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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

Appendix I

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ABSTRACT

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 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.

The objectives of the program are to assess the risks of severe accidents 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 objective of this volume of the report is to document the approach utilized in the level-1 internal events 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 (POSSs). The objective of the phase 1 study was to identify potential vulnerable plant configurations, to characterize (on a high, medium, or low basis) the potential core damage accident scenarios, 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 level-1 study includes plant damage state analysis, and uncertainty analysis. Volume 1 summarizes the results of the study. Internal events analysis is documented in Volume 2. It also contains an appendix that documents the part of the phase 1 study that has to do 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 consequences analysis.

In the phase 2 study, system models applicable for shutdown conditions were developed and supporting thermal hydraulic analysis were performed to determine both the timing of the accidents and success criteria for systems. Initiating events that may occur during mid-loop operations were identified and accident sequence event trees were developed and quantified. In the preliminary quantification of the mid-loop accident sequences, it was found that the decay heat at which the accident initiating event occurs is an important parameter that determines both the success criteria for the mitigating functions and the time available for operator actions. In order to better account for the decay heat, a "time window" approach was

developed. In this approach, time windows after shutdown were defined based on the success criteria established for the various methods that can be used to mitigate the accident. Within each time window, the decay heat and accident sequence timing are more accurately defined and new event trees developed and quantified accordingly. Statistical analysis of the past outage data was performed to determine the time at which a mid-loop condition is reached, and the duration of the mid-loop operation. Past outage data were used to determine the probability that an accident initiating event occurs in each of the time windows. This probability is used in the quantification of the accident sequences.

The mean core damage frequency of the Surry plant due to internal events that may take place during mid-loop operations is $5\text{E-}06$ per year, and the 5th and 95th percentiles are $5\text{E-}07$ and $2\text{E-}05$ per year, respectively. This can be compared with the mean core damage frequency from internal events of $4\text{E-}05$ per year estimated in the NUREG-1150 study for full power operations.

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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
- 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

Foreword (continued)

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
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APPENDIX I

Compilation of Work Done in Phase 1A - Coarse Screening Analysis

Appendix I Compilation of Work Done in Phase 1A - Coarse Screening Analysis

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Appendix I.1 Introduction

In this appendix, the work performed during the phase 1 of the level-1 low power and shutdown study is compiled. In phase 1, a screening analysis was performed for all plant operational states (POSS) in the 4 outage types. Due to the wider scope of phase 1, a lot of the work performed is not included in the main report of this report. The complete phase 1 report is only in a draft form and is not published. It is available at the NRC public document room. The intent of this appendix is to include those analysis performed in phase 1 for POSS other than mid-loop. Some modifications and corrections to the report are also included.

Appendix I.2 is the summary of the phase 1 report with an additional section added. Appendix I.3 is a tabulation of the results of the phase 1 analysis with corrections and modifications. Appendix I.4 documents the thermal hydraulic analysis performed for the 15 POSS. Appendix I.5 documents the human reliability analysis approach use in phase 1. Appendix I.6 documents the event tree analysis for all POSS.

Appendix I.2 Summary of Phase 1 Study

In this appendix, the summary of the phase 1 report is provided. In addition, a new Section, Section 1.2.5, is added to document some quantitative analysis of the results.

1. SUMMARY

1.1 BACKGROUND

This document presents the preliminary internal events Level 1 results (including fire and flood) obtained as a result of a coarse screening analysis on the low power and shutdown accident frequencies of the Surry Nuclear Power Plant. The work was performed by Brookhaven National Laboratory (BNL) for the Nuclear Regulatory Commission Office of Nuclear Regulatory Research (RES). This coarse screening analysis was performed in order to meet the RES commitment to the Office of Nuclear Reactor Regulation (NRR), in support of the NRC staff's follow-up actions subsequent to the March 20, 1990 Vogtle incident. It is the first phase of a study that will produce final estimates on the core damage frequency of a pressurized water reactor (PWR) during low power and shutdown conditions. Phase 2 of the study, to be completed by May 1993, is to produce a final Level 1 PRA analysis, and will be guided by the Phase 1 results to concentrate the effort on the various plant operational states, the dominant accident sequences, and pertinent data items according to their importance to core damage frequency and risk.

Surry Unit 1 was chosen as the PWR that will be analyzed, in part because the Surry plant was previously analyzed in the Reactor Safety Study and NUREG-1150 and in part because Virginia Power offered their cooperation. This will facilitate comparison of the core damage frequency or risk during shutdown operations with that of power operations. The Surry Plant contains two units 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 to be analyzed in a parallel study performed by Sandia National Laboratories (SNL).

This study is concerned with 'traditional' internal events as well as internal fires and floods. External events are beyond the scope of BNL efforts.

A separate NRC-sponsored project entitled "Scoping Analysis of LWR Shutdown Accidents Initiated by Earthquakes," performed by Future Resources Associates Inc. and PRD Consulting, addresses seismic induced accidents. In that study, seismic induced loss-of-offsite-power transients were analyzed with the plant-specific fragilities generated in the NUREG-1150 seismic PRA for Surry, and the loss-of-offsite-power event tree developed in this study.

1.2 RESULTS AND INSIGHTS

Due to the necessarily conservative nature of the coarse screening analysis, it was decided that qualitative results will be stressed in this document. Qualitative discussions highlighting the results follow. Some quantification has been included in various areas of this report to provide some additional perspective on the relative ranking of the results. Chapter 10 provides the qualitative characterization of the low power and shutdown scenarios. A coarse grouping of the core damage sequences into plant damage states is also provided. The results of the fire and flood analyses are presented in Chapters 11 and 12 respectively.

1.2.1 Potentially Vulnerable Configurations

In the coarse screening analysis, some plant configurations in an outage were found to be more vulnerable than the others. These configurations are based on the operational practices at the Surry plant and are routinely entered during shutdown operations. Their descriptions follow:

Mid-loop Operations Approximately twice a year, the plant has gone into mid-loop operation with the vessel level maintained at ~9 inches above the reactor coolant loop hot leg mid-plane. This can occur within one day after shutdown with decay heat still as high as 12.4 Mw. Core uncover can occur as early as 1.5 hours after a loss of cooling event. In a refueling outage, one or more reactor coolant loops have been isolated for extended periods of time, including a large fraction (if not total) of the mid-loop operation period. This leaves the associated SG(s) ineffective as a heat sink.

Use of Temporary Seals at the Seal Table In-core flux thimbles (and core-exit thermocouples) are normally inserted up into the core region through the guide tubes that penetrate the bottom head of the reactor vessel. The space between the moveable thimble tubes and fixed guide tubes form part of the reactor coolant pressure boundary. This space is sealed against full system pressure at the seal table during normal operation. In a refueling outage, the in-core flux thimbles are retracted out of the core, and temporary seals are used at the seal table. These temporary seals can only withstand approximately 15-25 psig and become a weakness in the system boundary. A slight increase in system pressure could cause failure of the temporary seals inducing a loss of coolant accident. Further pressurization can lead to core uncover rather quickly. It was estimated that approximately 10 days during a refueling outage such seals are in place with the reactor coolant system (RCS) closed and vessel head in place. It is during this period that such a scenario can potentially occur.

Isolation of the Steam Generators During an outage, the secondary sides of the SGs are often filled to approximately 98-100 percent wide range level (called "wet layup"), and represent a good heat sink if connected to the RCS. As discussed above, in a refueling outage with drained maintenance, reactor coolant loops can be isolated for extended periods of time. This removes the one or more SGs as a potential heat sink.

