,OEP-VCF-LP-GRWR
672	9.196E-10	< 0.01	1-IE-SD-LOOP,1-EPS-DGN-FS-G4001____,1-EPS-DGN-FS-G4002____,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-9-POT-4,1-POT-4
673	9.149E-10	< 0.01	1-IE-SD-LOOP,1-LPI-XHE-XM-GFEED-LOOP,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR01H,1-POS-11-POT-4,1-POT-4,1-SWS-MOV-CF-1668A69A
674	9.110E-10	< 0.01	1-IE-SD-LOOP,1-DCP-BCH-FC-AA__BABB-CC,1-POS-8E-POT-4,1-POT-4
675	9.104E-10	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205____,1-ACP-CRB-CC-BA0301____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-10-POT-4,1-POT-4,/1-SD-FEED-G-SBO
676	9.100E-10	< 0.01	1-IE-SD-LO4160C,1-LPI-CKV-CC-009010____-CC,1-POS-8L-POT-4,1-POT-4
677	9.076E-10	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-BA0301____,1-EPS-SEQ-FO-1821U301,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5B-POT-4,1-POT-4,/1-SD-FEED-G-SBO
678	9.076E-10	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205____,1-EPS-SEQ-FO-1821U302,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5B-POT-4,1-POT-4,/1-SD-FEED-G-SBO

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
679	9.022E-10	< 0.01	1-IE-SD-LOOP,1-ACP-BAC-MA-BYB1____,1-EPS-XHE-XL-NR04H,1-NSCWCT-SPRAY,1-OEP-XHE-XL-NR02H,1-POS-4-POT-4,1-POT-4,OEP-VCF-LP-GRWR
680	8.976E-10	< 0.01	1-IE-SD-LOOP,1-EPS-SEQ-FO-1821U302,1-OEP-VCF-LP-CLOPT,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4,/1-SD-FEED-G-SBO,OEP-VCF-LP-GRWR
681	8.878E-10	< 0.01	1-IE-SD-LOINV-DS,1-POS-9-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC,1-RHR-XHE-XM-M5LOI-I
682	8.840E-10	< 0.01	1-IE-SD-LO4160C,1-EPS-SEQ-CF-FOAB,1-OEP-VCF-LP-CLOPT,1-POS-9-POT-4,1-POT-4
683	8.829E-10	< 0.01	1-IE-SD-LO4160C,1-EPS-SEQ-FO-1821U302,1-LPI-MDP-FS-RHRA____,1-POS-8E-POT-4,1-POT-4,1-RHR-XHE-XL-LTREC-LT5H
684	8.809E-10	< 0.01	1-IE-SD-LO4160C,1-LPI-MDP-FS-RHRB____,1-POS-8E-POT-4,1-POT-4,1-RHR-XHE-XL-TRA-POS8E
685	8.755E-10	< 0.01	1-IE-SD-LOINV,1-LOI-XHE-XM-ISO2-LT,1-POS-5B-POT-4,1-POT-4,1-RHR-XHE-XM-LOI-POS5B
686	8.680E-10	< 0.01	1-IE-SD-LOOP,1-ACP-SSD-MA-1821U302,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-4-POT-4,1-POT-4,OEP-VCF-LP-GRWR
687	8.676E-10	< 0.01	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002____,1-EPS-SEQ-FO-1821U301,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-10-POT-4,1-POT-4
688	8.676E-10	< 0.01	1-IE-SD-LOOP,1-EPS-DGN-FS-G4001____,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-10-POT-4,1-POT-4
689	8.646E-10	< 0.01	1-IE-SD-LOINV-OP,1-ACP-INV-MA-AD1111____,1-L2-BE-RHRVSTUCK-SD,1-LOI-XHE-XD-POS5A,1-LPI-XHE-XM-GFEED-LOI5A,1-OA-NSCWGAN---H,1-POS-5A-POT-4,1-POT-4
690	8.606E-10	< 0.01	1-IE-SD-LOOP,1-EPS-DGN-CF-FRUN1,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OA-ALIGNPW-G-H,1-OEP-XHE-XL-NR02H,1-OEP-XHE-XX-NR02H0,1-POS-8E-POT-4,1-POT-4
691	8.542E-10	< 0.01	1-IE-SD-LOOP,1-AFW-TDP-FS-P4001____,1-EPS-SEQ-CF-FOAB,1-POS-12-POT-4,1-POT-4
692	8.514E-10	< 0.01	1-IE-SD-LOOP,1-EPS-DGN-CF-FSUN1,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-13-POT-4,1-POT-4,OEP-VCF-LP-GRWR
693	8.468E-10	< 0.01	1-IE-SD-LOOP,1-DCP-BAT-MA-ND1B____,1-EPS-DGN-FS-G4002____,1-EPS-XHE-XL-NR04H,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR02H,1-POS-8E-POT-4,1-POT-4
694	8.446E-10	< 0.01	1-IE-SD-LOOP,1-LPI-MDP-FS-RHRA____,1-LPI-MDP-FS-RHRB____,1-POS-8E-POT-4,1-POT-4
695	8.446E-10	< 0.01	1-IE-SD-LOOP,1-POS-8E-POT-4,1-POT-4,1-SD-HEP-JOINT-93

Table B-1 Top Cut Sets Contributing 95% of Total CDF (cont.)

#	Prob/Freq	Total %	Cut Set
696	8.382E-10	< 0.01	1-IE-SD-LORHR,1-LPI-MDP-FS-RHRA____,1-LPI-MDP-FS-RHRB____,1-POS-9-POT-4,1-POT-4
697	8.348E-10	< 0.01	1-IE-SD-LOOP,1-EPS-DGN-FS-G4002____,1-EPS-SEQ-FO-1821U301,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-11-POT-4,1-POT-4
698	8.348E-10	< 0.01	1-IE-SD-LOOP,1-EPS-DGN-FS-G4001____,1-EPS-SEQ-FO-1821U302,1-LPI-XHE-XM-GFEED-LOOP,1-OEP-XHE-XL-NR01H,1-POS-11-POT-4,1-POT-4
699	8.173E-10	< 0.01	1-IE-SD-LOOP,1-EPS-MDP-FS-XFERPPS____CC,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-13-POT-4,1-POT-4,OEP-VCF-LP-GRWR
700	8.013E-10	< 0.01	1-IE-SD-LOOP,1-ACP-CRB-CC-AA0205____,1-EPS-DGN-FS-G4002____,1-EPS-XHE-XL-NR04H,1-OEP-XHE-XL-NR02H,1-POS-5B-POT-4,1-POT-4,/1-SD-FEED-G-SBO

B.2 Dominant Cutsets by Release Category

The following importance measures are calculated:

- Fussell-Vesely (F-V) – an indication of the percentage of the overall risk metric result (CDF) contributed by the cut sets containing the basic event.
- Risk Increase Ratio (RIR) – also referred to as risk achievement worth; an indication of how much the overall risk metric (CDF) would go up if the specific event had probability equal to 1.0, corresponding to totally unreliable equipment or action.
- Risk Reduction Ratio (RRR) – also referred to as risk reduction worth; an indication of how much the overall risk metric (CDF) would be reduced if the specific event probability equaled zero, corresponding to a totally reliable piece of equipment or action.
- Birnbaum – an indication of the sensitivity of the overall risk metric (CDF) with respect to the basic event of concern.

The importance measures for all significant basic events are included in Table B-2. Significant basic events are defined as those basic events that have a F-V importance greater than 0.005 or a risk achievement worth greater than 2 (ASME/ANS, 2014). In addition to the importance measures listed above, Table B-2 also includes the basic event name, the number of cut sets in which that basic event appears (listed under column heading *Count*), the calculated probability or frequency associated with the basic event, and the basic event description.

The importance measure results were generated using model version number 419.

Table B-2 Importance Measures

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-POT-4	19699	5.530E-02	8.84E-01	1.61E+01	8.616E+00	2.004E-04	PLANT OPERATING TYPE - 4 (REFUELING)
1-IE-SD-LOOP	14011	1.630E-01	6.81E-01	4.50E+00	3.136E+00	5.238E-05	LOSS OF OFF-SITE POWER - SHUTDOWN
1-EPS-XHE-XL-NR04H	6252	6.984E-01	4.55E-01	1.20E+00	1.835E+00	8.166E-06	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
1-OEP-XHE-XL-NR02H	9019	3.264E-01	4.33E-01	1.90E+00	1.765E+00	1.665E-05	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
OEP-VCF-LP-GRWR	4071	4.100E-01	3.48E-01	1.50E+00	1.534E+00	1.065E-05	SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
1-LPI-XHE-XM- GFEED-LOOP	4881	5.500E-01	2.84E-01	1.23E+00	1.396E+00	6.463E-06	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
1-POS-8E-POT-4	3333	9.370E-02	2.60E-01	3.51E+00	1.351E+00	3.475E-05	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
1-POS-8L-POT-4	2478	7.160E-02	1.93E-01	3.51E+00	1.240E+00	3.385E-05	REFUELING NEW CORE (POS-8L) FOR REFUELING (POT-4)
1-EPS-DGN-FR- G4002__	2158	3.297E-02	1.85E-01	6.43E+00	1.227E+00	7.039E-05	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
1-OEP-XHE-XX- NR02H1	1532	2.729E-01	1.69E-01	1.45E+00	1.204E+00	7.774E-06	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
1-POS-5A-POT-4	1461	4.130E-02	1.63E-01	4.79E+00	1.195E+00	4.952E-05	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
1-ACP-CRB-CF- A205301	214	3.498E-04	1.29E-01	3.70E+02	1.149E+00	4.628E-03	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
1-ACP-CRB-CC- BA0301__	1528	5.350E-03	1.08E-01	2.11E+01	1.121E+00	2.535E-04	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
1-IE-SD-ODML-EARLY	417	5.880E-01	9.79E-02	1.07E+00	1.109E+00	2.087E-06	OVER DRAINING IN MID-LOOP - Before Refueling
1-POS-6-POT-4- MLDRL	417	4.070E-03	9.79E-02	2.50E+01	1.109E+00	3.015E-04	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)- (demand-based)

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-RHR-XHE-XL-LTREC	311	1.550E-02	9.29E-02	6.90E+00	1.102E+00	7.517E-05	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
1-OA-ALIGNPW-G-H	2281	9.200E-02	7.81E-02	1.77E+00	1.085E+00	1.064E-05	OPERATOR FAILS TO ALIGN Alternate offsite AC - General case
1-LPI-MDP-CF-START	62	4.878E-05	7.20E-02	1.47E+03	1.078E+00	1.836E-02	CCF OF RHR PUMPS A & B TO START
1-EPS-SEQ-FO-1821U302	1309	3.330E-03	6.76E-02	2.12E+01	1.072E+00	2.544E-04	SEQUENCER B FAILS TO OPERATE
1-EPS-SEQ-CF-FOAB	238	2.148E-04	6.50E-02	3.03E+02	1.070E+00	3.788E-03	CCF OF SEQUENCERS TO OPERATE
1-IE-SD-LOINV-OP	432	6.070E-01	6.18E-02	1.04E+00	1.066E+00	1.277E-06	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
1-L2-BE-RHRVSTUCK-SD	171	6.694E-02	6.14E-02	1.86E+00	1.065E+00	1.150E-05	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
1-LOI-XHE-XD-POS5A	171	2.500E-02	6.14E-02	3.40E+00	1.065E+00	3.080E-05	OPERATOR FAILS TO DIAGNOSE LOI EARLY
1-POS-5B-POT-4	2107	5.510E-02	5.95E-02	2.02E+00	1.063E+00	1.353E-05	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
1-EPS-DGN-FS-G4002	1171	2.940E-03	5.78E-02	2.06E+01	1.061E+00	2.465E-04	DG1B RANDOMLY FAILS TO START
1-IE-SD-LORHR	1852	3.670E-01	5.37E-02	1.09E+00	1.057E+00	1.833E-06	LOSS OF RHR - SHUTDOWN
1-IE-LONSCW	164	1.000E+00	5.10E-02	1.00E+00	1.054E+00	6.394E-07	LOSS OF NSCW
1-RHR-XHE-XL-TRN-MLDRL	84	1.250E-02	4.76E-02	4.76E+00	1.050E+00	4.776E-05	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN -MLDRL
1-IE-SWS-MDP-CR-123456	8	3.031E-05	4.22E-02	1.38E+03	1.044E+00	1.732E-02	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-POS-10-POT-4	1623	3.440E-02	3.94E-02	2.11E+00	1.041E+00	1.435E-05	MID-LOOP OPERATION (POS-10) AFTER REFUELING (POT-4)
1-RHR-XHE-XM-MLDRL-I	85	1.020E-02	3.83E-02	4.72E+00	1.040E+00	4.707E-05	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING -ML
1-POS-9-POT-4	1804	4.130E-02	3.72E-02	1.86E+00	1.039E+00	1.129E-05	DRAINING REACTOR TO MID-LOOP (POS-9) AFTER REFUELING (POT-4)

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-RHR-XHE-XL-LTREC-LT5H	432	1.505E-01	3.36E-02	1.19E+00	1.035E+00	2.802E-06	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
1-POS-12-POT-4	1055	7.440E-02	3.36E-02	1.42E+00	1.035E+00	5.656E-06	REACTOR COOLANT SYSTEM HEATUP/DRAW BUBBLE IN PRESSURIZER (POS-12) AFTER REFUELING (POT-4)
1-POS-6-POT-4	1433	1.930E-02	3.23E-02	2.64E+00	1.033E+00	2.098E-05	MID-LOOP OPERATION (POS-6) FOR REFUELING (POT-4)
1-IE-SD-LO4160C	2842	3.400E-01	2.99E-02	1.06E+00	1.031E+00	1.101E-06	LOSS OF 4160 VAC BUS - SHUTDOWN
1-RHR-XHE-XM-TRIP-MD	13	1.460E-01	2.75E-02	1.16E+00	1.028E+00	2.357E-06	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS (MODERATE DEPENDENCY)
1-POS-11-POT-4	1306	3.310E-02	2.18E-02	1.64E+00	1.022E+00	8.258E-06	REFILLING REACTOR (POS-11) AFTER REFUELING (POT-4)
1-OEP-XHE-XL-NR01H	2304	5.196E-01	2.10E-02	1.02E+00	1.021E+00	5.070E-07	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 1 HOUR
1-HPI-XHE-XM-FFEED-MLDRL	8	7.950E-03	2.04E-02	3.55E+00	1.021E+00	3.220E-05	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-MLDRL
1-IE-SD-ODML-LATE	241	5.100E-01	1.82E-02	1.02E+00	1.019E+00	4.468E-07	OVER DRAINING IN MID-LOOP - After Refueling
1-POS-10-POT-4-D	241	4.070E-03	1.82E-02	5.45E+00	1.019E+00	5.599E-05	MID-LOOP OPERATION (POS-10) AFTER REFUELING (POT-4)-(demand-based)
1-EPS-DGN-FR-G4001____	1486	3.297E-02	1.66E-02	1.49E+00	1.017E+00	6.328E-06	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
1-POS-7-POT-4	1747	2.070E-02	1.63E-02	1.77E+00	1.017E+00	9.861E-06	FILLING CAVITY (POS-7) FOR REFUELING (POT-4)
1-POS-4-POT-4	497	5.510E-03	1.35E-02	3.44E+00	1.014E+00	3.071E-05	COOLDOWN TO AMBIENT TEMP WITH RHR SYSTEM (POS-4) FOR REFUELING (POT-4)
1-POS-13-POT-4	630	2.750E-02	1.24E-02	1.44E+00	1.013E+00	5.647E-06	REACTOR COOLANT SYSTEM HEATUP TO 350 °F (POS-13) AFTER REFUELING (POT-4)
1-LPI-XHE-XM-GFEED-LD	29	4.520E-01	1.17E-02	1.01E+00	1.012E+00	3.233E-07	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-(LOW DEPENDENCY)-LOI

