Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1

Summary of Results

Manuscript Completed: September 1995
Date Published: October 1995

Edited by T.L. Chu and W. T. Pratt

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NRC FIN L1922

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ABSTRACT

During 1989, the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied by Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL), respectively. The objectives of the program are to assess 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 as assessed in NUREG-1150. The scope of the program includes that of a Level 3 PRA for internal events and a Level 1 PRA for seismically induced and internal fire and flood induced core damage sequences.

The objective of this report, NUREG/CR-6144, is to document the approach utilized in the PRA for the Surry plant, and discuss the results obtained. A phased approach was used in the Level 1 program. In Phase 1, which was completed in Fall 1991, a coarse screening analysis examining accidents initiated by internal events (including internal fire and flood) was performed for all plant operational states. The objective of the Phase 1 study was to identify potential vulnerable plant configurations, to characterize (on a high, medium, or low CDF basis) the potential core damage accident scenarios and risk, and to provide a foundation for a detailed Phase 2 analysis.

In Phase 2, mid-loop operation was selected as the plant configuration to be analyzed based on the results of the Phase 1 study. The objective of the Phase 2 study is to perform a detailed analysis of the potential accident scenarios that may occur during mid-loop operation, and compare the results with those of NUREG-1150. The scope of the Phase 2 study includes a Level 3 PRA. This volume summarizes the approach and results of the study. The internal events analysis is documented in Volume 2, which also contains an appendix that documents those parts of the Phase 1 study that deal with POSSs other than mid-loop operation. Internal fire and internal flood analyses are documented in Volumes 3 and 4. A separate study on seismic analysis, documented in Volume 5, was performed for the NRC by Future Resources Associates, Inc. Volume 6 documents the accident progression, source terms, and risk analysis.

The core damage frequencies (per year) of the Surry plant due to different initiating events that may take place during mid-loop operations are estimated as follows:

Initiating Events	Mean	5th Percentile	95th Percentile
Internal Events	5E-06	5E-07	2E-05
Internal Fires	2E-05	1E-06	8E-05
Internal Floods	5E-06	2E-07	2E-05
Seismic Events EPRA Hazard Curves	9E-08	3E-10	4E-07
Seismic Events LLNL Hazard Curves	4E-07	1E-09	1E-06

This can be compared with the mean core damage frequency from internal events of 4E-05 per year estimated in the NUREG-1150 study for full power operations.

The offsite risk estimates for latent health effects of accidents during mid-loop operation were similar to the risk estimates for full power operation. The early health consequences are much lower than the full power results primarily due to the long time after reactor trip when the accidents occur in mid-loop operation (i.e., because of the natural decay of the short-lived isotopes of iodine and tellurium, which are primarily associated with early health effects). The uncertainties in risk for accidents during mid-loop operations are largely due to uncertainties associated with isolating the containment and achieving a pressure retaining capability.

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FOREWORD

(NUREG/CR-6143 and 6144)

Low Power and Shutdown Probabilistic Risk Assessment Program

Traditionally, probabilistic risk assessments (PRA) of severe accidents in nuclear power plants have considered initiating events potentially occurring only during full power operation. Some previous screening analysis that were performed for other modes of operation suggested that risks during those modes were small relative to full power operation. However, more recent studies and operational experience have implied that accidents during low power and shutdown could be significant contributors to risk.

During 1989, the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. The program includes two parallel projects performed by Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL), with the seismic analysis performed by Future Resources Associates. Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied.

The objectives of the program are to assess the risks of severe accidents due to internal events, internal fires, internal floods, and seismic events initiated during plant operational states other than full power operation and to compare the estimated core damage frequencies, important accident sequences and other qualitative and quantitative results with those accidents initiated during full power operation as assessed in NUREG-1150. The scope of the program includes that of a Level 3 PRA.

The results of the program are documented in two reports, NUREG/CR-6143 and 6144. The reports are organized as follows:

For Grand Gulf:

NUREG/CR-6143 — Evaluation of Potential Severe Accidents during Low Power and Shutdown Operations at Grand Gulf, Unit 1

Volume 1: Summary of Results

Volume 2: Analysis of Core Damage Frequency from Internal Events for Operational State 5 During a Refueling Outage

Part 1: Main Report

Part 1A: Sections 1 - 9

Part 1B: Section 10

Part 1C: Sections 11 - 14

Part 2: Internal Events Appendices A to H

Part 3: Internal Events Appendices I and J

Part 4: Internal Events Appendices K to M

FOREWORD (continued)

- Volume 3: Analysis of Core Damage Frequency from Internal Fire Events for Plant Operational State 5 During a Refueling Outage
- Volume 4: Analysis of Core Damage Frequency from Internal Flooding Events for Plant Operational State 5 During a Refueling Outage
- Volume 5: Analysis of Core Damage Frequency from Seismic Events for Plant Operational State 5 During a Refueling Outage
- Volume 6: Evaluation of Severe Accident Risks for Plant Operational State 5 During a Refueling Outage
 - Part 1: Main Report
 - Part 2: Supporting MELCOR Calculations

For Surry:

NUREG/CR-6144 — Evaluation of Potential Severe Accidents during Low Power and Shutdown Operations at Surry, Unit-1

- Volume 1: Summary of Results
- Volume 2: Analysis of Core Damage Frequency from Internal Events during Mid-loop Operations
 - Part 1: Main Report
 - Part 1A: Chapters 1 - 6
 - Part 1B: Chapters 7 - 12
 - Part 2: Internal Events Appendices A to D
 - Part 3: Internal Events Appendix E
 - Part 3A: Sections E.1 - E.8
 - Part 3B: Sections E.9 - E.16
 - Part 4: Internal Events Appendices F to H
 - Part 5: Internal Events Appendix I
- Volume 3: Analysis of Core Damage Frequency from Internal Fires during Mid-loop Operations
 - Part 1: Main Report
 - Part 2: Appendices
- Volume 4: Analysis of Core Damage Frequency from Internal Floods during Mid-loop Operations
- Volume 5: Analysis of Core Damage Frequency from Seismic Events during Mid-loop Operations
- Volume 6: Evaluation of Severe Accident Risks during Mid-loop Operations
 - Part 1: Main Report
 - Part 2: Appendices

ACKNOWLEDGMENTS

The authors wish to thank the following persons for their contributions to this study: the various staff members at the Surry Nuclear Power Plant for their help in obtaining technical information and for their assistance during the respective walkdowns (fire, flood, and seismic,) all of which made this analysis possible; Richard C. Robinson, Jr. of the NRC for his help and guidance throughout the course of this study. Kenneth Russell of Idaho Nuclear Engineering Laboratory for his help in using IRRAS and for providing excellent code support during its use; and members of the Senior Consulting Group for their review and suggested improvements to the project. Finally, we want to thank Kathy Ryan and Barbara Kponou for their patience and secretarial support, and Cheryl Conrad for her technical support throughout the course of this study.

ACRONYMS

Acronym	Definition
ADHR	Auxiliary Decay Heat Removal system
APB	Accident Progression Bin
APET	Accident Progression Event Tree
ASEP	Accident Sequence Evaluation Program
ATWS	Anticipated Transient Without Scram
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CCW	Component Cooling Water
CDF	Core Damage Frequency
CS	Containment Spray
CSD	Cold Shutdown
CVCS	Chemical and Volume Control System
CVT	Cable Vault and Tunnel
CW	Circulating Water
EF	Error Factor
EPRI	Electric Power Research Institute
ESGR	Emergency Switchgear Room
FMEA	Failure Mode and Effect Analysis
HAZOP	Hazard and Operability Study
HEP	Human Error Probability
HHSI	High Head Safety Injection
HRA	Human Reliability Analysis
HSD	Hot Shutdown
INEL	Idaho National Engineering Laboratory
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
IRRAS	Integrated Reliability and Risk Analysis System
ISLOCA	Interfacing Systems LOCA
LLNL	Lawrence Livermore National Laboratory
LER	Licensee Event Report
LHS	Latin Hypercube Sampling
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection system
LP&S	Low Power and Shutdown
LPCS	Low Pressure Core Spray system
MCC	Motor Control Center
MFW	Main Feedwater
MW	Megawatt
NRC	Nuclear Regulatory Commission
PDS	Plant Damage State
PORV	Pilot Operated Relief Valve
POS	Plant Operational State
PRA	Probabilistic Risk Assessment
PRZR	Pressurizer

Acronyms (continued)

Acronym	Definition
PSF	Performance Shaping Factor
PSIG	Pound Per Square Inch Gauge
PWR	Pressurized Water Reactor
RCIC	Reactor Core Isolation Cooling system
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSS	Reactor Safety Study (WASH-1400)
RWCU	Reactor Water Cleanup system
RWST	Refueling Water Storage Tank
SDC	Shutdown Cooling
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SNL	Sandia National Laboratories
SV	Safety Valve
SW	Service Water
TS	Technical Specification

1 BACKGROUND

Traditionally, probabilistic risk assessments (PRA) of severe accidents in nuclear power plants have considered initiating events (IEs) potentially occurring only during full power operation. Some previous screening analyses that were performed for other modes of operation suggested that risks during those modes were small relative to full power operation. However, more recent studies and operational experience have implied that accidents during low power and shutdown could be significant contributors to risk.

During 1989, the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. The program includes two parallel projects being performed by Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL). Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied.

Surry Unit 1 was chosen for this study in part because the Surry plant was previously analyzed in the Reactor Safety Study^[1] and NUREG-1150,^[2] and in part because Virginia Power offered to cooperate. The core damage frequency (CDF) during low power and shutdown calculated in this study is compared with that calculated in NUREG-1150 for accidents during full power. The Surry plant has two Pressurized Water Reactors (PWRs), each rated at 788 megawatts (electrical) capacity, and is located near Surry in Virginia. Grand Gulf, a boiling water reactor, was selected as the plant for a parallel analysis performed by Sandia National Laboratories.

A phased approach was taken in this project. In Phase 1, a broadly scoped screening analysis,^[3] which included internal fire, flood, and seismic events was completed in late 1991. The objective of the Phase 1 study was to identify potential vulnerable plant configurations, to characterize (on a high, medium, or low CDF basis) the potential core damage accident scenarios, and to provide a foundation for a detailed Phase 2 analysis. This analysis produced a preliminary Level 1 PRA for accidents initiated during low power and shutdown (LP&S) and also gave insights on potential accident scenarios and potentially vulnerable configurations during low power and shutdown conditions.

In Phase 1, plant outages were grouped into 4 outage types: refueling, drained maintenance, non-drained maintenance with use of the residual heat removal (RHR) system, and non-drained maintenance without the RHR system. Due to the continuously changing plant configuration in any outage, plant operational states (POSs) were defined and characterized within each outage type. Each POS represents a unique set of operating conditions (e.g., temperature, pressure, and configuration). For example, in a refueling outage, up to 15 POSs were used, representing the evolution of the plant throughout a refueling from low power down to cold shutdown and refueling, and back-up to low power. An extensive effort was made to collect Surry-specific data to characterize each POS, that included reviewing operating and abnormal procedures for shutdown operations, the shift supervisor's log books, and the monthly operating reports, and performing supporting thermal-hydraulic calculations.

The Phase 1 study analyzed a total of 947 accident sequences or scenarios. Each of them was quantified and ranked according to the CDF of the scenario, timing of onset of core damage, and containment status. Figures 1 and 2 summarize the results of the quantification. Figure 1 shows the contribution of the 15 POSs to the total core damage frequency during low power and shutdown operation. POS 12, representing a cold shutdown condition late in a refueling outage, has the highest core damage frequency. This is due to high maintenance unavailabilities of equipment needed to mitigate an accident and conservatism introduced by the simplified model. Figure 2 shows the

1 Background

Figure 1 Contribution of the 15 POSs to the total core damage frequency during low power and shutdown operation – Phase 1

Figure 2 Distribution of the potentially high risk sequences – Phase 1

distribution of the 460 potentially high risk sequences. A total of 11 sequences were found to have high frequency with early core damage when the containment is open, and 5 out of the 11 sequences occur during mid-loop operation.

To accurately address each of the conditions identified in the Phase 1 study in detail would represent a very large effort. Consequently, the NRC decided to perform a detailed Phase 2 analysis on mid-loop operation. This configuration was selected because many incidents have occurred during mid-loop operations throughout the world. In addition, recent studies,^[4] including Phase 1 of this program, found that the core damage frequency during mid-loop operation appears to be comparable to that of power operation. The Phase 2 internal event analysis is documented in Volume 2 of this report. The internal fire and internal flood analysis are documented in Volumes 3 and 4, respectively. A separate study on seismic analysis, documented in Volume 5, was performed for the NRC by Future Resources Associated, Inc.

A phased approach was also used in the Level 2/3 program and both phases addressed the risk from only mid-loop operation. The first phase of the Level 2/3 PRA was initiated in late 1991 and consisted of an Abridged Risk Study. This study was completed in May 1992 and was focused on accident progression and consequences, conditional on core damage. Phase 2 is a more detailed study in which an evaluation of risk during mid-loop operation was performed. The results of the Phase 2 Level 2/3 study are the subject of Volume 6.

This volume of the report summarizes information contained in the documentation of the detailed analyses (refer to Volumes 2 through 6 of NUREG/CR-6144) performed for the Surry plant during mid-loop operation. A companion project for the Grand Gulf plant (a BWR) during cold shutdown was conducted by Sandia National Laboratories and is documented in NUREG/CR-6143.^[5]

2 OBJECTIVE

The primary objective of this study was to perform a detailed analysis of potential accidents that could occur at Surry while the plant is in mid-loop operation. The initiating events to be examined included: (1) internal initiators - including fire and flood, and (2) seismic initiators.

The objectives of the Level 1 study are:

- (1) Estimate the frequencies of severe accidents that might be initiated during mid-loop operation,
- (2) Compare the estimated core damage frequencies, important accident sequences, and other qualitative and quantitative results of this study with those of accidents initiated during full power operation (as assessed in NUREG-1150), and
- (3) Demonstrate methodologies for accident sequence analysis for plants in modes of operation other than full power.

The objectives of the Level 2/3 study are:

- (1) Develop methods to compute the risk of the Surry plant during mid-loop operation and perform the study, utilizing to the extent possible the component analyses developed as part of the NUREG-1150 program,
- (2) Identify those factors that have the most impact on the risk estimates and highlight unique features of the risk analysis performed, and
- (3) Compare the results of the study against the risk of full power operation as evaluated in the NUREG-1150 study of Surry and the NRC safety goals.

3 APPROACH AND LIMITATIONS

Due to the changing plant configuration during low power and shutdown operation, it was necessary to define different outage types, and different plant operational states (POSs) within each outage type. Within each POS, the plant configuration continues to change with time, and the decay heat continues to decrease. These factors significantly affect scenario frequencies. Therefore, a "time window" approach was developed in which different time windows were defined representing different levels of decay heat and success criteria. Within each time window, the approach used in performing the PRA for a particular POS in a particular outage type is similar to that used in the NUREG-1150 study. The following is a summary of the approach used in the key tasks of this study. The approach can be readily adopted for studies of POSs other than mid-loop and for other PWRs. The key assumptions are discussed at the end of each subsection. A more detailed description of approaches that can be used to perform a Level 1 low power and shutdown PRA is given in an Appendix to this report.

3.1 Level 1

3.1.1 Traditional Internal Events

Outage Types, Plant Operational States and Time Windows

The outage types and POSs defined in Phase 1 of this study were also adopted in Phase 2. Three mid-loop POSs, in which the reactor coolant system (RCS) level is lowered to the mid-plane of the hot leg, were selected for detailed analysis. Two of them occur in a refueling outage, POSs R6 and R10, and one in a drained maintenance outage, POS D6. The states are characterized by different levels of decay heat, and different plant configurations, such as the number of RCS loops that are isolated, and whether or not the RCS has a large vent. R6 represents a mid-loop operation that takes place early in a refueling outage allowing the RCS loops to drain quickly to permit eddy current testing of the steam generator tubes. R10 takes place after the refueling operation is completed to allow additional maintenance of equipment in the RCS loops. D6 represents mid-loop operation in which maintenance activities require the plant to go to mid-loop, and is characterized by the highest level of decay heat among the three mid-loop POSs.

To more accurately define the decay heat level when an accident is initiated, a time-window approach was developed. Four time windows after shutdown were defined, each with a unique set of success criteria reflecting the decay-heat level. For POSs R6 and D6, all four windows were needed. For POS R10, only time windows 3 and 4 were applicable. A statistical analysis was made on the time to and duration of mid-loop to determine the probability that a given accident occurs in a particular time window, conditional on the accident occurring. In this approach, an event tree was developed for each accident initiating event, POS and time window. One hundred and sixty event trees were developed for 16 initiating events.

During the Surry Unit 1 refueling outage that started on February 28, 1992, the utility changed previous practice and avoided going to mid-loop operation. Although the plant staff intend to continue this new practice mid-loop operation cannot be totally avoided in the future. Therefore, with NRC concurrence, BNL developed the PRA model

3 Approach and Limitations

based on outages (that included mid-loop operation) before the February 1992 refueling. Since the results are also presented on a per-unit-time basis, the results can be used to draw conclusions on the management of mid-loop outages.

Initiating Event Analysis

To identify initiating events, the existing studies, licensee event reports, (LERs), published NRC documents, and current Surry operating procedures were reviewed. This approach should ensure that any incident that has occurred or any scenario that has been studied will be considered in the present study. However, a systematic approach, such as a failure mode and effect analysis (FMEA) or a hazard and operability study (HAZOP), to further assure that all possible initiating events in all possible operating states were identified was not performed.

Event Tree Analysis

In Phase 1 of this study, accident scenarios were developed for all Low Power and Shutdown POSs. For those POSs that are similar to power operations, (e.g., low power operations), the relevant NUREG-1150 event trees developed for Surry were reviewed and modified (if necessary) to reflect the current plant design and operation. For other POSs, event trees were developed in group discussions, involving typically four or more BNL staff members with expertise in PWR operations, PRA, human reliability analysis (HRA), and thermal-hydraulics. Communications with the staff at Virginia Power clarified questions on the plant's design and operations.

In Phase 2, the event trees developed for the mid-loop POSs were reviewed and modified to incorporate additional information obtained in the system analysis, and to reflect current understanding of the expected operator responses to the accidents. A two-day meeting with Virginia Power operations personnel was held to discuss potential accident scenarios, and the expected responses of the plant and operators.

System Analysis

The fault tree models, developed as part of NUREG-1150 study, were reviewed and modified (when necessary) to develop two fault tree models for the plant applicable to shutdown and to low power operation for each system. The system configuration during shutdown was identified by reviewing the operating procedures used during shutdown, shift supervisor's log books, and the system training manual.

Definition of Core Damage

In the NUREG-1150 study, core damage was defined to be when the RCS level reached the top of active fuel. Due to relatively high decay heat levels, the difference between this time and the time of cladding failure is small. Under low power and shutdown conditions, the decay heat level will be significantly lower, and therefore, the difference between these two times becomes more significant. In this study, core damage was therefore defined to be the collapsed RCS level reaching 2.5 feet above the bottom of the core. This is based on the result of a MELCOR calculation of the RCS level in the core region when the cladding temperature reaches 1340°F. Phenomena such as clad oxidation and ballooning have an impact on core behavior above 1340°F. Time to core damage is used in the Level 1 study to determine the time available for operator actions such as initiating safety injection. The more realistic estimate of the time available has the tendency of lowering the associated human error probabilities.

Supporting Thermal-Hydraulic Analysis

The main purpose of the thermal-hydraulic analysis was to support the development of event trees and quantification of accident sequences. Thermal-hydraulic considerations are the basis of the time-window approach. Basically, the time windows were defined by the times when the success criteria of important mitigating functions change significantly.