During a RCS cooldown prior to initiating the RHR system, the motor operated valves (MOVs) in the individual SG feed lines from the auxiliary feedwater (AFW) system are closed. After the RCS temperature decreases to 228-250° F, the main steam trip valves (and non-return valves) are closed. Therefore, the secondary side of the SGs are isolated from the main steam system. This configuration makes station blackout (SBO) with the plant shutdown very difficult to cope with. Basically, the AFW lines to each individual SG are isolated with the MOVs located inside the containment. The SG relief valves (atmospheric dumps) fail closed on loss of air and can not be opened manually at the valves (quite unique at the Surry plant).

The loss-of-all-AC-power emergency procedure instructs the operators to manually open the bypass valves (locally at the valves) around the main steam trip valves, close the condenser vacuum breaker, "evacuate (the) turbine deck", and dump steam to the main condenser (and ultimately to the turbine building above the turbine deck). The timely success in manually dumping steam into the turbine building is essential to the mitigation of an SBO at Surry. This is especially true because the operating RHR removal system is a weakness in the RCS pressure boundary, and the RCS side relief valves are not capable of relieving the large volume of steam that would be generated in the vessel.

It was estimated that an over pressurization of the RHR system may occur as early as 0.7 hours after a SBO occurs. The RHR system has a design pressure of 600 psig. It is connected to the RCS prior to the SBO.

The relief valves that may be available are the pressurizer PORVs, RHR relief valve and possibly the relief valve in the CVCS letdown line. During a SBO, these valves are expected to relieve either liquid or two phase flow. It was estimated that the combined capacity of the valves is 2550 gpm at 600 psig, equivalent to the volumetric vapor generation rate of 5.8 Mw of decay heat. This corresponds to the decay heat approximately 17 days after shutdown. After 17 days, the relief valves should be able to relieve. However, they may still be inoperable or fail to open upon demand.

1.2.2 Maintenance Unavailability and Simultaneous Maintenance of Redundant Equipment

As expected, preliminary analysis of the data collected indicates that maintenance unavailabilities at shutdown are much higher than that during power operation. With the plant at cold shutdown, few Surry Technical Specification requirements are applicable. In terms of inventory makeup to the RCS, one high head injection train and one low head injection train are required to be operable while the RCS level is below 15.3' (i.e., reduced inventory), which is less than 3 feet above the 12.5' mid-loop level. This requirement is not in the Surry Technical Specifications, but is written in the operating procedure as a result of Surry's response to Generic Letter 88-17.

One concern regarding maintenance unavailability is simultaneous maintenance of redundant equipment. Preliminary review of the Surry data indicates that simultaneous maintenance of motor driven AFW pumps has taken place. In principle, simultaneous maintenance may take place if not otherwise prohibited. In this coarse screening analysis, the assumption that simultaneous maintenance can occur was applied and simultaneous maintenance turned out to be the dominant cause for core damage in many sequences. A more realistic maintenance model based upon the actual plant practice will be developed in Phase 2 of this study. For example, discussion with the plant personnel indicated that, in practice, no two emergency diesel generators (EDGs) could be maintained at the same time. (Note: there are only three EDGs total for the Surry Plant, that is both units).

1.2.3 Reactivity Accident Scenarios Identified in the French PRA

Two reactivity accident scenarios identified in the French PRA were analyzed taking into consideration the Surry-specific design and operation. The first scenario involves startup of a reactor coolant pump after improper dilution during a loss of offsite power transient. The second scenario involves a dilution event while the plant is in the mid-loop condition. Conservative assumptions were made in the coarse screening analysis, and the results indicate that the scenarios have high frequency. It is believed that some of the assumptions are very conservative. For example, in the first scenario, it was assumed that the decay heat is so low that natural circulation does not cause mixing of the unborated water with the reactor coolant. It is expected that more deterministic analysis is needed to resolve the issue in support of the Phase 2 effort.

1.2.4 Findings of Fire and Flood Analysis

The results of the analysis show that fire and flood risk at shutdown could be potentially significant. More detailed analysis should be done for the following list of fire and flood areas.

FireFlood

Cable vault and tunnels YesNo

Emergency switchgear and
relay room YesYes

Control room YesNo
Containment YesYes
Aux. Building YesYes
Safeguard areaYesYes
Turbine buildingYesYes

In the process of performing this Phase 1 study, some issues that only apply to shutdown conditions were identified:

1. Surry's compliance with Appendix R requirements emphasizes power operations not shutdown conditions. For example, Appendix R states that

"Both trains of equipment necessary to achieve cold shutdown may be damaged by a single fire ... but damage must be limited so that at least one train can be repaired or made operable within 72 hours using onsite capability."

Based on this statement, the Surry implementation of Appendix R relies on repair or replacement of the residual heat removal (RHR) pump cables rather than cable separation. According to the associated analysis, cable replacement will take 35 hours. Therefore, if the fire starts while the plant is operating, there is enough time to achieve cold shutdown within the allowed 72 hours. However, in this study, it is assumed that the plant is already in cold shutdown when the fire occurs. In this case, the 35 hour replacement time appears to be too long for the scenarios analyzed. The basic difference here is that, from power, the steam generators can be utilized to delay the need for RHR. At shutdown, this delay capability is not available.

Arguing in a similar manner, the credit taken by the Appendix R submittal for repair or replacement of the component cooling water (CCW) pump motors and cables, while acceptable for fires starting when the plant is at power, appears to be inappropriate for fires starting when the plant is shutdown.

Two other situations which appear to have been overlooked due to the emphasis on fires during operation involve the steam generator pilot-operated relief valves (SG PORVs) and Main Control Room fires. In the case of a fire that causes the loss of the SG PORVs, the alternative shutdown method prescribed by the Appendix R submittal employs the SG code safety valves. Although this is a viable approach at power, the setpoints of the valves are too high to be useful for decay heat removal while at cold shutdown. In the case of Main Control Room fires, the operators may have to use the Auxiliary Shutdown Panel (in the Emergency Switchgear Room) to control the plant. However, as seen from Tables 5-2a and 5-2b of Appendix R the submittal, this panel does not have control circuits for the RHR pumps.

It should be pointed out that the above finding regarding Appendix R's coverage of shutdown fires is generic; plants that rely on alternative shutdown methods rather than separation may need to develop means to maintain a plant at cold shutdown, given that a fire occurs when the plant is shut down (or in the process of shutting down).

2. Fire or flood barriers that are available during power operations may be removed during shutdown. They include cable and pipe penetrations, fire doors, flood dikes, floor plugs, and backflow preventers in the floor drain system. Surry did have some reported incidents in which one flood dike and two backflow preventers were removed.

1.2.5 Evaluation of the Changing Configuration throughout an Outage

Two applications of the quantitative results are discussed. The purpose of the applications is to demonstrate how the PRA model can be used to evaluate the changing plant configurations throughout an outage. It should be mentioned that the quantitative results of the phase 1 study are conservative and should not be considered as realistic representation of the risk.

Case 1- Evaluation of changing configuration within a POS

The PRA model developed for a given POS assumes that the plant configuration is well defined. In reality, the plant configuration continues changing within the POS. For example, Figure 1-1 shows the how the status of a few components varied in POS 12 of a refueling outage. It is estimated based on the outage log-book for the refueling outage in October of 1990. In the base case PRA model, the maintenance unavailabilities of the components were estimated by the fraction of time that the component is down within the POS. That is, the average unavailability throughout the POS is used. It is assumed also that the maintenance events are independent. The core damage frequency of two important sequences for POS 12 is quantified using the PRA and plotted as the dashed line in Figure 1-2. The scale of the core damage frequency is shown on the right and that for the conditional core damage probability is shown on the left. Conditional core damage probability is the core damage frequency divided by the frequency of the initiating event.

To more realistically evaluate the changing component availability, the core damage cutsets were re-evaluated using the known conditions of the components shown in Figure 1-1, and the result is plotted as the step function in Figure 1-2. It can be seen that a very large variation within the POS is observed and the average over the step function, as shown by the solid line, is not the same as that obtained from the PRA.