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-NSCWCT-SPRAY	1858	9.040E-01	1.12E-02	1.00E+00	1.011E+00	1.553E-07	NSCW CTS IN SPRAY MODE (FRACTION OF TIME)
1-RHR-XHE-XL-TRN-POS6	87	1.250E-02	1.10E-02	1.87E+00	1.011E+00	1.101E-05	OPERATORS FAIL TO RECOVER ONE RHR TRAIN
1-LPI-XHE-XM-GFEED-MD	81	5.050E-01	1.06E-02	1.01E+00	1.011E+00	2.639E-07	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-(MODERATE DEPENDENCY)-MLDRL
1-ACP-CRB-CC-AA0205__	1107	5.350E-03	1.04E-02	2.93E+00	1.010E+00	2.426E-05	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
1-OA-OSW-----H	140	2.000E-02	8.83E-03	1.43E+00	1.009E+00	5.532E-06	OPERATORS FAIL TO ESTABLISH SINGLE PUMP NSCW PUMP OPERATION
1-OEP-VCF-LP-CLOPT	2066	5.300E-03	8.41E-03	2.58E+00	1.008E+00	1.989E-05	CONSEQUENTIAL LOSS OF OFFSITE POWER - TRANSIENT
1-SWS-MOV-CC-1669A__	541	3.530E-04	7.00E-03	2.08E+01	1.007E+00	2.487E-04	NSCW CT B SPRAY VALVE HV1669A FAILS TO OPEN ON DEMAND
1-LPI-MDP-CF-RUN	31	3.940E-06	6.84E-03	1.72E+03	1.007E+00	2.152E-02	CCF OF RHR PUMPS A & B TO RUN
1-EPS-DGN-CF-FRUN1	218	3.237E-04	6.82E-03	2.21E+01	1.007E+00	2.639E-04	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN
1-EPS-SEQ-FO-1821U301	939	3.330E-03	6.72E-03	3.01E+00	1.007E+00	2.531E-05	SEQUENCER A FAILS TO OPERATE
1-OEP-XHE-XX-NR01H1	536	2.258E-01	6.61E-03	1.02E+00	1.007E+00	3.672E-07	CONVOLUTION FACTOR FOR 1FTR-OPR (1HR AVAIL)
1-LOI-XHE-XM-ISO2-LT	14	2.440E-02	5.83E-03	1.23E+00	1.006E+00	2.994E-06	FAIL TO ISOLATE LOI AFTER SUCCESSFUL INJECTION
1-LPI-MDP-FS-RHRB__	723	1.000E-03	5.79E-03	6.79E+00	1.006E+00	7.261E-05	RHR PUMP B RANDOMLY FAILS TO START
1-AFW-TDP-FR-P4001	633	3.802E-02	5.32E-03	1.14E+00	1.005E+00	1.753E-06	TDAFWP (P4-001) FAILS TO RUN
1-ACP-INV-FC-BD1112	333	2.148E-04	4.23E-03	2.07E+01	1.004E+00	2.468E-04	INVERTER 1BD1112 RANDOMLY FAILS
1-ACP-XHE-XL-ALTAC-6H	444	2.000E-03	4.12E-03	3.05E+00	1.004E+00	2.580E-05	FAIL TO ALIGN ALTERNATE OFFSITE POWER BY PW VIA SAT OR OPPOSITE TRAIN RAT IN 6 HOURS
1-RHR-XHE-XD-ML	102	1.250E-03	4.11E-03	4.28E+00	1.004E+00	4.119E-05	OPERATORS FAIL TO DIAGNOSE LORHR BEFORE BOILING DURING MID-LOOP

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-EPS-DGN-FS-G4001	799	2.940E-03	4.05E-03	2.37E+00	1.004E+00	1.727E-05	DG1A RANDOMLY FAILS TO START
1-RHR-XHE-XL-TRN-ML-P10	4	1.250E-03	3.96E-03	4.17E+00	1.004E+00	3.973E-05	OPERATORS FAIL TO RECOVER ONE RHR TRAIN
1-RHR-XHE-XM-POS10-I	6	1.200E-03	3.84E-03	4.19E+00	1.004E+00	4.008E-05	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING -ML
1-LPI-MDP-FS-RHRA	665	1.000E-03	3.64E-03	4.64E+00	1.004E+00	4.569E-05	RHR PUMP A RANDOMLY FAILS TO START
1-ACP-SSD-MA-1821U302	115	1.870E-04	3.41E-03	1.92E+01	1.003E+00	2.287E-04	SEQUENCER B UNAVAILABLE DUE TO MAINTENANCE
1-IE-SD-LOINV	69	6.400E-05	3.39E-03	5.39E+01	1.003E+00	6.628E-04	LOSS OF INVENTORY - SHUTDOWN - RCS vented, break causes RCS leak
1-ACP-DCP-FC-1B_PS1__	302	1.574E-04	3.10E-03	2.07E+01	1.003E+00	2.467E-04	FAILURE OF 48V SEQUENCER POWER SUPPLY PS-1
1-ACP-DCP-FC-1B_PS4__	302	1.574E-04	3.10E-03	2.07E+01	1.003E+00	2.467E-04	FAILURE OF 28V SEQUENCER POWER SUPPLY PS-4
1-RHR-XHE-XL-TRN-3H	116	1.250E-03	3.01E-03	3.41E+00	1.003E+00	3.018E-05	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN
1-EPS-DGN-CF-FSUN1	125	3.675E-05	2.94E-03	8.10E+01	1.003E+00	1.003E-03	CCF OF UNIT 1 DGNs G4001/G4002 TO START
1-EPS-MDP-FS-XFERPPS__CC	155	3.528E-05	2.69E-03	7.71E+01	1.003E+00	9.534E-04	CCF OF DG FUEL TRANSFER PUMPS TO START
1-DCP-BCH-FC-AAABBABB-CC	24	1.525E-06	2.25E-03	1.47E+03	1.002E+00	1.835E-02	CCF OF BCHs 1AD1CA, 1AD1CB, 1BD1CA, & 1BD1CB
1-RHR-XHE-XL-TRN-POS10	6	1.250E-03	1.92E-03	2.54E+00	1.002E+00	1.929E-05	OPERATORS FAIL TO RECOVER ONE RHR TRAIN
1-EPS-TNK-MA-DFOSTKB_	74	4.000E-04	1.68E-03	5.20E+00	1.002E+00	5.269E-05	TR B. DIESEL FUEL OIL STORAGE TANK 1-2403-T4-002 IN MAINTENANCE
1-ACP-BAC-FC-BA03	399	4.776E-05	1.59E-03	3.43E+01	1.002E+00	4.177E-04	4.16KV BUS 1BA03 FAILS
1-ACP-BAC-FC-BB16	393	4.776E-05	1.59E-03	3.43E+01	1.002E+00	4.173E-04	480V SWGR 1BB16 RANDOMLY FAILS
1-DCP-FUS-OP-BD104__	212	7.464E-05	1.47E-03	2.06E+01	1.001E+00	2.461E-04	SUPPLY CURRENT FUSE BETWEEN CRB 1BD104 & INVERTER FAILS

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-RHR-XHE-XL-TRB-POS8E	3	2.500E-05	1.35E-03	5.51E+01	1.001E+00	6.783E-04	OPERATORS FAIL TO RECOVER OPERATING RHR TRAIN B
1-CCW-HTX-CF-TRNAB	31	7.157E-07	1.24E-03	1.72E+03	1.001E+00	2.152E-02	CCF OF CCW HTXs E4-001 & 002
1-IE-SWS-MDP-CR-24	4	5.667E-05	1.21E-03	2.23E+01	1.001E+00	2.665E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-LPI-CKV-CC-009010__-CC	31	6.760E-07	1.17E-03	1.72E+03	1.001E+00	2.152E-02	CCF OF RHR PUMPS DISCHARGE CKVs 009 & 010 TO OPEN
1-ACP-BAC-FC-AA02	400	4.776E-05	1.13E-03	2.46E+01	1.001E+00	2.962E-04	4.16KV BUS 1AA02 FAILS
1-ACP-BAC-FC-AB15	384	4.776E-05	1.11E-03	2.43E+01	1.001E+00	2.923E-04	480V SWGR 1AB15 RANDOMLY FAILS
1-SWS-MOV-CF-1668A69A	167	1.187E-05	1.02E-03	8.70E+01	1.001E+00	1.078E-03	CCF OF NSCW CT SPRAY VALVES HV1668A & 1669A TO OPEN
1-RHR-XHE-XL-TRA-POS8E	227	5.000E-04	1.02E-03	3.04E+00	1.001E+00	2.557E-05	Operators fail to restore standby RHR train A
1-ACP-BAC-FC-MCCBBB	170	4.776E-05	9.61E-04	2.11E+01	1.001E+00	2.521E-04	480V MCC 1BBB RANDOMLY FAILS
1-RHR-XHE-XD-3H	76	1.250E-04	9.34E-04	8.47E+00	1.001E+00	9.370E-05	OPERATORS FAIL TO DIAGNOSE LORHR BEFORE BOILING STARTS (SHUTDOWN)
1-ACP-BAC-FC-BYB1	197	4.776E-05	9.33E-04	2.05E+01	1.001E+00	2.448E-04	120/240V PANEL 1BYB1 FAILS
1-ACP-BAC-FC-BB07	152	4.776E-05	9.21E-04	2.03E+01	1.001E+00	2.416E-04	480V SWGR 1BB07 RANDOMLY FAILS
1-ACP-BAC-FC-MCCBBF	152	4.776E-05	9.21E-04	2.03E+01	1.001E+00	2.416E-04	480V MCC 1BBF FAILS
1-ACP-BAC-MA-MCCBBB	59	2.150E-04	9.11E-04	5.24E+00	1.001E+00	5.313E-05	480V MCC 1BBB IN MAINTENANCE
1-ACP-BAC-MA-MCCBBF	58	2.150E-04	9.07E-04	5.22E+00	1.001E+00	5.286E-05	480V MCC 1BBF IN MAINTENANCE
1-ACP-BAC-MA-BYB1	78	2.150E-04	9.04E-04	5.21E+00	1.001E+00	5.272E-05	120/240V PANEL 1BYB1 IN MAINTENANCE
1-DCP-BCH-FC-AA__BABB-CC	10	1.079E-06	6.85E-04	6.34E+02	1.001E+00	7.933E-03	CCF OF BCHs 1AD1CA, 1BD1CA, & 1BD1CB
1-SWS-SWT-FC-TY16689B-CC	74	1.170E-05	6.33E-04	5.51E+01	1.001E+00	6.781E-04	CCF OF NSCW RETURN WATER TEMP SWITCHES TY1668B&1669B

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-EPS-MDP-FR-XFERPPS_-CC	83	7.259E-06	5.50E-04	7.67E+01	1.001E+00	9.491E-04	CCF OF DG FUEL TRANSFER PUMPS TO RUN
1-ACP-BAC-MA-AB15	140	2.150E-04	5.18E-04	3.41E+00	1.001E+00	3.019E-05	480V SWGR 1AB15 IN MAINTENANCE
1-ACP-TFW-FC-BB16X	239	1.526E-05	5.03E-04	3.40E+01	1.001E+00	4.131E-04	XFMR 1BB16X FAILS
1-SWS-MOV-CC-1668A	343	3.530E-04	4.93E-04	2.40E+00	1.000E+00	1.750E-05	NSCW CT A SPRAY VALVE HV1668A FAILS TO OPEN ON DEMAND
1-SWS-RLY-FC-AX3_69AB	142	2.480E-05	4.81E-04	2.04E+01	1.000E+00	2.432E-04	NSCW RELAY AX3 FOR OPENING/CLOSING 1HV1669A/B FAILS
1-IE-SWS-MDP-CR-124	4	2.077E-05	4.42E-04	2.23E+01	1.000E+00	2.665E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-IE-SWS-MDP-CR-234	4	2.077E-05	4.42E-04	2.23E+01	1.000E+00	2.665E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-IE-SWS-MDP-CR-245	4	2.077E-05	4.42E-04	2.23E+01	1.000E+00	2.665E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-IE-SWS-MDP-CR-246	4	2.077E-05	4.42E-04	2.23E+01	1.000E+00	2.665E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-SWS-RLY-FC-162_1X69	107	2.480E-05	4.33E-04	1.84E+01	1.000E+00	2.186E-04	RELAY 162-1X FOR OPENING HV1669A/B RANDOMLY FAILS
1-RHR-XHE-XL-TRN-POS9	15	1.250E-04	4.17E-04	4.34E+00	1.000E+00	4.183E-05	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN
1-IE-SWS-MDP-CR-1234	8	1.438E-05	4.00E-04	2.88E+01	1.000E+00	3.488E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-IE-SWS-MDP-CR-1236	8	1.438E-05	4.00E-04	2.88E+01	1.000E+00	3.488E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-IE-SWS-MDP-CR-1245	8	1.438E-05	4.00E-04	2.88E+01	1.000E+00	3.488E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-IE-SWS-MDP-CR-1256	8	1.438E-05	4.00E-04	2.88E+01	1.000E+00	3.488E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-IE-SWS-MDP-CR-1346	8	1.438E-05	4.00E-04	2.88E+01	1.000E+00	3.488E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-IE-SWS-MDP-CR-1456	8	1.438E-05	4.00E-04	2.88E+01	1.000E+00	3.488E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-IE-SWS-MDP-CR-2345	8	1.438E-05	4.00E-04	2.88E+01	1.000E+00	3.488E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-IE-SWS-MDP-CR-2356	8	1.438E-05	4.00E-04	2.88E+01	1.000E+00	3.488E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-IE-SWS-MDP-CR-3456	8	1.438E-05	4.00E-04	2.88E+01	1.000E+00	3.488E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-ACP-BAC-MA-AA02	84	2.150E-04	3.97E-04	2.85E+00	1.000E+00	2.317E-05	BUS 1AA02 IN MAINTENANCE
1-IE-SWS-MDP-CR-12346	17	1.350E-05	3.77E-04	2.89E+01	1.000E+00	3.501E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-IE-SWS-MDP-CR-12345	15	1.350E-05	3.77E-04	2.89E+01	1.000E+00	3.498E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-IE-SWS-MDP-CR-12356	8	1.350E-05	3.76E-04	2.88E+01	1.000E+00	3.488E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-IE-SWS-MDP-CR-12456	8	1.350E-05	3.76E-04	2.88E+01	1.000E+00	3.488E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-IE-SWS-MDP-CR-13456	8	1.350E-05	3.76E-04	2.88E+01	1.000E+00	3.488E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-IE-SWS-MDP-CR-23456	8	1.350E-05	3.76E-04	2.88E+01	1.000E+00	3.488E-04	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-SD-HEP-JOINT-93	3	1.000E-06	3.60E-04	3.60E+02	1.000E+00	4.502E-03	Joint HEP minimum threshold for HFE combination 93
1-ACP-DPL-FC-BY2B	114	1.802E-05	3.53E-04	2.06E+01	1.000E+00	2.452E-04	PANEL 1BY2B FAILS
1-ACP-TFW-FC-AB15X	228	1.526E-05	3.52E-04	2.40E+01	1.000E+00	2.887E-04	XFMR 1AB15X FAILS
1-SWS-MDP-CF-FS-ABCDEF	57	4.211E-06	3.36E-04	8.08E+01	1.000E+00	1.000E-03	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-LPI-MDP-FR-RHRB	253	5.842E-05	3.27E-04	6.60E+00	1.000E+00	7.024E-05	RHR PUMP B RANDOMLY FAILS TO RUN (24 HR MISSION)
1-HPI-XHE-XR-XVM207	95	1.000E-04	3.20E-04	4.20E+00	1.000E+00	4.012E-05	OPERATORS FAIL TO RESTORE RWST XVM 207 AFTER T&M
1-ACP-INV-FC-AD1111	210	2.148E-04	3.08E-04	2.44E+00	1.000E+00	1.799E-05	INVERTER 1AD1111 RANDOMLY FAILS

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-ACP-TFW-FC-1BSEQT1	106	1.526E-05	2.98E-04	2.05E+01	1.000E+00	2.447E-04	SEQUENCER XFMR T1 FAILS
1-ACP-TFW-FC-1BSEQT2	106	1.526E-05	2.98E-04	2.05E+01	1.000E+00	2.447E-04	SEQUENCER XFMR T3 FAILS
1-ACP-TFW-FC-BBB03X	109	1.526E-05	2.94E-04	2.03E+01	1.000E+00	2.418E-04	480V MCC XFMR 1BBB03X FAILS
1-ACP-TFW-FC-BB07X	89	1.526E-05	2.92E-04	2.01E+01	1.000E+00	2.397E-04	XFMR 1BB07X FAILS
1-HPI-CKV-CF-CLALL	29	1.625E-07	2.82E-04	1.72E+03	1.000E+00	2.152E-02	CCF OF COLD LEG CKVs 083, 084, 085, & 086 TO OPEN
1-LPI-CKV-CF-CLALL	29	1.625E-07	2.82E-04	1.72E+03	1.000E+00	2.152E-02	CCF OF RHR COLD LEG CKVs 147, 148, 149, & 150 TO OPEN
1-SD-HEP-JOINT-95	3	1.000E-06	2.75E-04	2.76E+02	1.000E+00	3.441E-03	Joint HEP minimum threshold for HFE combination 95
1-SD-HEP-JOINT-62	4	1.000E-06	2.73E-04	2.74E+02	1.000E+00	3.423E-03	Joint HEP minimum threshold for HFE combination 62
1-DCP-BAT-FC-BD1B	98	1.404E-05	2.73E-04	2.05E+01	1.000E+00	2.441E-04	BATTERY 1BD1B RANDOMLY FAILS (125V)
1-LPI-MDP-FR-RHRA	228	5.842E-05	2.47E-04	5.22E+00	1.000E+00	5.293E-05	RHR PUMP A RANDOMLY FAILS TO RUN (24 HR MISSION)
1-ACP-DCP-FC-1A_PS1	186	1.574E-04	2.24E-04	2.43E+00	1.000E+00	1.786E-05	FAILURE OF 48V SEQUENCER POWER SUPPLY PS-1
1-ACP-DCP-FC-1A_PS4	186	1.574E-04	2.24E-04	2.43E+00	1.000E+00	1.786E-05	FAILURE OF 28V SEQUENCER POWER SUPPLY PS-4
1-EPS-MOT-CF-START	47	2.801E-06	2.22E-04	8.03E+01	1.000E+00	9.946E-04	CCF OF DG ROOM VENT FANS FAIL FROM COMMON CAUSE TO START
1-DCP-DPL-FC-BD11	142	5.640E-06	1.81E-04	3.31E+01	1.000E+00	4.029E-04	DISTRIBUTION PANEL 1BD11 FAILS
1-ACP-CRB-CO-BA0309	138	5.400E-06	1.73E-04	3.31E+01	1.000E+00	4.021E-04	FEEDER CRB 1BA03 SPURIOUSLY OPENS - 1BA03 TO 1BB16X
1-ACP-CRB-CO-BB1601	138	5.400E-06	1.73E-04	3.31E+01	1.000E+00	4.021E-04	SUPPLY CRB 1BB16 SPURIOUSLY OPENS - 1BB16X TO 1BB16
1-DCP-CRB-CO-BD105	138	5.400E-06	1.73E-04	3.31E+01	1.000E+00	4.021E-04	SUPPLY CRB 1BD105 FROM BUS 1BD1 TO 1BD11 SPURIOUSLY OPENS