In the Phase 1 study, assumptions were made based on simple "back of the envelope" calculations. It was found that more detailed calculations were needed to confirm the simple calculations, and support the assumptions made. In the Phase 2 study, a more detailed calculation was done to determine the timing of a feed and bleed operation during mid-loop operation. The calculation also gave information on the amount of water from the refueling water storage tank (RWST) needed to sustain the feed and bleed operation, as well as the timing of core uncover for different initial conditions.

The MELCOR code^[6] was used to assess whether or not gravity feed from the RWST could be used to provide long term cooling (i.e., 24 hour decay heat removal). It was found that although gravity feed is sufficient only when the decay heat is low, it can buy a few hours for restoring other means of removing decay heat even when the decay heat is high.

For reflux cooling, the studies at the Idaho National Engineering Laboratory (INEL),^{[7],[8]} Westinghouse,^[9] and Virginia Power^[10] were used to determine the success criteria. The analysis of feed and spill, gravity feed and reflux cooling were used to determine the boundary of the time windows.

Quantification

A Bayesian approach^[11] was used to estimate the initiating event frequencies. The basic event data for hardware failures were derived from the NUREG-1150 data base for Surry. The IRRAS computer code^[12] was used to quantify the fault tree and event tree, with a truncation limit of 10^{-10} . An uncertainty analysis of the total core damage frequency was performed by propagating the uncertainty of the parameters used in the model.

Human Reliability Analysis

Two types of human error events were identified and modeled in this study: pre- and post-accident errors. The pre-accident errors identified in the full power study for Surry² were adopted, together with others identified in the system analysis task and added to the system fault trees.

To evaluate human actions and recovery actions that follow an initiator, the event scenario, required actions, important factors affecting operator performance, and the consequences of the action being unsuccessful were qualitatively defined. Two types of post accident human errors were modeled, failure to diagnose and failure to carry out the needed action given successful diagnosis. They were used in the fault trees for the top events of the event trees. It was assumed that, given failure to diagnose, the operator would fail to perform the needed actions; therefore, core damage would result. The same basic event representing failure to diagnose was used in all fault trees for a given event tree. On the other hand, failure to carry out the action given successful diagnosis would only fail the specific top event of the event tree.

3 Approach and Limitations

The qualitative evaluation of the actions and the important parameters that affect operator's performance were used to derive the human error probabilities (HEPs) by adapting the success likelihood index methodology.^[13] This methodology^[14] assumes that the likelihood of operator error in a particular situation depends on the combined effects of a small set of performance-shaping factors (PSFs) that influence the operator's ability to accomplish the action.

To quantify the HEPs, the PSFs were rated with weights that reflect the relative influence of each PSF on the likelihood of the success of the action, and a score that reflects whether the PSF helps or hinders the operator in carrying out the actions. With the rating for PSF, the numerical model was calibrated using well defined actions obtained from analysis for other PRAs. Calibration ensures that the error probabilities are realistic and consistent with the data, observed human behavior, and the results from comparable expert evaluations of similar activities.

Data Base Analysis

An extensive effort was devoted to collecting data to characterize the plant during shutdown.

- (1) A data base of initiating events was compiled for the initiating event analysis.
- (2) The shift supervisor's log books, outage schedules, minimum equipment list, and monthly operating reports were reviewed to collect the data needed to estimate the frequency of shutdown, duration of plant operational states, and maintenance unavailabilities.
- (3) The shift supervisor's log books were reviewed to determine the time that the plant is in different configurations. For example, the reactor coolant loops were found to be isolated for a long period in a refueling.

Key Assumptions:

The following is a discussion of the various modeling issues and how they were treated.

Changing Plant Practices and Information

Staff at the Surry plant are aware of the potential safety concerns of reduced inventory operations and they are constantly making improvements regarding such operations. This is reflected in improvements in the operating procedures and abnormal procedures used during shutdown, as well as changes in plant practice. The most significant change in plant practice started in the refueling outage of Unit 1 in 1992, during which mid-loop operation was totally avoided; this appears to be the new policy. Another way of reducing the risk is to carry out reduced inventory operation while the fuel in the core is removed during refueling.

Modifications made to the model developed for this study account for procedures and other plant information available as of April 30, 1993. Regarding the plant's policy of avoiding mid-loop operation, it was decided that this study would use the data collected from past outages before the Unit 1 refueling outage of 1992. Consequently, the core damage frequency reported in this study could be overestimated by making this assumption. However, it is emphasized that the core damage frequency calculated in the current study was reduced significantly by changes made before April 1, 1993.

Changing Plant Configuration

Due to the activities taking place during shutdown, the plant configuration changes with time, which affects the likelihood of initiating events and the ability to mitigate the accidents. In this study, the constantly changing plant configuration was approximated by a number of discrete configurations, namely: different outage types, POSs, and time windows. The changing configuration was also reflected by the different basic events and different event trees for each of these configurations. The following is a description of the basic events and how they varied.

Initiating Event Frequency

The initiating events were assumed to occur with constant rates independent of the outage type or POS. The conditional probability that an initiating event occurs in a POS was calculated as the product of the rate and the duration of the POS. The initiating event frequency is the frequency of the POS multiplied by the conditional probability. The frequency that it occurs in a given time window of a given POS of a given outage type is the initiating event frequency times the conditional probability of the time window of the given POS.

Loop Isolation Probability

Isolating the loops makes it impossible to establish reflux cooling. Loop isolation probability was estimated by judgment using the information from the log books for outages, and an outage plan for a refueling outage. It was estimated as a function of the outage types, and time windows.

Removal of Pressurizer Safety Valves

The fraction of time that the safety valves are removed in a given time window of a given POS in a given outage type was estimated by judgment, using information from the log books for outages, and an outage plan for a refueling outage. With the safety valves removed, it is possible to use gravity feed from the RWST, but not reflux cooling because of inventory loss through the opening.

Maintenance Unavailabilities

Maintenance unavailability was estimated as a function of the POSs of a refueling outage by reviewing the log books for three refueling outages. The data was collected for the period when the plant was at mid-loop. Due to lack of sufficient information, it was assumed to be independent of the time windows. It was also assumed that the data for a refueling outage is applicable to a drained maintenance outage.

Success Criteria

The success criteria for shutdown conditions were determined by reviewing various studies, and performing supporting thermal-hydraulic analysis based on the Surry-specific design. The changing level of decay heat was accounted for by defining four time windows after shutdown, each with its own set of success criteria. In general, whenever the success criteria for one system or mitigating function changes, a new time window needs to be defined, and potentially, more than four time windows could be needed. The use of four windows, therefore, was a trade-off between the accuracy of the model and the level of effort needed to arrive at a solution. It was determined that four time windows give an adequate representation.

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During development of the time window approach, it was recognized that the procedure for loss of RHR (AP-27.00) is conservative with respect to the success criteria for reflux cooling and feed-and-spill, and does not include all possible methods of establishing recirculation. These are the areas in which the plant model used in this study deviated from the abnormal procedure for the plant. The following describes how these issues were treated.

Reflux Cooling

In AP 27.00, the number of steam generators (SGs) needed for reflux cooling is given as a function of the decay heat (e.g., 3 SGs are needed for the first 75 hours after shutdown). This value is based on thermal-hydraulic considerations documented in a Virginia Power technical report.^[10] From a review of existing studies performed by INEL^[7,8] and Westinghouse^[9] on reflux cooling, and independent BNL calculations, it was determined that one steam generator is sufficient for reflux cooling. Therefore, based on more detailed calculations, it appears that one SG is sufficient, while the abnormal procedure states that three would be needed. The issue is, how much credit should be given to reflux cooling when less than three SGs are available. In this study, the conservative success criterion based on the procedure was used in the logic model, and whenever reflux cooling was failed due to insufficient SGs, a recovery action was entered with a failure probability of 0.1.

Feed-and-Spill

In AP27.00 and its supporting study,^[10] the number of pumps and PORVs needed for this operation was determined based on the flow from the RWST needed to maintain sub-cooling, the capacity and shut-off head of the pumps, and the relieving capability of the PORVs. For example, during the first 129 hours after shutdown, 2 charging pumps and 2 PORVs would be needed. This success criterion was derived from the requirement to maintain sub-cooling, and is more stringent than the criterion needed for feed-and-bleed during an accident that occurs during full power operation. An alternative to feed-and-spill, (i.e., feed-and-steam), which is also discussed in reference [10], is much less demanding in terms of the needed flow. However, feed and steam is not the recommended method because of the difficulty in maintaining the level, and the potential for over pressurization. In this study, a success criterion of 1 charging pump and 1 PORV was used, based on the understanding that this is sufficient to prevent core damage.

Recirculation

AP 27.00 instructs the operators to establish high pressure recirculation by using the low-pressure injection pump to take suction from the containment sump, and discharge to the suction of the high pressure injection system: this requires that a low head injection pump and a charging pump are available. In the fault tree model for recirculation, two alternative methods also are modeled, low-pressure feed-and-steam (by taking suction from the containment sump), and low-pressure feed and spill (by taking suction from the sump and using spray recirculation). In these modes, low-head injection is needed. The feed-and-steam mode requires that the safety valves be removed to provide an adequate vent path, and does not require cooling of the sump water. The feed-and-spill operation requires operation of the spray recirculation systems to cool the water in the sump, so that sub-cooling in the reactor vessel can be established.

Operator Response

Operator actions modeled in this study were identified when developing the event trees. The identification process involved reviewing abnormal and emergency procedures, and discussing the accident scenarios with plant personnel.

In most cases, the operator's responses to various accidents are identified in the procedures. For example, abnormal procedures for loss of RHR, loss of instrument air, and loss of offsite power give guidance on what to do in case of respective losses during shutdown. However, the latter two procedures are not written specifically for shutdown conditions. In case of a station blackout, the procedure written for power operation (1-ECA-0.0), does not address shutdown conditions. Therefore, only the relevant steps in the procedure are applicable to shutdown conditions. Similarly, for other initiating events, such as loss of component cooling, spurious safety injection, and loss of a vital bus, there is no specific procedure for shutdown conditions, and the ability of the operators to use the relevant steps in the procedures for power operation becomes very important. As discussed under success criteria, some of the operator actions modeled in this study are not explicitly spelled-out in the existing plant procedures, and some recovery actions modeled are extension of the existing procedures.

The operator actions needed to mitigate an accident are included in the high level fault trees. A high level fault tree models one method of mitigating the accident, e.g., feed-and-spill operation which typically contains two human error events, and one transfer to the fault tree for hardware failures. One human error event represents the failure of the operator to diagnose, so that the correct actions cannot be decided upon; the other represents failure to carry out the action after correct diagnosis. Assuming that the failure to diagnose would lead to core damage, many of the dominant core damage cutsets are caused by such events. Human error probabilities (HEPs) were quantified using the method of failure likelihood index that involves assessing weights and scores on various performance-shaping factors, and calibration using the HEPs from existing studies.

3.1.2 Internal Fires

Fire Area Identification and Scenario Development

Fire areas in which important equipment is located were identified. The areas are defined such that a fire in a certain location within the area might cause an initiating event by disrupting RHR operation. Cable routing information within the area was also included in the analysis (both for power and control cables) for important systems. Important systems were identified in the internal event PRA as providing alternate core cooling paths. Systems and components when damaged by a fire could result in an initiator and possibly a degraded configuration of an alternate core cooling path. This information was eventually included in the event trees and fault trees developed for the internal event PRA. Whether or not a certain cable would be damaged by a fire in a certain location would be decided by the fire growth calculations. The probability of damage would be determined by the fire suppression model.

Fire Frequency Calculations

Fire frequency calculations were performed for important fire areas and for equipment found in these areas. Equipment classes were broken into subclasses (e.g., switchgear panels were divided up into bus panels, monitor control center (MCC) panels, un-interruptible power supply panels, etc.). Cable fires and transient fires were also considered. Large fires and small fires were calculated separately. Generic and Surry-specific fire events were obtained from the updated Sandia Fire data base,^[15] which considers events through 1989. These events were used to calculate Surry specific fire frequencies in specific categories. These fire frequencies were then prorated to account for the fraction of fires in that category that can occur at a specific location.

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Fire Growth Modeling

Once a certain critical location was identified (e.g., cables for important equipment pass nearby), the fire growth model was used to determine if a fire in that location would damage the cable (which may be located at a certain horizontal or vertical distance from the fire source). If damage does occur, the time to damage was also calculated. The model used for fire frequency calculations is the computer code COMPBRN-IIIe.^[16] While this model is an important advance in fire analysis the predications of the code remain uncertain because a number of aspects are not modeled well and simplifying assumptions have to be made.^{[16],[17]}

Fire Suppression and Damage Fraction Calculation

After fire initiation (and if detection occurs) suppression efforts will start to bring the fire under control. Detection and suppression may consist of combinations of manual and automatic systems. These systems may suppress the fire before damage to a critical cable occurs. (The time to damage is input from the fire growth calculations). The probability of damage in a certain scenario is the damage fraction, and is input as a basic event representing damage to the particular component in the PRA model, given the occurrence of a fire in that location. The suppression model uses transition rates among various states in the suppression path toward the eventual control of the fire.^{[18],[19],[20]} The transition rates and a particular combination of suppression steps determine the probability of suppression within a certain time and in the plant area that contains the detection and suppression equipment.

Fault Tree and Event Tree Modifications

The damage fraction calculated was input into the modified internal PRA model as a basic event representing a possible failure mode in a given component following a fire. The internal PRA model for a given initiator (caused by fire) was modified to account for additional equipment that may be damaged in a fire. Failure modes due to fire (e.g., spurious operation or power loss) were added to the internal failure modes of the component. The event tree shows core damage sequences given a fire in a certain location and given occurrence of the initiating event. The sequence frequencies were evaluated by IRRAS 5.0,^[12] the computer program used in the internal PRA quantification.

Key Assumptions:

- (1) Multiple initiators were not considered (e.g., loss of offsite power subsequent to a fire) for probabilistic reasons. For example, loss of offsite power probability in any 24 hour period is about $7E-4$ for the Surry plant at mid-loop. Multiplying this by the mid-loop time fraction of 0.066 means that the fire frequency at a given location would have to be a few times 10^{-3} in order for it to survive screening. No such scenarios were found.
- (2) Barrier failure was not considered as it was determined to be probabilistically unimportant. For initiator-causing fires that occur near an important barrier (very few examples were found), the combination of fire frequency, probability of equipment failures causing an initiator, the mid-loop POS fraction, and barrier failure probability (on the order of 10^{-2} - 10^{-3})^[17] was such that the fire scenario frequency was below the screening value. Also, comparing this probability to the conditional core damage probability for most scenarios (0.1) leads to screening of this consideration. It should be noted that during the plant visits (including during shutdown), it was observed that all normally closed fire doors were kept closed.

- (3) Certain cables were not traced due to lack of resources and/or because the scenarios would be probabilistically unimportant, would not lead to an initiator, or would be an irrelevant post-initiator for the particular initiators possible in the fire area (various compressed air systems, diesel generator cables, ventilation system cables, etc.). In some instances assumptions were made regarding the location of these and other cables (i.e., cables in conduits for which no precise tracing system is available at Surry).
- (4) It was assumed, for purposes of fire frequency calculations, that other plants are similar to Surry in terms of the amount of equipment of a certain type in the plant fire areas (e.g., number of switchgear cabinets in the emergency switchgear room).
- (5) Relay fires were screened probabilistically, because the combination of a relay fire frequency, mid-loop fraction, and the conditional probability of initiator occurrence was below the screening value.
- (6) It should be noted that both power and control cables were traced and spurious operation following hot shorts was treated. Lack of electrical coordination between the 4160 V emergency buses and the 480 V load centers was also taken into account.

3.1.3 Internal Floods

The approach used in the internal flood event analysis was based on approaches used in past PRAs, especially the methodology used in the full power study^[2] and the plant-specific Individual Plant Examination (IPE)^{[21],[22]} of the Surry plant. The basic steps in the analysis involve the following:

Identification of Flood Areas Taking Into Consideration Spatial Interaction of Flood Areas

In the Phase 1 study,^[3] a screening analysis was performed to identify the potential flood areas. These areas were used as the basis for the second phase effort. The potential pathways for flood propagation were identified using the information obtained during a plant visit as well as information available from the NUREG-1150 and IPE studies. The spatial interactions and the various pathways between high risk areas (i.e., Turbine Building and Emergency Switchgear Room and Auxiliary Building) were inspected and evaluated for potential flood damage.

Flood Event Frequency Analysis

The initiating frequency data was developed based on actual flooding events and updated to include plant-specific features and data. The data was gathered from various sources including the IPE Surry flood analysis, industry sources, licensing event reports (LERs). In some cases, plant-specific models were developed especially with regard to the most important circulating water (CW) and service water (SW) lines in the Turbine Building. These two sources of flood dominate the risk, and therefore required special treatment. The most dominant failure mechanism for large pipes is pipe rupture due to water hammer events. Generic pipe failure data and models^[23] were used when plant-specific data was unavailable with the exception of the CW and SW lines. The effects of flood barriers and other mitigating mechanisms were also considered in the development of damage states.

The equipment damage due to internal flood was determined as a function of flood level. The flood level depends on various factors such as the leakage rate, area of interconnections, drainage pathways, and potential mitigating

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actions. In order to take these factors into account, simple flood damage categories or damage states were developed. The damage states effectively incorporate the time dependency of the flood event and any potential mitigating or recovery actions. The damage states reflect a particular flood-exceedance frequency at a particular location with a predefined equipment damage level. This allows the incorporation of any partial failures due to flooding or differentiation between equipment damages at a particular location.

The flood initiating frequencies were developed to incorporate simple recovery actions as expected from the existing experience or data base. In addition, the data base was examined for its applicability to the Surry plant and modified to take into account certain plant-specific features.

Location and Scenario Identification

The identification of potential accident scenarios and flood locations was accomplished by initially using the general arrangement diagrams and the RHR system fault tree developed for the internal event analysis. The fault tree identifies all equipment and support components required for the RHR system operation. These were located and identified on the plant drawings and correlated with the flood areas as identified during the analysis. The scenario development also included the identification of potential propagation pathways and their physical nature (doors, stairways, etc.) which may limit the flood rate. In addition, drainage potential and other mitigating features which could help in the recovery process were also identified. Once the flood scenarios were identified, the effect on other front line systems or their components was determined. This was limited to only those systems which were required to prevent core damage and were identified by examining the internal event trees for the loss of RHR function and/or station blackout. The appropriate equipment and/or its support were located and placed in the appropriate flood areas.

The quantification takes into account the rate of water flow into the area or height of the flood and establishes generally a critical flood level in each specific plant area. The time available for any recovery action was estimated by defining the critical volume and the applicable flow rate. This approach allows the incorporation of simple recovery actions in the quantification without greatly modifying the existing fault trees and internal event trees.

Plant Response Analysis

The internal flood analysis utilized the system fault trees and event trees developed for the internal event study. The flood events were categorized as to their effect on the residual heat removal system and were mapped to the corresponding event tree for internal events. The event trees were modified to take into account the specific accident evolution of the flood event. In addition, the system fault trees were also modified according to the effects of the flood on equipment used to mitigate the accident. The relevant human error probabilities were also modified as timing and potential spatial interactions complicated the accident progression.

3.1.4 Seismic Events

In this study, only a Level 1 seismic PRA was performed. The key elements are:

- (1) Seismic hazard analysis - estimation of the frequency of various levels of seismic ground motion (acceleration) occurring at the site.

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- (2) Seismic fragility evaluation - estimation of the conditional probabilities of structural or equipment failure for given levels of ground acceleration.
- (3) Systems/accident sequence analysis - modeling of the various combinations of structural and equipment failures that could initiate and propagate a seismic core damage accident sequence.
- (4) Evaluation of core damage frequency - assembly of the results of the seismic hazard, fragility, and systems analyses to estimate the frequencies of core damage for various accident sequences and for the plant as a whole.