Case 2- Plot of conditional core damage frequency as a function of the POSs

Figure 1-3 is a plot of the core damage frequency as a function of the POSs for a refueling outage. The conditional core damage frequency for a POS is calculated by dividing its core damage frequency by the estimated fraction of a year that the plant is in the POS. This plot provides a measure of the relative vulnerability of the POSs to core damage.

Figure I-1 Downtime of Selected Components in POS 12 of Oct. 1990 Refueling Outage

Figure I-2 Time Dependent Analysis of Conditional Core Damage Probability in POS 12 of the Oct. 1990 Refuel Outage

Figure I-3 Conditional Core Damage Frequency as a Function of POSs for a Refueling Outage

Appendix I.3 Results of the Phase 1 Study

In this appendix, an updated Chapter 10 of the phase 1 study is provided. It tabulates the core damage sequences and ranks their likelihood in terms of "high", "medium", and "low". Also provided are the timing of the accident scenarios in terms of "early" and "late", and containment status in terms of "open", and "closed". The tables were updated to reflect the changes and corrections made since the phase 1 report was completed. A Venn diagram showing the breakdown of the core damage sequences is added.

10 RESULTS

Tables 10-1 to 10-7 summarize the results of the accident sequence quantification. The results of the fire and flood analysis are tabulated in Chapters 11 and 12, respectively. Due to the preliminary nature of the analysis no quantitative result is given. The sequences are ranked according to the estimated core damage frequency, the timing of the accident, and the condition of the containment.

Ranking of Core Damage Sequences (H,M,L)

The core damage sequences are ranked according to the estimated core damage frequency into three classes: high (H), medium (M), and low (L).

Early vs Late Core Damage (E,L)

The timing of the core damage sequences was estimated for each core damage sequence. If core damage is expected within the first 2 hours after the initiating event, the sequence is an early sequence (E). Otherwise, it is a late sequence (L). This timing is used in determining the possible condition of the containment. It is assumed that in a late core damage sequence the containment closure can be achieved after the initiating event has occurred.

Containment Open or Closed (O,C)

Containment integrity is required whenever the Reactor Coolant System (RCS) temperature is above 200° F. Therefore, the containment can be assumed to be closed in those Plant Operational States (POSS) with RCS temperature above 200° F. Any accident sequence occurring in one of these POSSs has a containment closed classification (C). Such classification does not account for the possibility that containment may fail as a result of the accident. For those core damage sequences that occur when the RCS is below 200° F, containment is assumed to be initially open. If more than 2 hours is available prior to core damage in a sequence and AC is available (not station blackout (SBO)), it is assumed that sufficient time is available to close the containment and a containment closed classification is used. Otherwise, a containment open classification is used (O).

Accident Sequence Severity Estimation

The results summarized in Tables 10-1 through 10-7 have been further analyzed and Plant Damage State (PDS) groups were created. The discussion below outlines the approach used in this procedure.

The results presented in Tables 10-1 through 10-7 list the dominant accident sequences ending with a core damage (Level I portion of PRA). All these sequences were then grouped into PDS groups in terms of potential severity of the radioactive release. Grouping was based on three characteristics specific to each sequence (descriptions of the characteristics can be found in the following sections):

1. status of the containment (open/closed),
2. status of the vessel head (removed/in place),
3. timing of core damage (early/late).

The fourth characteristic used later in further categorization was the sequence likelihood. Details of the approach and results of the analysis are presented below.

Plant Damage State (PDS) Formation

The Level I PRA study produced nineteen categories of accident sequences based on initiating events (from nineteen classes of event trees). Table 10-8 lists the categories and total number of sequences in each category.

Definitions

The three characteristics of PDS groups (containment status, vessel head status, and core damage timing) were used to determine the potential severity of release (consequence) associated with those groups. This classification is coarse and should be used only for the preliminary prioritization of sequences in the process of selecting the PDSs for more detailed analysis during the Phase II of the PRA study.

Assumptions in Interpretation of Data

The following assumptions have been made while interpreting the PDS data from the Level I analysis.

- a. In case of V-Sequence Accidents: "O & C" is equivalent to "Open" (Table 10.2c).
- b. "Bypassed Containment" in SGTR sequences is equivalent to "Open" for both Early and Late CD sequences.
- c. "Upper Head Removed" state is assumed for POSs seven through nine.
- d. Some scenarios/sequences that may occur in more than one outage type or more than one POS are assumed single sequences (see Tables 10.2a,b,c, 10.5, and 10.6).
- e. Initially "Open Containment" with a Late CD sequence is equivalent to "Closed".
- f. Initially "Open Containment" with SBO is equivalent to "Open" for both Early and Late CD sequences.

Note also that the LOCA sequences are combined for all four initial states: Refueling, Drained Maintenance, Non-Drained Maintenance with RHR, and Non-Drained Maintenance without RHR. Finally, in the Reactivity Accidents group the sequence numbers for LOCAs are not numerals (several event trees): literals "a" and "b" are used instead.

All sequences were grouped into plant damage states (PDS) based on characteristics of each accident sequence and status of the plant that could affect the severity of radionuclide release. In this study, five plant damage state (PDS) groups have been defined based on the following three parameters:

1. **Status of the containment.** If the containment is initially open and the core damage (CD) timing is early, the PDS is referred to a high severity PDS group. The same is valid if the containment is initially open and an SBO is in progress; in this case the PDS timing is irrelevant, and the PDS falls into one of the high severity groups. However, in cases where the containment is initially open, AC power is available, and the PDS timing is LATE, the operators are assumed to be able to close the containment, and the corresponding PDSs are referred to the low severity group 5. The PDSs with a containment initially closed are referred to the low severity group.

The latter simplification is based on the NUREG-1150 results for Surry 1 (NUREG/CR-4551, Vol. 3, Rev. 1, Part 1) which indicate (Figure S.2) that only about 19% of accidents at full power lead to a containment failure or development of a containment leak.

2. **Status of the reactor vessel upper head.** Two possibilities are considered: upper head removed or in place. At this time, the consequence severity does not depend on the status of the upper head. However, the upper head status is taken into account in subdividing the PDSs with high severity consequences.

3. **PDS timing.** Presently, as with the status of the upper head, the PDS timing is only taken into account in subdividing the PDSs within the class of high severity consequences. The EARLY and LATE PDS times refer to the time span from the accident initiation to the time of core damage (CD):

EARLY, if CD TIME $\leq$ 2 hours,
LATE, if CD TIME $>$ 2 hours.

Table 10-9 provides the definitions of the consequence severity for five PDS groups. In terms of the potential consequences, the PDS groups were divided in two major classes: HIGH and LOW consequences classes. Although a medium consequences class could be introduced (for example, including PDS groups two and four), it was not done in order to be consistent with the counterpart of this analysis at Sandia National Laboratory on the Grand Gulf plant.

The PDSs were divided into two groups based on the potential severity of the accident. The first group consisted of accidents that have the potential to have large consequences. This group includes all the accidents that have the containment open before core damage. For these accidents, an early release to the environment will occur. Although the vessel integrity and the core damage time will affect the magnitude of the source term and the potential for recovery, the fact that the containment is open (or bypassed) in these accidents is sufficient to place all of these accidents in the high severity group. The second group consisted of accidents that are expected to have low consequences. This group consisted of accidents in which the containment is closed.

In addition to the three characteristics used for the PDS consequence severity grouping, each sequence was also classified according to its frequency of occurrence. At this level of the study, instead of using the actual values of frequencies for the subsequent analysis, all sequences were binned into three **accident likelihood groups**: **Low Likelihood (L)**, **Medium Likelihood (M)**, and **High Likelihood (H)**, according to the sequence's frequency of occurrence.