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-SWS-MDP-CF-FS-ABCD	43	2.000E-06	1.59E-04	8.04E+01	1.000E+00	9.955E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-ABCF	43	2.000E-06	1.59E-04	8.04E+01	1.000E+00	9.955E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-ABDE	43	2.000E-06	1.59E-04	8.04E+01	1.000E+00	9.955E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-ABEF	43	2.000E-06	1.59E-04	8.04E+01	1.000E+00	9.955E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-ACDF	43	2.000E-06	1.59E-04	8.04E+01	1.000E+00	9.955E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-ADEF	43	2.000E-06	1.59E-04	8.04E+01	1.000E+00	9.955E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-BCDE	43	2.000E-06	1.59E-04	8.04E+01	1.000E+00	9.955E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-BCEF	43	2.000E-06	1.59E-04	8.04E+01	1.000E+00	9.955E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-CDEF	43	2.000E-06	1.59E-04	8.04E+01	1.000E+00	9.955E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-BD	69	8.309E-06	1.59E-04	2.01E+01	1.000E+00	2.394E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-BF	69	8.309E-06	1.59E-04	2.01E+01	1.000E+00	2.394E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-DF	69	8.309E-06	1.59E-04	2.01E+01	1.000E+00	2.394E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-DCP-BDC-FC-BD1	112	5.640E-06	1.52E-04	2.79E+01	1.000E+00	3.367E-04	125V BUS 1BD1 FAILS
1-SWS-MDP-CF-FS-ABCDE	41	1.870E-06	1.48E-04	8.03E+01	1.000E+00	9.945E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-ABCDF	41	1.870E-06	1.48E-04	8.03E+01	1.000E+00	9.945E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-ABCEF	41	1.870E-06	1.48E-04	8.03E+01	1.000E+00	9.945E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-SWS-MDP-CF-FS-ABDEF	41	1.870E-06	1.48E-04	8.03E+01	1.000E+00	9.945E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-ACDEF	41	1.870E-06	1.48E-04	8.03E+01	1.000E+00	9.945E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-BCDEF	41	1.870E-06	1.48E-04	8.03E+01	1.000E+00	9.945E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-DCP-CRB-CO-BD1101__	107	5.400E-06	1.44E-04	2.77E+01	1.000E+00	3.350E-04	CRB BD1101 SPURIOUSLY OPENS ON LOAD SHED LOGIC CIRCUITS
1-SWS-MDP-CF-FR-ABCDEF	45	8.363E-08	1.44E-04	1.71E+03	1.000E+00	2.138E-02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-ACP-BAC-FC-AB05__	142	4.776E-05	1.41E-04	3.96E+00	1.000E+00	3.707E-05	480V SWGR 1AB05 RANDOMLY FAILS
1-DCP-BCH-FC-AD1CA__	79	6.504E-05	1.29E-04	2.98E+00	1.000E+00	2.479E-05	BCH 1AD1CA RANDOMLY FAILS
1-SWS-RLY-FC-AX36869_-CC	64	1.538E-06	1.22E-04	8.00E+01	1.000E+00	9.899E-04	CCF OF AX3 RELAYS FOR OPEN/CLOSE NSCW MOVs 1HV1668A/B & 1669A/B AFTER LOSP
1-ACP-CRB-CO-AA0210__	131	5.400E-06	1.18E-04	2.29E+01	1.000E+00	2.748E-04	FEEDER CRB 1AA02 SPURIOUSLY OPENS - 1AA02 TO 1AB15X
1-ACP-CRB-CO-AB1501__	131	5.400E-06	1.18E-04	2.29E+01	1.000E+00	2.748E-04	SUPPLY CRB 1AB15 SPURIOUSLY OPENS - 1AB15X TO 1AB15
1-SWS-RLY-FC-162_1X89-CC	40	1.538E-06	1.10E-04	7.27E+01	1.000E+00	8.985E-04	CCF OF RELAYS 162-1X FOR OPENING HV1668A /BAND 1669A /B AFTER LOSP FAILS
1-ACP-CRB-CO-BB1609__	62	5.400E-06	1.06E-04	2.06E+01	1.000E+00	2.450E-04	FEEDER CRB 1BB16 SPURIOUSLY OPENS - 1BB16 TO 1BBB
1-DCP-FUS-OP-AD104__	123	7.464E-05	1.04E-04	2.40E+00	1.000E+00	1.750E-05	SUPPLY CURRENT FUSE BETWEEN CRB 1AD104 & INVERTER FAILS
1-ACP-CRB-CO-BY2B02__	60	5.400E-06	1.03E-04	2.01E+01	1.000E+00	2.396E-04	CRB 1BY2B02 BETWEEN INVERTER 1BD1112 & 1BY2B SPURIOUSLY OPENS
1-DCP-CRB-CO-BD101__	60	5.400E-06	1.03E-04	2.01E+01	1.000E+00	2.396E-04	CRB FROM BATTERY 1BD1B TO BUS 1BD1 SPURIOUSLY OPENS
1-DCP-CRB-CO-BD104__	60	5.400E-06	1.03E-04	2.01E+01	1.000E+00	2.396E-04	CRB 1BD104 BETWEEN INVERTER 1BD1112 & 1BD1 SPURIOUSLY OPENS

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-DCP-CRB-CO-BY2B08__	60	5.400E-06	1.03E-04	2.01E+01	1.000E+00	2.396E-04	CRB SPURIOUSLY OPENS (BY2B08 TO SEQUENCER B)
1-DCP-CRB-CO-BD111024	59	5.400E-06	1.03E-04	2.00E+01	1.000E+00	2.386E-04	CRB BD1110 TO FAN CONTROL LOGIC SPURIOUSLY OPENS
1-DCP-CRB-CO-BD1104__	55	5.400E-06	1.03E-04	2.00E+01	1.000E+00	2.380E-04	CRB BD1104 SPURIOUSLY OPENS ON LOAD SHED LOGIC CIRCUITS
1-ACP-CRB-CO-BBB45__	69	5.400E-06	1.02E-04	1.99E+01	1.000E+00	2.373E-04	480V MCC CRB 1BBB45 SPURIOUSLY OPENS
1-ACP-CRB-CO-BYB116__	69	5.400E-06	1.02E-04	1.99E+01	1.000E+00	2.373E-04	120/240V CRB 1BYB116 SPURIOUSLY OPENS
1-ACP-CRB-CO-BA0304__	51	5.400E-06	1.02E-04	1.98E+01	1.000E+00	2.356E-04	FEEDER CRB 1BA03 SPURIOUSLY OPENS - 1BA03 TO 1BB07X
1-ACP-CRB-CO-BB0701__	51	5.400E-06	1.02E-04	1.98E+01	1.000E+00	2.356E-04	SUPPLY CRB 1BB07 SPURIOUSLY OPENS - 1BB07X TO 1BB07
1-ACP-CRB-CO-BB0714__	51	5.400E-06	1.02E-04	1.98E+01	1.000E+00	2.356E-04	CRB FROM 480V SWGR 1BB07 TO 480V MCC 1BBF SPURIOUSLY OPENS
1-EPS-PND-CF-1205X	36	1.216E-06	9.60E-05	7.99E+01	1.000E+00	9.884E-04	CCF OF DG VENT DAMPERS
1-ACP-INV-FC-AD11BD12-CC	37	1.207E-06	9.53E-05	7.99E+01	1.000E+00	9.894E-04	CCF OF INVERTERS 1AD1111/1BD1112
1-ACP-BAC-FC-MCCABA__	66	4.776E-05	9.35E-05	2.96E+00	1.000E+00	2.454E-05	480V MCC 1ABA RANDOMLY FAILS
1-SD-HEP-JOINT-63	2	1.000E-06	8.05E-05	8.15E+01	1.000E+00	1.009E-03	Joint HEP minimum threshold for HFE combination 63
1-EPS-MOT-CF-RUN	35	1.011E-06	7.96E-05	7.98E+01	1.000E+00	9.874E-04	CCF OF DG ROOM VENT FANS TO RUN
1-SD-HEP-JOINT-92	3	1.000E-06	7.94E-05	8.04E+01	1.000E+00	9.956E-04	Joint HEP minimum threshold for HFE combination 92
1-SWS-MDP-FR-P4_001__	63	3.816E-05	7.47E-05	2.96E+00	1.000E+00	2.454E-05	NSCW PUMP 1 FAILS TO RUN
1-SWS-MDP-FR-P4_003__	63	3.816E-05	7.47E-05	2.96E+00	1.000E+00	2.454E-05	NSCW PUMP 3 FAIL TO RUN
1-SWS-MOV-CF-116-ABCDEF	36	8.698E-07	6.88E-05	8.00E+01	1.000E+00	9.905E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MDP-CF-FR-ABCD	40	3.967E-08	6.81E-05	1.70E+03	1.000E+00	2.130E-02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-SWS-MDP-CF-FR-ABCF	40	3.967E-08	6.81E-05	1.70E+03	1.000E+00	2.130E-02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ABDE	40	3.967E-08	6.81E-05	1.70E+03	1.000E+00	2.130E-02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ABEF	40	3.967E-08	6.81E-05	1.70E+03	1.000E+00	2.130E-02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ACDF	40	3.967E-08	6.81E-05	1.70E+03	1.000E+00	2.130E-02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ADEF	40	3.967E-08	6.81E-05	1.70E+03	1.000E+00	2.130E-02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-BCDE	40	3.967E-08	6.81E-05	1.70E+03	1.000E+00	2.130E-02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-BCEF	40	3.967E-08	6.81E-05	1.70E+03	1.000E+00	2.130E-02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-CDEF	40	3.967E-08	6.81E-05	1.70E+03	1.000E+00	2.130E-02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ABCDE	40	3.724E-08	6.39E-05	1.70E+03	1.000E+00	2.130E-02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ABCDF	40	3.724E-08	6.39E-05	1.70E+03	1.000E+00	2.130E-02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ABCEF	40	3.724E-08	6.39E-05	1.70E+03	1.000E+00	2.130E-02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ABDEF	40	3.724E-08	6.39E-05	1.70E+03	1.000E+00	2.130E-02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ACDEF	40	3.724E-08	6.39E-05	1.70E+03	1.000E+00	2.130E-02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-BCDEF	40	3.724E-08	6.39E-05	1.70E+03	1.000E+00	2.130E-02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-CCW-HTX-PG-TRNB	107	1.217E-05	6.26E-05	6.15E+00	1.000E+00	6.450E-05	CCW HTX E4-002 IS UNAVAILABLE
1-SWS-CTF-CF-R-ABCDEFGH	40	3.632E-08	6.23E-05	1.70E+03	1.000E+00	2.130E-02	System Generated Event based upon Rasp CCF event : 1-SWS-CTF-CF-R
1-ACP-BAC-FC-MCCABB	87	4.776E-05	6.10E-05	2.28E+00	1.000E+00	1.600E-05	480V MCC 1ABB RANDOMLY FAILS

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-ACP-BAC-FC-MCCABF	90	4.776E-05	6.08E-05	2.27E+00	1.000E+00	1.596E-05	480V MCC 1ABF FAILS
1-SWS-RLY-FC-162_1ALL-CC	35	7.440E-07	5.87E-05	7.99E+01	1.000E+00	9.893E-04	RELAYS 162-1 ASSOC WITH OPENING OF HV-11600
1-ACP-BAC-FC-AYB1	100	4.776E-05	5.75E-05	2.20E+00	1.000E+00	1.510E-05	120/240V PANEL 1AYB1 FAILS
1-SD-HEP-JOINT-31	4	1.000E-06	5.68E-05	5.77E+01	1.000E+00	7.112E-04	Joint HEP minimum threshold for HFE combination 31
1-LPI-CKV-CC-010_____	92	1.070E-05	5.39E-05	6.04E+00	1.000E+00	6.319E-05	RHRP B Discharge CKV 010 RANDOMLY FAILS TO OPEN
1-SWS-MDP-CF-FS-ABD	35	2.907E-06	5.38E-05	1.95E+01	1.000E+00	2.322E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-ABF	35	2.907E-06	5.38E-05	1.95E+01	1.000E+00	2.322E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-ADF	35	2.907E-06	5.38E-05	1.95E+01	1.000E+00	2.322E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-BCD	35	2.907E-06	5.38E-05	1.95E+01	1.000E+00	2.322E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-BCF	35	2.907E-06	5.38E-05	1.95E+01	1.000E+00	2.322E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-BDE	35	2.907E-06	5.38E-05	1.95E+01	1.000E+00	2.322E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-BDF	35	2.907E-06	5.38E-05	1.95E+01	1.000E+00	2.322E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-BEF	35	2.907E-06	5.38E-05	1.95E+01	1.000E+00	2.322E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-CDF	35	2.907E-06	5.38E-05	1.95E+01	1.000E+00	2.322E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-DEF	35	2.907E-06	5.38E-05	1.95E+01	1.000E+00	2.322E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SD-HEP-JOINT-60	1	1.000E-06	5.36E-05	5.46E+01	1.000E+00	6.718E-04	Joint HEP minimum threshold for HFE combination 60
1-DCP-BDC-FC-AD1_____	89	5.640E-06	5.03E-05	9.92E+00	1.000E+00	1.118E-04	125V BUS 1AD1 FAILS