Key Assumptions:

- (1) The seismic study adopted all of the assumptions made in the internal events analysis without exception. Most importantly, the loss-of-offsite-power (LOOP) event trees were adopted directly. This included the definitions of the top events and underlying thermal-hydraulic and other assumptions that support these event trees, namely those corresponding to mid-loop operation.
- (2) Only refueling outages were considered in this analysis. Indeed, only certain specific operating states within the standard refueling outages at Surry were considered. Outages for other reasons frequently occur at nuclear power plants, and they are of two broad types: controlled shutdowns and uncontrolled (rapid) shutdowns. Although these outages for other reasons can produce different plant operating states and configurations unique to the reason for the shutdown, resources in this analysis did not permit examining outage configurations other than those for refueling. In any event, refueling outages contribute a majority of the shutdown time.
- (3) It was assumed that the only seismic events of concern are those that cause loss-of-offsite-power (LOOP) transients. Seismic events of lower acceleration than those causing LOOP are expected to have a negligible probability of causing severe plant accidents for two reasons: (i) Critical plant equipment, including the residual heat removal (RHR) system, can withstand significantly higher accelerations than that which is sufficient to cause LOOP.^{[24],[25]} Thus, loss of core-cooling capability will have a very low probability for seismic events too small to cause a LOOP. (ii) With offsite power available, sources of water sufficient to cool the core from alternative pumping sources will generally be available even if the RHR system fails.
- (4) It was also assumed that a seismic-initiated LOOP is non-recoverable in the time frame of interest in this study (from about one to several hours). This is a reasonable and only slightly conservative assumption, because the LOOP initiator is most likely to arise from the seismic-caused failure of the ceramic insulators in the plant substation.^[24,25] Replacement of these insulators would likely require several hours at a minimum, and probably much longer. Additionally, other damage caused by the earthquake (for example, to offsite transmission systems and offsite switch yards) would likely hamper efforts to recover offsite power.
- (5) The engineering methodology used to develop probabilistic seismic fragility curves employed a successive-screening approach, in which structures and equipment that are judged to be very strong under earthquake loading were screened out first. In this manner, the number of items for which actual numerical fragilities had to be developed was limited. Also, for a few items the study team was unable to obtain enough information,

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either because access to some parts of the plant was limited or because engineering information in the appropriate form was not available. For these items, generic fragilities were used.

- (6) The seismic failures of identical equipment in similar locations were assumed to be fully correlated. This means, for example, that when a postulated earthquake causes one diesel generator to fail, it was assumed that the other diesel generator(s) would also fail. This simplifying assumption is probably conservative in many cases, but perhaps not overly so for truly identical equipment. Sensitivity studies performed in several past seismic PRAs have shown that the bottom-line results are sensitive to this assumption at a level of about a factor of plus-or-minus two.
- (7) Equipment failure from seismic-induced relay chatter is outside the scope of this analysis. While relay chatter can be important,^[26] it is a complicated issue, and the resources to study it were not available within this project. In any event, relay chatter is being studied at every U.S. nuclear plant as part of the current IPEEE program,^[27] and it is expected that almost all of the relays that are especially sensitive to relay chatter will be identified and, if appropriate, modified in the course of the IPEEE studies.
- (8) Both the Lawrence Livermore National Laboratory (LLNL)^[28] and Electric Power Research Institute (EPRI)^[29] seismic hazard curves were used to produce two parallel results.

3.2 Level 2/3

A phased approach was also used in the Level 2/3 program; however, both phases addressed the risk from only mid-loop operation. The first phase of the Level 2/3 PRA was initiated in late 1991 and consisted of an Abridged Risk Study. This study focused on accident progression and consequences, conditional on core damage. Phase 2 is a more detailed study in which an integrated evaluation of risk due to internal events during mid-loop operation was performed. Risk estimates were not made for accidents initiated by internal fires, internal floods, and seismic events. As this study is limited to accidents during mid-loop operation, it is not a complete risk estimate for accidents that could occur during low power and shut down operation. The approach adopted in the Phase 2 study is mostly based on the NUREG-1150 approach. The sections below therefore describe only those elements of the methodology which are different from the NUREG-1150 approach.

Accident Frequency Analysis

The Level 1 analysis, including fault trees, event trees, recovery actions, etc., of the significant accident sequences leading to core damage and their frequencies, was carried out by the IRRAS code.^[12] A newly added feature of IRRAS which became available recently was used to group the minimal cut sets into the plant damage states (PDSs). Seven characteristics were used to construct the plant damage states.

The first characteristic identifies the time frame in which the accident occurs. A major difference between a PRA at full power and at mid-loop operation is the extended time period following shutdown during which an accident can occur. This time period allows for a significant decay of the power level, extends the time available for various phenomena and for recovery actions and leads to a lower value of the radionuclide inventory which can potentially be released. This feature of the shutdown PRA was modeled in the Level 1, accident frequency analysis through the

construction of four "time windows" for various time periods following shutdown. Each time window has its own decay heat level and success criteria for accomplishing various recovery actions prior to core damage. The first PDS characteristic therefore identifies the time window in which the accident occurs.

The second characteristic provides the status of ac power. Of particular interest is whether or not ac power is available and if it is not available whether it can be recovered. Human error is an important contributor to the core damage accident frequency for mid-loop operation. The third characteristic therefore identifies if human error contributed to the accident and if it did what was the type of error. The status of the reactor coolant system can significantly impact accident progression and this is therefore addressed in the fourth PDS characteristic. For some plant damage states recovering coolant injection after the start of core damage can prevent further core damage and terminate the accident prior to the core melting through the reactor vessel. The fifth characteristic deals with the issue of restoring coolant injection. If core damage occurs and containment integrity is lost then operation of the containment sprays can reduce the airborne fission product aerosol concentration and reduce the amount of radionuclides released to the environment. The sixth characteristic gives the status of the spray system. Finally the status (injected or not injected) of the refueling water storage tank is given in the seventh characteristic.

Using the above characteristics, the Level 1 cutsets were binned into 48 plant damage states. These plant damage states were further binned into four PDS groups to be processed by the accident progression event tree (APET) analysis. The main consideration in binning was the similarity of accident progression after core damage.

PDS Group 1 consists of station blackout accidents.

PDS Group 2 consists of accidents attributable to human error.

PDS Group 3 consists of accidents in which core cooling is lost during recirculation.

PDS Group 4 consists of accidents where core cooling is lost because of loss of the 4 kV bus.

Accident Progression Analysis

The accident progression event tree developed for mid-loop operation has been developed largely based on the APET developed for full power operation in the NUREG-1150 study. Some questions were removed and other questions pertinent to mid-loop operation (such as time windows and containment closure) were added. The APET for mid-loop operation consists of 40 questions compared to 71 questions in the full power APET. Due to resource limitations, a formal expert elicitation procedure could not be implemented to construct ranges of values and distributions for key variables. Assignments of these values and ranges were therefore made internally at BNL. The timing of key events in the accident progression is based on calculations carried out with the MELCOR code.^[6] A very important issue which has a major impact on the result is the status of the containment at accident initiation. Assignments of the possible values of this status were made based on discussions and exchange of written communications with the Surry plant personnel. The APET was quantified using the EVNTRE^[30] code as in the full power study. It was again necessary to combine the numerous outcomes of the APET into accident progression bins (APBs) for input to the source term analysis. A similar approach to that used in the full power study was also used in the current analysis. The only additional information needed for the current study was the time window in which the accident occurred. Thus information identifying the time window was carried throughout all of the constituent analyses.

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Source Term Analysis

The source term analysis used for mid-loop operation was similar to the approach used for full-power operation. The SORSOR^[31] code was reviewed for its applicability to shutdown conditions by an expert group consisting of staff from BNL and SNL. The source term ranges in SORSOR were also compared against predictions with the MELCOR code. In general SORSOR was considered appropriate for use in the current study. The APBs were therefore processed through the SORSOR code in a similar manner to the full power study. The output from SORSOR is a larger number of source terms which need to be grouped into representative source terms. The process was done in the full power study using PARTITION.^[32]

Methodologically, the important difference between NUREG-1150 and the present study is a reworking of the partition approach to reflect the long time interval and consequent decay of the inventory in the current study. In effect, accident progression bins and source term groups are defined for each time window. The partitioning of the source terms and the assignment of health effect weights is carried out through a partition procedure designated PARTITION-LPS.

Consequence Analysis

The consequence measures for the average source term in each source term group were calculated by the MACCS code.^[33] The latest release of MACCS, Version 1.5.11.1, which incorporates the important BEIR V update to the latent cancer-dose relationship,^[34] was used to compute consequences. In contrast, the NUREG-1150 study used an earlier version of MACCS, Version 1.5.11, to compute consequences. The more recent version of MACCS gives a higher (by approximately a factor of 3) number of latent cancers, than the earlier MACCS version for the same value of population dose.

Risk Integration and Uncertainty

The integrated risk was obtained by combining the individual results of each of the constituent analyses. The approach was similar to the NUREG-1150 approach. Distributions were assigned to important variables (some distributions were identical to those used in NUREG-1150, others were developed specifically for mid-loop operation) and samples were then created by randomly picking values from the distributions using Latin Hypercube Sampling.^[35] For each sample the values assigned to each variable were propagated through the integrated analysis to determine risk estimates for each consequence measure. By repeating the calculation for 100 samples (or observations) distributions of risk estimates were obtained from which the uncertainty in risk was determined.

Key Assumptions (Level 2/3 Study):

- (1) Discretization of time windows and decay heat levels: Decay heat level is a key parameter in the accident analysis due to the long time interval, depending on the POS and outage type, over which an accident can potentially occur during mid-loop operation. Four time windows with corresponding decay heat levels were constructed in the Level 1 analysis and it was assumed that the decay heat level (which varies continuously) of each time window can be adequately represented by its value at the mid-point of the time window.
- (2) Containment Status: Several assumptions had to be made on the status of the containment during mid-loop operation. The pressure capability ranges from no pressure retaining capability (leakage at inception of release)

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to full design capability (as at full power operation). The ability of containment to retain fission products released from damaged fuel is the dominant factor affecting risk.

- (3) **Source Term:** It was assumed that the source term code, SRSOR, which was developed for the full power study would adequately apply to low power and shutdown conditions as well. This assumption was checked through spot comparisons with calculations based on the mechanistic code MELCOR and by a review performed by a Source Term Advisory Group comprised of BNL and SNL staff.
- (4) **Consequences:** The consequence calculation assumed the same emergency response for the offsite population in the low population zone surrounding Surry and the same long-term protective actions as the NUREG-1150 study.
- (5) **Onsite Doses:** The scoping calculation of onsite doses assumed that the releases were directly from the containment to the environment through the equipment hatch and not through the personnel hatch so no in-building doses were calculated.

Strengths and Limitations (Level 2/3 Study)

The strengths of the Level 2/3 study are:

- (1) It is a systematic and integrated evaluation of risk during mid-loop operation at the Surry Unit 1 plant, including accident frequency, accident progression, source term and consequence analysis with a determination of uncertainty in each of the component analyses and in the final risk measures.
- (2) The integrated analysis takes into account the long time after shutdown that the accidents can occur and the impact of the consequent decay in power level and radio nuclide inventory on the risk. In particular, new latent cancer weights were derived for source term partitioning and used in the consequence calculation.
- (3) The newest version of the MELCOR code, Version 1.8.2, was used to calculate the timing of key events in the accident progression which were then used in the accident progression event tree.
- (4) The accident progression event tree has sufficient detail to account for a significant portion of the likely paths of evolution of the accident.
- (5) The study includes a scoping calculation of onsite dose rates at locations in the vicinity of the plant during the accident.

The Level 2/3 study has the following main limitations:

- (1) There was no formal expert elicitation process used, as in the NUREG-1150 study, to provide values and distributions for key variables in the accident progression. Assignments for these variables had to be made internally at BNL or derived from analogy with full power conditions. The selection of the key variables themselves was also made internally at BNL. Thus the uncertainty analysis is not as robust as it could have been with input from an expert panel.

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- (2) The scope of the Level 2/3 study is limited to accidents initiated by internal events (due to equipment failure or human error) during mid-loop operation. Risk estimates were not made for accidents initiated by internal fires, internal floods and seismic events. The final risk numbers should therefore not be interpreted to reflect the risk of all plant operational states during low power and shutdown operation.

4 RESULTS

4.1 Level 1 Results

4.1.1 Quantitative Results from Traditional Internal Events

Table 1 summarizes the results of the event tree quantification, showing the core damage frequency as a function of the initiating events and POSs. The core damage frequency is the frequency that core damage occurs while the reactor is at mid-loop, and includes the fraction of a year that the reactor is at mid-loop. POS 6 of a drained maintenance outage (D6), and POS 6 of a refueling outage are the most dominant POSs. The characteristics of these POSs are high decay heat level and a relatively short time available for operator action. In contrast, POS 10 of a refueling outage has a very low decay heat, and its core damage frequency is approximately one order of magnitude lower.

Table 2 compares the results of this study with those of NUREG-1150 and the individual plant examination (IPE) performed by the utility for Surry. The results are displayed in two ways. The core damage frequency, shown in the first row, is the frequency that core damage occurs when the plant is at mid-loop (the core damage frequencies in the parentheses are the contributions due to over-draining events), and the core damage frequency, shown in the third row, is the core damage frequency (minus the contribution of over-draining events) divided by the fraction of time the plant is at mid-loop (second row). The former accounts for the fact that the plant is at mid-loop only a small fraction of the time, while the latter is the core damage frequency conditional on the plant being at mid-loop. The core damage frequency of mid-loop operation is approximately an order of magnitude lower than the CDF estimated in NUREG-1150 for power operation. However, the plant is only at mid-loop operation approximately 7% of a calendar year. The numbers in the parentheses of the third row of the table are the conditional probability of core damage due to over-draining events, given that the plant enters mid-loop operation in the POS.

The core damage frequencies shown in the first row of Table 2 are additive. That is, the sum of the core damage frequencies of the three POSs is the total core damage frequency of mid-loop operation. This total ($5 \text{ E-}06$ per year) can be added to the core damage frequency of power operation (e.g., $4 \text{ E-}05$ per year for NUREG-1150). Therefore, the sum of $4.5 \text{ E-}05$ per year is the frequency per year that core damage occurs while the plant is at full power or mid-loop operation.

The core damage frequency shown in the third row of Table 2 is conditional on the plant being at mid-loop. The fact that the conditional core damage frequency of mid-loop operation ($7.6 \text{ E-}05$ per year) is higher than that of full power operation ($4 \text{ E-}05$ per year) indicates that mid-loop operation may be more vulnerable to core damage than full power operation. However, the plant is at mid-loop only a small fraction of the time.

Table 3 lists the conditional core damage frequency as a function of the time windows and POSs. The conditional core damage frequency is the rate at which core damage occurs given that the plant is in the time window of the POS. It is obtained by dividing the core damage frequency by the fraction of time the plant is in the time window of the POS. The conditional core damage frequency is a measure that can be used to compare the vulnerability of the time windows and POSs with respect to core damage. It can be seen, from Table 3, that for each POS the conditional

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Table 1 Summary of Results: Core Damage Frequency by Initiating Event and Plant Operational States

	Initiating Event	Point Estimate Core Damage Frequency (per year)			
		R6	R10	D6	Total
1.	Loss of RHR				
	RHR2A-Over Draining	1.8E-7	5.3E-8	2.6E-7	4.9E-7
	RHR2B-Failure to Maintain Level	2.1E-08	2.0E-8	2.9E-8	7.0E-8
	RHR3-Non-Recoverable Loss of RHR	1.5E-7	8.4E-9	3.0E-7	4.6E-7
	RHR4-Non-Recoverable Loss of Operating Train of RHR	7.6E-9	1.2E-9	2.3E-8	3.2E-8
	RHR5-Recoverable Loss of RHR	4.0E-8	4.1E-09	9.3E-8	1.4E-7
2.	LOOP-Loss of Offsite Power				
	L1-Both 1H and 1J Energized	3.3E-7	7.0E-8	7.6E-7	1.2E-6
	L2-1H and 2H energized, not 1J	1.0E-7	1.3E-8	1.7E-7	2.9E-7
	L3-1H energized, not 1J, unit 2 blackout	4.2E-8	1.3E-8	9.9E-8	1.5E-7
	B1-Unit 1 Blackout	4.8E-8	1.1E-8	1.7E-7	2.3E-7
	B2-2 Unit Blackout	3.8E-8	4.2E-8	1.1E-7	1.9E-7
3.	4KV-Loss of 4 kV Bus	1.4E-7	1.9E-8	2.4E-7	4.0E-7
4.	VITAL-Loss of Vital Bus	2.8E-8	5.1E-9	7.3E-8	1.1E-7
5.	AIR-Loss of Outside Instrument Air	7.9E-10	—	3.2E-9	4.0E-9
6.	CCW-Loss of CCW	6.3E-8	1.1E-10	2.1E-7	2.7E-7
7.	SWGR-Loss of Emergency Switchgear Room Cooling	3.6E-8	1.2E-8	7.4E-8	1.2E-7
8.	ESFAS-Inadvertent Safety Feature Actuation	2.7E-7	2.7E-8	6.8E-7	9.8E-7
9.	Dilute-Boron Dilution (CDF)				6.8E-08
TOTAL		1.5E-6	3.0E-7	3.3E-6	5.1E-6*

Notes: * Not including boron dilution
R6 = Plant operational State 6 of a refueling outage
R10 = Plant operational State 10 of a refueling outage
D6 = Plant operational State 6 of a drained maintenance

Table 2 Comparison of Total Core Damage Frequency (CDF) with NUREG-1150 and IPE

Study	Results				
		R6	R10	D6	TOTAL
PWR Low Power and Shutdown Study (Mid-Loop POSs, Internal Event Only, Point Estimate)	CDF* per year	1.6E-06 (1.8E-07)**	3.1E-07 (5.5E-08)**	3.3E-06 (2.7E-07)**	5.1E-06 (5.0E-07)**
	Fraction of a year the plant is in mid-loop	1.6E-02	1.5E-02	3.5E-02	6.6E-02
	Conditional CDF*** per year (CDP)	8.3E-05 (3.0E-07)	1.7E-05 (1.8E-07)	8.7E-05 (2.3E-07)	7.6E-05**** (2.4E-07)
NUREG-1150 (Internal Events Only, at Power, Mean Value)	4.0E-05				
IPE (Internal Event Only, Point Estimate)	7.4E-05				

* CDF reflects the fraction of time the plant is at mid-loop

** Contribution of over-draining events

*** Frequency of core damage given that the plant is at mid-loop

**** Average over the POSs

(CDP) probability of core damage due to over-draining upon entering the POS

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Table 3 Conditional Core Damage Frequency As a Function of the Time Windows and POSs (per year)

	R6	R10	D6	Average
Window 1 (13hr-75hr)	1.0E-03	—	3.4E-04	3.8E-04
Window 2 (75hr-240hr)	7.6E-05	—	5.9E-05	7.3E-05
Window 3 (240-768hr)	5.5E-05	6.5E-05	5.2E-05	5.6E-05
Window 4 (> 768hr)	1.9E-05	1.6E-05	1.1E-05	1.8E-05
Average	8.3E-05	1.7E-05	8.7E-05	7.0E-05

Notes:

R6 = Plant Operational State 6 of a refueling outage

R10 = Plant Operational State 10 of a refueling outage

D6 = Plant Operational State 6 of a drained maintenance

core damage frequency decreases with time window. This is due to the relaxed success criteria and more time available for operator actions in the later time windows. The conditional core damage frequency for R6 or R10 is higher than for D6 mainly because the RCS loops have a high probability of being isolated in a refueling outage which makes reflux cooling impossible. For example, in window 1, the probability that the loops are isolated in a refueling outage is 0.3, and the probability that reflux cooling fails in a drained maintenance outage is 0.1 (modeled as a recovery action). The difference between R6 and R10 in windows 3 and 4 is due to the difference in maintenance unavailabilities.