This technique was employed since the actual values of frequencies have been obtained in a very approximate way, subject to re-evaluation in a more detailed study.

Plant Damage State Severity Estimation: Results

The results of the severity grouping are summarized in Table 10-10. Column one shows the PDS number, column two shows the severity class, columns three through five show the number of sequences arranged by their likelihood of appearance, and columns seven through nine show the same information as a percentage within each likelihood group.

The following three observations can be made:

- a) As it follows from Table 10-10, only 2.3% (22/947) constitute the HIGH likelihood group, 17.4% (165/947) fall into the MEDIUM likelihood group, and the remaining 80.3% of the sequences (760/947) fall into the LOW likelihood group.
- b) In terms of severity of the accident consequences, the majority of the sequences fell into the LOW severity class for each of the three likelihood groups (50.0%, 46.6%, and 78.3%, respectively).
- c) Among the HIGH severity class PDS groups, PDS group #4 has a dominating weight: its fractions are 0%, 17.0%, and 15.8%, respectively, within each likelihood group. According to the Table 10-9 definitions, this group is composed of the accident sequences with a late core damage developing in the plant with an open or bypassed containment and the vessel upper head in place.

As it was mentioned before, the present estimates of the accidents sequence severity is based on a largely qualitative analysis of the event trees generated in this study. It is expected, however, that the analysis of results will lead to a rough identification of the accident sequences with highest severity potential for which a detailed Level I thermal hydraulics analysis followed by the Level II and III analyses will be performed.

Figure 10-1 is a Venn diagram showing the breakdown of the sequences based on the parameters used to define the plant damage states.

Figure 10-1 Phase 1 Surry PRA, Distribution of Core Damage Sequences (947)

Table 10-1a

**Results of Accident Sequence Quantification for
T₁ Transients (Refueling Outage)**

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
2	5	L	L	C
	6	L	L	C
	8	L	L	C
	9	L	L	C
	10	L	L	C
	11	M	L	C
	12	M	L	C
14	5	L	L	C
	6	M	L	C
	8	L	L	C
	9	L	L	C
	10	L	L	C
	11	M	L	C
	12	M	L	C

Table 10-1b

**Results of Accident Sequence Quantification for
T_{2a} Transients (Refueling Outage)**

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
1	5	L	L	C
	6	L	L	C
	8	L	L	C
	9	L	L	C
	10	L	L	C
	11	L	L	C
	12	L	L	C
15	5	L	L	C
	6	L	L	C
	8	L	L	C
	9	L	L	C
	10	L	L	C
	11	M	L	C
	12	M	L	C

Table 10-1c

**Results of Accident Sequence Quantification for
T, Transients (Refueling Outage)**

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
2	6	L	L	C
	7	L	L	C
	9	L	L	C
	10	L	L	C
	11	L	L	C
	12	L	L	C
	13	L	L	C
14	6	L	L	C
	7	L	L	C
	9	L	L	C
	10	L	L	C
	11	L	L	C
	12	L	L	C
	13	L	L	C

Table 10-1d

**Results of Accident Sequence Quantification for
T₂₀ Transients (Refueling Outage)**

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
1	6	L	L	C
	7	L	L	C
	9	L	L	C
	10	L	L	C
	11	L	L	C
	12	L	L	C
	13	L	L	C
15	6	L	L	C
	7	L	L	C
	9	L	L	C
	10	L	L	C
	11	L	L	C
	12	L	L	C
	13	L	L	C

Table 10-2a

Results of Accident Sequence Quantification for LOCAs

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
A	1	2	L	L	C
		4	L	L	C
		5	L	L	C
		6	L	E	C
		7	L	E	C
	2	2	L	L	C
		4	L	L	C
		5	L	L	C
		6	L	E	C
		7	M	E	C
	3	2	L	L	C
		3	L	E	C
		4	L	E	C
		5	L	E	C
	4	2	M	L	C
		3	L	E	O
		4	L	E	O
		5	L	E	O
	12	2	M	L	C
		3	L	E	O
		4	M	E	O
		5	L	E	O
	13	2	L	L	C
		3	L	E	C
		4	L	E	C
		5	L	E	C

Table 10-2a (continued)

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
	14	2	L	L	C
		3	L	E	C
		4	L	E	C
		5	L	E	C
	15	2	L	L	C
		3	L	E	C
		4	L	E	C
S1	1	2	L	L	C
		3	L	E	C
		5	L	L	C
		6	L	E	C
		7	L	L	C
		8	L	E	C
		9	L	E	C
		2	L	L	C
		3	L	E	C
	2	5	L	L	C
		6	L	L	C
		7	L	L	C
		6	L	E	C
		7	L	L	C
		8	L	E	C
		9	L	E	C
		10	M	E	C
	3	2	L	L	C
		3	L	E	C
		5	L	L	C
		6	L	E	C
		7	L	E	C
		8	L	E	C

Table 10-2a (continued)

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
		9	L	E	C
	4	2	M	L	C
		3	L	E	O
		5	L	L	C
		6	L	E	O
		7	L	E	O
		8	M	E	O
		9	L	E	O
	12	2	M	L	C
		3	L	E	O
		4	M	E	O
		5	L	E	O
	13	2	L	L	C
		3	L	E	C
		5	L	L	C
		6	L	E	C
		7	L	E	C
		8	L	E	C
		9	L	E	C
	14	2	L	L	C
		3	L	E	C
		4	L	E	C
		5	L	E	C
		6	L	E	C
	15	2	L	L	C
		3	L	E	C
		4	L	E	C
		5	L	E	C

Table 10-2a (continued)

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
S2	1	2	M	L	C
		4	M	L	C
		5	M	L	C
		7	M	L	C
		8	M	L	C
		10	L	L	C
		11	L	L	C
		13	L	L	C
		14	L	L	C
		16	L	L	C
		17	L	L	C
		18	L	L	C
		19	M	E	C
		20	M	E	C
	2	2	M	L	C
		4	M	L	C
		5	M	L	C
		7	M	L	C
		8	M	L	C
		10	L	L	C
		11	L	L	C
		13	L	L	C
		14	L	L	C
		16	L	L	C
		17	L	L	C
		18	L	L	C
		19	M	E	C
		20	M	E	C

Table 10-2a (continued)

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
		21	M	E	C
	3	2	L	L	C
		4	L	L	C
		5	L	L	C
		7	L	L	C
		9	L	L	C
		10	L	L	C
		12	L	L	C
		14	L	L	C
		15	L	L	C
		17	L	L	C
		18	L	L	C
		19	L	E	C
		20	L	E	C
		21	L	E	C
		22	L	E	C
	4	2	M	L	C
		4	L	L	C
		5	L	L	C
		7	L	L	C
S2	4	9	L	L	C
		10	L	L	C
		12	L	L	C
		14	L	L	C
		15	L	L	C
		17	L	L	C
		18	L	L	C
		19	L	E	O

Table 10-2a (continued)

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
		20	L	E	O
		21	M	E	O
		22	L	E	O
	12	3	L	L	C
		4	L	L	C
		5	L	E	O
		6	M	E	O
		7	L	E	O
	13	3	L	L	C
		4	L	L	C
		5	L	E	C
		6	L	E	C
		7	L	E	C
	14R&D	2	M	L	C
		4	M	L	C
		5	M	L	C
		7	L	L	C
		8	L	L	C
S2		9	M	E	C
		10	M	E	C
	14N1	2	M	L	C
		4	M	L	C
		5	M	L	C
		7	L	L	C
		8	L	L	C
		9	L	E	C
		10	M	E	C
		11	M	E	C

Table 10-2a (continued)