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-DCP-DPL-FC-AD11	89	5.640E-06	5.03E-05	9.92E+00	1.000E+00	1.118E-04	DISTRIBUTION PANEL 1AD11 FAILS
1-SWS-RLY-FC-AX4_69AB	36	2.480E-05	4.88E-05	2.97E+00	1.000E+00	2.465E-05	RELAY AX4 FOR OPENING NSCW 1HV1669A/B RANDOMLY FAILS
1-ACP-TFW-FC-AB05X	96	1.526E-05	4.86E-05	4.18E+00	1.000E+00	3.989E-05	XFMR 1AB05X FAILS
1-DCP-CRB-CO-AD105____	87	5.400E-06	4.80E-05	9.89E+00	1.000E+00	1.115E-04	SUPPLY CRB 1AD105 FROM BUS 1AD1 TO 1AD11 SPURIOUSLY OPENS
1-CCW-HTX-PG-TRNA	81	1.217E-05	4.63E-05	4.80E+00	1.000E+00	4.769E-05	CCW HTX E4-001 IS UNAVAILABLE
1-SWS-MOV-CF-116-ABCDE	34	4.696E-07	3.70E-05	7.98E+01	1.000E+00	9.877E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-ABCDF	34	4.696E-07	3.70E-05	7.98E+01	1.000E+00	9.877E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-ABCEF	34	4.696E-07	3.70E-05	7.98E+01	1.000E+00	9.877E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-ABDEF	34	4.696E-07	3.70E-05	7.98E+01	1.000E+00	9.877E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-ACDEF	34	4.696E-07	3.70E-05	7.98E+01	1.000E+00	9.877E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-BCDEF	34	4.696E-07	3.70E-05	7.98E+01	1.000E+00	9.877E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-LPI-CKV-CC-009_____	68	1.070E-05	3.68E-05	4.44E+00	1.000E+00	4.308E-05	RHR PUMP A DISCHARGE CKV 009 RANDOMLY FAILS TO OPEN
1-SWS-MDP-CF-FR-ABD	9	5.730E-08	3.64E-05	6.34E+02	1.000E+00	7.937E-03	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ABF	9	5.730E-08	3.64E-05	6.34E+02	1.000E+00	7.937E-03	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ADF	9	5.730E-08	3.64E-05	6.34E+02	1.000E+00	7.937E-03	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-BCD	9	5.730E-08	3.64E-05	6.34E+02	1.000E+00	7.937E-03	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-BCF	9	5.730E-08	3.64E-05	6.34E+02	1.000E+00	7.937E-03	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-SWS-MDP-CF-FR-CDF	9	5.730E-08	3.64E-05	6.34E+02	1.000E+00	7.937E-03	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FS-ABDF	23	2.000E-06	3.62E-05	1.91E+01	1.000E+00	2.271E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-BCDF	23	2.000E-06	3.62E-05	1.91E+01	1.000E+00	2.271E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-BDEF	23	2.000E-06	3.62E-05	1.91E+01	1.000E+00	2.271E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SD-HEP-JOINT-64	1	1.000E-06	3.11E-05	3.21E+01	1.000E+00	3.892E-04	Joint HEP minimum threshold for HFE combination 64
1-SWS-MOV-CF-116-ABDE	33	3.852E-07	3.03E-05	7.96E+01	1.000E+00	9.852E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-ABDF	33	3.852E-07	3.03E-05	7.96E+01	1.000E+00	9.852E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-ABEF	33	3.852E-07	3.03E-05	7.96E+01	1.000E+00	9.852E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-ACDE	33	3.852E-07	3.03E-05	7.96E+01	1.000E+00	9.852E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-ACDF	33	3.852E-07	3.03E-05	7.96E+01	1.000E+00	9.852E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-ACEF	33	3.852E-07	3.03E-05	7.96E+01	1.000E+00	9.852E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-BCDE	33	3.852E-07	3.03E-05	7.96E+01	1.000E+00	9.852E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-BCDF	33	3.852E-07	3.03E-05	7.96E+01	1.000E+00	9.852E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-BCEF	33	3.852E-07	3.03E-05	7.96E+01	1.000E+00	9.852E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-RLY-FC-AX3_68AB	69	2.480E-05	2.80E-05	2.13E+00	1.000E+00	1.417E-05	RELAY AX3 FOR OPENING/CLOSING NSCW 1HV1668A/B RANDOMLY FAILS
1-SWS-RLY-FC-162_1X68	56	2.480E-05	2.64E-05	2.07E+00	1.000E+00	1.335E-05	RELAY 162-1X FOR OPENING HV1668A/B RANDOMLY FAILS

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-ACP-DPL-FC-AY2A	68	1.802E-05	2.57E-05	2.42E+00	1.000E+00	1.786E-05	PANEL 1AY2A FAILS
1-SWS-MDP-CF-FR-ABDF	9	3.967E-08	2.52E-05	6.34E+02	1.000E+00	7.937E-03	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-BCDF	9	3.967E-08	2.52E-05	6.34E+02	1.000E+00	7.937E-03	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-SWT-FC-TY1669B_	31	9.864E-06	2.29E-05	3.32E+00	1.000E+00	2.907E-05	NSCW TR B RETURN WATER TEMP SWITCH TY1669B RANDOMLY FAILS
1-ACP-TFW-FC-1ASEQT1	64	1.526E-05	2.15E-05	2.41E+00	1.000E+00	1.761E-05	SEQUENCER XFMR T1 FAILS
1-ACP-TFW-FC-1ASEQT2	64	1.526E-05	2.15E-05	2.41E+00	1.000E+00	1.761E-05	SEQUENCER XFMR T3 FAILS
1-DCP-CRB-CO-AD1101__	59	5.400E-06	1.94E-05	4.59E+00	1.000E+00	4.500E-05	CRB AD1101 SPURIOUSLY OPENS ON LOAD SHED LOGIC CIRCUITS
1-DCP-BAT-FC-AD1B	58	1.404E-05	1.93E-05	2.37E+00	1.000E+00	1.719E-05	BATTERY 1AD1B RANDOMLY FAILS (125V)
1-SWS-MOV-OC-1669A__	30	7.008E-07	1.73E-05	2.57E+01	1.000E+00	3.096E-04	NSCW TR B SPRAY VALVE HV1669A SPURIOUSLY CLOSES
1-SWS-MOV-CF-116-DE	13	9.760E-07	1.72E-05	1.86E+01	1.000E+00	2.203E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-DF	13	9.760E-07	1.72E-05	1.86E+01	1.000E+00	2.203E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-EF	13	9.760E-07	1.72E-05	1.86E+01	1.000E+00	2.203E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-ACP-TFW-FC-ABB03X__	50	1.526E-05	1.61E-05	2.06E+00	1.000E+00	1.325E-05	480V MCC XFMR 1ABB03X FAILS
1-ACP-CRB-CO-AA0221__	46	5.400E-06	1.50E-05	3.77E+00	1.000E+00	3.473E-05	FEEDER CRB 1AA02 SPURIOUSLY OPENS - 1AA02 TO 1AB05X
1-ACP-CRB-CO-AB0501__	46	5.400E-06	1.50E-05	3.77E+00	1.000E+00	3.473E-05	SUPPLY CRB 1AB05 SPURIOUSLY OPENS - 1AB05X TO 1AB05
1-SWS-MOV-CF-116-ADE	12	7.136E-07	1.25E-05	1.85E+01	1.000E+00	2.192E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-ADF	12	7.136E-07	1.25E-05	1.85E+01	1.000E+00	2.192E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-SWS-MOV-CF-116-AEF	12	7.136E-07	1.25E-05	1.85E+01	1.000E+00	2.192E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-BDE	12	7.136E-07	1.25E-05	1.85E+01	1.000E+00	2.192E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-BDF	12	7.136E-07	1.25E-05	1.85E+01	1.000E+00	2.192E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-BEF	12	7.136E-07	1.25E-05	1.85E+01	1.000E+00	2.192E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-CDE	12	7.136E-07	1.25E-05	1.85E+01	1.000E+00	2.192E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-CDF	12	7.136E-07	1.25E-05	1.85E+01	1.000E+00	2.192E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-CEF	12	7.136E-07	1.25E-05	1.85E+01	1.000E+00	2.192E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-DEF	12	7.136E-07	1.25E-05	1.85E+01	1.000E+00	2.192E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-RLY-FC-AX46869_CC	24	1.538E-06	1.12E-05	8.29E+00	1.000E+00	9.142E-05	CCF OF RELAYS AX4 FOR OPENING NSCW 1HV1668A/B & 1669A/B AFTER LOSP
1-SWS-MOV-OC-1668A__	32	7.008E-07	1.10E-05	1.67E+01	1.000E+00	1.962E-04	NSCW TR A RETURN ISOLATION VALVE HV1668A SPURIOUSLY CLOSES
1-DCP-BAT-CF-ALL	24	1.235E-07	9.93E-06	8.14E+01	1.000E+00	1.007E-03	CCF OF 125V BATTERIES
1-ACP-CRB-CO-AB0502__	15	5.400E-06	9.30E-06	2.72E+00	1.000E+00	2.159E-05	FEEDER CRB 1AB05 SPURIOUSLY OPENS - 1AB05 TO 1ABA
1-ACP-CRB-CO-ABA04__	15	5.400E-06	9.30E-06	2.72E+00	1.000E+00	2.159E-05	CRB 1ABA04 BETWEEN 1ABA & BCH 1AD1CA OPENS SPURIOUSLY
1-DCP-CRB-CO-AD106__	15	5.400E-06	9.30E-06	2.72E+00	1.000E+00	2.159E-05	CRB 1AD106 FROM BCH 1AD1CA TO 1AD1 SPURIOUSLY OPENS
1-SWS-MDP-CF-FS-AC	36	8.309E-06	8.85E-06	2.07E+00	1.000E+00	1.335E-05	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-AE	36	8.309E-06	8.85E-06	2.07E+00	1.000E+00	1.335E-05	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
1-SWS-MDP-CF-FS-CE	36	8.309E-06	8.85E-06	2.07E+00	1.000E+00	1.335E-05	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-DCP-BCH-FC-____BABB-CC	32	2.036E-06	8.20E-06	5.02E+00	1.000E+00	5.045E-05	CCF OF BCHs 1BD1CA & 1BD1CB
1-SD-HEP-JOINT-48	2	1.000E-06	8.17E-06	9.17E+00	1.000E+00	1.025E-04	Joint HEP minimum threshold for HFE combination 48
1-SD-HEP-JOINT-72	2	1.000E-06	8.17E-06	9.17E+00	1.000E+00	1.025E-04	Joint HEP minimum threshold for HFE combination 72
1-CCW-XVM-OC-HV_11814	31	2.021E-06	7.97E-06	4.95E+00	1.000E+00	4.947E-05	LOCKED OPEN MANUAL VALVE HV-11814 FAILS CLOSED
1-CCW-XVM-OC-HV_11815	31	2.021E-06	7.97E-06	4.95E+00	1.000E+00	4.947E-05	LOCKED OPEN MANUAL VALVE HV-11815 FAILS CLOSED
1-SWS-CTF-CF-S-ABCDEFGH	21	1.067E-07	7.89E-06	7.49E+01	1.000E+00	9.267E-04	System Generated Event based upon Rasp CCF event : 1-SWS-FAN-CF-S
1-EPS-TNK-RP-DFOSTKB_	11	4.334E-07	7.53E-06	1.84E+01	1.000E+00	2.177E-04	TR B. DIESEL FUEL OIL STORAGE TANK 1-2403-T4-002 RUPTURES
1-EPS-TNK-RP-DGBDAY	11	4.334E-07	7.53E-06	1.84E+01	1.000E+00	2.177E-04	DG B FUEL OIL DAY TANK RUPTURES
1-SWS-MOV-CF-116-ADEF	11	3.852E-07	6.69E-06	1.84E+01	1.000E+00	2.177E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-BDEF	11	3.852E-07	6.69E-06	1.84E+01	1.000E+00	2.177E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SWS-MOV-CF-116-CDEF	11	3.852E-07	6.69E-06	1.84E+01	1.000E+00	2.177E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116
1-SD-HEP-JOINT-67	7	1.000E-06	6.26E-06	7.26E+00	1.000E+00	7.853E-05	Joint HEP minimum threshold for HFE combination 67
1-EPS-RLY-FC-RUN24____CC	11	3.571E-07	6.20E-06	1.84E+01	1.000E+00	2.177E-04	CCF OF DG RUNNING RELAYS 2 & 4
1-ACP-CRB-CO-AY2A02__	32	5.400E-06	6.08E-06	2.13E+00	1.000E+00	1.412E-05	CRB 1AY2A02 BETWEEN INVERTER 1AD1111 & 1AY2A SPURIOUSLY OPENS
1-DCP-CRB-CO-AD101__	32	5.400E-06	6.08E-06	2.13E+00	1.000E+00	1.412E-05	CRB FROM BATTERY 1AD1B TO BUS 1AD1 SPURIOUSLY OPENS
1-DCP-CRB-CO-AD104__	32	5.400E-06	6.08E-06	2.13E+00	1.000E+00	1.412E-05	CRB 1AD104 FROM INVERTER 1AD1111 TO 1AD1 SPURIOUSLY OPENS

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-DCP-CRB-CO-AY2A08__	32	5.400E-06	6.08E-06	2.13E+00	1.000E+00	1.412E-05	CRB SPURIOUSLY OPENS (AY2A08 TO SEQUENCER A)
1-DCP-CRB-CO-AD1104__	29	5.400E-06	5.66E-06	2.05E+00	1.000E+00	1.315E-05	CRB AD1104 SPURIOUSLY OPENS ON LOAD SHED LOGIC CIRCUITS
1-ACP-CRB-CO-AB0514__	31	5.400E-06	5.66E-06	2.05E+00	1.000E+00	1.314E-05	CRB FROM 480V SWGR 1AB05 TO 480V MCC 1ABF SPURIOUSLY OPENS
1-DCP-CRB-CO-AD111024	31	5.400E-06	5.66E-06	2.05E+00	1.000E+00	1.314E-05	CRB AD1110 TO FAN CONTROL LOGIC SPURIOUSLY OPENS
1-CCW-XVM-OC-HV_11806	23	2.021E-06	5.43E-06	3.69E+00	1.000E+00	3.367E-05	LOCKED OPEN MANUAL VALVE HV-11806 SPURIOUSLY FAILS CLOSED
1-CCW-XVM-OC-HV_11807	23	2.021E-06	5.43E-06	3.69E+00	1.000E+00	3.367E-05	LOCKED OPEN MANUAL VALVE HV-11807 SPURIOUSLY FAILS CLOSED
1-RHR-XHE-XD-POS8E	21	2.500E-06	5.36E-06	3.15E+00	1.000E+00	2.689E-05	OPERATORS FAIL TO DIAGNOSE LORHR BEFORE BOILING POS 8E
1-DCP-BCH-FC-AAAB__-CC	19	2.036E-06	4.83E-06	3.37E+00	1.000E+00	2.973E-05	CCF OF BCHs 1AD1CA & 1AD1CB
1-EPS-RLY-FC-RUN1234_-CC	16	6.919E-08	4.78E-06	7.00E+01	1.000E+00	8.655E-04	CCF OF DG RUNNING RELAYS 1, 2, 3, & 4
1-SD-HEP-JOINT-21	4	1.000E-06	4.59E-06	5.59E+00	1.000E+00	5.759E-05	Joint HEP minimum threshold for HFE combination 21
1-SWS-RLY-FC-162_1PPS-CC	15	6.498E-08	3.98E-06	6.23E+01	1.000E+00	7.683E-04	CCF OF NSCW PPS TDE RELAYS 162-1 - OVERALL CCF FOR CCFG=6
1-DCP-BCH-FC-__ABBABB-CC	20	1.079E-06	3.64E-06	4.37E+00	1.000E+00	4.227E-05	CCF OF BCHs 1AD1CB, 1BD1CA, & 1BD1CB
1-ACP-DPL-MA-BY2B	10	2.120E-07	3.60E-06	1.80E+01	1.000E+00	2.131E-04	PANEL 1BY2B IN MAINTENANCE
1-EPS-CKV-CC-FXFERP_-CC	13	5.764E-08	3.54E-06	6.24E+01	1.000E+00	7.702E-04	CCF OF CVS IN DG FUEL XFER PUMPS TRAINS TO OPEN (047, 044, 053,050)
1-SWS-MDP-CF-FR-BD	11	1.563E-07	3.42E-06	2.29E+01	1.000E+00	2.742E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-BF	11	1.563E-07	3.42E-06	2.29E+01	1.000E+00	2.742E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-SWS-MDP-CF-FR-DF	11	1.563E-07	3.42E-06	2.29E+01	1.000E+00	2.742E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-DCP-BCH-FC-AA__BA__CC	11	2.036E-06	3.32E-06	2.63E+00	1.000E+00	2.046E-05	CCF OF BCHs 1AD1CA & 1BD1CA
1-DCP-BCH-FC-AA__BB-CC	11	2.036E-06	3.32E-06	2.63E+00	1.000E+00	2.046E-05	CCF OF BCHs 1AD1CA & 1BD1CB
1-HPI-TNK-RP-RWST	6	4.334E-07	2.30E-06	6.32E+00	1.000E+00	6.662E-05	TANK RUPTURES
1-SD-HEP-JOINT-91	2	1.000E-06	2.24E-06	3.24E+00	1.000E+00	2.809E-05	Joint HEP minimum threshold for HFE combination 91
1-LPI-AOV-PG-HV0607__	14	7.200E-07	2.03E-06	3.82E+00	1.000E+00	3.529E-05	RHR HTX OUTLET AIR OPERATED VALVE HV0607 TRANSFERS CLOSED
1-DCP-BCH-FC-AAABBA__CC	10	1.079E-06	2.00E-06	2.85E+00	1.000E+00	2.319E-05	CCF OF BCHs 1AD1CA, 1AD1CB, & 1BD1CA
1-DCP-BCH-FC-AAAB__BB-CC	10	1.079E-06	2.00E-06	2.85E+00	1.000E+00	2.319E-05	CCF OF BCHs 1AD1CA, 1AD1CB, & 1BD1CB
1-SWS-MDP-CF-FR-AC	9	1.563E-07	1.85E-06	1.28E+01	1.000E+00	1.479E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-AE	9	1.563E-07	1.85E-06	1.28E+01	1.000E+00	1.479E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-CE	9	1.563E-07	1.85E-06	1.28E+01	1.000E+00	1.479E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-EPS-TFL-FC-XFERPSIG-CC	9	3.229E-08	1.71E-06	5.40E+01	1.000E+00	6.643E-04	DG FUEL XFER PUMP SIGNAL LT CCF
1-SD-HEP-JOINT-65	3	1.000E-06	1.29E-06	2.29E+00	1.000E+00	1.621E-05	Joint HEP minimum threshold for HFE combination 65
1-LPI-AOV-PG-HV0606__	8	7.200E-07	1.25E-06	2.73E+00	1.000E+00	2.168E-05	RHR HTX A OUTLET AIR OPERATED VALVE HV0606 TRANSFERS CLOSED
1-EPS-RLY-FC-RUN124__CC	5	8.134E-08	1.13E-06	1.49E+01	1.000E+00	1.740E-04	CCF OF DG RUNNING RELAYS 1, 2, & 4
1-EPS-RLY-FC-RUN234__CC	5	8.134E-08	1.13E-06	1.49E+01	1.000E+00	1.740E-04	CCF OF DG RUNNING RELAYS 2, 3, & 4