The averages in Table 3 represent the averaged (weighted by the frequency that initiating events occur) conditional core damage frequency. For example, the averaged conditional core damage frequency for R6 is 8E-05 per year, while that for D6 is 9E-05 per year. This does not contradict the comparison made earlier for a given time window of the POSs, because given that plant is in D6 the plant is more likely to be in the earlier time windows that have higher conditional core damage frequency. The averaged conditional core damage frequency over the POSs, shown in the last column of Table 3, does show the trend of decreasing with decay heat.

Table 4 lists the key uncertainty characteristics of the core damage frequencies for mid-loop operation and power operation, and shows that the core damage for mid-loop operation has a larger dispersion than that of power operation. Note also that the mean CDF in Table 4 is slightly different for the total CDF in Tables 1 and 2. This is because the numbers in Tables 1 and 2 are point estimates whereas the information in Table 4 reflects an uncertainty analysis.

Table 4 Result of the Level 1 Uncertainty Analysis and Comparison with Full Power Operation (per year)

	Study		Mean	5th Percentile	50th Percentile	95th Percentile	Error Factor
Internal Events	Full Power Operation - NUREG 1150 (per year)		4.0E-05	6.8E-06	2.3E-05	1.3E-04	4.4
	Full Power Operation- IPE		7.4E-05*	—	—	—	—
	Mid-Loop Operation (per year)		4.9E-06	4.8E-07	2.1E-06	1.5E-05	5.7
Internal Fires	Full Power Operation - NUREG 1150 (per year)		1.1E-05	—	—	—	—
	Full Power Operation- IPE		**	—	—	—	—
	Mid-Loop Operation (per year)		2.2E-05	1.4E-06	9.1E-06	7.6E-05	7.2
Internal Flood	Full Power Operation - NUREG 1150 (per year)		***	—	—	—	—
	Full Power Operation- IPE		5.0E05**	—	—	—	—
	Mid-Loop Operation (per year)		4.8E-06	2.2E-07	1.7E-06	1.8E-05	9.0
Seismic Events	Full Power Operation - NUREG 1150 (per year)	LLNL	1.2E-04	—	—	—	33
		EPRI	4.0E-05	—	—	—	4.4
	Full Power Operation- IPE		**	—	—	—	—
	Mid-Loop Operation (per year)****	LLNL	3.5E-07	1.3E-09	4.0E-08	1.4E-06	32
		EPRI	8.6E-08	2.5E-10	9.7E-09	3.7E-07	37

* point estimate

** not available

*** below truncation of 1.0E-08 per year

**** refueling outage only (no drained maintenance)

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4.1.2 Insights from Internal Event Analysis

The following insights were gained from this study. They are based on the Surry-specific design and operation. Their applicability and significance with respect to other PWRs would have to be assessed separately.

Operator Response

The dominant cause of core damage was the operator's failure to mitigate the accident. (There is very large uncertainty in the human error probabilities used in this study.) In general, it would be beneficial to have good training, procedures, and instrumentation to ensure that the utility's staff can respond to shutdown accidents.

Procedures for Shutdown Accidents

Very few procedures are available for accidents during shutdown; the procedure for loss of decay heat removal, AP 27.00, is the only one that was written specifically for the shutdown scenarios analyzed in this study. The procedure is conservative with regard to the equipment needed to establish reflux cooling and feed-and-bleed. In this study, the use of fewer than the number of steam generators specified in the procedure for reflux cooling was treated as a recovery action, and a more realistic success criteria was used for feed-and-bleed when the decay heat is high. In most cases, the information in the procedures for power operation is helpful, for shutdown accidents. For example, the procedure for station blackout, ECA-0.0, gives instructions for dumping steam to the condenser. Credit for this procedure was taken into account in this study. However, some procedures written for power operation would mislead the operator if followed during shutdown. For example, the procedure for loss of offsite power, AP 10.00, states that "When the EDG is the only source of power to an emergency bus, the Component Cooling Pump should NOT be in service". During shutdown, CCW flow to the RHR heat exchanger is necessary for decay heat removal. Therefore, following this procedure under these circumstances would not be the most appropriate operator response.

Instrumentation

The level instrumentation used during mid-loop operation (i.e., standpipe level instrumentation and ultra-sonic level instrumentation) has limited applicability during a shutdown accident. The standpipe system indicates the correct level only when there is no build-up of pressure in the system. The ultra-sonic level instrumentation only provides level indication when the level is within the reactor coolant loops, and therefore, may not be useful during a feed and bleed operation.

Supporting Thermal-Hydraulic Analysis

The thermal-hydraulic behavior of the reactor coolant system is rather complex, mainly because the pressurizer is usually the relief path for coolant or steam, and the vessel head does not have a large vent. When performing thermal-hydraulic analysis in support of the PRA effort, consideration must be given to longer term system behavior, at least 24 hours into the accident. In this study, such calculations were done for feed-and-bleed operation using a charging pump, and with gravity feed from the RWST. Additional calculations would be helpful to better understanding the effectiveness of reflux cooling, and feed and bleed using a low pressure injection pump. In this study, the conservative results of the Virginia Power Technical Report # 865 (Revision 1, dated July 3, 1992)^[10] were used to determine the number of steam generators needed as a function of time after shutdown, because such criteria

are explicitly written in the procedure for loss of RHR. In the event trees and fault trees, it was conservatively assumed that if there were too few steam generators, then no credit was given to reflux cooling. In this case, reflux cooling still would help. In fact, a review of the studies performed by Westinghouse^[9] and the Idaho National Engineering Laboratory^[7,8] found that one steam generator is sufficient for any level of decay heat. To take some credit for reflux cooling in this case, a recovery action with failure probability 0.1 was applied to those core damage cutsets that involve failure of reflux cooling due to insufficient steam generators. It was assumed that hot-leg injection using a low head injection pump is adequate to prevent core damage. Due to the low shut-off head of the pumps, approximately 150 psig, the concern is that if boiling takes place in the system, the low head pump may be unable to inject.

Maintenance Unavailability

A review of shift supervisor's log books and minimum equipment lists for three refueling outages showed that the maintenance unavailabilities of equipment that can be used to mitigate an accident were very high. For example, two out of three charging pumps were found to be tagged out practically throughout the whole mid-loop period. The two low-head injection pumps also were unavailable a large fraction of the time. Generic Letter 88-17, requires the plant to have one high-head pump and one low-head pump available. For the purpose of quantification it was assumed that charging pump A, charging pump cooling water pump A, and low head injection pump B are available. From the check list used for reduced inventory conditions, it was assumed that the maintenance of diesel generators, 4 kV emergency buses, and stub buses is not allowed.

Maintenance unavailability was found to be the dominant cause of equipment unavailability. In combination with human errors, maintenance of the charging pump cooling water pump, the charging pump, and the low head injection pump appear in the dominant cutsets for some of the core damage sequences.

Isolation of Reactor Coolant Loops

Isolation of the RCS loops is an important contributor to the core damage frequency. Review of the plant shutdown experience indicated that the reactor coolant loops are isolated for extended periods in a refueling outage, making the steam generators unavailable for decay heat removal upon loss of RHR. In a cold shutdown condition, the steam generators are usually maintained in the wet lay-up condition with the secondary side filled with water. During mid-loop operation, the availability of the SGs makes reflux cooling a possible method of mitigating a loss of RHR; this might be the only mitigation function available in a station blackout.

Single Failures of the RHR System

The RHR system at Surry has no active safety function (i.e., it does not perform the safety injection function in scenarios initiated at full power). Consequently, many single component failures can cause loss of RHR. In the RHR system, a single suction line from the loop A hot leg and a single flow control valve HCV-1758 are used. During its operation, a single CCW header provides cooling to both RHR pump seal coolers and the operating RHR heat exchanger, and two CCW return lines from the RHR system are used. Hence, a failure of the trip valve 109A or B in one of the two CCW return lines can cause loss of the system. These trip valves also fail closed on loss of instrument air, or vital bus. Closure of the TV-109 valves is a significant contributor to loss of RHR. It was assumed that the opening of the RHR flow control valve HCV-1758 as a result of loss of vital bus III will cause RHR pump run out: this also was a significant contributor to loss of RHR.

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Valve Arrangement of Auxiliary Feedwater System and Main Steam System during Shutdown

The auxiliary feedwater system has six motor operated valves in the flow path to the steam generators, that are normally closed during shutdown. They are difficult to locate during a station blackout. Similarly, the main steam non-return valves are normally closed during shutdown, and have to be opened to use steam dump to the condenser. These valves depend on offsite power and would be very difficult to open without it.

Potential for Plugging the Containment Sump when Recirculation is Needed

Because of various activities, transient material and equipment are brought into containment during shutdown. For example, large plastic Herculite sheets are often used to separate work areas from the rest of the containment. When an accident requiring recirculation from the containment sump occurs (as in time windows 1 and 2) the material would increase the potential for plugging the containment sump.

4.1.3 Quantitative Results from Internal Fire Analysis

Table 5 summarizes the point estimate results from the internal fire analysis. The results of an uncertainty analysis on the core damage frequency due to fires is included in Table 4. The quantification indicates that scenarios in the emergency switchgear room (ESGR), the cable vault and tunnel, and the containment are important contributors to the CDF. The most dominant scenarios occur in the cable vault and tunnel (due to proximity of many emergency cables from both divisions in a closed, constrained space) and in the ESGR, where many emergency cables from both divisions come together in close proximity (before entering the control room). In the containment, the relatively high CDF is due to a relatively high scenario frequency combined with non-separation of RHR trains over significant distances. Other scenarios are also important, due to moderate damage from the fire combined with a relatively high scenario frequency.

POSSs D6 and R6 are much more important than R10 (as R10 occurs in later time windows). D6 is more important than R6 due to constraints imposed by a drained maintenance outage and its tendency to occur in earlier time windows.

The earlier time windows are more important than the later ones, with window 4 being relatively unimportant. Windows 1 and 2 are of the highest importance, with window 2 being significantly more important than window 1. While the decay heat is higher and the success criteria are more stringent in window 1, this window is relatively short duration and the outages tend to occur in the later time windows. The most risk significant fire initiator occurs in the cable vault tunnel area, in window 2 and POS D6, followed by a few scenarios in the ESGR, in the same window and POS.

Before the recovery actions were applied, the core damage frequency due to fire events at mid-loop was $2.7\text{E-}05/\text{yr}$. After application of recovery actions, the core damage frequency was reduced to $1.7\text{E-}05/\text{yr}$. Note that this point estimate differs from the total core damage frequency of Table 5, i.e., $1.8\text{E-}05/\text{yr}$. This is because the total core damage frequency of Table 5 was calculated simply as the sum of the frequencies of the sequences, while the point estimate of $1.7\text{E-}05/\text{yr}$ was calculated using the minimum-cutset-upper-bound method implemented in the IRRAS computer code. The recovery actions do not reduce the core damage frequency significantly, because they are only

effective in windows 3 and 4 (in order to satisfy the 24 hour success criterion), whereas windows 1 and 2 are more dominant.

Table 5 Summary of Point Estimate Core Damage Frequencies for Fire Events (per year)

Fire Area	R6	D6	R10	Total
Emergency Switchgear Room	4.1E-06	8.2E-06	2.1E-07	1.3E-05
Containment	7.0E-08	5.5E-07	5.0E-09	6.3E-07
Cable Vault and Tunnel	1.3E-06	2.7E-06	7.4E-08	4.0E-06
Normal Switchgear Room	1.5E-08	3.5E-08	1.4E-09	5.1E-08
Main Control Room	7.0E-08	5.3E-07	4.4E-09	6.0E-07
Total	5.5E-06	1.2E-05	2.9E-07	1.8E-05

Notes:

R6 = Plant Operational State 6 of a refueling outage

R10 = Plant Operational State 10 of a refueling outage

D6 = Plant Operational State 6 of a drained maintenance

Table 4 summarizes the result of the uncertainty analysis for core damage accidents. The results on internal fires were obtained by performing uncertainty analysis using 500 Latin Hypercube Sampling (LHS) samples. Also shown in the table is the uncertainty analysis results of the internal event analysis as well as the mean value of the internal fire analysis of NUREG-1150.

4.1.4 Insights from Internal Fire Analysis

Fires during shutdown are not as prevalent as fires during power operation (after the construction events are taken out). However, a greater potential does exist for fires in certain categories (e.g., transient or welding igniting cables or other equipment fires) during shutdown operations, but the possibility of some types of fires is reduced (e.g., deenergized equipment, oil dripping on hot piping). A fire at shutdown is liable to be detected much sooner and extinguished in its early phases, because of increased floor traffic. (Credit is taken for this by disallowing events that were discovered in the smoking stage (without flames) or early enough such that deenergizing equipment extinguished the fire.) At Surry, a fire watch is in place during welding operations and the fire doors are kept closed.

Human error events are not prominent contributors individually in terms of the Fussell-Vesely importance measure (a few percent). Part of the reason for lack of importance of individual HEPs is that there are many events, each applicable in a small fraction of sequences; another reason is in the values assigned; the third reason is that in many important scenarios hardware failures dominate due to heavy damage by fire.

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Table 4 provides a comparison of the fire induced core damage frequency during mid-loop operation with that of power operation. Although the plant spends much less time in mid-loop, the core damage frequency is comparable to that of power operation. The main reason for this result is the routing of cables needed to support RHR operation or to mitigate an accident during mid-loop operation. The cable routing is such that a single fire at a few critical locations can damage almost all the equipment need to mitigate the accident, while during power operation much fewer critical locations exist.

4.1.5 Quantitative Results from Internal Flood Analysis

The main results of the flood analysis are presented in Table 6, listing the point estimate core damage frequencies of the analyzed operating states. The results of the uncertainty analysis are shown in Table 4, indicating the uncertainty bounds of the core damage frequency due to internal floods. It was found that the most dominant contributors to core damage due to internal floods are accident scenarios initiated in the Turbine Building leading to the draining of the intake canal. This potentially could result in a flood encompassing the plant Emergency Switchgear Rooms (ESGR) leading to a two unit loss of all emergency power (F1 and F2 scenarios). The scenarios account for approximately 85% of the total core damage frequency (CDF) due to internal floods. This result is mainly due to the specific features of the Surry circulating water system and may not be applicable to other plants. The second most dominating flood scenario involves flooding of the Safeguard/Auxiliary Building in combination with the unavailability of the Refueling Water Storage Tank (RWST). The contribution of these scenarios (F4 and F5) is approximately 13% of the total internal CDF.

Table 6 Summary of Point Estimate Core Damage Frequencies for Flood Events (per year)

Scenario	Core Damage Frequency with Recovery			
	POS 6 Refueling	POS 6 Drained	POS 10 Refueling	Total
Turbine Building F1	1.9E-06	9.3E-08	1.5E-06	3.5E-06
Turbine Building F2	4.5E-07	2.2E-08	3.6E-07	8.3E-07
Auxiliary Building F3	4.7E-08	4.3E-08	1.2E-08	1.0E-07
Auxiliary Building F4	1.6E-07	5.7E-08	6.7E-08	2.8E-07
Safeguard Area F5	2.0E-07	8.9E-08	9.4E-08	3.8E-07
Spray in Containment F6	—	—	—	—
Mechanical Equipment Room No. #3 - F7	1.0E-08	1.5E-08	7.8E-09	3.3E-08
Total-Flood	2.8E-06	3.2E-07	2.0E-06	5.1E-06

4.1.6 Insights from Internal Flood Analysis

The internal flood CDF is dominated by Turbine Building flood events. These events are primarily initiated by either valve or expansion joint failures in the main inlet lines of the circulating water system. These failures may lead to pipe ruptures upstream of the condenser water box and inlet valves. At Surry the circulating water system is gravity fed from a very large capacity intake canal and its isolation may not be accomplished in a timely manner. This is in contrast with other common design arrangements where dedicated pumps provide the required cooling water flow through the system. In these designs, stopping the pumps would effectively isolate the system limiting potential water outflow.

The potential draining of the intake canal inventory in the Turbine Building is dominant due to a plant specific spatial interdependence. For both units the Emergency Switchgear Rooms are located in the Service Building on the same elevation as the Turbine Building basement. These areas are separated by a fire door with 2-foot high flood dikes in front of them. A large scale flood could potentially overflow the dikes and enter into the two unit ESGR, leading to the potential loss of emergency power in both units, including the loss of Residual Heat Removal (RHR) stub busses. The normal off-site power supply to the plant would not be affected since the normal SGR is located at higher elevation in the Service Building.

Another important contributor to the internal flood CDF is due to flood events originated or entering into the Auxiliary Building. These flood scenarios, mainly from supply pipe ruptures from the RWST, result in the loss of all Component Cooling Water (CCW) and consequently the RHR function at the plant. This coupled with the unavailability of the RWST inventory to be injected into the reactor core leads to core damage. Again, the plant-specific spatial arrangement of piping and equipment is the main reason for the development of the accident scenario and its risk significance.

4.1.7 Quantitative Results from Seismic Analysis

Table 4 shows the base case results. The base case consists of the Surry plant (systems and fragilities) at the Surry site with EPRI and LLNL seismic hazard curves. In this table, the mean, median, 5 percentile and 95 percentile frequencies of the two plant operating states are shown. It is seen from the table that mean annual frequency of the two plant operating states is less than 10^{-6} per year using either the LLNL or the EPRI seismic hazard curves. Therefore, it can be concluded that the seismic contribution to the mean annual core damage frequency during both POS 6 and POS 10 is very small at Surry Unit 1.

The comparison of core damage-frequency results is also shown in Table 4. From examining the table, several important observations emerge:

- During shutdown conditions, the total annual mean CDF arising from earthquakes is small compared to the CDF arising from internal initiators: a factor of about 15 smaller for the LLNL seismic hazard curves and a factor of about 60 smaller using the EPRI hazard curves.

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- The seismic mean CDF during shutdown is small compared to the mean CDF at full power from seismic initiators from NUREG-1150: a factor of about 350 times smaller for the LLNL hazard curves and about 300 times smaller for the EPRI hazard curves.
- The Error Factor (EF) in this seismic study is significantly greater than the EF in the CDF from internal initiators during shutdown. This is primarily due to the large uncertainty in the seismic hazard curves but another contribution arises from the uncertainty in the seismic fragilities.

4.1.8 Insights from Seismic Analysis

A number of important insights emerge from this Surry analysis, including:

Core Damage Frequency

The core damage frequency for earthquake-initiated accidents during refueling outages in POS 6 and POS 10 is found to be low in absolute terms, below 10^{-6} /year. The reasons for this are (i) Surry's seismic capacity in responding to earthquakes during shutdown is excellent, well above its design basis and similar to its ability to respond to earthquakes during full-power conditions; (ii) the Surry site is one of the least seismically active locations in the United States; (iii) the Surry plant is only in POS 6 and POS 10 (combined) for an average (mean) of 6.6% of the time.

The core damage frequencies are also low relative to the frequencies during POS 6 and POS 10 for internal initiators. This can be seen in Table 4.

The Results are Plant Specific

The results for Surry are highly plant-specific, in the sense that the seismic capacities, the specific sequences that are found to be most important, and the seismicity of the site are all difficult to generalize to other reactors elsewhere.

Shutdown Seismic Sequences are Similar to Full-Power Seismic Sequences

Nevertheless, it is important to observe that all of the sequence types, components, and human errors that emerge in the key sequences in this analysis are similar or identical to sequences, components, and human errors that appear in typical full-power seismic PRAs. That is, nothing that has arisen as important in this study appears to be unique to earthquakes occurring during shutdown conditions. Whether this observation can be generalized to other reactors at other sites is unknown.