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
	15R&D	2	M	L	C
		4	M	L	C
		5	M	L	C
		7	L	L	C
		8	L	L	C
		9	M	E	C
	15N1	2	M	L	C
		4	M	L	C
		5	M	L	C
		7	L	L	C
	15N1	8	L	L	C
		9	L	E	C
		10	M	E	C
S3	1	3	L	L	C
		5	L	L	C
		6	L	L	C
S3	1	9	L	L	C
		11	L	L	C
		12	L	L	C
		14	L	L	C
		15	L	L	C
		17	L	L	C
		18	L	L	C
		19	L	L	C
		20	L	E	C
		22	L	E	C
	2	3	L	L	C
		5	L	L	C

Table 10-2a (continued)

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
		6	L	L	C
		9	L	L	C
		11	L	L	C
		12	L	L	C
		14	L	L	C
		15	L	L	C
		17	L	L	C
		18	L	L	C
		19	L	L	C
		20	L	E	C
		22	L	E	C
		23	M	E	C
	3	10	L	E	C
	4	10	M	E	O
S3	12	2	M	E	O
	13	2	L	E	C
	14	3	L	L	C
		5	L	L	C
		6	L	L	C
		9	L	L	C
		11	L	L	C
		12	L	L	C
		14	L	L	C
		15	L	L	C
		16	L	E	C
		18	L	E	C
		19	L	E	C

Table 10-2a (continued)

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
	15	3	L	L	C
		5	L	L	C
		6	L	L	C
		9	L	L	C
		11	L	L	C
		12	L	L	C
		14	L	L	C
		15	L	L	C
		16	L	E	C
		18	M	E	C
Cavity seal failure	8	All	M	L	C
Cavity drain valve open	7	All	M	L	C

Table 10-2b

Results of Accident Sequence Quantification for SGTR
SGTR Core Damage Sequences

Initiating Event	POS	Sequence	CDF	Early or Late	Containment Open - O Closed - C Bypassed - B
SGTRI	1	5	L	L	C
		6	L	L	B
		7	L	L	B
		9	L	L	B
		11	L	L	C
		12	L	L	C
		13	L	L	B
		14	L	L	C
		16	L	L	B
		17	L	L	B
		18	L	L	B
		19	L	L	B
		20	L	L	B
SGTRI/SGRCT	1	2/4 15/4	L L	E E	C C
SGRT2	2	5	L	L	C
		6	L	L	B
		7	L	L	B
		9	M	L	B
		11	L	L	C
		12	L	L	C
		13	L	L	B
		17	L	L	C
		18	L	L	C
		19	L	L	B
		21	L	L	C

Table 10.2b (continued)

Initiating Event	POIS	Sequence	CDF	Early or Late	Containment Open - O Closed - C Bypassed - B
SGTR2	2	23	L	L	C
		24	L	L	C
		25	L	L	C
		26	L	L	B
		28	L	L	B
		29	M	L	B
		30	M	L	B
		32	L	L	B
		33	L	L	B
		34	L	L	B
SGTR2/SGRCT	2	2/4	L	E	C
		14/4	L	E	C
		27/4	L	E	C
SGTR3	3	7	L	L	C
		8	L	L	B
		9	L	L	B
		11	L	L	B
		13	L	L	C
		14	L	L	C
		15	L	L	B
		19	L	L	C
		20	L	L	B
		21	L	L	B
		23	L	L	B
		25	L	L	C
SGTR3	3	26	L	L	C
		27	L	L	B
SGTR3	3	28	L	L	C

Table 10.2b (continued)

Initiating Event	POS	Sequence	CDF	Early or Late	Containment
					Open - O Closed - C Bypassed - B
		30	L	L	B
		32	L	L	B
		33	L	L	B
		34	L	L	B
		36	L	L	B
		37	L	L	B
		38	L	L	B
		39	L	L	B
SGTR3/SGRCT	3	2/4	L	E	C
		4/4	L	E	C
		16/4	L	E	C
		29/4	L	E	C
SGTR4	4	7	L	L	O
		8	L	L	B
		9	L	L	B
		11	L	L	B
		13	L	L	O
		14	L	L	O
		15	L	L	B
		19	L	L	O
		20	L	L	B
		21	L	L	B
		23	L	L	B
		25	L	L	O
		26	L	L	O
		27	L	L	B
SGTR4	4	28	L	L	O
		30	L	L	B

Table 10.2b (continued)

Initiating Event	POS	Sequence	CDF	Early or Late	Containment Open - O Closed - C Bypassed - B
		33	L	L	B
		34	L	L	B
		36	L	L	B
		37	L	L	B
		38	L	L	B
SGTR4/SGRCT	4	39	L	L	B
		2/4	L	E	O
		4/4	L	E	O
		16/4	L	E	O
		29/4	L	E	O
SGTRI2	12	5	L	L	B
		7	L	L	B
		8	L	L	B
SGTRI2/SGRCT	12	2/4 6/4	L L	E E	O O
SGTRI3	13	5	L	L	B
		7	L	L	B
		8	L	L	B
SGT13/SGRCT	13	2/4 6/4	L L	E E	C C
SGRT14	14	5	L	L	C
		6	L	L	B
		7	L	L	B
		9	M	L	B
SGTRI4	14	11	L	L	C
		12	L	L	C
		13	L	L	B
		17	L	L	C

Table 10.2b (continued)

Initiating Event	POS	Sequence	CDF	Early or Late	Containment Open - O Closed - C Bypassed - B
		17	L	L	C
		18	L	L	C
		19	L	L	B
		21	L	L	C
		23	L	L	C
		24	L	L	C
		25	L	L	C
		26	L	L	B
		28	L	L	B
		29	M	L	B
		30	L	L	B
		32	L	L	B
		33	L	L	B
		34	L	L	B
		35	L	L	B
SGTRI4/SGRCT	14	2/4 14/4	L L	E E	C C
		27/4	L	E	C
SGTRI5	15	5	L	L	C
		6	L	L	B
		7	L	L	B
		9	M	L	B
		11	L	L	C
		12	L	L	C
SGTRI5	15	13	L	L	B
		14	L	L	C
		16	L	L	B
		17	L	L	B

Table 10.2b (continued)

Initiating Event	POS	Sequence	CDF	Early or Late	Containment
					Open - O Closed - C Bypassed - B
		20	L	L	B
SGTR15/SGRCT	15	2/4	L	E	C
		15/4	L	E	C

Table 10-2c

Core Damage Frequencies Due to "V-Events"

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Open or Closed O/C
LPI "V"	1		L	L	O
	2		L	L	O
	14		L	L	O
	15		L	L	O
RHR Suction/	1		L	L	O and C*
Discharge/	15		L	L	O and C*
Letdown "V"	2		L	L	O and C*
	3		L	L	O
	4		L	L	O
	12		L	L	O
	13		L	L	O
	14		L	L	O and C*

* In these cases LOCA occurs outside and inside containment.