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-EPS-XVM-PG-041FXFR_	5	7.200E-08	1.00E-06	1.49E+01	1.000E+00	1.740E-04	DGB FUEL TRANSFER LINE MANUAL VALVE 041 PLUGS
1-SWS-MDP-CF-FR-BDE	5	5.730E-08	9.51E-07	1.76E+01	1.000E+00	2.081E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-BDF	5	5.730E-08	9.51E-07	1.76E+01	1.000E+00	2.081E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-BEF	5	5.730E-08	9.51E-07	1.76E+01	1.000E+00	2.081E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-DEF	5	5.730E-08	9.51E-07	1.76E+01	1.000E+00	2.081E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-CCW-TNK-RP-T4_002	8	4.334E-07	8.55E-07	2.97E+00	1.000E+00	2.472E-05	CCW SURGE TANK T4-002 FAILS (RUPTURES)
1-CCW-MDP-FR-P4002004-CC	8	4.277E-07	8.43E-07	2.97E+00	1.000E+00	2.472E-05	CCF OF CCW PUMPS P4-002 & P4-004 TO RUN
1-CCW-MDP-FR-P4002006-CC	8	4.277E-07	8.43E-07	2.97E+00	1.000E+00	2.472E-05	CCF OF CCW PUMPS P4-002 & P4-006 TO RUN
1-CCW-MDP-FR-P4004006-CC	8	4.277E-07	8.43E-07	2.97E+00	1.000E+00	2.472E-05	CCF OF CCW PUMPS P4-004 & P4-006 TO RUN
1-SWS-MDP-CF-FR-BDEF	5	3.967E-08	6.59E-07	1.76E+01	1.000E+00	2.081E-04	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-ACP-XSW-FC-ITCHYARD	2	1.670E-07	5.32E-07	4.18E+00	1.000E+00	3.990E-05	LOSS OF OFFSITE POWER FAILURE DURING MISSION TIME (24 hrs): prob=LOSP yr freq
1-SWS-MDP-CF-FR-ABC	3	5.730E-08	3.71E-07	7.47E+00	1.000E+00	8.107E-05	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ABE	3	5.730E-08	3.71E-07	7.47E+00	1.000E+00	8.107E-05	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ACD	3	5.730E-08	3.71E-07	7.47E+00	1.000E+00	8.107E-05	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ACE	3	5.730E-08	3.71E-07	7.47E+00	1.000E+00	8.107E-05	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ACF	3	5.730E-08	3.71E-07	7.47E+00	1.000E+00	8.107E-05	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR

Table B-2 Importance Measures (cont.)

Name	Count	Prob	FV	RIR	RRR	Birnbaum	Description
1-SWS-MDP-CF-FR-ADE	3	5.730E-08	3.71E-07	7.47E+00	1.000E+00	8.107E-05	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-AEF	3	5.730E-08	3.71E-07	7.47E+00	1.000E+00	8.107E-05	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-BCE	3	5.730E-08	3.71E-07	7.47E+00	1.000E+00	8.107E-05	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-CDE	3	5.730E-08	3.71E-07	7.47E+00	1.000E+00	8.107E-05	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-CEF	3	5.730E-08	3.71E-07	7.47E+00	1.000E+00	8.107E-05	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-HPI-XVM-PG-207	2	7.200E-08	3.12E-07	5.33E+00	1.000E+00	5.432E-05	MANUAL VALVE 207 PLUGS
1-SWS-MDP-CF-FR-ABCE	2	3.967E-08	1.89E-07	5.77E+00	1.000E+00	5.975E-05	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ACDE	2	3.967E-08	1.89E-07	5.77E+00	1.000E+00	5.975E-05	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1-SWS-MDP-CF-FR-ACEF	2	3.967E-08	1.89E-07	5.77E+00	1.000E+00	5.975E-05	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR

APPENDIX C

SOURCES OF MODEL UNCERTAINTY

This appendix documents the low power and shutdown Level 1 PRA key sources of model uncertainty in Table C-1.

Table C-1 Key Sources of Model Uncertainty

Technical Element	Topic	Description	Characterization
LPOS	Plant evolutions and outage types	The low power and shutdown (LPSD) probabilistic risk assessment (PRA) model estimates risk only from refueling outages (POT-4). Other outage types are discussed but not modeled.	Inclusion of other outage types, particularly unplanned forced outages, could introduce a significant contribution to low power and shutdown risk. The conditions that can result in entry into an unplanned forced outage can involve unavailability or failure of risk significant structures, systems, and components (SSCs).
	Low Power and Hot Standby POS	Low power and hot standby plant operating states (POSs) are defined but not modeled.	The low power and hot standby POSs are relatively short duration (for planned refueling outages), and the plant response is generally assumed to be similar to the at-power model. However, some plant conditions can be significantly different than at-power conditions. Also, initiating event frequencies are potentially higher during these POSs. Modeling of low power and hot standby POSs could identify additional contributors to outage risk. The potential CDF contributions are evaluated in a sensitivity study in Section 10.3.2. The sensitivity study suggests that the overall low power and hot standby POS contributions to CDF are much lower than those for the shutdown POSs.
	POS Definition	POS conditions are intended to be defined with the most severe and limiting conditions applied. Several of the POSs represent transitions, such as changing of reactor coolant system (RCS) coolant level. In these cases, the lowest level is assumed to apply throughout the POS. Similar limiting assumptions are applied to decay heat levels, initial temperature, equipment unavailabilities, and containment status.	Assumptions on the POS conditions are generally conservative. This may result in over-estimating the risk, but further division of the POSs to account for a range of conditions is not expected to significantly impact the results.

Table C-1 Key Sources of Model Uncertainty (cont.)

Technical Element	Topic	Description	Characterization
LIE	LPSD initiating event frequency data	A comprehensive collection and analysis of recent low power and shutdown initiating event (IE) data does not exist. The most recently published shutdown IE frequencies are based on data from year 2000 and earlier (EPRI, 2001). The authors of this study are unaware of any reference on low power event frequencies. This requires relying on the limited available data even though it might not be complete or the most recent. Plants should have access to their own plant specific data, but this may not be sufficient for rare events.	Improved LPSD IE data collection and analysis would help improve our understanding of these events and reduce uncertainties. It is not clear what impact improved data would have on the mean value of frequency estimates.
	Low power and hot standby IE analysis	Low power and hot standby IEs are addressed qualitatively and semi-quantitatively through sensitivity studies.	Sensitivity studies suggest some scenarios could have risk significant contributions.
	Shutdown IE-POS combinations	Not all IE-POS combinations are represented for the modeled shutdown POSs.	Excluding IE-POS combinations that have not been shown to meet the screening criteria defined in the trial use LPSD PRA standard supporting requirement LIE-C6 (ASME/ANS, 2014) or other appropriate quantitative screening criteria could underestimate the risk for those POSs.
	Other shutdown IEs	Other shutdown IEs, including heavy load drops, reactivity control events, and fuel handling events are not addressed in the analysis.	These events are not expected to have significant contributions to risk, but this has not been demonstrated through qualitative analysis or quantitative screening. Operating experience has shown that heavy load drops can have significant impacts on plant safety, but the frequency of events and applicability of different types of drops are uncertain.

Table C-1 Key Sources of Model Uncertainty (cont.)

Technical Element	Topic	Description	Characterization
LIE	Fault tree modeling for support system IEs	The LPSD PRA model does not use fault tree modeling for most support system IEs, but it may be appropriate to use a fault tree approach for some systems (e.g, component cooling water and residual heat removal (RHR) in shutdown cooling mode).	Fault tree modeling could provide a more realistic representation of the contributions of common-cause failures and the impacts of different POS configurations (e.g., one RHR pump normally operating versus two operating). Note that fault tree modeling is used for loss of nuclear service cooling water IEs.
	LOINV scenario selection	The loss of inventory IE is defined very broadly to include a variety of breaks, human error-induced flow diversions, and opening of relief valves. The model LOINV scenarios have assume a break in the operating RHR loop. The analysis should consider other break locations and release pathways.	Consideration for different LOINV break types and locations can provide insight on different inventory loss rates and release pathways. Also, while human error-induced flow diversions and over-draining are modeled, these events are limited to POSs were draining the RCS is being performed. Other maintenance errors or inadvertent actions could result in draining the RCS during other POSs (for example, inadvertent draining RCS at Wolf Creek, licensee event report 482/1994-013). These types of errors of commission are beyond the current modeling state-of-practice. Additional sensitivity analyses could provide insight on the potential significance of such events.

Table C-1 Key Sources of Model Uncertainty (cont.)

Technical Element	Topic	Description	Characterization
LAS	Post-refueling POS accident sequence modeling	POS-specific heatup and boiling calculations were not performed for some POSs. In these cases, bounding assumptions were applied, but these do not provide a realistic representation of the POS conditions, particularly for post-refueling POSs with low decay heat.	The modeling simplifications for post-refueling POSs are expected to give conservative results for those scenarios. In some cases, post-refuel may not reach core damage for a very long time (> 7 days). These types of sequences were evaluated as part of the Level 2 PRA analysis. Accident sequence simulations indicate that under some conditions (e.g., ECCS injection available and event occurring greater than 20 days after shutdown) core damage can be significantly delayed or avoided. While modeling assumptions are not expected to significantly impact the CDF, release category frequencies, or the consequence results, some of the post-refueling sequences could be re-evaluated to determine if no core damage would be a more appropriate end state.
	LOOP accident sequence modeling	Loss of offsite power (LOOP) accident sequence modeling is simplified by using the same model logic as LORHR for the sequences where emergency power or normal offsite power has been restored.	The modeling simplifications for LOOP sequences with power recovery are not expected to have a significant impact on the results. The plant and operator responses are similar. However, the specific RHR restoration actions can be different. Reflecting these differences with LOOP-specific human failure events (HFEs) would improve the realism of the model. Other considerations could be differences in sequence timing, staff resources (as operators are responding to LOOP and restoring RHR), and equipment dependencies.

Table C-1 Key Sources of Model Uncertainty (cont.)

Technical Element	Topic	Description	Characterization
LAS	POS 8E and 8L maintenance modeling	Significant maintenance unavailabilities are expected during POSs 8E and 8L when one normal source of offsite power (i.e., reserve auxiliary transformer) is taken out of service for maintenance. Much of the equipment on the associated safety-related train will be unavailable. After refueling the protected train is swapped. The current modeling approach does not reflect the swapping of trains but assumes one train is unavailable and both trains are symmetric.	The modeling simplification does not account for asymmetries in safety-related trains. It also does not account for differences in the specific maintenance configurations when the trains are swapped. However, the assumptions about the availability of offsite power sources (e.g., a reserve auxiliary transformer is out of service for maintenance) have a dominant impact on the sequence frequencies and minimize the impacts of uncertainties about the specific configuration of the under-maintenance train.
LSC	Thermal-hydraulic calculations	A sample of thermal-hydraulic (T-H) calculations was performed for some POSs, but not for all. Additional calculations would provide more realistic timing and success criteria.	If no representative T-H calculations were performed for an IE-POS scenario, then POS-specific heatup and boiling calculations were performed. In some cases, no POS-specific calculations were performed, and conservative assumptions were made. However, these assumptions are expected to have limited impact on the overall shutdown CDF. Refer to Section 5.3 for a summary of the POS-specific calculations that were performed.
	Feed and bleed cooling success path	Controlled injection from the refueling water storage tank (RWST) is not considered as a means of establishing a safe, stable state because the RWST inventory will eventually be exhausted and another heat removal method must be established.	If sufficient procedural guidance is established that long-term cooling can be sustained with controlled injection, then sequences with successful controlled injection from RWST could be considered to reach a safe, stable state. Consideration for makeup to the RWST should also be evaluated in a long-term cooling strategy.

Table C-1 Key Sources of Model Uncertainty (cont.)

Technical Element	Topic	Description	Characterization
LSC	Mission time	The mission time for the LPSD PRA is 24 hours. However, for some accident sequences during post-refueling conditions with low decay heat, successful pumped injection could extend the sequence time beyond 24 hours. This would allow additional time for restoring RHR cooling. The LPSD PRA currently does not credit additional time beyond 24 hours for the longer mission time sequences.	The potential for additional time beyond the 24-hour mission time is not expected to significantly impact the results. Successful injection in response to a loss of RHR cooling event during post-refueling mid-loop could extend the sequence to approximately 27 hours until the RWST inventory is exhausted. The small amount of additional time available is not expected to change any of the sequence modeling assumptions and is not credited. For most of the risk significant sequences occurring prior to refueling, the additional time associated with successful injection is not sufficient to justify extending the mission time beyond 24 hours.
LSY	Maintenance and testing unavailability	Detailed information about timing and duration of equipment testing and maintenance are not available. It was generally assumed the most limiting conditions apply throughout the POS.	Assumptions may be conservative for key equipment. In some cases, equipment outages might not be captured if not in key systems.
	RHR repair	RHR repair is modeled, but supporting data is sparse.	Additional modeling detail on specific actions required for restoration and repair of RHR should be considered, and the limited repair data should only be used where applicable.
LHR	HFE Delineation	The human reliability analysis (HRA) approach was used to identify the HFEs that are to be modeled and the conditions under which they are characterized. There are hundreds of individual HFEs that could be modeled for different POS conditions.	The HRA for the L3PRA Level 1 LPSD model did not evaluate HFEs for every possible set of POS conditions, but instead modeled the HFEs to be applied to similar POSs such that the estimated human error probability (HEP) values are expected to be applicable across the range of POS conditions.

Table C-1 Key Sources of Model Uncertainty (cont.)

Technical Element	Topic	Description	Characterization
LHR	Dependency between human-induced IEs and subsequent operator response	The enhanced reliance on operator actions during LPSD operations and the potential for human-induced IEs during plant evolutions could result in dependencies between the IE occurrence and the subsequent operator response. This could negatively influence the ability to respond to the event.	The HFE evaluations consider the context of the IEs, including the timing, cues, and other influencing factors. This approach characterizes the initial diagnosis of the event by the operators. However, there is no formal dependency model applied between IE HFEs and subsequent HFEs.
	Operator Action Time Windows	It is difficult to determine exact timing of occurrence of cues and operator actions.	There are often multiple cues and indicators available to the crew for LPSD events. A best estimate time window was obtained based on the cue most likely to be used and when it would occur in relation to the IE.
	Crew-to-Crew Variability	Crew-to-crew variability was not included as part of the HRA, including the following: • Overall experience • Experience with event(s) • Staffing level (minimum versus maximum) • Back shift maintenance resources • Crew personalities • Creativity	The staffing level and crew experience were generally assumed to be standard. This assumption was informed by the plant visit and staff interviews. A more detailed analysis considering crew-to-crew variability is not done in most HRAs.
	Organizational Interfaces	The impact of the plant-specific organization during an event may be difficult to capture in the HRA and may strongly depend on the personalities involved, including: • Operations/Maintenance • Staff/Management • Control Room/Technical Support Center • Ex-Plant (e.g., Grid Operator)	The organizational culture of the plant and individual differences between crew members are not captured within the HRA analysis, which is consistent with the current state-of-practice.
	Multi-Unit Events	Multiple units may provide both significant benefit— by virtue of the sharing of equipment and personnel—and significant challenges if all units require accident mitigation simultaneously.	This analysis was focused on only one unit of a two-unit site. Help or interference by the second unit was not considered but is expected to be addressed as part of the L3PRA project integrated site risk task.

Table C-1 Key Sources of Model Uncertainty (cont.)