Sensitivities

Sensitivity studies reveal that if the Surry reactor were moved to the Zion site in Illinois (a typical midwestern site) or the Pilgrim site in Massachusetts (one of the most seismically active sites among all of the reactor sites in the eastern U.S.), the mean annual CDF from this study would increase by factors of about 1.8 and 10, respectively.

Uncertainties

While there are significant uncertainties in the numerical values of core damage frequencies found in this study (see Table 4), the above conclusions are relatively robust—they do not depend on the detailed numerical values found.

4.2 Level 2/3 Results

4.2.1 Plant Damage State Frequency

The CDF analysis is coupled to the accident progression analysis through the plant damage states (PDS). Of particular interest is the characteristics of the PDS groups and their relative contribution to the core damage frequency estimates. The PDS characteristics are important because they strongly influence the subsequent accident progression. Table 7 displays the PDS contributors to the core damage frequency for this study. Four statistical measures (namely the 5th percentile, median, mean and 95th percentile) on the distributions of the various PDS groups are given in Table 7. Similar results from the full power study are also included in the table.

Accident sequences in which the operators did not correctly diagnose the situation or take proper actions were the largest contributor (approximately two-thirds of the total) to the mean core damage frequency for mid-loop operation. Accident sequences that lead to station blackout during mid-loop operation (loss of the 4 kV Bus is similar to a station blackout) contribute about 10 percent to the mean CDF. Other accidents were identified that resulted in loss of core cooling after depletion of the refueling water storage tank and failure of recirculation. The leading cause of recirculation failure was found to be plugging of the suction from the sump. These accidents contribute about 20 percent to the mean core damage frequency.

Station Blackout accidents were the largest contributor (approximately two-thirds of the total) to the mean core damage frequency for accidents initiated by internal events during power operation. Other accidents initiated by loss-of-coolant accidents (LOCAs), transient events and anticipated transient without scram (ATWS) contributed about 25 percent to the mean CDF. Accidents that result in containment bypass (steam generator tube ruptures (SGTR) and interfacing systems LOCAs) contributed less than 10 percent to the mean CDF.

The plant damage states in Table 7 cannot be directly compared because the plant configuration during mid-loop operation is different than the configuration during full power operation. For example a Station Blackout during full power operation will have a different accident progression than a Station Blackout during mid-loop operation. An important difference is that the containment may not be isolated during mid-loop operation whereas the containment was found to be isolated for most of the accidents at full power. Differences in the status of containment integrity during mid-loop and full power operation have an important influence on the accident progression analysis and risk estimates. In the following sections differences in the plant configuration (and hence plant damage states) between mid-loop and low power are indicated.

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**Table 7 Comparison of the PDS Core Damage Frequencies (per reactor year)
for Mid-Loop and Full-Power Operation (Internal Events Only)**

Full-Power Operation				
PDS	5th Percentile	Median	Mean	95th Percentile
Station Blackout				
Short Term	1.2E-7	1.5E-6	5.4E-6	2.1E-5
Long Term	1.6E-6	1.1E-5	2.2E-5	6.4E-5
ATWS	2.9E-8	4.2E-7	1.4E-6	6.5E-6
LOCAs	1.2E-6	3.9E-6	6.1E-6	2.0E-5
Interfacing LOCA	3.6E-11	4.9E-8	1.6E-6	8.2E-6
SGTR	4.5E-7	1.4E-6	1.8E-6	4.7E-6
Total	9.8E-6	2.5E-5	4.1E-5	1.0E-4

Mid-Loop Operation				
PDS	5th Percentile	Median	Mean	95th Percentile
Station Blackout	1.9E-8	1.2E-7	4.0E-7	1.7E-6
Human Errors	2.1E-7	1.0E-6	2.8E-6	1.3E-5
Loss of Recirculation	5.3E-8	4.1E-7	7.8E-7	3.3E-6
Loss of 4 kV Bus	8.9E-9	8.9E-8	2.2E-7	1.3E-6
Total	3.2E-7	2.0E-6	4.2E-6	1.9E-5

4.2.2 Accident Progression Analysis

The plant damage states developed for the mid-loop and full power studies cannot be directly compared. An attempt was therefore made to summarize the results of the accident progression analyses performed for the two studies in such a way that differences in containment status could be ascertained for each of the plant damage states. Table 8 summarizes the probability of accident progression bins (APB) conditional on the various PDS groups for full power operation and for mid-loop operation. The table has been constructed in such a way that APBs have the same meaning in both studies. For example accidents that "bypass" containment were identified in the full-power study but not in the mid-loop study, whereas "containment not isolated" was an important APB for accidents during mid-loop operation but not for full power.

The most significant difference in the results given in Table 8 relates to the probability of the containment not being isolated. In the full power study the probability for the containment not being isolated was very small because during power operation the Surry containment is maintained at a subatmospheric pressure and therefore containment leakage would be detected. However, the probability of the containment not being isolated was determined to be high for most of the plant damage states during mid-loop operation. In fact, the plant damage state that is the largest contributor to the mean core damage frequency (PDS 2 - Human Error) has a very high conditional probability for the containment being open. This is because it was determined in the accident progression analysis that if operator error due to failure to diagnose the accident led to core damage then the operators probably would not have taken measures to isolate containment.

Another difference between the results in Table 8 relates to accidents that bypass the containment. In the full power study accidents that bypass the containment contribute less than 10 percent to the mean CDF but, because they are high consequence events, they are large contributors to the risk estimates. However, in the mid-loop study accidents that bypass the containment (such as SGTRs or interfacing systems LOCAs) were not included because the configuration of the plant precludes such events.

The probability for early containment failure caused by such phenomena as hydrogen combustion, direct containment heating and steam explosions was found to be very low for all PDS groups in the full study. This is because the failure pressure of the containment was determined to be much higher than the design pressure and the loads predicted from the phenomena were generally lower than the failure pressures. The probability for early containment failure was also found to be small for accidents during mid-loop operation except for accidents involving station blackout and loss of the 4 kV bus. For these accident sequences the mean conditional probability for early containment failure was determined to be between 0.1 and 0.2. The cause of early containment failure was determined to be hydrogen combustion, which is a problem during mid-loop operation for two reasons. Firstly, if the operators are able to isolate containment during an accident in mid-loop operation there is a possibility that they may not be able to achieve full pressure retaining capability in the time available. The higher containment failure probability in the mid-loop study therefore reflects the lower pressure retaining capability of the containment relative to the capability expected during power operation. Secondly, for accidents involving station blackout it is unlikely that an ignition source would be available to ignite the hydrogen until power is recovered. This means that large quantities of unburned hydrogen could accumulate in containment. The higher early containment failure probability for station blackout accidents during mid-loop operation therefore also reflects the possibility that power will be restored after a large quantity of hydrogen has accumulated in containment.

Table 8 Comparison of the Mean Probabilities of APBs Conditional on PDS Groups for Mid-Loop and Full-Power Operation (Internal Events Only)

Accident Progression Bin Groups	Plant Damage State Groups†										
	Full-Power Operation*						Mid-Loop Operation**				
	SBO (2.8E-5)	ATWS (1.4E-6)	Transients (1.8E-6)	LOCAs (6.1E-6)	Bypass (3.4E-6)	All*** (4.1E-5)	SBO (4E-7)	Human Error (2.8E-6)	Loss of Recirculation (7.8E-7)	Loss of 4 kV Bus (2.2E-7)	All*** (4.2E-6)
Early Containment Failure	.01	Neg.	Neg.	.01	—	.01	.17	—	.03	.11	.03
Late Containment Failure	.08	.05	.01	.05	—	.06	—	—	—	—	—
Containment Bypass	Neg.	.08	.01	—	1.0	.12	—	—	—	—	—
Containment Not Isolated	—	—	—	—	—	—	.28	.95	.10	.28	.69
No VB, No CF	.31	.52	.22	.59	—	.34	.37	.02	—	.39	.07
VB, No CF	.6	.35	.76	.35	—	.47	.18	.03	.87	.22	.21

* Reproduced from NUREG/CR-4551, Volume 3

** Reproduced from NUREG/CR-6144, Volume 6

*** The values given under the "All" column are frequency-weighted averages and not totals.

† Values given in parentheses are core damage frequency per reactor year for each plant damage state group.

VB Vessel Breach

CF Containment Failure

Neg. Negligible (<0.01)

The conditional probability of late containment failure, caused by the core debris penetrating the basemat or by overpressurizing the containment (due to the accumulation of steam and noncondensable gases) was determined to be between 0.01 and 0.1 for accidents during full power operation. Both of these failure mechanisms were eliminated for accidents during mid-loop operation based on deterministic calculations. The calculations indicated that the decay heat levels for accidents during mid-loop operation were not sufficiently high to cause late containment failure by basemat penetration or containment overpressurization.

Finally, the mean conditional probability of the containment being intact (i.e., isolated, not bypassed, no excessive leakage, and no containment failure) was determined to be high (i.e., between 0.8 and 0.9) for all PDS groups in the full power except for the PDS group containing bypass accidents. As noted above, bypass accidents contribute less than 10 percent to the mean CDF in the full power study. The mean conditional probability of the containment being isolated varied over a wide range for accidents during mid-loop operation. The range varied from 0.05 (Human Error PDS) to about 0.9 for accidents involving loss of recirculation. However, as the Human Error PDS is the largest contributor to the mean CDF the probability of the containment being intact conditional on the mean CDF for all internal events during mid-loop operation was less than 0.3. This compares with a probability of the containment being intact conditional on the mean CDF for accidents during power operation of over 0.8. This difference in containment integrity during mid-loop and full power operation has an important influence on the risk estimates.

4.2.3 Results of the Source Term Analysis

The source term model (SORSOR) used in the full power study was considered suitable for use in the mid-loop study with only minor modifications. This suitability was based on comparisons with calculations from a deterministic code, MELCOR, and the views expressed by an expert review panel drawn from staff at Sandia and Brookhaven National Laboratories. Therefore, as the same source term model was used in the two studies the source terms are similar for similar accident progression bins. Although the source term calculations are similar for the two studies the risk estimates for mid-loop operation are influenced by the changing radionuclide inventories for the various accidents because they can occur a long time after shutdown. In order to account for the changing radionuclide inventory the partitioning method used in the full power study to combine the source terms into a smaller number of representative source term groups had to be modified for the mid-loop study. Therefore, a direct comparison of the source term groups determined for the two studies would be difficult because of the changing inventory associated with accidents during mid-loop operation.

4.2.4 Consequence Analysis

The approach used to calculate offsite consequences was similar in both studies. The major difference was that the latest version of the MACCS code was used to evaluate the offsite consequence measures in the mid-loop study. The latest version of MACCS incorporates the BEIR V update to the latent cancer versus dose relationship, whereas the full power study used in an earlier version of MACCS, which did not include the latest BEIR V update. The latest BEIR V update gives a factor of approximately three times higher latent cancers for the same value of population dose.

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4.2.5 Risk Analysis

Table 9 presents statistical measures of the distributions for seven consequence measures for accidents during mid-loop operation obtained from this study. Similar statistical measures for full power operation obtained from the NUREG-1150 study of Surry are also included in the table.

Table 9 indicates that the mean risk of offsite early health effects is over two orders of magnitude lower for accidents during mid-loop operation than for full power in spite of the lack of mitigative features. This is due to the natural decay of the radionuclide inventory (because the accidents occur a long time after shutdown) particularly the short-lived isotopes of iodine and tellurium, which are primarily associated with early health effects. The statistical measures for latent cancer fatalities (only 1000 miles was reported in Ref. 1) differ by a factor of approximately three, although the statistical measures for population dose (1000 miles) for mid-loop and full power operations are similar. This difference is largely explained by differences in the latent cancer versus dose relationship in the different versions of MACCS (as noted above) used in the two studies.

PDS 2 is the largest contributor to risk for all consequence measures. The reason for this is that for accidents where operator errors, such as failure to diagnose or take proper action, play a major role in determining the progression of the accident, it was judged unlikely that actions to isolate containment would be taken. As shown in Table 8, the mean conditional probability of the containment being unisolated was estimated to be almost 0.95 for the accident sequences belonging to PDS group 2.

4.2.6 Comparison with the Safety Goals

Comparison of the results of this study against the NRC safety goals is done for the two quantitative health objectives identified in the Commission's policy statement of August 1986. These objectives deal with individual early fatality and latent cancer fatality risks within 1 mile and 10 miles of the site, respectively. The numerical value of these objectives are given in Table 9. The 95th percentile of the distribution for individual latent cancer fatality risk falls more than an order of magnitude below the objective. The 95th percentile of the distribution for individual early fatality risk falls over two orders of magnitude below the corresponding health objective. The health objectives, however, apply to the total risk of the Surry Plant. The risk estimates of this study are for accidents initiated by internal events during mid-loop operation and therefore reflect only a fraction of the total risk at Surry.

4.2.7 Insights

The main finding of the study is that during mid-loop operation the risk of consequence measures related to long-term health effects, latent cancer fatalities and population dose, are high, comparable to those at full power, despite the much lower level of the decay heat and the radionuclide inventory. The reason for this is that containment is likely to be unisolated for a significant fraction of the accidents initiated during mid-loop operation so the releases to the environment are potentially large and the radionuclide species which mostly contribute to long-term health effects (such as cesium) have long half-lives. Accident sequences involving failure to correctly diagnose the situation or take proper actions are the largest contributors to the integrated risk. Another finding of the study is that the risk of early fatalities is low despite the unisolated containment due to the decay of the short-lived radionuclide species

Table 9 Comparison of Distributions of Risks for Mid-Loop and Full-Power Operation
(All Values per Reactor Year; Population Doses in P-Sv per Year)

	5th Percentile		Median		Mean		95th Percentile		Sigma	
	Mid-Loop	Full-Power	Mid-Loop	Full-Power	Mid-Loop	Full-Power	Mid-Loop	Full-Power	Mid-Loop	Full-Power
Early Fatalities	1.3E-10	7.6E-10	3.6E-09	7.0E-08	4.9E-08	2.0E-06	1.6E-07	5.4E-06	1.7E-07	N.A.
Latent Fatalities within 50 mi	1.6E-04	N.A.	8.3E-04	N.A.	2.5E-03	N.A.	8.8E-03	N.A.	3.7E-03	N.A.
Latent Fatalities within 1000 mi	8.0E-04	3.1E-04	5.3E-03	2.2E-03	1.6E-02	5.2E-03	5.5E-02	1.9E-02	2.5E-02	N.A.
Population Dose within 50 mi	3.8E-03	5.9E-03	2.0E-02	2.7E-02	5.8E-02	5.8E-02	1.9E-01	2.5E-01	8.8E-02	N.A.
Population Dose within 1000 mi	1.9E-02	1.9E-02	1.2E-01	1.3E-01	3.7E-01	3.1E-01	1.3E+00	1.2E+00	5.9E-01	N.A.
Individual Early Fatalities Risk within 1 mi*	6.0E-12	1.4E-11	1.3E-10	8.7E-10	1.7E-09	1.6E-08	6.9E-09	4.9E-08	5.5E-09	N.A.
Individual Latent Fatalities Risk within 10 mi*	1.2E-10	1.6E-10	7.5E-10	4.9E-10	2.1E-09	1.7E-09	7.1E-09	8.1E-09	3.0E-09	N.A.

N.A. – Not Available

*NRC quantitative health objectives:

- Individual early fatality risk within one mile to be less than 5×10^{-7} per reactor year.
- Individual latent cancer fatality risk within 10 miles to be less than 2×10^{-6} per reactor year.

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such as iodine and tellurium which contribute to early fatality risk. The integrated risk estimates have a range of uncertainty extending over approximately two orders of magnitude from the 5th to the 95th percentile of the distribution.

Containment Status

The major factor driving the risk is the status of containment during mid-loop operation. It was determined that there is a high probability that the containment is either unisolated or that it would not have full-pressure retaining capability during mid-loop operation. This is particularly the case if the operators fail to diagnose the accident as it was judged unlikely that they would take action to isolate containment or could succeed in doing so within the available time frame. This factor played a significant role in influencing the risk estimates of mid-loop operation. During the course of the study, Surry plant personnel made available new procedures for containment closure during mid-loop operation. However, it was difficult to assess the adequacy of these procedures in ensuring the pressure retaining capability of the containment within the time frame encompassed by this study. This feature contributed significantly to the uncertainty in containment status and the estimate of risk.

Availability of Containment Sprays

There is no requirement at Surry for the containment sprays to be available during shutdown. Plant records show that the spray systems could be inoperable because of maintenance. Spray availability was modeled as an uncertainty parameter in the integrated risk analysis. Since the sprays perform an important safety function in mitigating the effects of releases, spray unavailability contributed both to the risk and its uncertainty.

Possibility of Core Damage Arrest

The inclusion of the possibility of arresting the core degradation process before vessel failure is an important feature of this analysis as it was for the full power study. Termination of the accident in-vessel can significantly reduce some of the fission product releases and thus the risk. The potential for core recovery depends on the nature of the accident progression and is different for the various PDS groups. Overall, the conditional probability of core damage arrest ranged from 0.23 (5th percentile) to 0.44 (95th percentile) with a mean of 0.35.

Comparison with Full Power Study

The mean core damage frequency for accidents during mid-loop operation is about an order of magnitude lower than the mean frequency of accidents caused by internal events at full power. However, the risk distributions obtained for comparable long term health consequences are very similar in the two studies. What this finding implies is that the lower decay heat and lower radionuclide inventory of the mid-loop operating state, compared with full power, is offset by the likelihood of containment being unisolated. Finally, the mean risk of early health effects is over two orders of magnitude lower for accidents during mid-loop operation than for accidents during full power operation. This is due to the natural decay of those radionuclide species which have the greatest impact on early fatality risk because accidents during mid-loop operation occur a long time after shutdown.

5 CONCLUSIONS

This study was successful in developing a methodology to estimate the risk associated with the operation of a PWR during mid-loop operation. The methodology developed and the lessons learned from its application provide the NRC with new tools that could be used in subsequent analyses. The study concentrated the effort on mid-loop operation only. The core damage frequency contributions of other low power and shutdown POSs were analyzed in the coarse screening analysis of the Phase 1 study. The following sections summarize the conclusions of the study.

5.1 Level 1 Conclusions

Internal Events

This study shows that the contribution to the total core damage frequency due to internal events during mid-loop operation at the Surry plant is lower by an order of magnitude than that of power operation. This is mainly due to the much smaller fraction of time that the plant is at mid-loop. The core damage frequency conditional on the plant being at mid-loop is actually higher than that of power operation by nearly a factor of two.

Only a few procedures are available for mitigating accidents that may occur during shutdown. Procedures written specifically for shutdown accidents would be useful. Realistic thermal-hydraulic analysis should be used as the basis of the procedures.

It was assumed that the reduced-inventory checklist was followed, and the maintenance unavailability of equipment not on the list were found to be dominant contributors to system unavailability. However, the checklist is believed to be sufficient for ensuring the availability of essential equipment. The dominant cause of core damage is due to operator errors. It is recognized that there is very large uncertainty in the human error probabilities used in this study.

The time window approach developed in this study provides a more realistic approach to account for changing decay heat levels during shutdown. Without using this approach, the core damage frequency estimates would be an order of magnitude higher.

Internal Fires

A comparison of the fire induced core damage frequency during mid-loop operation with that of power operation shows that, although the plant spends much less time at mid-loop, the core damage frequencies are comparable. The main reason for this finding is that the routing of the cables needed to support RHR operation or mitigate an accident during mid-loop operation is such that a single fire at a few critical locations can damage almost all the equipment need to mitigate the accident. During power operation, considerably fewer critical locations exist.

Risk significant scenarios were found mainly in the emergency switchgear room (ESGR), the cable vault and tunnel (CVT), and in the containment. In the ESGR, several important scenarios (which are also the most risk significant ESGR scenarios) occur in locations where many cables for the H and the J emergency divisions come together in close proximity. The tunnel part of the CVT is a constrained space, where damage would quickly propagate to both divisions (serving many different emergency equipment). In the containment, the risk significance stems from the relatively high fire frequency and non-separation of the two RHR divisions.