** For "V-Events" "O" means Outside Containment.

Table 10-3a

**Results of Accident Sequence Quantification for
Loss of Offsite Power Initiating Event (Refueling Outage)**

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
1	5	L	L	C
	6	L	L	C
	8	L	L	C
	9	L	L	C
	10	L	L	C
	11	L	L	C
	12	L	L	C
2	5	L	L	C
	6	L	L	C
	8	L	L	C
	9	L	L	C
	10	L	L	C
	11	M	L	C
	12	M	L	C
3	9	M	E	C
4	9	M	E	O
5	9	M	E	O
6	5	L	E	O
	9	M	E	O
7	5	L	L	C
	9	H	L	C
8	5	M	L	O
9	5	L	L	C
	9	H	L	C
10	5	L	L	C
	9	M	L	C

Table 10-3a (Cont'd)

**Results of Accident Sequence Quantification for
Loss of Offsite Power Initiating Event (Refueling Outage)**

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
11	5	M	L	C
	9	L	L	C
12	9	M	L	C
13	9	L	L	C
14	5	L	L	C
	6	M	L	C
	8	L	L	C
	9	L	L	C
	10	L	L	C
	11	M	L	C
	12	M	L	C
15	5	L	L	C
	6	L	L	C
	8	L	L	C
	9	L	L	C
	10	L	L	C
	11	M	L	C
	12	M	L	C

Table 10-3b

**Results of Accident Sequence Quantification for
Station Blackout at Unit 1 (Refueling Outage)**

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
1	3	L	L	C
	5	L	L	C
	7	L	L	C
	9	L	L	C
	11	L	L	C
	13	L	E	C
	15	L	L	C
	17	L	E	C
	19	L	L	C
	21	L	L	C
	23	L	E	C
	25	L	L	C
	27	L	E	C
	29	L	L	C
	31	L	L	C
	33	L	L	C
	35	L	L	C
	36	L	E	C
	38	L	L	C
	39	L	E	C
	41	L	L	C
	43	L	L	C
	45	L	L	C
	47	L	L	C
	49	L	L	C
	51	L	L	C
	53	L	L	C
	55	L	L	C

Table 10-3b (continued)

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
2	57	L	L	C
	59	L	L	C
	61	L	L	C
	63	L	L	C
	65	L	E	C
	3	M	L	C
	5	M	L	C
	7	L	L	C
	9	L	L	C
	11	L	L	C
	13	L	E	C
	15	L	L	C
	17	L	E	C
	19	L	L	C
	21	L	L	C
	23	L	E	C
	25	L	L	C
	27	L	E	C
	29	L	L	C
	31	L	L	C
2	33	L	L	C
	35	L	L	C
	36	L	E	C
	38	L	L	C
	39	L	E	C
	41	L	L	C
	43	L	L	C
	45	L	L	C
	47	L	L	C
	2	L	L	C

Table 10-3b (continued)

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
	49	L	L	C
	51	L	E	C
	53	L	L	C
	55	L	L	C
	57	L	L	C
	59	L	L	C
	61	L	E	C
	63	L	L	C
	65	L	E	C
3	2	L	L	C
	4	L	L	C
	6	L	L	C
	8	L	E	C
4	2	M	L	O
	4	M	L	O
	6	M	L	O
	8	M	E	O
5	2	M	L	O
	4	M	L	O
	6	M	L	O
	8	M	E	O
6	3	M	L	O
	5	L	E	O
	9	H	E	O
7	2	M	L	O
	4	M	L	O
	5	L	L	O
8	2	M	L	O
	4	M	L	O

Table 10-3b (continued)

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
9	5	M	L	O
	2	M	L	O
	4	M	L	O
	5	L	L	O
10	3	M	L	O
	5	L	L	O
	9	M	L	O
11	4	L	L	O
12	4	M	L	C
13	4	L	L	C
14	4	M	L	C
	6	L	L	C
	8	L	E	C
15	4	M	L	C
	6	L	L	C
	8	L	E	C
	8	L	E	C

Table 10-3c

Results of Accident Sequence Quantification for
Station Blackout at Both Units (Refueling Outage)

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
1	3	L	L	C
	5	L	L	C
	7	L	E	C
	9	L	L	C
	11	L	L	C
	13	L	L	C
	15	L	E	C
	17	L	L	C
	19	L	E	C
2	3	L	L	C
	5	L	L	C
	7	L	E	C
	9	L	L	C
	11	L	L	C
	13	L	L	C
	15	L	E	C
	17	L	L	C
	19	L	E	C
3	2	L	L	O
	4	L	L	O
4	2	M	L	O
	4	M	L	O
5	2	L	L	O
	4	L	L	O
6	4	L	E	O
	7	M	E	O
7	2	L	L	O
7	4	M	L	O

Table 10-3c (continued)

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
	5	L	L	O
8	2'	M	L	Q
	4	L	L	O
	5	L	L	O
9	2	L	L	O
	4	L	L	O
	5	L	L	O
10	4	L	L	O
	7	L	L	O
11	3	L	L	O
12	3	L	L	O
13	3	L	L	C
14	3	M	L	C
	5	L	E	C
	7	L	E	C
15	3	L	L	C
	5	L	E	C
	7	L	E	C

Table 10-3d

**Results of Accident Sequence Quantification for
Loss of Offsite Power Initiating Event (Drained Maintenance Outage)**

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
1	5	L	L	C
	6	L	L	C
	8	L	L	C
	9	L	L	C
	10	L	L	C
	11	L	L	C
	12	L	L	C
2	5	L	L	C
	6	L	L	C
	8	L	L	C
	9	L	L	C
	10	L	L	C
	11	L	L	C
	12	L	L	C
3	9	M	E	C
4	9	H	E	O
5	9	M	E	O
6	9	M	E	O
11	5	L	L	C
	9	L	L	C
12	9	M	L	C
13	9	L	L	C
14	5	L	L	C
	6	L	L	C
	8	L	L	C
	9	L	L	C
	10	L	L	C
	11	M	L	C

Table 10.3d (continued)

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
14	12	M	L	C
15	5	L	L	C
	6	L	L	C
	8	L	L	C
	9	L	L	C
	10	L	L	C
	11	M	L	C
	12	L	L	C

Table 10-3a

**Results of Accident Sequence Quantification for
Station Blackout at Unit 1 (Drained Maintenance Outage)**

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
1	3	L	L	C
	5	L	L	C
	7	L	L	C
	9	L	L	C
	11	L	L	C
	13	L	E	C
	15	L	L	C
	17	L	E	C
	19	L	L	C
	21	L	L	C
	23	L	E	C
	25	L	L	C
	27	L	E	C
	29	L	L	C
	31	L	L	C
	33	L	L	C
	35	L	L	C
	36	L	E	C
	38	L	L	C
	39	L	E	C
	41	L	L	C
	43	L	L	C
	45	L	L	C
	47	L	L	C
	49	L	L	C
	51	L	L	C
	53	L	L	C
	55	L	L	C

Table 10-3e (continued)

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
1	57	L	L	C
	59	L	L	C
	61	L	L	C
	63	L	L	C
	65	L	E	C
2	3	M	L	C
	5	M	L	C
	7	L	L	C
	9	L	L	C
	11	L	L	C
	13	L	E	C
	15	L	L	C
	17	L	E	C
	19	L	L	C
	21	L	L	C
	23	L	E	C
	25	L	L	C
	27	L	E	C
	29	L	L	C
	31	L	L	C
	33	L	L	C
	35	L	L	C
	36	L	E	C
	38	L	L	C
	39	L	E	C
	41	L	L	C
	43	L	L	C
	45	L	L	C
	47	L	L	C

Table 10-3e (continued)

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
	49	L	L	C
	51	L	E	C
	53	L	L	C
	55	L	L	C
	57	L	L	C
	59	L	L	C
	61	L	E	C
	63	L	L	C
	65	L	E	C
3	2	M	L	C
	4	L	L	C
	6	L	L	C
	8	L	E	C
4	2	M	L	O
	4	M	L	O
	6	M	L	O
	8	M	E	O
5	2	M	L	O
	4	M	L	O
	6	M	L	O
	8	L	E	O
6	3	M	L	O
	5	L	E	O
	9	M	E	O
11	4	L	L	O
12	4	M	L	O
13	4	L	L	C
14	4	M	L	C
	6	M	L	C

Table 10-3e (continued)

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
	8	M	E	C
15	4	L	L	C
	6	L	L	C
	8	L	E	C