Technical Element	Topic	Description	Characterization
LHR	Distractions (e.g., Fatigue, Problems Outside of Work, etc.)	The crew work schedule and individual crew member conditions are not generally included as part of the shaping factors of the HRA.	The consideration of these issues is typically beyond the state-of-practice.
	Crew Awareness to Conditions	Training can alter crew awareness. The awareness of the crew to specific accident conditions varies with the training cycle and current industry experiences that are promulgated to the crews.	
	Circadian Clock	Time of day is not generally included in the HRA despite evidence that the most serious crew errors occur between midnight and 6 a.m.	
	Scenario-Dependent Recovery and Repair	The accident sequence level of discrimination with regard to plant conditions, timing, operator interface, and use of non-safety systems can significantly impact associated uncertainty. The finite nature of the level of delineation collapses the continuum of possible sequences to a limited set. Repair and recovery of failures is an area of significant judgment in the PRA model. It involves the designation of sufficient time, access, personnel, and guidance to either recovery (manual action) or repair of a failed SSC.	Recovery/repair of failed SSCs is not considered in the model and is beyond the state-of-practice.
LDA	LPSD-specific data	The parameter estimates for equipment failures are based on the same generic data used for the at-power PRA. System alignments and failure modes are adjusted to represent the LPSD operating configurations, but the failure data are generally taken from at-power operating experience. The environmental and service conditions that may be unique to LPSD are not addressed. There is generally a lack of consistent data that would support POS-specific operating hours, demands, and failures that could be used to develop POS-specific parameter estimates.	The parameter estimates used for equipment failures during shutdown are not expected to differ significantly from those used in the at-power PRA. While some environmental and service conditions may differ, these are not expected to significantly influence failure rates. There is no clear bias (that would suggest either higher or lower failure rates) that is introduced by this modeling choice. However, LPSD-specific data should be collected to verify this.

Table C-1 Key Sources of Model Uncertainty (cont.)

Technical Element	Topic	Description	Characterization
LDA	POS demands and operating hours	POS-specific demands and operating hours are estimated based on available operating experience and assumptions about the modeled refueling outage. Data are not generally available that align exactly with the PRA definition of POS windows. The data on outage timing and duration are usually tied to key critical path activities required for completing the outage. These activities do not always align with the POS definitions modeled in the PRA.	The POS demands and operating hours can have a significant impact on core damage frequency (CDF) estimates. The CDF results should be viewed with the assumptions about POS durations in mind. For this study, the POS durations were defined to represent a realistic refueling outage consistent with the reference plant's past experience, but every refueling outage can be different. For example, CDF estimates would be lower if the fueled mid-loop condition was not entered, and there are examples of refueling outages where fueled mid-loop was not entered.
	Plant transition between states and POS operating hours	The POS durations are defined to completely cover a representative refueling outage. The durations are defined in Section 2. However, in some cases (POS 7 and POS 9) only a fraction of the POS is modeled because only a fraction is spent in the representative conditions. These POSs represent significant transitions that are occurring with the plant (e.g., raising RCS level from reduced inventory to refueling cavity flooded).	This modeling simplification leaves some portions of the POS un-modeled because only a fraction of the total POS duration is included. This leads to an underestimation of risk. This could be improved by further sub-dividing the POSs and covering the entire duration of the outage in the model. The overall impact of these omissions is expected to be small because the modeled portion represents the limiting conditions of the POSs (e.g., POS 9 represents the portion of the POS with reduced RCS inventory and omits earlier portions of the POS with higher inventory levels).
	Test and maintenance durations	Data for test and maintenance durations are not collected. Conservative assumptions are applied to key systems and components to assume the equipment is unavailable for the entire duration of the POS in which the maintenance occurs (unless restoration is modeled with an HFE.)	Improved data collection on outage-specific test and maintenance durations would improve the realism of the modeling. For many of the key systems and components, the duration may be lower than the assumed duration, as the equipment is assumed unavailable for the entire POS.

Table C-1 Key Sources of Model Uncertainty (cont.)

Technical Element	Topic	Description	Characterization
LQU	Success terms	There is only one success term that is explicitly accounted for in the model quantification. Several other failure events have relatively high failure probabilities, and their success terms could impact the quantification.	Eight top events were identified with failure probabilities greater than 2.5×10^{-2} . The success term for the highest failure probability (1-SD-FEED-G-SBO) is explicitly accounted for in the quantification. Sensitivity studies were run on the remaining seven and found to not have a significant impact on the results.
	Uncertainty in POS durations	Parameter uncertainty is accounted for all equipment failures and HFEs, but POS duration fractions do not include uncertainty.	The POS durations can have a significant impact on CDF estimates. The CDF results should be viewed with the assumptions about POS durations in mind. For this study, the POS durations were defined to represent a realistic refueling outage consistent with past experience, but every refueling outage can be different. This is a source of uncertainty that is not addressed in the overall CDF parameter uncertainty distribution.
	HEP value for combined HFEs	There are several HFEs that are significant contributors to the shutdown CDF, and many cut sets that contain multiple HFEs. A dependency analysis was performed to assess all HFE combinations. Still, the HEP quantification approach involves much interpretation and many assumptions. To account for uncertainty, a minimum joint HEP value of 10^{-6} is applied to all cut sets containing multiple HFEs.	The minimum joint HEP value of 10^{-6} is applied to cut sets with multiple HFEs. Applying this post-processing rule did not have a significant impact on the results. Most of the significant HFE combinations have joint HEP values that are greater than 10^{-6} .

APPENDIX D REVIEW OF LOW POWER AND SHUTDOWN OPERATING EXPERIENCE

A high-level review of operating experience with low power and shutdown (LPSD) events for large commercial operating nuclear reactors was undertaken with the following objectives:

- Ensure the project has accounted for the major insights obtained from prior efforts
- Identify more recent events that may not be part of past reviews of operating experience
- Look for unique initiating events and plant configurations
- Review events where the initiating event is human-induced to identify any impact on operator mitigation actions.

A review of past operating experience of LPSD events is an important activity to identify the types of failures that can occur and review the lessons learned from previous events. Past operating experience can provide important lessons that are relevant to LPSD operations. The nuclear power plant accident at Chernobyl occurred during a power reduction to support a test of plant emergency power systems (see (NRC, 2002) for a description of the event). Also, the Fukushima Daiichi accident included two reactor units, Unit 5 and Unit 6, that were shut down at the time of the earthquake and tsunami. Only one air-cooled emergency diesel generator (EDG) remained available after the event, which was used to power loads to restore core cooling in the shutdown units. If the EDG had been lost like the others at the site, then the shutdown units may have also experienced significant damage (ANS, 2012).

This operating experience review is meant to be limited in scope and depth, focusing on major qualitative insights, rather than a detailed quantitative review. For example, the assessment of initiating event frequencies applicable to LPSD PRA modeling is **not** within the scope of this review. In addition, insights from recent significance determination process (SDP) findings may be highly dependent on the conditions of the event itself and may or may not have wider implications for a baseline LPSD PRA.

D.1 Summary of Findings

While past studies have been performed to collect and evaluate LPSD events, no recent studies have been performed. Several data sources were used to identify potential LPSD events. Still, there are open questions about the completeness of the data. Often events could be identified through one data source but not corroborated through other sources. The on-going collection and analysis of LPSD events is not as well-established as the collection and analysis of events for full power operations. Despite these concerns about completeness, the LPSD operating experience review has provided useful insights and qualitative information about event occurrences and actions to respond and recover from the events. Notable observations include:

- Many events include some aspect of human failures, for example, maintenance errors, communication issues, and inadvertent actions.
- Operating experience suggests restoration actions can be accomplished in a short time after the event occurrence.

- Heavy load drops may be a concern for shutdown operations. The event at ANO1 is an example of the types of failures that can occur with heavy load drops.
- Broad communications about LPSD events were emphasized during the 1990s and earlier. Examples include Generic Letter 88-17 and NUREG-1449. Although LPSD events still occur, there has been less emphasis on generic communication activities. There have been no recent NRC information notices or significant reports from INPO related to shutdown events. The last report on shutdown initiating events from the Electric Power Research Institute (EPRI) was issued in 2010.
- Potential incompleteness in the event identification and incompleteness in the tracking of LPSD -exposure time (i.e., the total time that plants are in LPSD operating states) make it difficult to use the operating experience to support initiating event frequency estimates.

It is important to note that there were many potential LPSD events identified in licensee event reports (LERs), but not all the LERs were reviewed as part of this effort. Additional screening would be needed to determine if LPSD initiating events could be identified from the LERs.

This review shows the challenges with establishing a complete set of LPSD events. The information collected is not sufficient to support initiating event frequency estimates, but it does provide insights about the types of events that can occur and what types of actions should be modeled in response to the events.

D.2 Sources

There are at least several major reports related to aspects of LPSD that have been published in the past (note that this is not meant to be an exhaustive reference list, it just indicates the sources of information for this document):

- Information from Idaho National Laboratory (INL) prepared to support shutdown initiating event frequency estimates for this study, as discussed in Section 8.1
- EPRI 1021176, "An Analysis of Loss of Decay Heat Removal and Loss of Inventory Event Trends (1990–2009)", December 2010 (EPRI, 2010)
- EPRI 10031113, "An Analysis of Loss of Decay Heat Removal Trends and Initiating Event Frequencies (1989 – 2000)", November 2001 (EPRI, 2001)
- EPRI 105802, "An Analysis of Loss of Decay Heat Removal Trends (1989-1994)", December 1995 (EPRI, 1995)
- NUREG/CR-6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1", Volume 2, June 1994 (NRC, 1994a)
- NUREG/CR-6093, "An Analysis of Operational Experience During Low Power and Shutdown and a Plan for Addressing Human Reliability Assessment Issues", June 1994 (NRC, 1994b)
- NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States", September 1993 (NRC, 1993)

- Evaluation of LER database
- Compilation of insights from various recent findings evaluated under the SDP
- Operating experience communications

D.3 Past Studies

A summary of the major insights and additional sources of information from each of the major references above is included.

The EPRI Report 105802 (EPRI, 1995) compiled and analyzed operating experience information from 1989 through 1994 based on LERs, operating experience (OE) and daily plant status entries into Nuclear Network®, and Institute for Nuclear Power Operations (INPO) event summary reviews. A database was created containing each loss of reactor coolant system (RCS) inventory or decay heat removal (DHR) incident. The database also included incidents that resulted in a loss of spent fuel pool inventory and cooling and incidents that would be considered precursors¹ to a loss of DHR. The database included a total of 202 actual loss of reactor coolant system inventory or loss of DHR incidents. For loss of DHR, an incident severity ranking was determined by how much thermal margin prior to boiling was used during the incident. For loss of inventory, the incident severity was determined by how much of the inventory margin was used prior to a loss of DHR occurring. The report notes that the database includes several RCS and DHR precursor incidents (i.e., did not result in a loss of DHR, but could have resulted in a loss if the problem was not detected or if other conditions existed) and a few spent fuel pool inventory or DHR incidents. Trend analyses were done for one set for losses of RCS inventory and DHR occurrences, as well as a separate set that also included precursor incidents. The report states that, while utilities more consistently report RCS actual incidents compared to precursor incidents and spent fuel pool incidents, the results with precursor and spent fuel pool incidents do not change the overall trend distributions for actual events (while recognizing that perhaps not all actual losses of inventory and DHR incidents are included in the database due to differences in reporting criteria used by utilities). The database contains 202 occurrences from 1989 through 1994 that resulted in either an unplanned loss of RCS inventory or loss of DHR. The trend analyses indicate a decrease in the number of events during the early part of the range considered (1989–1992) versus the later part (1993–1994). Consideration of precursor incidents indicates maintenance related activities were the major activities involved at the time of the incidents. The type of maintenance activities involved when the incidents occurred indicated that calibration and testing activities, including valve stroke testing, were the dominant contributors, followed by corrective or preventive activities, and modifications. The report describes in more detail which type of maintenance activities were involved in these incidents. The type of operational activities involved during incidents are indicate that incidents which occurred during system alignment changes were the largest contributor, followed by pump starts or pump swaps with no system alignment changes, initiating and removing clearances. The report also describes in more detail which type of operational activities were involved in these incidents. A study of the causes indicated that written communications was the largest contributor, followed by planning and scheduling of activities, work practices, and verbal communications. While this report can be considered dated due to the changing performance of LPSD activities during the time period of consideration for the nuclear fleet, it provides a basis to consider maintenance/operational activities, and potential causes for risk drivers in LPSD PRA modeling.

The EPRI Report 10031113 (EPRI, 2001) extends the time window of assessment for loss of Decay Heat Removal trends to 1989–2000 and is also an update to another EPRI report (EPRI 113051) which includes information from 1997 and 1998. This report uses the INPO operating experience sources to create a database and to update shutdown initiating event frequencies using Bayesian methods, as well as to calculate time-dependent recovery probabilities. For purposes of updating the information, data from 1989–1993 was supplemented with data from 1981–1988 to develop generic priors which were then updated with data from 1994–2000. This report indicates that the improvement observed since the early 1990s leveled off around 1995 with a long-term improvement trend. The report also indicates that precursors to DHR incidents significantly increased since 1994 with increased reporting being a potential cause. This report follows the same format of its predecessors and provides equivalent insights on maintenance/operational activities and potential causes for risk drivers in LPSP PRA modeling. The report also performs an initiating event recovery analysis. Appendix A of (EPRI, 2001) contains a list of both pressurized-water reactor (PWR) (Table A-1) and boiling-water reactor (BWR) (Table A-2) shutdown initiating events in the range 1989–2000 (as supplied by INPO). Appendix B of (EPRI, 2001) contains a list of both PWR (Table B-1) and BWR (Table B-2) shutdown initiating events in the range 1981–1988.

The EPRI Report 1021176 (EPRI, 2010) is an update of the previous efforts that acknowledges an observed increase in plant events during shutdown conditions. This report documents a long-term study of loss of DHR and loss of inventory events during shutdown conditions spanning the 20-year period 1990–2009. Loss of spent fuel pool (SFP) inventory and events that can be considered near-misses or precursors to a loss of inventory or DHR are currently not evaluated. This report provides detailed analyses of the dominant contributors to the causes of particular types of events. The sources of information were the INPO plant events database, along with LERs, information notices, and plant notifications, as well as other INPO documents (OE reports and significant event reports). Some high-level insights in this report include an observation of an increase in the trend of annual loss of DHR events of the 3-year rolling average after 2004, and a higher number of loss of inventory events in 2007–2009 than in the prior 3 years, 2004–2006. This report includes, for the first time, loss of inventory events as a new category for evaluation (i.e., it includes all inventory loss events, whether or not DHR is lost). Operations-related activities were involved in nearly 75 percent of all loss of inventory events, with maintenance having a relatively smaller contribution at only 17 percent. A large percentage of the events associated with loss of inventory occurred during system alignment changes (e.g., swapping trains) and tagging activities (e.g., clearing tags with drains or vents still open). Testing and calibration were the primary maintenance-related activities. Work practices were the most prevalent cause for both loss of DHR and loss of inventory events. Work practices include operator or technician errors, failure to follow plant processes or procedures, and general human performance failures (e.g., lack of conservative decision-making). Work practices contributed to approximately 35 percent of both types of events. Other event causes of note are (1) inadequate or incorrect procedures (accounting for nearly 30 percent of all loss of inventory events, and 20 percent of loss of DHR events) and (2) planning and scheduling (e.g., scheduling two incompatible evolutions simultaneously or improper sequencing of evolutions that impacts availability of DHR).

NUREG-1410 (NRC, 1990) describes a significant event with a station blackout occurring at the Vogtle plant during mid-loop operations. Prior to the event occurrence, maintenance on one reserve auxiliary transformer made the normal source of offsite power to one safety bus unavailable. The EDG associated with the bus was also unavailable due to maintenance. The event was initiated by a refueling truck that was supporting maintenance activities in the

switchyard backing up into a support pole for the 230-kV feeder line to the operable reserve auxiliary transformer. This resulted in a cracked insulator and a faulted feeder line. The reserve auxiliary transformer breakers opened. The EDG started automatically but tripped after 1-2 minutes. The diesel generator was restored and powered the bus 36 minutes after the initiating event. This event highlights many areas that are important for modeling shutdown accident scenarios. First, the maintenance configuration prior to the event significantly diminished the redundancy of available power sources to support DHR equipment. The on-going maintenance activities in the switchyard contributed to the event occurrence. The reliance on manual actions was an important aspect of the event response. The EDG started and loaded onto the safety bus automatically, as designed, but required manual action to be restored after tripping. Manual actions were also taken to close the containment equipment hatch, which was open when the event occurred. Current plant operation uses a qualitative shutdown risk monitoring approach, which would make it unlikely to enter the plant conditions that contributed to the event. Following the shutdown risk approach, the maintenance activities are now much more controlled during mid-loop operation, and maintenance activities are scheduled not to coincide with the mid-loop window. The containment equipment hatch also is typically closed prior to entering mid-loop.

NUREG-1449 (NRC, 1993) provides a summary of LPSD issues, including PRA implementation aspects at the time of its publication (1993). It has a description of typical LPSD operations (which may be dated in some respects), summary of past PRA studies and other insights. Table 2.7 in this report includes the cold-shutdown and refueling events analyzed by the Accident Sequence Precursor program (the table is organized by docket/LER).

NUREG/CR-6093 (NRC, 1994b), published in 1994, identified unique aspects of human performance during LPSD conditions and provided a program plan for research and development necessary to improve existing human reliability analysis (HRA) methodologies. It contains an assessment of influences on HRA for LPSD items, which includes an analysis of PWR events, input from NUREG-1449, and general HRA insights. Appendices A, B, and C contain directly relevant information (albeit potentially dated) that focused on detailed insights, consideration of past events (with breakdowns of human actions, defenses, and conditions), HRA framework aspects, and recommendations.