5 Conclusions

Three mid-loop POSs, in which reactor coolant system level is lowered to the mid-plane of the hot leg, were analyzed. Two of them occur in a refueling outage (POSs R6 and R10) and one in a drained maintenance outage (POS D6). POSs D6 and R6 are much more important than R10 (as R10 occurs in later time windows). D6 is more important than R6 due to constraints imposed by a drained maintenance outage and its tendency to occur in earlier time windows. The earlier time windows are more important than the later ones, with window 4 being relatively unimportant. Windows 1 and 2 are of the highest importance, with window 2 being significantly more important than window 1. While the decay heat is higher and the success criteria are more stringent in window 1, this window is of relatively short duration, and mid-loop tends to occur in the later time windows. The most risk significant fire initiator occurs in the cable vault tunnel area, in window 2 and POS D6, followed by a few scenarios in the ESGR, in the same window and POS.

Internal Floods

In general, the risk contribution from flood events is relatively significant and is dominated by potential flood events in the ESGR coupled with loop isolation. The internal flood CDF is dominated by Turbine Building flood events. These events are primarily initiated by either valve or expansion joint failures in the main inlet lines of the circulating water system. These failures may lead to pipe ruptures upstream of the condenser water box and inlet valves. At Surry the circulating water system is gravity fed from a very large capacity intake canal and its isolation may not be accomplished in a timely manner. This is in contrast with other common design arrangement where dedicated pumps provide the required cooling water flow through the system.

The potential draining of the intake canal inventory in the Turbine Building is dominant because of a plant-specific spatial interdependence. For both units the Emergency Switchgear Rooms are located in the Service Building on the same elevation as the Turbine Building basement. These areas are separated by a fire door with 2-foot high flood dikes in front of them. A large scale flood could potentially overflow the dikes and enter into the two unit ESGR, leading to the potential loss of emergency power in both units, including the loss of stub busses that support the RHR pumps. The normal off-site power supply to the plant would not be affected since the normal switch gear room is located at higher elevation in the Service Building.

The flood initiating event analysis has, in general, indicated that shutdown (and specifically mid-loop operation) does not pose a significant increase in flood risk. A possible exception is flood events coupled with loop isolation, which can occur in later Time Windows 2, 3 and 4.

Seismic Events

The core damage frequency for earthquake-initiated accidents during refueling outages in POS R6 and POS R10 was found to be low in absolute terms, below 10^{-6} /year. The reasons for this are (i) Surry's seismic capacity in responding to earthquakes during shutdown is excellent, well above its design basis and similar to its ability to respond to earthquakes during full-power conditions; (ii) the Surry site enjoys one of the least seismically active locations in the United States; (iii) the Surry plant is only in POS R6 and POS R10 (combined) on average for a relatively short period.

The seismic mean CDF during shutdown is small compared to the mean CDF at full power from seismic initiators:^[2] a factor of about 350 times smaller for the LLNL hazard curves and about 300 times smaller for the EPRI hazard curves.

5.2 Level 2/3 Conclusions

A systematic evaluation of risk has been performed for mid-loop operation at the Surry Unit 1 plant. The analysis includes accident frequency, accident progression, source terms, consequences, risk, and a determination of the uncertainty in each of the component analyses and in the final risk measures. The analysis takes into account the long time after shutdown that the accidents can occur and the impact of the consequent decay in power level and radionuclide inventory on the risk. The inclusion of this time factor in a risk analysis is a new development in PRA technology and represents a strength of the study from the methodological standpoint.

The results contained in this report are based on the analysis of accident frequency during mid-loop operation where the accident sequences leading to core damage were binned into four plant damage states (PDS) groups: PDS1 (station blackout events), PDS 2 (human errors, failure to diagnose or take proper actions on the part of the operators), PDS 3 (recirculation cooling failure), and PDS 4 (loss of 4 kV bus). These PDSs served as the entry point for the further analysis of accident progression, the determination of potential recovery actions, and the evaluation of source terms, consequences, and risk.

The main finding of the study is that during mid-loop operation the risk of consequence measures related to long-term health effects, latent cancer fatalities, and population dose, are high, comparable to those at full power, despite the much lower level of the decay heat and the radionuclide inventory. The reason for this is that there is a lack of mitigative features for a significant fraction of the accidents initiated during mid-loop operation so the releases to the environment are large and the radionuclide species which mostly contribute to long-term health effects (such as cesium) have long half-lives. PDS 2 (diagnostic and corrective action failures) makes the largest contribution to the risk. Another finding of the study is that the risk of early fatalities is low despite the unisolated containment due to the decay of the short-lived radionuclide species, such as iodine and tellurium which contribute to early fatality risk. The risk estimates have a range of uncertainty extending over approximately two orders of magnitude from the 5th to the 95th percentiles of the distribution.

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APPENDIX

APPROACHES FOR PERFORMING A LEVEL 1 LOW POWER AND SHUTDOWN PRA

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APPENDIX – APPROACHES FOR PERFORMING A LEVEL 1 LOW POWER AND SHUTDOWN PRA

A.1 Objective

The objective of this appendix is to provide a summary of the methods that can be used for performing a Level 1 LP&S PRA. As guidance^[1,2] already exists for those methods typically used in performing a full power PRA, they will not be discussed in detail in this appendix. Only those methods that are unique to LP&S conditions are discussed. It is assumed that the readers are PRA practitioners who are familiar with the methods used in performing a full power PRA. This appendix focuses on internal events only, excluding internal fires and floods. The approach for analyzing internal fires or floods is the same as that for a full power PRA. The only difference is that the event trees and fault trees developed during the internal event analysis for LP&S conditions have to be used. This appendix is based on the methods used in the three latest LP&S PRAs^[3-5] performed in the United States. In most cases, different methods exist for the tasks, and it is not possible at this time to prescribe the best method to use. Therefore, a few methods will be discussed with some comparison provided. The most important tasks of a shutdown PRA are identification of the shutdown configuration, and determination of success criteria and how they vary with decay heat. The documentation needed for each of the tasks for a LP&S PRA is also discussed.

A.2 Characterization of Outages and Plant Operational States

Characterization of plant operational states defines the initial conditions for accident scenario development. Table A.2.1 lists the plant information that can be used to identify and characterize the shutdown POSs. The applications of each item to a shutdown PRA are also listed. Table A.2.2 lists important parameters that need to be determined for each POS and how they are used in a shutdown PRA. Tables A.2.3 and A.2.4 list the POSs defined for the Surry and Grand Gulf plants, respectively.

Table A.2.1 List of Plant Information That Can Be Used to Identify and Characterize the Shutdown POSs

Plant Information	Description	Application
Outage Plan	Planned major activities and plant configurations during an outage	1. Definition of plant configuration 2. Duration of POSs 3. Maintenance unavailability
Shift Supervisor's Log Books	Actual detailed activities during an outage, status of major components and systems	1. Definition of plant configuration 2. Duration of POSs 3. Maintenance unavailability
Operating Procedures for Shutdown Conditions	Detailed steps of how operators change the conditions of the plant	1. Definition of plant configuration 2. Duration of POSs 3. Maintenance unavailability 4. Identification of initiating events
Test and Maintenance Procedures	Detailed steps of operators in performing the test and maintenance	1. Maintenance unavailability 2. Identification of initiating events 3. Definition of plant configuration
Monthly Operating Reports	Actual timing of major activities and plant configurations	1. Definition of plant configuration 2. Duration of POSs
Minimum Equipment Lists	Actual condition of major equipment	1. Definition of plant configuration 2. Maintenance unavailability
System Training Manual	General discussion of conditions of systems and components during different plant conditions	1. Success criteria of systems 2. Operability of systems in different accident sequences in different POSs

Table A.2.2 Parameters Used in the Definition and Characterization of POSs

Parameter	Source of Information	PRA Application
Frequency	Graybook, shift supervisor's log books, outage plans	Quantification of accident scenarios
Plant Configuration	Operating procedures, shift supervisor's log books	Initial condition for accident scenarios development
System Availability	Shift supervisor's log books, system training manuals	Fault tree quantification
Shutdown Activities	Operating procedures, shift supervisor's log books, outage plan	Definition of POS Boundaries, System availability
Time to Core Uncovery/Damage	--	Determination of success criteria, quantification of HEPs
Maintenance Unavailability	Shift supervisor's log books	Fault tree quantification
RCS Integrity	Operating procedures, shift supervisor's log books, outage plan	Scenario development
Containment Integrity	Operating procedures, shift supervisor's log books, outage plan	Quantification of containment event tree
Reactivity	Operating procedures, shift supervisor's log books, outage plan	Scenario development
Reactor Coolant Temperature	Operating procedures, shift supervisor's log books, outage plan	Initial condition for accident scenarios development
Reactor Coolant Pressure	Operating procedures, shift supervisor's log books, outage plan	Initial condition for accident scenarios development
Reactor Vessel Level	Operating procedures, shift supervisor's log books, outage plan	Initial condition for accident scenarios development
Time after Shutdown	Operating procedures, shift supervisor's log books, outage plan	Initial condition for accident scenarios development
Duration of POSs	Operating procedures, shift supervisor's log books, outage plan	Quantification of accident frequency

Table A.2.2 (continued)

Parameter	Source of Information	PRA Application
PWR POS Dependent Parameters		
Fraction of Time the Steam Generators Are Drained	Operating procedures, shift supervisor's log books, outage plan	Availability of steam generators as a heat sink, definition of POS boundaries
Fraction of Time the Steam Generator Nozzle Dam Is in Place	Operating procedures, shift supervisor's log books, outage plan	Quantification of accident scenarios
Fraction of Time the Pressurizer Safety Valves Are Removed	Operating procedures, shift supervisor's log books, outage plan, test procedures	Availability of reflux cooling and gravity injection from RWST
Fraction of Time the RCS Loops Are Isolated	Operating procedures, shift supervisor's log books, outage plan	Availability of cooling using steam generators, definition of POS boundaries
Fraction of Time the Steam Generator Manways Are Off	Operating procedures, shift supervisor's log books, outage plan	Availability of cooling using steam generators, definition of POS boundaries
Fraction of Time the Vessel Head Is Off	Operating procedures, shift supervisor's log books, outage plan	Definition of POS boundaries, quantification of accident scenarios
BWR POS Dependent Parameters		
Fraction of Time the Hydraulic Test Is in Progress	Operating procedures, shift supervisor's log books, outage plan	Determination of initial conditions, availability of safety relief valves
Fraction of Time the Suppression Pool Is Empty	Operating procedures, shift supervisor's log books, outage plan	Availability of suppression pool function
Fraction of Time the Suppression Pool Is at Reduced level	Operating procedures, shift supervisor's log books, outage plan	Availability of suppression pool function
Fraction of Time the Vessel Level Is Sufficient for Natural Recirculation Cooling	Operating procedures, shift supervisor's log books, outage plan	Operability of natural recirculation function
Fraction of Time the Vessel Level Is Sufficient for Forced Recirculation Cooling	Operating procedures, shift supervisor's log books, outage plan	Operability of forced recirculation function
Fraction of Time the Containment Is Open	Operating procedures, shift supervisor's log books, outage plan	Control of radioactivity release

Table A.2.3 Plant Operational States for a Refueling Outage at Surry Unit 1

		Activities	Level	K	T (°F)	P (psig)	Standard T.S. Mode
1	Low Power Operation & RX Shutdown	<ul style="list-style-type: none"> • Transfer to manual RX control • Transfer to manual SG level control 	In Przr.	=1.0 (<15% Power)	547	2235	1,2
				$\leq 0.9823TS^{(3)}$	547	2235	
2	Cooldown With SG	<ul style="list-style-type: none"> • Borate RCS • Stop 2 RCPs • Block auto SI & accumulators • Establish LTOP <350°F 		<1.0			3
			In Przr.	<0.96Adm ⁽⁴⁾	345	345	
3	Cooldown With RHR	<ul style="list-style-type: none"> • Disable Engineered Safeguards • Initiate RHR 					4
			In Przr.	<0.96Adm ⁽⁴⁾	200	345	
4	Cooldown With RHR	<ul style="list-style-type: none"> • No longer need Ctmt Integrity or Ctmt vacuum ⁽⁵⁾ • Secure SGs 					5
			In Przr.	<0.96Adm ⁽⁴⁾	140	345	
5	Drain RCS To Mid-loop	<ul style="list-style-type: none"> • Deenergize Przr heaters • Stop last RCP • Cooldown Przr by fill up 					5,6
			Mid- loop	<0.96Adm ⁽⁴⁾	<140	0	
6	Mid-loop Operation	<ul style="list-style-type: none"> • Possible loop isolation 					5,6
			Mid- loop	<0.96Adm ⁽⁴⁾	<140	0	

Table A.2.3 (continued)

		Activities	Level	K	T (°F)	P (psig)	Standard T.S. Mode
7	Fill for Refueling	<ul style="list-style-type: none"> Remove vessel head & upper internals structure Establish Refueling Ctmt Integrity 					6
			Refuel Cavity Full	<0.95TS ⁽²⁾	<140	0	
8	Refueling	<ul style="list-style-type: none"> Move fuel assemblies per refueling sequence 					6
			Refuel Cavity Full	<0.95TS ⁽³⁾	<140	0	
9	Drain RCS to Mid-loop After Refueling	<ul style="list-style-type: none"> Install vessel upper internals structure & head No longer need Refueling Ctmt. Integrity 					6
			Mid-loop	0.96Adm ⁽⁴⁾	<140	0	
10	Mid-loop Operation After Refueling	<ul style="list-style-type: none"> Possible further loop isolation 					5,6
			Mid-loop	<0.96Adm ⁽⁴⁾	<140	0	
11	Refill RCS completely	<ul style="list-style-type: none"> Pressurize RCS & jog RCP for RCS vent & Fill Unisolate any and all loops 					5,6
			RCS Full	<0.96Adm ⁽⁴⁾	<140	<125	
12	RCS Heatup Solid and Draw Bubble	<ul style="list-style-type: none"> Pressurize RCS, start RCP & energize Przr. httrs Establish Ctmt. Integrity and Ctmt. vacuum 					5
			In Przr.	<0.98Adm ⁽⁴⁾	200	345	

Table A.2.3 (continued)

		Activities	Level	K	T (°F)	P (psig)	Standard T.S. Mode
13	RCS Heatup	<ul style="list-style-type: none"> • Start other RCPs • Enable Engineered Safeguards • Secure RHR 					4
			In Przr.	$<0.9823TS^{(3)}$	350	≈ 345	
14	RCS Heatup With SGs	<ul style="list-style-type: none"> • Flow test MDAFWs • Disable LTOP protection $>350^{\circ}\text{F}$ • Unblock accumulators • Verify auto SI reestablished • Dilute RCS 		<1.0			3
			In Przr.	$\leq 0.9823TS^{(3)}$	547	2235	
15	RX Startup & Low Power Operation	<ul style="list-style-type: none"> • Flow test TDAFW • Transfer to auto SG level • Transfer to auto RX control 		≈ 1.0 ($<15\%$ power)	547	2235	1,2
Full Power Operation							

- (1) The time is based on the October 1986 refueling outage of Unit 2. Both duration of POS and cumulative time after shutdown are shown.
- (2) The operating procedures listed are based on the most updated information as of yet. Some of these procedures' numberings are different from those prevailing in October 1986.
- (3) Shutdown Margin Requirements - Per Technical Specifications (1.0 Definitions)
CSD (Cold Shutdown) Condition - RX subcritical by at least 1% $\Delta K/K$ ($\therefore K \leq \approx 0.99$) and T_{avg} is $\leq 200^{\circ}\text{F}$.
ISD (Intermediate Shutdown) Condition - RX subcritical by $\geq 1.77\% \Delta K/K$ ($\therefore K \leq \approx 0.9823$) and T_{avg} is $>200^{\circ}\text{F}$ but $<547^{\circ}\text{F}$.
HSD (Hot Shutdown) Condition - RX subcritical by $\geq 1.77\% \Delta K/K$ ($\therefore K \leq \approx 0.9823$) and T_{avg} is $>547^{\circ}\text{F}$.
- (4) Shutdown Margin Administrative Limits - Per 1-OP-1C, Shutdown Margin Calculation
A further requirement states that except for during the approach to criticality, the minimum boron concentration allowed in a shutdown core must be the greater of the following:
Critical Boron for the 0% power insertion limit OR
The boron required to maintain the administratively (Adm^*) required shutdown margin > 5500 pcm ($K=0.945$) from BOL (beginning of life) to 9000 MWD/MTU and >4000 pcm ($K \leq \approx 0.960$) after 9000 MWD/MTU.
- (5) As in the recently implemented OP-RC-005, before entering Reduced Inventory within 150 hours of shutdown, containment integrity must be established.

Table A.2.4 Plant Operational States for a Refueling Outage at Grand Gulf

POS	Plant Condition	Activities	Heat Removal	Level Control	K	T(^o F)	P (psig)	Standard T.S. Mode	Notes
1	Low Power Operation & Reactor Shutdown	Transfer to manual RX control	Turbine Generator or Turbine bypass Valve(steam dump to condenser)	Feedwater	=1.0 (<15% Power)	550	1000	1,2 (Mode Switch in Run or Startup/Hot Standby)	
2	Hot Shutdown	Cooldown	Turbine bypass Valve (steam dump to condenser)	Feedwater	<1.0	>200	>500	3 (Mode Switch in Shutdown)	
3	Hot Shutdown	Initiation of RHR on SDC	Turbine bypass Valve(steam dump to condenser)	Feedwater	<1.0	>200	>100	3	Recirculation Pumps no longer needed
4	Hot Shutdown		RHR on SDC	RWCU	<1.0	>200	100	3	RCIC Auto Isolation at 60 psig
5	Cold Shutdown	Vessel Head Removal	RHR on SDC	RWCU	<1.0	<200	0	4 (Mode Switch in Shutdown) 5 (Vessel Head Detensioned)	
6	Refueling (Vessel Head off)	Preparation for Refueling	RHR on SDC	Level at Steam Lines	<1.0	<140	0	5	
7	Refueling Operation	Refueling Operation	RHR on SDC	Upper Pool Filled	<1.0	<140	0	5	Refueling Transfer Tube Open

A.3 Identification of Initiating Events and Accident Scenarios

Definition of Initiating Event

An initiating event during shutdown can be defined as one of the following:

- loss of decay heat removal,
- pressure transient that can potentially challenge the RCS integrity,
- loss of RCS inventory,
- and reactivity transients.

Two approaches have been used in identifying and quantifying the initiating events. They are discussed below:

(1) Data Analysis Approach

References [3] and [4] used an approach in which data on incidents that have actually occurred, and studies of accident scenarios that have been identified were reviewed in order to identify initiating events. The collected data was used to estimate the frequencies of the initiating events. Table A.3.1 summarizes the approach used in identifying accident initiating events that may occur at PWRs and provides references to relevant documents. Table A.3.2 lists the initiating events of a PWR. Table A.3.3 lists the POS specific initiating event frequencies for a PWR. Tables A.3.4 to A.3.6 provide similar information for a BWR.

(2) Procedure Event Tree Approach

References [5] and [6] used the procedure event tree approach to define and quantify initiating events. Figure A.3-1 is an example of such an event tree. The top events of the trees are either major operator actions identified in the operating procedures for shutdown conditions or failure of hardware used during shutdown conditions. Failure of the operator or hardware may lead to different types of initiating events represented by the sequences in the procedure event tree. The failure probability of the operator is quantified using an applicable human reliability analysis approach, and hardware failure is modeled by fault trees with component hardware failure data. The end states of the procedure event trees transfer to the applicable accident event trees that model the mitigation of the accidents.