Table 10.3 F

**Results of Accident Sequence Quantification for
Station Blackout at Both Units (Drained Maintenance Outage)**

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
1	3	L	L	C
	5	L	L	C
	7	L	E	C
	9	L	L	C
	11	L	L	C
	13	L	L	C
	15	L	e	C
	17	L	L	C
	19	L	E	C
2	3	L	L	C
	5	L	L	C
	7	L	E	C
	9	L	L	C
	11	L	L	C
	13	L	L	C
	15	L	E	C
	17	L	L	C
	19	L	E	C
3	2	L	L	O
	4	L	L	O
4	2	M	L	O
	4	L	L	O
5	2	L	L	O
	4	L	L	O
6	4	L	E	O
	7	M	E	O
11	3	L	L	O
12	3	M	L	O

Table 10.3 F (continued)

POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
13	3	L	L	C
14	3	M	L	C
	5	L	E	C
	7	L	E	C
15	3	L	L	C
	5	L	E	C
	7	L	E	C

Table 10-4a

**Results of Accident Sequence Quantification for
Loss of Residual Heat Removal (RHR)
(Refueling Outage)**

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
RHR4	3	8	L	L	C
RHRT1	3	8	L	L	C
RHR6B	3	8	L	L	C
RHR4	84	8	H	E	O
RHRT1	4	8	M	E	O
RHR6B	4	8	M	E	O
RHR4	5	8	M	E	O
RHRT1	5	8	M	E	O
RHR6B	5	8	M	E	O
RHR2A	6	12	L	E	O
RHR2A	6	15	L	E	O
RHR2A	6	17	M	E	O
RHR2A	6	19	H	E	O
RHR2B	6	12	L	E	O
RHR2B	6	15	L	E	O
RHR2B	6	17	M	E	O
RHR2B	6	19	M	E	O
RHR4	6	5	M	E	O
RHR4	6	8	H	E	O
RHRT1	6	5	L	E	O
RHRT1	6	8	M	E	O
RHR6B	6	5	L	E	O
RHR6B	6	8	M	E	O
RHR4	7	5	M	L	C
RHR4	7	8	H	L	C

Table 10-4a (continued)

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
RHRT1	7	5	L	L	C
RHRT1	7	8	M	L	C
RHR6B	7	5	M	L	C
RHR6B	7	8	M	L	C
RHR4	9	5	M	L	C
RHR4	9	8	H	L	C
RHRT1	9	5	L	L	C
RHRT1	9	8	H	L	C
RHR6B	9	5	L	L	C
RHR6B	9	8	M	L	C
RHR2A	10	12	L	L	C
RHR2A	10	15	M	L	C
RHR2A	10	17	M	L	C
RHR2A	10	19	H	L	C
RHR2B	10	12	L	E	C
RHR2B	10	15	K	L	C
RHR2B	10	17	L	L	C
RHR2B	10	19	M	L	C
RHR4	10	5	L	L	C
RHR4	10	8	H	L	C
RHRT1	10	5	L	L	C
RHRT1	10	8	H	L	C
RHR6B	10	5	L	L	C
RHR6B	10	8	M	L	C
RHR4	11	8	M	L	C
RHRT1	11	8	M	L	C

Table 10-4a (continued)

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
RHR6B	11	5	L	L	C
RHR6B	11	8	M	L	C
RHR4	12	8	H	L	C
RHRT1	12	8	H	L	C
RHR6B	12	8	M	L	C
RHR4	13	8	M	L	C
RHRT1	13	8	L	L	C
RHR6B	13	8	M	L	C

Table 10-4b

**Results of Accident Sequence Quantification for
Loss of Residual Heat Removal (RHR)
(Drained Maintenance)**

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
RHR4	3	8	M	L	C
RHRT1	3	8	L	L	C
RHR6B	3	8	L	L	C
RHR4	4	8	H	E	O
RHRT1	4	8	M	E	O
RHR6B	4	8	M	E	O
RHR4	5	8	M	E	O
RHRT1	5	8	L	E	O
RHR6B	5	8	M	E	O
RHR2A	6	15	M	E	O
RHR2A	6	17	M	E	O
RHR2A	6	19	H	E	O
RHR2B	6	15	L	E	O
RHR2B	6	17	M	E	O
RHR2B	6	19	M	E	O
RHR4	6	8	M	E	O
RHRT1	6	8	M	E	O
RHR6B	6	8	M	E	O
RHR4	11	8	M	O	O
RHRT1	11	8	M	E	O
RHR6B	11	8	M	O	O
RHR4	12	8	M	O	O
RHRT1	12	8	M	O	O
RHR6B	12	8	M	E	O
RHR4	13	8	M	L	C

Table 10-4b (continued)

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
RHRT1	13	8	L	L	C
RHR6B	13	8	L	L	C

Table 10-4c

**Results of Accident Sequence Quantification for
Loss of Residual Heat Removal (RHR)
(Non-Drained Maintenance)**

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
RHR4	3	8	L	L	C
RHRT1	3	8	L	L	C
RHR6B	3	8	L	L	C
RHR4	4	8	H	E	O
RHRT1	4	8	M	E	O
RHR6B	4	8	M	E	O

Table 10-5**Reactivity Accidents during Shutdown**

Initiating Event	POS	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
Blowdown of accumulator during shutdown	5	L	E	O
	6	L	E	O
	7	L	E	O
	8	L	E	O
	9	L	E	O
	10	L	E	O
	11	L	E	O
	12	L	E	O
Leaking accumulator isolation MOVs	8	L	L	C
	9	L	L	C
	10	L	L	C
	11	L	L	C
	12	L	L	C
Large LOCA with diluted accumulator water	1	L	E	C
	2	L	E	C
	14	L	E	C
	15	L	E	C
Inadvertent safety injection during shutdown	1	L	E	C
	2	L	E	C
	13	L	E	C
	14	L	E	C
	15	L	E	C
Leaking of RWST water due to improper valve alignment	9	L	L	C
	10	L	L	C
	11	L	L	C

Table 10.5 (continued)

Initiating Event	POS	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
Large LOCA with diluted RWST water	1	L	E	C
	2	L	E	C
	3	L	E	C
	4	L	E	C
	12	L	E	C
	13	L	E	C
	14	L	E	C
	15	L	E	C
Boron dilution during mid-loop operations	6	H	E	O
	10	M	E	O
Startup of RCP after improper boron dilution	14	H	E	C
Inadvertent criticality due to misloading of fuel assemblies	8	M	E	O

Table 10-6

Special Initiators

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Open or Closed O/C
Thimble Tube Seal	7		H	E	O
LTOP	3		L	E	C
	13		L	E	C
LTOP	4		L	E	C
	12		L	E	C
PTS	1		L	E	C
	15		M	E	C

Table 10-7

Results of Accident Sequence Quantification for ATWS

Initiating Event	POS	Sequence	CDF	Early or Late (E/L)	Containment Closed or Open (C/O)
ATWS	1	3	L	E	C
		5	L	E	C
		6	L	E	c
		7	L	E	C
		9	L	E	C
		11	L	E	C
		12	L	E	C
	15	2	L	E	C
		3	L	E	C
		5	L	E	C
		6	L	E	C
		7	L	E	C
		9	L	E	C
		11	L	E	C
		12	L	E	C

Table 10-8
Accident Categories

CATEGORY NUMBER	INITIATING EVENT CATEGORY	NUMBER OF SEQUENCES
1	Transient with Loss of MFW, No Challenge to RPS (T2)	14
2	Transient with Loss of MFW, Challenge to RPS (T2a)	14
3	Transient with MFW Available, No Challenge to RPS (T3)	14
4	Transient with MFW Available, Challenge to RPS (T3a)	14
5	Loss of Coolant Accidents (LOCA)	234
6	Steam Generator Tube Rupture (SGTR)	143
7	Interfacing LOCA ("V-Events")	12
8	Loss of Off-Site Power-Refueling	44
9	Station Blackout at Unit 1-Refueling	102
10	Station Blackout at Both Units-Refueling	46
11	Loss of Offsite Power - Drained Maint.	35
12	SBO at Unit 1 - Drained Maint.	90
13	SBO at Both Units - Drained Maint.	35
14	Loss of Residual Heat Removal (RHR) During Refueling Outage	60
15	Loss of Residual Heat Removal (RHR) During Drained Maintenance	27
16	Loss of Residual Heat Removal (RHR) During Non-Drained Maintenance	6
17	Reactivity Accidents	37
18	Special Initiators	6
19	Anticipated Transient Without Scram (ATWS)	14
TOTAL	All Sequences	947