NUREG/CR-6144 (NRC, 1994a) was a follow on effort from NUREG-1150, which assesses the risks of severe accidents initiated during plant operational states (POSs) other than full power operation and to compare the estimated core damage frequencies (CDFs), important accident sequences, and other qualitative and quantitative results with those accidents initiated during full power operation. The focus was on the Surry plant. As part of this study an extensive effort was undertaken to collect data to characterize the plant during shutdown operations. The data collection included:

- A data base of initiating events was compiled to support the initiating event analysis.
- Plant-specific log books, outage schedules, minimum equipment lists, and monthly operating reports were reviewed to collect data needed to estimate the frequency of shutdown, duration of plant operational states, and maintenance unavailabilities.
- The plant log books were reviewed to determine the time that the plant is in different configurations.

The report uses LERs as the primary source of events used for initiating event frequency. Some of the events reported in NUREG-1410 were also included. In total 351 events were identified and used for the initiating event frequency analysis. However, there is limited discussion of the details of the events and the role of human actions contributing to those events, which makes it difficult to assess applicability of these past events to current studies. The report does acknowledge the impacts that human-induced initiating events may have on subsequent human failures, but this type of dependency is not explicitly addressed in the study. The report notes that the research program laid out in NUREG/CR-6093 intends to address this issue.

Input for this study was drafted by INL for estimating rates of selected shutdown initiating events. The draft report evaluates a more recent time-window (2000–2015) that includes 93 shutdown initiating events in INL's database for the 16-year period. This effort is based on a request by the NRC to update the data that was evaluated in EPRI reports 1021176 and 1003113 and provide similar initiating event estimates using more recent data. The number of loss of offsite power (LOOP), loss of alternating current (AC) bus, and loss of residual heat removal (RHR) events for the recent nuclear reactor fleet are identified, but the collected information does not include descriptions of the events. The draft report deals with the statistical treatment of initiating event estimates, rather than focus on event narratives.

Although the INL work does not directly characterize the details of the events, the work does provide some lessons on the challenges of collecting operating experience for shutdown states. The INL database that was used for the study compiles event records from several sources that are directly report to NRC by the nuclear industry. Data sources include LERs, event notifications, and the Equipment Performance Information Exchange (EPIX). The events identified in INL's database were compared to the events reported in EPRI Report 1021176 for the period where the two studies overlap, 2000–2009. The reports showed good agreement in identifying events for the LOOP and loss of AC bus categories. However, the INL database did not include several loss of RHR and loss of inventory events that were reported in EPRI 1021176. The INL database could not be confirmed to provide a complete data set for loss of RHR and loss of inventory events. For this study, the authors decided to use the data from the EPRI reports for loss of RHR and loss of inventory events because it appears to provide the most complete collection of shutdown events. This decision also means that the data for those event categories was restricted to events occurring in 2009 and earlier.

D.4 Evaluation of Licensee Event Report Database

Between the EPRI reports (notably the tables in 10031113) and the NUREGs, a list of the major events from 1988 to 2009 can be derived (e.g., Diablo Canyon Unit 2 loss of RHR while in mid-loop in 1987, Vogtle 1 loss of offsite and onsite ac power in 1990) and their information obtained from direct sources (including associated LERs). A recent study was completed to review LERs to develop an event classification scheme (Schroer, 2013). The focus of that study was to identify LERs that impacted multiple reactor units, but the study identified some multi-unit events occurring at shutdown. Additional review of LERs with a focus on LPSD events may yield more events and insights that are relevant to this current study.

A cursory investigation in the LER database indicates that there is no simple, direct way to distinguish LPSD events from at-power precursor incidents. A more challenging gap is the inclusion of events at full power that may have implications for LPSD operations (or its compliment, the exclusion of events in LPSD operations which do not necessarily apply to LPSD operations directly), which would require a full review of all LERs. Hence, observing and

understanding these gaps is an inherent feature of any LER search inquiry. The database does allow for specific searches per operating mode (i.e., hot shutdown, refueling, startup, cold shutdown, and hot standby). A search that includes all non-power operation modes produces 19,188 search results. This search does not provide sufficient granularity to assess quantitative values and/or detailed insights into human aspects of LPSD (for those, the insights from previous cited reports would be more beneficial).

From a simple inquiry, a similar trend from the one observed in prior publications also emerges. For the period of 1981–1993, a total of 13,971 LERs are recorded; with a decreasing trend in the ranges of 1994–1999 (2,370 LERs), 2000–2009 (797 LERs), and January 2010 to March 2016 (450 LERs). It is also possible to capture the most recent events listed in the LER database to have a general input from the most recent operating experience not fully covered/detailed in the current literature review. The LER search results include some events that could be considered as major events, such as the stator drop during a refueling outage at ANO Unit 1 in 2013 (with subsequent loss of RHR) are included. It is also notable that the Catawba consequential LOOP event in 2012 which resulted in a LOOP to both Units (with Unit 2 in Mode 5) was only reported as a full power event (i.e., a clear example of a gap in reporting that impacts LPSD relevant insights). Hence, additional work would be needed to provide a detailed operating experience insight based on a more rigorous evaluation of the LER database. In order to provide an initial cross-check, insights from various recent SDPs are provided next.

D.5 Insights from Various Findings Analyzed under the Significance Determination Process

Recent regulatory activities related to SDPs were compiled in order to highlight events or conditions related to LPSD. It should be noted that SDPs involve specific inspection activities bounded by performance deficiencies that may or may not be relevant for a LPSD baseline PRA. For activities that involve specific events such as loss of RHR (LORHR) and LOOP, the SDPs (and associated documentation, such as NRC inspection reports) will contain information on the event as well as an analysis of the safety significance of a specific finding (which again may or may not be directly translatable for the shutdown Level 3 PRA model).

Kewaunee: Inability to Rapidly Close Containment Equipment Hatch (2004):

The plant entered a refueling outage on October 9, 2004. On October 10, the containment equipment hatch was removed, and the rail system was installed on October 11. The rail system consisted of two sections, one installed inside containment and the other exterior to containment. It was intended that the rail system be rapidly removed in the event of a need to close the equipment hatch. The containment equipment hatch closure plan called for the removal of the exterior rail only, whereas the interior rail system was intended to be left in place inside containment. On October 14, in preparation for lifting the reactor vessel head, an attempt to close the equipment hatch revealed that the interior rail system interfered with and prevented the hatch from being closed. To eliminate the interference, the interior rail system was cut, and the hatch was subsequently closed. Resolution of the issue took approximately 8 hours.

An inspection finding was identified with respect to the inability to rapidly close the containment equipment hatch during cold shutdown conditions due to an interference. The interference was caused by the inadequate design of a rail system that was installed in the containment to facilitate reactor vessel head replacement activities. During the time that the interference existed, the reactor was in cold shutdown with a high decay heat load and with the RCS vented

to the containment atmosphere. All emergency core cooling systems and charging systems were available. However, during a portion of the time that the interior rail system was installed, only one of two EDGs were available.

This event highlights potential complications that could occur when attempting to close containment. The reference plant uses a similar rail system at the containment hatch opening. Observing the hatch opening and equipment and procedures to support closure was a key focus of the site visit to the reference plant. The reference plant has guidance to avoid having any equipment near the hatch opening and equipment needed to remove the rail system is staged near the opening.

Columbia: Isolation of Operational Residual Heat Removal Pump (2006):

Columbia was shut down in Mode 4, approximately 72 hours into a forced outage with the reactor vessel level being maintained at 70 inches and water temperature at 114 °F. Operators were performing a procedure to transfer reactor protection system (RPS) B from its normal power supply (RPS MG set) to its alternate power supply. During the procedure, RHR shutdown cooling (SDC) inboard containment isolation valve was closed and SDC was interrupted (the valve was closed due to an error in the controlling procedure). Reactor recirculation pump A remained in service providing forced flow through the core. Operators reopened the valve and restored SDC. Reactor temperature reached 148 °F and vessel level reached 95 inches (bottom of main steam lines is 116 inches) while SDC was not in service. SDC was restored in 45 minutes (120 minutes time to boil from 120 °F) after closure of the valve. Reactor temperature and level were restored to their previous operating bands.

After evaluation of the event, the NRC determined that a procedure directed opening an incorrect electrical disconnect for the RHR valve. Opening the identified disconnect removed power from the motor operator of the RHR valve, but did not prevent formation of a sealed-in closure signal when the RPS B power supply was transferred to its alternate supply. Consequently, the valve closed when power was restored to the motor operator. A different electrical disconnect should have been identified that would also have prevented formation of the sealed-in closure signal.

Turkey Point Unit 3: Loss of Decay Heat Removal Event (2006):

Unit 3 was operating in DHR mode with one operating A-train RHR pump. At the time the RCS was being drained, there was a large decay heat load, the reactor coolant loops were not available, and the RCS temperature was less than 200 °F. Maintenance, which had been moved up in the outage schedule, resulted in a loss of all decay heat removal for 7 minutes, which caused reactor coolant temperature to increase from 113 °F to 140 °F. Plant operators took prompt action to restore core cooling which terminated the event.

Approximately 7 minutes elapsed from the time decay heat removal cooling was lost until it was restored. If the attempt to restore cooling using the B train had failed, operators would likely have restored the A train, which became available when the EDG provided power to the A bus. Because no core exit thermocouples were connected, actual conditions of the reactor core if the loss of decay heat removal were sustained, would not be readily known to control room personnel. The only indication of core temperatures was RCS loop temperature which was not in the area of highest heatup. Additionally, steam generators had been removed from service

due to RCS draining and an adequate vent path had not been established to prevent RCS pressurization on a sustained loss of decay heat removal.

Calvert Cliffs Unit 2: Loss of RCS Level Indication during Vacuum Fill (2007):

Calvert Cliffs Unit 2 was shut down in Mode 5. The unit was approximately 31 days into a refueling outage. The reactor had been refueled and the reactor vessel head reinstalled. In preparations for restart the reactor vessel level had been lowered to mid loop so that a vacuum fill of the RCS, including the steam generator tubes, could be conducted per procedure. RCS level was being monitored by two level instruments. At 11:02 am when RCS vacuum reached approximately 15 inches, both level indications began to operate erratically rendering them non-functional leaving the operators with no direct indication of RCS level. Licensee personnel continued to draw vacuum and began investigating the cause of the erratic level indication, which was suspected to be a vacuum leak. They monitored the operating decay heat removal pump from the control room by observing suction pressure, flow and motor amps and locally at the pump by an auxiliary operator. After reaching the minimum vacuum of 23 inches required for filling the RCS, they commenced gravity filling of the RCS via the high pressure injection piping as directed by procedure at a rate of 150 to 200 gpm. They were not able to obtain the expected vacuum of 25 inches supplying additional evidence of a vacuum leak. At 16:05, approximately five hours after losing all direct level indication, an instrument root valve was found inappropriately open. This valve is a drain valve common to both instruments relied on for mid-loop level indication is connected to local refueling water level indication during some conditions. Upon closing this valve, level indication began to function in a stable manner and in agreement with wide range pressurizer level indication. Prior to the event, no RCP was in service as is normal for this condition, one train of DHR was available in SDC mode with a reactor water temperature of 102 °F and a flow rate of 1500 gpm. Both trains of CCW were operable, secondary cooling via the steam generators was not available, and containment was open but closable in less than 30 minutes.

Sequoyah: Loss of Level Indication during a LOOP and at Reduced Inventory (2007):

NRC inspections identified a failure to establish an adequate procedure for RCS reduced inventory/mid-loop operations. Specifically, a procedure for RCS drain and fill operations did not establish adequate actions to maintain continuous RCS level indication during all possible plant conditions while in the reduced inventory/mid-loop configuration, specifically a LOOP event. Per procedure, two channels of level indication provide the two continuous independent level monitoring systems required for reduced inventory/mid-loop operations. The instruments are powered from 120 VAC wall receptacles in the main control room which are powered by the auxiliary building lighting boards. On a loss of off-site power the lighting boards would be de-energized and are not re-energized when the EDGs restore on-site vital power. Each unit has an uninterruptible power supply (UPS) battery to maintain the system for approximately 30 minutes. The procedure did not establish contingency actions to recover power to level indication systems or provide guidance for alignment of an alternate RCS level indication mechanism within the 30 minutes for which power was still available from the Mansell system unit battery.

Perry: Loss of Shutdown Cooling (2007):

The Unit was in Mode 4 (12 days into a shutdown), with the in-service B RHR loop providing DHR for the reactor vessel, when a maintenance technician disconnected a wire for a relay

associated with a RCIC system isolation function A during a surveillance test associated with the RCIC steam line flow high instrument channel. The resultant electrical transient caused an inadvertent actuation of the optical isolator and the trip of the RHR B pump, resulting in the loss of all reactor decay heat removal. The main control room received the RHR pump trip and low pressure alarms. Water level was above the normal at-power water level but below the main steam lines. Prior to the event, a licensee engineer had supplied the control room with a conservative estimate of the time to core boiling (TTB) given a loss of SDC of 2.6 hours. Both loops of reactor recirculation were out of service and RHR was supplying the required forced circulation as well as SDC. In addition, the control rod drive system was supplying a small amount of cold water into the reactor bottom head. Licensee personnel restored SDC water flow to the reactor by placing the A RHR loop in SDC service, after approximately two hours. Upon SDC startup, the A RHR pump suction temperature indicated 140 °F, while the bottom head drain temperature indicated 111 °F.

Oconee Unit 1: Loss of Inventory Event (2008):

During a refueling outage, Oconee Unit 1 had restored level from a mid-loop operation to install coldleg nozzle dams to below the reactor vessel flange. The head was detensioned in preparation for removal. As part of main generator voltage regulator modification testing, a main generator lockout signal was generated while the switchyard was back-feeding all Unit 1 electrical loads through the main transformer and the associated auxiliary transformer. This caused a slow transfer from the auxiliary transformer to backup transformer (CT1) from the switchyard. The resulting electrical transient caused a momentary loss of power to the running pumps performing SDC and, due to a complication, a relief valve in the letdown purification system opened and remained open as designed. This transient caused a loss of inventory (LOI) from the RCS to the miscellaneous waste holdup tank (MWHUT). The operators quickly recognized the LOI and entered the appropriate procedures. They had the relief valve isolated and makeup water going into the RCS within 17 minutes.

Prior to the event the purification system was aligned to the LPI system which was in-service in the SDC mode of operation. Both the A and B LPI pumps were in SDC. When the electrical system slowly transferred, the 600 volt AC system initially re-energized as designed. However, it tripped off on over-current from the initial current inrush. The over-current trip was caused by an improperly set instantaneous magnetic trip device. With the system de-energized, three air operated valves in the purification system did not reopen as designed due to a lack of power on the valve solenoids. This caused the purification system to see a pressure above the setpoint of the letdown relief valve's setpoint (discharging to the MWHUT).

Wolf Creek: Potential Loss of RHR Due to Inadequate Cooling of Suction Line (2008):

On October 3, 2008, Wolf Creek reported a condition which affected the residual heat removal (RHR) system. The condition existed from at least January 1990 to May 2008. This condition would have prevented the RHR system from performing its function due to system piping being saturated during heat up or cool down between Mode 3 and Mode 4.

Prior to identifying this condition, Wolf Creek would have one or both of the RHR trains lined up to the RCS for SDC during plant startups and they would secure them prior to entering Mode 3 from Mode 4 (Mode 3 starts when RCS temperature exceeds 350 °F). System procedures required the RHR system to be cooled down using the mini-flow recirculation line, following alignment for Emergency Core Cooling System (ECCS) injection mode. The physical location at

which the mini-flow piping connects to the suction of the RHR system prevented approximately 140 feet of RHR suction line, in each RHR train, between the RCS hot leg isolation valve and the mini-flow line location from being cooled, this piping would remain above 198 °F for approximately 29 hours. Licensee analysis indicates that if an RHR pump were started (either manually or from a Safety Injection System (SIS) signal) with water temperature above 198 °F water in the suction piping would flash to steam with the potential consequence of pump steam binding and/or piping water hammer. This would occur because a lowering of system pressure at the pump suction would flash the water into steam, depending upon the initial water temperature. As long as the saturation pressure in the RCS hot leg is higher than the static pressure from the Refueling Water Storage Tank (RWST), the check valve located in the supply line from the RWST will not open and no injection will occur. As the RHR pump is started, the pressure in the RCS hot leg will decrease which will cause the hot pressurized water to flash, before the pressure is low enough to open the check valve. The steam void could extend to the pump suction and steam bind the pump or water hammer could ensue failing the pumps. Likewise, if a Mode 4 Loss of Inventory (LOI), with water temperature between 198 °F and 350 °F, were to occur the operators would align the RHR system for injection and make the pumps susceptible to the failures discussed above.