A.3.1 Special Initiating Events or Issues to Be Addressed

RCS Pressure Relief Capability

During an outage, the relief valves that can potentially relieve RCS pressure have to be identified and their capacity evaluated for the shutdown condition. Typically for a PWR, the PORVs, and possibly the RHR relieve valve and the relief valves in the letdown flow path, can relieve pressure during shutdown. Whether or not the valves are available in different accident scenarios has to be evaluated.

Appendix

Weakness in RCS Pressure Boundary

During an outage (for some plants), a weakness in the RCS pressure boundary may be introduced, e.g., steam generator nozzle dam or thimble tube seals for PWRs. Any such weakness has to be specifically addressed in the PRA. Two types of scenarios are possible: (1) failure of the weakness as an initiating event and (2) failure as a result of some other scenarios that challenge the pressure boundary. Both possibilities have to be developed and quantified. When identifying weakness in the RCS pressure boundary, all systems and components that may be connected to the RCS should be taken into consideration. For example, the residual heat removal system at PWRs has a design pressure that is significantly lower than that of the RCS. In Appendix I of Reference [3], scenarios involving challenges to the RHR system were identified. In those scenarios, the RCS pressure may exceed the design pressure of the RHR even if the PORVs and RHR relief valve are operable.

French Scenario

Reference [7] identified a fast reactivity accident scenario that may occur during de-boration. Reference [8] evaluated the scenario for three PWRs and took into consideration a number of deterministic issues, such as reactivity effects and mixing of the injected unborated water with the reactor coolant. It was found that the mixing issue cannot be easily resolved and depends on the geometry of the connection between the injection line and the cold leg. This scenario has to be evaluated on a plant-by-plant basis.

Adverse Configuration

Reference [9] identified a plant configuration during shutdown that has the potential for rapid core uncover as a result of loss of RHR. In this configuration, the hot legs are isolated and an opening or weakness in the pressure boundary exists in the cold leg. A small pressure increase in the RCS can push the coolant out of the vessel and uncover the core. Generic Letter 88-17^[10] specifically addresses this issue and requires the plants to implement protection against the scenario. Each plant's implementation has to be evaluated in terms of its effectiveness against the likelihood of the scenario.

Reactivity Accidents

Reference [11] evaluated different categories of reactivity accidents for PWRs and BWRs. The following scenarios are judged to be significant and should be analyzed based on plant specific design:

- PWR-French scenario
- PWR-LOCA/SGTR with the secondary side coolant diluting the primary coolant
- BWR-Flushing of boron during an ATWS
- BWR-Reactivity event initiated while in unstable operation
- BWR-Refueling Accident

Spent Fuel Pool Accident

NUREG/CR-4982^[12] addresses spent fuel pool accidents and identifies failure of the refueling cavity seals and a fuel transfer cask drop as potential accident scenarios during a refueling outage.

Figure A.3-1 An Example Procedure Event Tree Taken from NSAC-84

Table A.3.1 Approach Used in Identifying PWR Initiating Events

I.	Loss of RHR Events
	Review of Shutdown Incident Reports
	NSAC-52, AEOD/C503, Seabrook Shutdown Study
	Search of Computerized Data Base-Sequence Coding Search System
II.	Review of Existing Shutdown Studies
	NSAC-84, Seabrook, French Study, NUREG/CR-5015
III.	Review of Procedures Used at Shutdown
	Operating Procedure, Test Procedure, Maintenance Procedure
IV.	Review of Initiating Events for Power Operations
	NUREG/CR-4550 Volume 1, Generic Transients for PWRs
	NUREG/CR-4550 Volume 3, Support System Failures, Subtle Interactions
	Surry Probabilistic Risk Assessment for the Individual Plant Examination
V.	Review of NRC Generic Letters, Information Notices, Bulletin, circulars
VI.	Review of Other Studies That Identified Scenarios at Shutdown
	NUREG/CR-4999 Low Temperature Over Pressurization
	NUREG/CR-5368 Reactivity Accidents
	NUREG-1269 Diablo Canyon Incident Augmented Investigation
	Team Report
	NUREG-1410 Vogtle Incident Investigation Team Report
	WCAP-11916 Loss of RHRS While the RCS is Partially Filled
	Westinghouse PRA of LOCAs in Modes 3 and 4
	NUREG/CR-4407 Pipe Break Frequency Estimation for NPP

Table A.3.2 PWR Initiating Events Categories

I.	Loss of RHR
	Spurious Closure of RHR Suction Valves
	Loss of RHR due to Low Vessel Inventory
	A. overdraining during drain down
	B. failure to maintain level
	Total Non-recoverable Loss of RHR
	Loss of operating Train of RHR
	Recoverable Loss of RHR
	Inability to Initiate RHR
II.	Loss of Offsite Power (LOOP) and Station Blackout
III.	Support System Failures
	Loss of a 4 kV Bus
	Loss of 480V Bus
	Loss of a DC Bus
	Loss of a 120 V Vital AC Bus
	Loss of Instrument Air
	Loss of Component Cooling Water
	Loss of Service Water
	Loss of Emergency Switchgear Room Cooling
IV.	LOCAs
	Pipe Ruptures in RCS
	A: Large LOCA
	S1: Medium LOCA
	S2: Small LOCA
	S3: Small-Small LOCA
	H LOCA-Diversion of Flow
	J LOCA-LOCAs in Connected Systems
	K LOCA-Maintenance Induced LOCAs
V.	Interfacing System LOCA
VI.	Steam Generator Tube Rupture

Table A.3.2 (continued)

VII.	Inadvertent Safety Feature Actuation
VIII.	Transients
	Transient with loss of MFW (no challenge to RPS)
	Transient with loss of MFW (challenge to RPS)
	Transient with MFW Available (no challenge to RPS)
	Transient with MFW Available (challenge to RPS)
IX.	Pressurized Thermal Shock
X.	Reactivity Accidents
	Addition of Diluted Accumulator Water
	Addition of Diluted RWST Water
	Dilution When Cleaning Cavity
	Dilution due to Steam Generator Maintenance
	Uncontrolled Boron Dilution from CVCS
	Boron Dilution via RHR
	Startup of RCP after Improper Boron Dilution
	Backfill Cooldown after STGR
	Rod Ejection
	Misloading of Fuel Assembly
	Uncontrolled Bank Withdraw
XI.	Heavy Load Drop Accident

Table A.3.3 Estimated Initiating Event Frequency

I. Loss of RHR

	Mean Value	Error Factor
	N.A.	
RHR2A	1.62E-02/DEMAND	1.97
RHR2B	1.220E-05/hr	3.27
RHR3	4.09E-06/hr	3.79
RHR4	5.28E-06/hr	3.82
RHR5	2.12E-05/hr	2.21
RHR6	1.29E-02/DEMAND	1.54

II. LOOP

POS 1,15 POS 2-14

9.13E-6/hr 6.96E-6/hr

III. Support System Failures

	Mean	Error Factor
Loss of a 4 kV Bus (4 kV)		
POS 6,10	2.10E-05/hr	2.4
POS 3,4,5,8,9,11,12,13	2.63E-05/hr	2.14
Loss of a 120 V Vital	5.58E-06/hr	3.94
Loss of Instrument Air-Outside Containment	2.12E-06/hr	2.86
Loss of Component Cooling Water	3.76E-06/hr	6.35
Loss of Emergency Switchgear Room Cooling		
POS 6,10	1.81E-08/hr	4.0
POS 3,4,5,7,8,9,11,12,13	6.13E-08/hr	4.0

Table A.3.3 (continued)

IV. Loss of Coolant Accident

	POS 1 & 15	POS 2 & 14	POS 3,4,12,13
A	5.0E-4/y	2.5E-5/y	1.7E-5/y
S1	1.0E-3/y	5.0E-5/y	3.3E-5/y
S2	1.0E-3/y	6.0E-5/y	3.3E-5/y
S3	1.3E-3/y	6.5E-4/y	4.3E-4/y

H - Recoverable Flow Diversion

via RHR 5.0E-3/y

via CVCS 1.0E-3/y

Drain

Valves 1.0E-3/y

J - LOCAs in Connected System

via RHR 5.0E-3/y

via CVCS 3.0E-3/y

K - Maintenance Induced LOCA

via RCS 2.0E-3/y

via RHR 1.0E-3/y

V. Interfacing LOCA

	POS 1	POS 2	POS 14	POS 15
LPSI (CDF)	3.00E-10/y	8.90E-9/y	1.22E-08/y	1.00E-08/y
RHR Suction & Discharge (CDF)	3.2E-8/y	9.3E-7/y	1.28E-6/y	1.0E-6/y
RHR-CVCS (CDF)	3.2E-12/y	9.5E-11/y	1.3E-10/y	1.1E-10/y
ACC	1.8E-7/y	5.4E-6/y	7.4E-5/y	6.1E-6/y

	POS 3	POS 4	POS 12, 13
RHR-RWST (CDF)	2.3E-8/y	2.9E-7/y	9.1E-7/y

Table A.3.3 (continued)

VI. Steam Generator Tube Rupture

POSS 1,2,14,15 1.00E-02/yr

POSS 3,4,12,13 1.00E-03/yr

VII. Inadvertent Safety Feature ActuationMean Error Factor

1.06E-04/hr 1.15

VIII. Transients

T2 1.09E-3/hr

T2a 1.82E-3/hr

T3 1.24E-3/hr

T3a 8.3E-3/hr

		<u>Mean</u>	<u>Error Factor</u>
LTOP	POS 3,4,12,13	1.06E-04/hr	1.15

IX. Pressurized Thermal Shock (CDF) 1.05E-07/y**X. Reactivity Accidents (CDF)**

Accumulator 8.0E-10/y

RWST 2.5E-10/y

CVCS (French) 6.8E-08/y

RCP (French) 2.8E-05/y

Refueling 1.2E-07/y

Table A.3.4 References Used in Identifying BWR Initiating Events

1. M.T. Drouin et al., "Analysis of Core Damage Frequency: Grand Gulf, Unit 1; Internal Events," NUREG/CR-4550, SAND86-2084, Vol. 6, Rev. 1, Part 1, September 1989.
2. D.J. Diamond et al., "Reactivity Accidents: A Reassessment of the Design-Basis Events," NUREG/CR-5368, BNL-NUREG- 52198, September 1989.
3. D.P. Mackowiak et al., "Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments," NUREG/CR-3862, EGG-2323, May 1985.
4. H. R. Booth, "Analysis of Refueling Incidents in Nuclear Power Plants," NSAC-129, December 1988.
5. D. M. Ericson, ed., "Analysis of Core Damage Frequency: Internal Events Methodology," NUREG/CR-4550, SAND86-2084, Vol. 1, Rev. 1, January 1990.
6. R.E. Wright et al., "Pipe Break Frequency Estimation for Nuclear Power Plants," NUREG/CR-4407, EGG-2421, May 1987.
7. G. Vine et al., "Residual Heat Removal Experience Review and Safety Analysis: Boiling Water Reactors," NSAC-88, March 1986.
8. T. A. Wheeler et al., "Analysis of Core Damage Frequency From Internal Events: Expert Judgment Elicitation," NUREG/CR-4550, SAND86-2084, Vol. 2, April 1989.
9. D. W. Whitehead, J. L. Darby, B. D. Staple, B. Walsh, T. M. Hake, and T. D. Brown, "BWR Low Power and Shutdown Accident Frequencies Project, Phase 1 - Coarse Screening Analysis," Vol. 1, Draft Letter Report, Sandia National Laboratories and Science and Engineering Associates, Inc., November 23, 1991 update, (Available at the USNRC Public Document Room).
10. P. Lam et al., "Operating Experience Feedback Report - Service Water System Failures and Degradation," NUREG-1275, Vol. 3, November 1988.

Table A.3.5 BWR Initiating Event Categories

T ₁ :	Loss of Offsite Power
T ₂ :	Transient with Loss of the Power Conversion System
T _{3A} :	Transient with PCS Initially Available
T _{3B} :	Transients Involving Loss of Feedwater
T _{3C} :	Loss of Plant Service Water
A:	Large LOCA
A _{HY} :	Large LOCA during Hydraulic Test
S ₁ :	Medium LOCS between 0.007 and 0.4 sq ft.
S _{1H} :	Medium LOCA during Hydraulic Test
S ₂ :	Small LOCA < 0.007 sq ft.
S _{2H} :	Small LOCA during Hydraulic Test
S ₃ :	Very Small LOCA
S _{3H} :	Very Small LOCA during Hydraulic Test
V:	Interfacing System LOCA
R:	Vessel Rupture
H ₁ :	Diversion of Vessel Inventory to the Suppression Pool via RHR
H ₂ :	Diversion to the Condenser via RWC
J ₁ :	LOCA in Operating Connected System (RCIC)
J ₂ :	LOCA in Operating Connected System (RHR)
K:	Test/Maintenance Induced LOCA
E _{1B} :	Isolation from Operating RHR-Shutdown Cooling Loop
E _{1C} :	Isolation of RWC as SDC
E _{1D} :	Isolation of ADHRS Only
E _{1T} :	Isolation of Common RHR Suction Line
E _{1V} :	Isolation of Common Suction Line for ADHRS
E _{2B} :	Loss of Operating RHR-Shutdown Cooling System
E _{2C} :	Loss of RWC as DHR
E _{2D} :	Loss of ADHRS only
E _{2T} :	Loss of Common RHR Suction Line
E _{2V} :	Loss of Common Suction Line for ADHRS
T _{4A} :	Rod Withdraw Error
T _{4B} :	Refueling Accident
T _{4C} :	Instability Event
T _{5A} :	Loss of All Standby Service Water
T _{5B} :	Loss of Turbine Building Cooling Water
T _{5C} :	Loss of All Plant Service Water (includes Radial Well)
T _{5D} :	Loss of All Component Cooling Water
T _{AB} :	Loss of 1E 4160 V AC Bus B
T _{DB} :	Loss of 1E 125 V DC Bus B
T _{IA} :	Loss of Instrument Air
T _{ORV} :	Inadvertent Open Relief Valve at Shutdown
T _{IOF} :	Inadvertent Overpressurization (Makeup Greater Than Letdown)
T _{IHP} :	Inadvertent Overpressurization via Spurious HPCS Actuation
T _{IOF} :	Inadvertent Overfill via LPCS or LPCI
T _{RPT} :	Loss of Recirculation Pump
T _{LM} :	Loss of Makeup

Appendix

Table A.3.6 BWR Initiating Event Frequencies and Applicability to POSs

Initiator	POS 1	POS 2	POS 3	POS 4	POS 5	POS 6	POS 7
T ₁	0.07	0.13	0.13	0.13	0.13	0.13	0.13
T ₂	1.63	0.44	0.44	NA	NA	NA	NA
T _{3A}	4.54	1.00	1.00	NA	NA	NA	NA
T _{3B}	1.06	0.25	0.25	NA	NA	NA	NA
T _{3C}	0.16	0.07	0.07	NA	NA	NA	NA
A	0.00	0.00	0.00	0.00	0.00	0.00	0.00
A _{HY}	NA	NA	NA	NA	0.00	NA	NA
S ₁	0.00	0.00	0.00	0.00	0.00	0.00	0.00
S _{1H}	NA	NA	NA	NA	0.00	NA	NA
S ₂	0.00	0.00	0.00	0.00	0.00	0.00	0.00
S _{2H}	NA	NA	NA	NA	0.00	NA	NA
S ₃	0.00	0.00	0.00	0.00	0.00	0.00	0.00
S _{3H}	NA	NA	NA	NA	0.00	NA	NA
V	NA	NA	NA	NA	NA	NA	NA
R	NA	NA	NA	NA	NA	NA	NA
H ₁	NA	NA	NA	0.06	0.06	0.06	0.06
H ₂	NA	NA	NA	NA	NA	NA	NA
J ₁	?	?	?	NA	NA	NA	NA
J ₂	NA	NA	NA	0.02	0.02	0.02	0.02
K	-	-	-	-	-	-	-
E _{1B}	NA	NA	NA	0.06	0.06	0.06	0.06
E _{1C}	NA	NA	NA	NA	0.00	NA	?
E _{1D}	NA	NA	NA	NA	0.06	0.06	0.06
E _{1T}	NA	NA	NA	0.36	0.36	0.36	0.36
E _{1V}	NA	NA	NA	NA	0.36	0.36	0.36
E _{2B}	NA	NA	NA	0.07	0.07	0.07	0.07
E _{2C}	NA	NA	NA	NA	0.00	NA	?

Table A.3.6 (continued)

Initiator	POS 1	POS 2	POS 3	POS 4	0.07	0.07	0.07
E _{2D}	NA	NA	NA	NA	0.07	0.07	0.07
E _{2T}	NA	NA	NA	0.04	0.04	0.04	0.04
E _{2V}	NA	NA	NA	NA	0.04	0.04	0.04
T _{4A}	NA	NA	NA	NA	NA	NA	NA
T _{4B}	NA	NA	NA	NA	NA	NA	NA
T _{4C}	NA	NA	NA	NA	NA	NA	NA
T _{5A}	NA	NA	NA	0.02	0.02	0.02	0.02
T _{5B}	0.02	0.02	0.02	0.02	0.02	0.02	0.02
T _{5C}	0.02	0.02	0.02	0.02	0.02	0.02	0.02
T _{5D}	NA	NA	NA	0.02	0.02	0.02	0.02
T _{AB}	NA	NA	NA	0.00	0.00	0.00	0.00
T _{DB}	NA	NA	NA	0.01	0.01	0.01	0.01
T _{IA}	0.18	0.18	0.18	0.18	0.18	0.18	0.18
T _{ORV}	NA	NA	NA	0.07	0.07	NA	NA
T _{IOP}	NA	NA	NA	0.00	0.00	NA	NA
T _{IHP}	NA	NA	NA	0.01	0.01	NA	NA
T _{IOF}	NA	NA	NA	0.02	0.02	NA	NA
T _{RPT}	NA	NA	NA	0.07	0.07	NA	NA
T _{LM}	NA	NA	NA	0.01	0.01	NA	NA

A.4 Determination of Success Criteria and Time Windows

For initiating events occurring during shutdown conditions, different systems or functions can potentially be used to mitigate the accident. The conditions under which a system or function are effective and their success criteria have to be determined based on thermal-hydraulic considerations. Tables A.4.1 and A.4.2 list the different systems and functions that have been identified for PWRs and BWRs respectively. Tables A.4.3 and A.4.4 list the studies that evaluate the effectiveness of these systems and functions for PWRs and BWRs respectively. It should be mentioned that for a specific plant, the plant specific design has to be taken into consideration in determining the timing and success criteria.

Decay heat is an important parameter that has to be correctly accounted for in a PRA for shutdown conditions. It affects the success criteria used and the timing of the accident scenarios. In a realistic analysis, the decay heat and accident timing have to be well defined. A way to account for decay heat was developed in Reference [3] in which a "Time Window" approach was used to determine the success criteria and timing for accidents that may occur during mid-loop operation at the Surry plant. The following is a generalization of the time window approach applied to a shutdown POS:

(1) Determination of Success Criteria as a Function of Decay Heat

The success criteria for the mitigating functions in the event tree analysis have to be determined based on the decay heat level, initial conditions of a POS, and preceding events in the accident sequences. For given initial conditions and preceding events in a sequence, the decay heat determines the success criterion for a mitigating function. Table A.4.5 is an example, taken from Reference [3], that shows how the success criteria change with decay heat for the Surry plant.

(2) Definition of Time Windows after Shutdown Based on the Changing Success Criteria with Decay Heat

The time windows are defined based on the time after shutdown when the success criteria of a system or function change with decay heat. In general, whenever the success criteria for one system or mitigating function changes, a new time window needs to be defined, and potentially, a large number of time windows could be needed. To reduce the number of time windows to a manageable number, several potential windows can be merged into a single time window. Table A.4.6 lists the four time windows that were used in Reference [3]. In this study, a total of more than ten time windows could have been used. The use of four windows (as shown in the table) is, therefore, a trade-off between the accuracy of the model and the level of effort needed to arrive at a solution. It is up to the analysts to demonstrate that the merging of the time windows gives an adequate representation without sacrificing the accuracy of the analysis. With the time windows defined, the decay heat level for each time window is better defined, and the timing of the accident scenarios can be better defined. The timing of the scenarios is an important factor that affect the performance of the operators and is an important input to the human reliability analysis.

(3) Collection of Data on the Time after Shutdown When a POS Is First Reached and Duration of the POS

The time after shutdown when a POS might start and the duration of the POS are data that have to be collected for the specific plant being analyzed. Data from other plants can potentially be use as the population data in the statistical analysis. The time when a POS is reached can be identified in the Monthly Report of Operation that each plant submits to the NRC. The duration of the POS can be identified in the shift supervisor's log books for the past outages. The POSs defined in a PRA represent a simplified model of a constantly changing plant configuration. Therefore, data on

the duration of a POS may involve the analyst's judgment on the plant condition when reviewing the shift supervisor's log books.

(4) Statistical Analysis of Time to and Duration of the POS

The data collected in the preceding step is used in the statistical analysis to derive the probability distributions for the time when a POS is reached and the duration of the POS.

(5) Determination of Distributions of Time When the Accident Initiators Occur

The time when the accident initiators occur can be expressed as the time when the POS is reached plus the time duration of the POS multiplied by a uniform distribution between 0 and 1. This is based on the assumption that the initiating event occurs with a constant rate within the POS. Therefore, its probability distribution can be easily determine from the distributions for the time at which the POS is reached and the duration of the POS. Some accidents may occur at the beginning of a POS, e.g., over-draining to mid-loop operation occurs at the beginning of a mid-loop POS. For these initiating events, the time when the initiating event occurs is the same as the time when the POS is reached.

(6) Determination of the Conditional Probabilities that the Accident Initiating Events Occur in the Time Windows

The conditional probability that the accident initiating event occurs in a time window, given that it occurred in the POS, is simply the probability that the accident initiating event occurs in the time window which can be evaluated using the probability distribution determined in Step 4.

(7) Use of the Conditional Probabilities in Quantifying the Accident Scenarios

The annual frequency that an initiating event occurs in a POS is equal to the frequency of the POS times the probability that the initiating event occurs given that the plant is in the POS. The frequency that an initiating event category, IE_k , occurs in a particular POS j of a particular type of outage, I , can be expressed as:

$$\text{Frequency (outage I) per year} \times \text{Probability (IE}_k \mid \text{POS j of outage I),}$$

where

$$\text{Probability (IE}_k \mid \text{POS j of outage I)} = \text{hourly rate (IE}_k) \times \text{duration of POS j of outage I.}$$

In the time window approach, the frequency that an initiating event occurs in a given time window, window ℓ , of a particular POS j of a particular type of outage type, I , is the above frequency multiplied by the conditional probability of the time window given that the initiating event occurred in the specified POS and outage type, i.e.,

$$\frac{\text{Frequency (outage I) per year} \times \text{Probability (IE}_k \mid \text{POS j of outage I)}}{\text{Probability (window } \ell \mid \text{IE}_k \text{ in POS j of outage I)}}$$

Table A.4.1 Systems and Functions for PWRs

(1)	Residual Heat Removal System
(2)	Low Pressure Injection System
(3)	High Pressure Injection System
(4)	Natural Circulation / Reflux Cooling
(5)	Gravity Feed from RWST
(6)	High Pressure and Low Pressure Recirculation Systems
(7)	Containment Spray System
(8)	Steam Generator Recirculation and Transfer System
(9)	Spent Fuel Pool Cooling System

Table A.4.2 Systems and Functions for BWRs

(1)	Control Rod Drive Pumps
(2)	Feedwater System
(3)	Condensate System
(4)	Recirculation System
(5)	Turbine Bypass Valves
(6)	Auxiliary Decay Heat Removal System
(7)	Reactor Core Isolation Cooling System
(8)	Safety Relief Valves
(9)	High Pressure Core Spray System
(10)	Low Pressure Core Spray System
(11)	Low Pressure Coolant Injection System
(12)	Spent Fuel Cooling System

Table A.4.3 References to Thermal Hydraulic Analyses for PWRs

1. Audreycheck, T.S., et al., "Loss of RHRs Cooling While the RCS is Partially Filled," WCAP - 11916, Westinghouse Electric Corporation, July 1988.
2. Naff, S.A., et al., "Thermal-Hydraulic Processes During Reduced Inventory Operation with Loss of Residual Heat Removal," NUREG/CR-5855, Idaho National Engineering Laboratory, April 1992.
3. Wald, L.W., et al., "Consequence of the Loss of Residual Heat Removal Systems in Pressurized Water Reactors," NUREG/CR-5820, Idaho National Engineering Laboratory, May 1992.
4. "Loss of Residual Heat Removal System," Diablo Canyon, Unit 2, April 10, 1987, NUREG-1269, U. S. NRC, June 1987.
5. "Loss of Vital AC Power and the Residual Heat Removal System During Mid-Loop Operations at Vogtle Unit 1 on March 20, 1990," NUREG-1410, U.S. NRC, June 1990.
6. Chu, T.L., et. al., "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry Unit 1, Analysis of Core Damage Frequency from Internal Events During Mid-Loop Operations," Brookhaven National Laboratory, NUREG/CR-6144, Volume 2, Chapter 5, June 1994.

Table A.4.4 Thermal Hydraulic Analyses Performed in Reference [4] for Grand Gulf

(1)	For different decay heat level, the following were determined: • injection flow needed to remove decay heat without steaming• injection to remove decay heat with steaming• time to boiling of vessel water• time to top of fuel due to boil off• time to suppression pool over heat
(2)	Capacity of a SRV to relieve steam at different system pressure
(3)	Number of SRVs needed to relieve water in support of ECCS injection
(4)	For different initial suppression pool levels, decay heat levels, initiating events and modes of ECCS injection, determine whether or not suppression pool makeup would be needed
(5)	Determine time when containment failure pressure is reached due to heating of flooded containment with no heat removal
(6)	Determine time to boil off of suppression pool inventory to where ECCS is lost.
(7)	Estimate the time available to restore makeup if makeup is lost and letdown is isolated, before recirculation from the core to the down comer is lost.

Table A.4.5 Success Criteria for Mitigating Features Used in Reference [3]

Feature	Time Window	Success Criteria	
		Short Term	Long Term
Reflux Cooling	< 75 hours	3 SGs	AFW to 3 SGs
	>= 75 hours and < 475 hours	2 SGs	AFW to 2 SGs
	>= 475 hours	1 SG	AFW to 1 SG
Feed-and- Spill	< 107 hour	1LHSI * (SV removed + 2 PORV)	
	> 107 hours	1LHSI * (SV removed + 1 PORV)	
	< 129 hours	2HHSI * (SV removed + 2 PORV)	
	> 129 and < 138 hours	1HHSI * (SV removed + 2 PORV)	
	>= 138 hours	1HHSI * (SV removed + 1 PORV)	
Gravity Feed	< 70 hours	at least 1 SV removed , less than 1 hour of subcooling	
	> 70 hours and < 150 hours	at least 1 SV removed, 2 hours of subcooling	
	> 150 hours and less than 768 hours	at least 1 SV removed, 3 hours of subcooling	
	>= 768 hours	at least 1 SV removed, sufficient for 24 hours	
Recirculation	<= 3 days	needed	
	> 3 days and <=10 days	not needed if RWSTs are cross tied, otherwise needed	
	> 10 days	not needed	
Spray Recirculation	<= 3 days	needed to prevent recirculation failure	
	> 3 days and <=10 days	not needed if RWSTs are cross tied, otherwise needed to prevent recirculation failure	
	> 10 days	not needed	

Appendix

Table A.4.6 Definition and Characterization of Time Windows Used in Reference [3]

	WINDOW 1	WINDOW 2	WINDOW 3	WINDOW 4
Definition	<= 75 hours	> 75 hours and <= 240 hours	> 240 hours and <= 32 days	> 32 days
Representative Decay Heat	13.23 MW (2days)*	10 MW (5 days)	7 MW (12 days)	5 MW(32 days)
Success Criteria				
Reflux Cooling	3 SGs	2 SG	2 SG	1 SG
Feed and Bleed				
LHSI	1LHSI*(SV removed + 2 PORV)	1LHSI*(SV removed + 2 PORV)	1LHSI *(SV removed + 1PORV)	1LHSI *(SV removed + 1PORV)
HHSI	1HHSI*(SV removed + 1 PORV)	1HHSI*(SV removed + 1 PORV)	1HHSI*(SV removed + 1 PORV)	1HHSI*(SV removed + 1 PORV)
Gravity Feed	1 SV removed * LHSI flow path provides 4.3 hours for operator actions (with less than 2 hours of subcooling)	1 SV removed * LHSI flow path provides 6.5 hours for operator actions (with 2 hours of subcooling)	1 SV removed * LHSI flow path provides 12 hours for operator actions (with 2 hours of subcooling)	1 SV removed * LHSI flow path provides sufficient cooling for 24 hours (with more than 3 hours of subcooling)
Recirculation	needed(HPR + LPF&Steam + LPF&Spill)	1 RWST, needed	not needed	not needed
		2 RWST, not needed		
Recirculation Spray	needed	1 RWST, NEEDED	not needed	not needed
		2 RWST, not needed		
Probability that IE Occurs in the Window				
D6	0.117 (0.31)*	0.436 (0.454)	0.375 (0.21)	7.20E-02 (2.6E-02)
R6	1.7E-02 (5.82E-02)	0.543 (0.7)	0.41 (0.24)	3.4E-02 (1.48E-03)
R10	0.0	0.0	0.016 (2.2E-02)	9.84E-01 (0.98)
* RHR2A based on time to mid-loop				

Table A.4.6 (continued)

	WINDOW 1	WINDOW 2	WINDOW 3	WINDOW 4
Decay Heat	13.23 MW(2 days)	10 MW(5 days)	7 MW(12 days)	5 MW(32 days)
Time to Boiling	15 min.	20 min.	27 min.	37 min
Time to Tygon Tube Rupture(40 psia)	23 min.	31 min.	43 min.	59 min.
Time to PRT Rupture(100 psig)	51 min.	63 min.	78 min.	96 min.
Time to 165 psia	41 min. with 2 PORV 43 min. with 1 PORV	63 min. with 2 PORV 60 min. with 1 PORV	227 min. with 2 PORV 89 min. with 1 PORV	352 min. with 2 PORV 147 min. with 1 PORV
Time to 615 psig	145 min. with 1 PORV - with two	-	-	-
Time to RWST Depletion	10 hrs	13.5 hrs	18.7 hrs	38.6 hrs
Time to AFW Initiation(with 25% SG inventory remaining)	743 min.	669 min.	925 min.	628 min.
Time to Core Uncovery	120 min.	157 min.	209 min.	273 min.
Time to Core Damage	219 min.	297 min.	411 min.	557 min.

A.5 System Analysis

System configurations during shutdown conditions can be identified by reviewing system training manuals, operating procedures used during shutdown conditions, and shift supervisor's log books. Typically, 2 or 3 different configurations may be needed to model a system. For example, for the residual heat removal system, one configuration for power operation and one for shutdown conditions would be needed. Discussions with systems engineers would be useful to better define the system configuration. Typically, the shutdown configuration of a system may differ from that of power operations in the following way:

- Valves may be in a different normal position than that of power operations.
- Automatic signals may not exist during shutdown, or may not operate in the way that supports the functions needed to mitigate an accident.
- A system or component may be in a condition that makes it impossible to operate. For example, the turbine driven auxiliary feedwater pump does not have the steam needed to run during a cold shutdown.
- Test or maintenance may make a system or component unavailable for a long time. For example, the steam generators at the Surry plant are isolated in a refueling outage for a long time.

The following are typical changes needed to convert the system fault trees developed for power operations into those suitable for shutdown conditions:

- Different failure modes of a valve may be applicable due to the different normal position.
- If the automatic signal for actuating a component does not exist during shutdown, then failure of the signal should not be a cause of failure of the component. If the component requires operator action for its actuation, the human error event has to be modeled in the fault trees.
- The physical condition of the plant may prohibit the operation of some items of equipment. The equipment should be identified and not included in the system analysis.
- Maintenance unavailabilities of equipment have to be estimated and modeled in the fault trees. In principle, they could vary from one POS to another.

A.6 Development of Event Trees and Accident Scenarios

The development of event trees and accident scenarios requires a good understanding of the plant response to the initiating event. For accidents that happen during LP&S conditions, there may not be any emergency procedures that instruct the operator as to an appropriate course of action. Whatever procedures and training are available have to be used. This includes procedures developed for accidents during power operations which may have relevant steps that are applicable to shutdown accidents. Ultimately, the scenarios developed have to be discussed with plant personnel such as operators and system engineers, to ensure they are realistic.

The event trees that delineate the accident scenarios can be developed in the same way event trees are developed for a PRA for power operation. The following is a summary of the approaches used in existing studies^[3-6]:

- **Surry LP&S Study^[3]** - In this study, the event trees were developed by using the mitigating functions defined in the abnormal procedure for loss of RHR as top events. The event trees are small and the fault trees for the top events include hardware failures of systems and human error events.
- **Grand Gulf Study^[4]** - In this study the event trees were developed in stages. First, generic functional event trees were developed, followed by the generic system-level event trees, and last, the initiating event specific system-level event trees were developed. The latter include many levels of transfers from one event tree to the other, and became difficult to solve. The advantage of such approach is that the sequences are more explicitly defined. The disadvantage is that solving the event trees is very time consuming.
- **Zion^[5] and Seabrook^[6] Studies** - These studies used the same type of event trees as the Surry LP&S study, except that the large event tree approach was used instead of the large fault tree approach.

A.7 Human Reliability Analysis

Human reliability analysis (HRA) is a very important part of a PRA for LP&S conditions. The human reliability analysis in a PRA consists of the following key tasks:

- **Identification of operator actions/errors**
Different types of human errors can be identified, and defined based on their impact on the plant. For example, diagnosis errors and action errors, errors committed before the initiating events and errors committed after the initiating events.
- **Inclusion of human error events in the logic model for accident scenario quantification**
Depending on the types of human errors, they can be included in different parts of a PRA model. Typically, pre-initiating event errors are included in system fault trees. Post-initiating events can be included in system fault trees or as top events of event trees. In Reference [3], diagnosis errors are consistently applied to all core damage sequences, and failure to diagnose is assumed to caused failure of all mitigating functions. Action errors can be included in the logic model for the hardware that they affect.
- **Quantification of human error probabilities**
For each clearly defined human error, different methods can be used to estimate the human error probability. Reference [3] used the success likelihood index methodology^[13], and Reference [4] used the Accident Sequence Evaluation Program Human Reliability Analysis Procedure (ASEP HRAP).^[14]

4.8 Presentation of Results

For each of the tasks of a LP&S PRA, the following documentation should be provided.

- **Outage Types and Plant Operational States**

The definition of outage types should be provided such that all possible outages are included. The boundaries between POSs should be clearly defined. Descriptions of major activities in each of the POS should be provided. A table showing the frequency and duration of the POSs should also include the source of data that was used.

- **Initiating Event Analysis**

A table of initiating events should be provided for each POS, providing quantitative estimates of the IE frequencies. Detailed discussions should be provided on the effects each IE has on the plant and the automatic response of the plant subsequent to the IE.

- **Success Criteria**

For each POS, a table of different methods of providing inventory makeup, decay heat removal, and reactivity control should be provided. The way in which the train level success criteria change with the decay heat should also be described. Supporting thermal-hydraulic analyses that determine the success criteria need to be provided. For each mitigating function, descriptions on the operator actions and the time needed to activate the function are needed.

- **System Analysis**

For each system that can be used to mitigate an accident, a description of the system configuration during each POS, including the conditions of the support systems, and actuation signals is needed. For each system included in the analysis, the physical condition needed for system operation should be described. It is also important to verify that the conditions are satisfied for those scenarios in which the system is operational. If operator actions are needed, they should be described and the amount of time estimated.

- **Event Tree Analysis and Accident Scenario Development**

The dominant scenarios that contribute to at least 90% of the total core damage frequency (including the basic events and their probabilities used in the quantification) should be described.

- **Human Reliability Analysis**

Human error events that appear in the dominant core damage scenarios, including the methods of the HEP quantification, should be described.

- **Results**

The results should be presented in terms of core damage frequency and conditional core damage frequency for each time window in each POS. A comparison of the conditional core damage frequencies among the POSs with CDF for power operation should be provided. The reasons for any differences in the results should be explained.

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<p>NRC FORM 336 (2-89) NRCM 1102, 3201,3202</p> <p style="text-align: center;">BIBLIOGRAPHIC DATA SHEET <i>(See Instructions on the reverse)</i></p>		<p>1. REPORT NUMBER <small>(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)</small> NUREG/CR-6144 BNL-NUREG-52399 Vol. 1</p>					
<p>2. TITLE AND SUBTITLE</p> <p>Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1:</p> <p>Summary of Results</p>		<p>3. DATE REPORT PUBLISHED</p> <table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td style="width: 50%; text-align: center;">MONTH</td> <td style="width: 50%; text-align: center;">YEAR</td> </tr> <tr> <td style="text-align: center;">October</td> <td style="text-align: center;">1995</td> </tr> </table>		MONTH	YEAR	October	1995
		MONTH	YEAR				
		October	1995				
<p>4. FIN OR GRANT NUMBER L-1922</p>							
<p>5. AUTHOR(S)</p> <p>Edited by T. L. Chu and W. T. Pratt</p>		<p>6. TYPE OF REPORT Technical</p>					
		<p>7. PERIOD COVERED (Inclusive Dates) 09/01/90 - 09/30/95</p>					
<p>8. PERFORMING ORGANIZATION - NAME AND ADDRESS <small>(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)</small></p> <p>Brookhaven National Laboratory Department of Advanced Technology, Bldg. 130 P.O. Box 5000 Upton, New York 11973-5000</p>							
<p>9. SPONSORING ORGANIZATION - NAME AND ADDRESS <small>(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)</small></p> <p>Division of Systems Technology Office of Nuclear Regulatory Research U.S. Nuclear Regulatory Commission Washington, DC 20555-0001</p>							
<p>10. SUPPLEMENTARY NOTES R. Robinson, NRC Project Manager</p>							
<p>11. ABSTRACT <small>(200 words or less)</small></p> <p>This document contains a summarization of the results and insights from the Level 1 accident sequence analyses of internally initiated events, internally initiated fire and flood events, seismically initiated events, and the Level 2/3 risk analysis of internally initiated events (excluding fire and flood) for Surry, Unit 1. The analysis was confined to mid-loop operation, which can occur during three plant operational states (identified as POSs R6 and R10 during a refueling outage, and POS D6 during drained maintenance). The report summarizes the Level 1 information contained in Volumes 2 - 5 and the Level 2/3 information contained in Volume 6 of NUREG/CR-6144.</p>							
<p>12. KEYWORDS/DESCRIPTORS <small>(List words or phrases that will assist researchers in locating the report.)</small></p> <p>Low Power and Shutdown, Probabilistic Risk Assessment, PRA, PWR, Risk, Surry, Plant Operational State, Insights, Mid-loop, Internal Events, Fire, Flood, Seismic</p>		<p>13. AVAILABILITY STATEMENT Unlimited</p>					
		<p>14. SECURITY CLASSIFICATION</p> <p><small>(This Page)</small> Unclassified</p> <p><small>(This Report)</small> Unclassified</p>					
		<p>15. NUMBER OF PAGES</p>					
		<p>16. PRICE</p>

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