The broader applicability of a similar vulnerability occurring at other plants depends on the plant-specific procedures, approach for aligning the RHR system, and the physical system alignment. No specific vulnerabilities have been identified for the Level 3 PRA project, but pump steam binding and/or water hammer should be considered as a possibility unless specific defenses are found to be in place to prevent occurrence.

St. Lucie Unit 2: Forced Outage to Repair RCP Seal Leak-off line (2009):

On July 13, 2009, St. Lucie Unit 2 reported a condition that affected one of the four reactor coolant pump (RCP) seal leak-off lines. This was established after an increase in unidentified RCS leakage was observed over a period of four days. At this point, a decision was made to down power and manually shutdown Unit 2 following plant procedures. While in Mode 3, the main feedwater pump unexpectedly tripped but was recovered prior to the actuation of auxiliary feedwater. No other major equipment failures were reported during shutdown operations and no other maintenance was performed during this forced outage.

Upon visual examination inside containment, the RCS leakage was identified and classified as reactor coolant pressure boundary leakage originating from one of the RCP seal leak-off lines due to a defective weld. The seal leak-off line is designed for 1 gpm leakage from the RCPs. Similar failures had been previously observed and repaired on other occasions. Following Technical specification requirements, Unit 2 entered cold shutdown (Mode 5). The licensee then proceeded to reduced inventory in order to perform repairs on the leak-off line. The cause of the weld failure in the seal line was identified as excessive vibration due to operating the RCP outside the band of acceptable levels. The condition and the associated POSs that were necessary to repair the condition include: (i) power operations, (ii) hot shutdown, (iii) transition from Mode 4 to Mode 5, (iv) Mode 5 with reduced inventory, and (v) transition to Mode 4 until 30% level is reached in the Pressurizer. Unit 2 spent a total of 64.5 hours (2.7 days) in Mode 5 with SDC in operation and reduced RCS inventory.

Prairie Island Unit 2: Reactor Trip during Preparation for a Refueling Outage (2012):

Prairie Island Unit 2 was performing a normal shutdown in preparation for a refueling outage. With Unit 2 at approximately 11 percent power, the reactor was manually tripped due to a feedwater heater (FWH) Hi-Hi alarm response procedure. An evaluation determined that the third stage low-pressure FWH bypass line to the condenser was potentially restricting flow to the dump valve and that the moisture separator reheater control valves were ramped closed earlier than necessary and went fully closed by approximately 20 percent reactor power. This caused excessive moisture in the extraction steam at low power and resulted in more water accumulating in the low-pressure feedwater heaters.

Catawba Unit 2: LOOP due to Inadequate Design of Switchyard Protective Relaying (2012):

The licensee had implemented a Zone G Switchyard Relay Modification for Catawba Unit 1 in 2011. The purpose of this modification was to replace electromechanical and static main generator relays with multifunction, microprocessor-based relays. These relays designed to detect faults and other abnormal conditions and isolate any element of the power system that could jeopardize the continued operation or integrity of the remainder of the electrical system. The original design used one train of protective relays mostly arranged in a two-out-of-two scheme for each protective relaying function. The new design provided two redundant trains of relays connected in a two-out-of-two scheme for each train. One of the functions was an under-frequency relay to protect the generator by opening the switchyard breakers connected to the generator. This function allowed for the unit to be isolated from the grid while the main generator continued to power station loads in cases where the grid experiences a significant disturbance such as a load rejection. The design requirements included an “off-line” block for the generator under-frequency relay functions. The “off-line” condition was based on the position of generator output breakers 1A and 1B. If these breakers were open, the unit was considered offline and the generator under-frequency function was intended to be blocked.

In April 2012, Catawba Nuclear Station Unit 1 was operating at full power, while Unit 2 was in cold shutdown with fuel in the reactor and RHR system in operation. Power to Unit 2 vital busses was being supplied through Unit 1 via the cross connections at the 4,160V AC buses (labeled ETA and ETB at Catawba). A ground fault occurred on the Unit 1 ‘D’ reactor coolant pump and the electrical breaker supplying the pump opened causing a reactor trip on low RCS loop flow. The reactor trip also caused a turbine trip which then opened the generator output breakers, per the licensee’s design of the system. However, powered circuit breakers (PCBs) in the switchyard also opened causing a total LOOP to Unit 1 and a loss of power to the Unit 2 vital busses due to the electrically cross-connected systems. The 1B EDG automatically started due to loss of power to the 1ETB essential bus. The 1A, 2A, and 2B EDGs automatically started 15 seconds later due to loss of power to their respective essential busses. The licensee started the 2A RHR pump to restore decay heat removal on Unit 2 and a Notice of Unusual Event (NOUE) was declared. The standby shutdown facility (SSF) diesel generator was also started to supply power to the security system batteries but was declared inoperable 6 minutes later due to low output voltage. The EDGs started and powered the essential busses, which limited the impacts of the event. The normal source of offsite power was restored to one essential bus on both Unit 1 and Unit 2 after 17 hours since the event initiated.

This modification was also installed on Unit 2 during a refueling outage. The licensee used the same vendor to program the Unit 2 relays and the same procedures used on Unit 1; therefore, the programming error also was undetected on Unit 2. If Unit 2 had been restarted and operated

at power, then any turbine trip would have also resulted in a LOOP on Unit 2. However, the LOOP on Unit 1 allowed the licensee to identify and correct the programming error on Unit 2 prior to restart.

ANO Unit 1: Loss of Shutdown Cooling due to Stator Drop (2013):

On March 31, 2013, ANO Unit 1 lost SDC during an outage due to a heavy load drop (i.e., main generator stator, ~525 tons). The stator drop damaged beyond repair both non-safety 4,160V AC buses which were the normal and only path for offsite power to unit safety buses. The EDG 1 auto started immediately as designed, and EDG 2 auto started after short delay (~10 seconds). This occurred 7 days into the refueling outage where the reactor cavity was flooded. The licensee calculated that the time to boil was approximately 12 hours and the time to core uncover was approximately 4.8 days.

Prior to the event, both trains of SDC were in service and the plant electrical lineup was in a plant shutdown configuration to support maintenance and testing, which included a de-energized 4,160V AC non-safety bus A2, cross-tied safety-related 4,160V AC buses A3 and A4 supplied by non-safety related 4,160V AC bus A1, among other unusual system lineups. After the event, both EDGs were operating but the SBO EDG was unavailable as its cabling into Unit 1 was severed by the stator drop (fire water, the primary water source for B5b equipment, was also unavailable for the same reason). Both LPI/SDC pumps and all three trains of HPI were available with the restoration of electrical power. Gravity feed unavailable as water level in BWST was lower than water level in flooded RCS. It took several days for recovery of offsite power (at the Regulatory Conference, the licensee indicated that connection between the switchyard and safety buses were replaced and tested within 4.4 days). It should be noted that offsite electrical power was always available in the switchyard but that the connection between the switchyard and the safety related Unit 1 buses was destroyed by the stator drop. Recovery of offsite power was, therefore, feasible within the time to core uncover but estimating a human failure probability within the conditions of the event was a challenge. The licensee articulated three "success paths" for re-establishing electrical power from the switchyard to the safety buses: (1) cable from startup 1 transformer (SUI) to the 1A3 4,160V AC safety bus, (2) cable from alternate AC (AAC) diesel generator breaker to the 1A3 4,160V AC safety bus, and (3) cable from Unit 2 non-vital switchgear to the 1A3 4,160V AC safety bus. At the time of the event, the licensee formulated these contingency plans conceptually but, because the EDGs functioned successfully, the licensee did not fully develop these contingencies beyond the planning stages (i.e., they did not develop engineering packages, work plans or detailed instructions).

Operating experience from heavy load drop events shows that these can be important contributors to plant risk. However, risk assessments of heavy load drops do not have a well-established state of practice. The risk assessment can depend on many factors that can vary from plant-to-plant and outage-to-outage. The type of lift, equipment located in the vicinity, crane design, plant configuration, and procedures for the lift all can influence the risk. Additional study of risk associated with heavy lifts is suggested as a candidate for future work in

APPENDIX A

DC Cook Unit 1: Level Indicator Problem during RCS Drain Down (2013):

In April 2012, during refueling outage of Unit 2, the licensee started the drain down to vacuum fill. A camera used for level indicator, which monitors the site glass, failed. The licensee

replaced the camera and recommenced the drain down. Later, operations noted level anomalies with a level indicator and broke vacuum established in the RCS. The licensee could not readily determine the cause for the anomalies. The licensee determined that alternate level indications were suitable substitutes for non-available indications and made procedural changes to allow their use. Eventually, the licensee restarted the evolution. At 22 inches of vacuum, level anomalies occurred on two level indicators. The licensee decided to continue the evolution while monitoring RCS level on multiple indications. The licensee successfully completed the evolution. The apparent cause evaluation performed by the licensee concluded that the length of instrument tubing runs caused the level anomalies.

In May 2013, during refueling outage of Unit 1, the licensee commenced an RCS vacuum fill. As vacuum increased, the licensee noted that one level indicator was drifting high while two other level indicators drifted low. Because of the anomalies, the licensee broke RCS vacuum. The licensee then retried to draw vacuum. Similar anomalies occurred and the licensee broke vacuum. Maintenance personnel tightened fittings associated with the vacuum fill equipment in order to correct any potential vacuum leaks. The licensee started a third attempt at vacuum fill, which was stopped due to level deltas between instruments. Trouble shooting by maintenance identified a loose connection internal to the box for one level indicator. The licensee replaced the transducer and re-commenced drain down. During this attempt, two level indicators exceeded the maximum level deviation and the licensee refilled the RCS to above the reduced inventory level. In consultation with the vendor, the licensee performed a zero adjustment to bring the level indications into agreement. After the adjustment, the licensee performed the drain down and vacuum fill evolution successfully. The licensee used commonality review, event and causal factor charts, and barrier analysis to determine the root cause. The licensee concluded that the root cause of the issues with vacuum fill was lack of station ownership of the instrumentation associated with vacuum fill. The licensee noted that condensation in one of the transducers, loose fittings in another and a broken wiring harness all pointed to mishandling of the equipment. As contributing causes, the licensee also noted poor station understanding of the sensitivity of the transducers and acceptance of minor level deviations during stable RCS conditions.

Nine Mile Point Unit 1: Loss of Shutdown Cooling Event (2013):

In preparation for a refueling outage, Nine Mile Point Unit 1 was in cold shutdown with the water level at the reactor flange. The reactor head vent piping had been removed and work was ongoing to de-tension the reactor vessel head. Simultaneous to this work, the station was conducting a LOCA/LOOP test. In addition to LOCA/LOOP testing and work to remove the reactor head, preparations were being made to start work on an electromechanical relief valve (ERV) modification. A contractor verifying the safety tag-out on the ERV modification incorrectly opened the breaker cabinet for the vital DC bus 12 (the wrong division). The vital 125V DC bus 12 cabinet door contains a mechanical interlock which opens battery breaker 12 and the static battery charger DC output breaker. This de-energized the DC switchgear when the door was opened. Upon opening the cabinet door and hearing breakers trip, the contractor realized he was in the incorrect cabinet and immediately contacted the control room. The loss of DC power generated a SDC pump 12 high temperature pump breaker trip signal, but without DC control power the breaker did not trip and the pump continued to run and to cool the core.

Operators failed to recognize the bus 12 signal present on the alarm log and the plant process computer displays prior to attempting to restore bus 12. The presence of the trip signal was also indicated by a control room annunciator which was locked in since the loss of battery bus 12.

Two unsuccessful attempts were made by operations to re-energize the bus at by closing battery breaker 12. A third attempt was initiated utilizing a different method, which used battery charger 171. This momentarily energized the DC system allowing the previously created SDC pump trip signal to trip the running SDC pump 12. The DC system then tripped again leaving the system de-energized. The operator at the controls took several minutes to identify the loss of SDC as there were no additional alarms received when the pump tripped. The operator did identify the loss of SDC by observing a decreasing reactor building closed cooling (RBCLC) temperature. Temperature was decreasing because no heat was being transferred from the SDC system to the RBCLC system. The operators restored SDC by racking in the breakers for SDC pumps 11 and 13 and starting those pumps. SDC was restored when a SDC temperature control valve was opened.

Millstone Unit 2: Loss of Inventory Event (2015):

While operating in Mode 4, an SDC system suction relief valve lifted and stayed open (unknown to the operators) during activities to start 'A' low pressure safety injection (LPSI) pump at Millstone Unit 2. This valve remained open throughout the duration of this event. Operators had no direct indication of the valve position and had attributed the initial lowering of pressurizer level to shrinkage caused by RCS temperature initially dropping several degrees when relatively cold SDC water entered the RCS upon SDC injection. Further lowering of pressurizer level was attributed to the subsequent 8 °F/hour cooldown of the RCS caused by known leakage past a closed SDC Temperature Control Valve.

Based on the review of plant process computer traces, pressurizer level began to decrease approximately 40 minutes prior to the operators commencing a controlled cooldown via SDC system at approximately 25 °F/hour. The cooldown rate was reduced to about 13 °F/hour, when operators conducted the first brief to address indications of a possible RCS leak by assessing changes in pressurizer level. Operators entered an abnormal operating procedure (AOP) and stopped the cooldown; and the event was declared an unusual event (UE) based on RCS identified leakage exceeding 25 gallons per minute (gpm). Later, a plant equipment operator identified the location of the leak by reporting that the relief valve was hot to the touch. Operators then secured the 'A' LPSI pump and isolated the SDC cooling system from the RCS, exiting the UE. Millstone Unit 2 remained in Mode 4 and maintained RCS temperature using the steam generators. The SDC suction relief valve was eventually replaced and the SDC system was subsequently returned to service, to place Unit 2 on cold shutdown conditions (Mode 5). The leak had been in progress for a total of three hours and 14 minutes. Following the event, the licensee estimated that approximately 16,570 gallons of reactor coolant were discharged to the equipment drain sump tank.

D.6 Review of Recent Operating Experience

It should be noted that, even with a cross-check between LERs and SDP findings, a complete set of LPSD events cannot be assured. A rigorous process of reviewing additional sources may be needed to ensure completeness. For example, recent operating experience communications indicates two additional shutdown events:

Perry: Loss of Bus Event (2016):

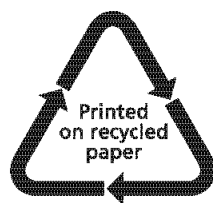
On February 11, 2016, during a forced outage at Perry, the Division 1 4,160V AC safety bus lost power as the result of a failed bus potential transformer secondary fuse. Failure of the fuse was

attributed to a manufacturing defect. Loss of the safety bus resulted in SDC pump 'A' tripping and the Division 1 EDG starting and loading the Division 1 safety bus as designed. The emergency service water 'A' pump that supplies cooling water to the EDG did not start because of the sensed loss of bus voltage (the failed fuse supplies power to this circuitry), and operators took manual action to secure the EDG. Reactor coolant temperature rose approximately 26 °F in the ensuing 40 minutes while operators established SDC via the Division 2 safety bus and SDC pump 'B.' NRC Region III inspectors performed a special inspection in response to this event.

River Bend: Loss of SDC (2016):

In January 2016, River Bend was in Mode 4 with the "A" train of SDC in service. As part of the procedure for placing SDC in service, field operators installed a jumper to bypass one of the system isolation functions. Contrary to the procedure, an operator used the wrong type of jumper, and during jumper installation, a fuse blew which interrupted power to the SDC isolation logic and caused valves in the system to close automatically. The loss of a suction source caused the "A" RHR pump to trip, resulting in a loss of SDC. Because they were unable to reset the SDC logic, the licensee had to dispatch operators to manually reestablish SDC flow. This event occurred in part because the field operator installing the jumper did not fully appreciate the reason for specifying a retractable-sheath jumper. Further investigation by the licensee showed that supervisors did not provide appropriate oversight while the operator was working in the energized panel, and at the time of the event, the operator had not received proper authorization to begin this step in the procedure.

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11. ABSTRACT (200 words or less) The U.S. Nuclear Regulatory Commission performed a full-scope site Level 3 probabilistic risk assessment (PRA) project (L3PRA project) for a two-unit pressurized-water reactor reference plant. The scope of the L3PRA project encompasses all major radiological sources on the site (i.e., reactors, spent fuel pools, and dry cask storage), all internal and external hazards, and all modes of plant operation. A full-scope site Level 3 PRA for a nuclear power plant site can provide valuable insights into the importance of various risk contributors by assessing accidents involving one or more reactor cores as well as other site radiological sources. This report, one of a series of reports documenting the models and analyses supporting the L3PRA project, specifically addresses the reactor, at-shutdown, Level 2 PRA model for internal events for a refueling outage for a single unit. The analyses documented herein are based information for the reference plant as it was designed and operated as of 2012 and does not reflect the plant as it is currently designed, licensed, operated, or maintained.									
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