Table 10-9

PDS Consequence Severity Classification

PDS GROUP	CONTAINMENT CONDITION	UPPER HEAD REMOVED (POS 7,8,9)	PDS TIMING	CONSEQUENCE SEVERITY CLASS
1	OPEN	YES	EARLY	HIGH
2	OPEN	NO	EARLY	HIGH
3	OPEN	YES	LATE	HIGH
4	OPEN	NO	LATE	HIGH
5	CLOSED	YES/ NO	EARLY/ LATE	LOW

Table 10.10

Likelihood of Accident Sequences

PDS GROUP	SEVERITY CLASS	ACCIDENT LIKELIHOOD GROUP (Number of Sequences)				ACCIDENT LIKELIHOOD GROUP ("Frequency", %)		
		High	Med.	Low	Total	High	Med.	Low
1	HIGH	1	1	0	2	4.5	0.6	0
2	HIGH	10	48	35	93	45.5	29.1	4.6
3	HIGH	0	11	10	21	0	6.7	1.3
4	HIGH	0	28	120	148	0	17.0	15.8
5	LOW	11	77	595	683	50.0	46.6	78.3
TOTAL		22	165	760	947	100	100	100

Appendix I.4 Support Thermal Hydraulic Analysis of the Phase 1 Study

In this appendix, the thermal hydraulic analysis performed in the phase 1 study is documented. It is extracted from the phase 1 report.

5.2 Thermal Hydraulic Calculations

The thermal hydraulic calculations supporting the event trees quantification were performed in a "back-of-envelope" manner by using a number of simplifying assumptions as described below. More detail calculations will be required to confirm the results obtained by using supposedly conservative assumptions.

The main purpose of the thermal hydraulic calculations is to estimate the time of the plant reaching a potential core damage condition. This time is further used in the event tree quantification calculations. In this report, the core damage (CD) condition is declared when the collapsed water level drops to the top of active fuel (TAF). Another CD condition is assumed at the time of LB LOCA (rupture in RHR system).

Figures 5.1 through 5.15 show graphically the thermal hydraulic conditions corresponding to the fifteen POSs analyzed in the report. The schematics also show the key elevations of the loops and major components.

5.2.1 Assumptions in Thermal Hydraulic Calculations

The assumptions listed below have been used for the thermal hydraulic estimates of the time available to the operator to prevent the core damage in the Loss of Power and Station Blackout Accident scenarios.

1. The transient phenomena have been ignored and the quasi-steady state conditions were assumed in the heat-up and boil-off calculations.
2. The decay heat power was conservatively assumed constant throughout each particular plant operational state (POS). When the starting time of a POS was not readily available, a conservative assumption was made: the earliest possible time was selected for the decay power estimation. The only exceptions were POSs 1 & 2 where the steam generator boil-off was estimated using either the decay power level corresponding to the time of scram or to several hours after scram depending on the particular sequence.
3. As mentioned above, the time of LB LOCA (rupture in the RHR system) was identified as the CD time in relevant sequences.
4. Residual Heat Removal (RHR) system is not isolated during the loss of off-site power accidents. Thus, RHR is the vulnerable system during primary pressurization.
5. Natural circulation flow in the primary system always develops when the secondary side is available and the water level in the vessel is above the cold leg. Heat transfer between the primary and secondary is assumed sufficient to maintain the primary and secondary at equilibrium; no boiling in primary develops at these conditions.
6. Seal break LOCA develops after 1.5 hours of the seals exposure to the elevated temperature and pressure. The flow rate is 750 gpm at the Pressurizer Safety Valve setpoint pressure (2485 psig).
7. RHR safety valve setpoint is 600 psig; the valve's capacity is 750 gpm of liquid at this pressure.
8. PORV setpoint in POSs 3 through 13 is 365 psig. PORV capacity for liquid discharge is assumed to be 700 gpm at 365 psig.
9. In SBO accidents, the battery depletion time is four hours. Following this time, auxiliary feedwater (AFW) to steam generators stops, control of PORVs is lost (PORVs fail closed), and charging of primary (when in effect) is terminated.

5.2.2 Loss of Off-Site Power and Station Blackout

This section provides background for the timing data used in the event tree quantification. The information is presented in Table 5.2.2-1. The first column indicates POS, the second column contains descriptions of the processes limiting the time available to a plant operator before a core damage (CD) condition develops. The third column presents the time before CD in hours (duration of the critical processes).

Table 5.2.2-1 Summary of Loss of Offsite Power and Station Blackout Calculations

POS	PROCESS	TIME TO CD
1	Timing data for quantification calculations for this state were taken from NUREG-1150.	
2	Timing data for quantification calculations for this state were taken from NUREG-1150.	
3	<p>The scenario at this state assumes a failure to depressurize the secondary side (station blackout). The primary and secondary will heat up together from 174 C to 214 C. Total liquid mass (primary and secondary) is 192,000 + 225,000 = 417,000 kg. A conservative value for the liquid mass in the secondary of the steam generators (225,000 kg) has been chosen for the estimates. Decay power is 20 MW. The heat-up time to boiling in primary is 1 hour.</p> <p>Thermal expansion of liquid in the primary: available steam volume in the pressurizer is 17 m³. The approximate expansion rate is 6.7 10⁻³ m³/s (104 gpm). With this expansion rate, the primary system will be filled with single-phase liquid in approximately 0.7 hours. This is the limiting condition based on the assumption that the RHR system will develop a break in the course of the "solid" system pressurization.</p>	0.7 hours
4	Conditions at this state are similar to the conditions at POS 3. Initial liquid temperature is 93 C. Primary and secondary heat-up to the primary saturation temperature takes longer. However, thermal expansion leading to a solid RCS condition (primary filled with single-phase liquid) is the limiting condition.	0.7 hours
5	Initial condition: primary system is full of liquid; liquid temperature is 60 C; pressure is 0 psig. Subcooled expansion begins after loss of RHR and PORVs relieving pressure to close to atmospheric. The limiting process is the primary and secondary heat-up to saturation temperature and boiling in the primary.	1.0 hour

Table 5.2.2-1 Summary of Loss of Offsite Power and Station Blackout Calculations (Continued)

POS	PROCESS	TIME TO CD
6	<p>POS 6 is the mid-loop plant operation state. The limiting processes are the primary liquid heat-up (0.17 hours) and boil-off (1.37 hours) at atmospheric pressure. Liquid mass above the core at mid-loop operation is 27,120 kg (WCAP-11916, July 1988, page 3-55). Initial temperature is 32 C. Decay power is 12.4 MW. Figure 5.16 provides the boil-off timing as a function of decay power.</p> <p>Successful reflux cooling can prevent CD condition. Calculations show that the reflux mode can establish at RCS pressure below the failure limit of the tygon tube.</p>	1.5 hours
7	Conditions at this state are similar to conditions at POS 6; decay power is 8.7 MW.	2.2 hours
8	Liquid mass in the refueling bay is 850,000 kg (220,000 gal) at 32 C. Decay power is 7.4 MW. The limiting processes are the liquid heat-up (9 hours) and boil-off at atmospheric pressure (72 hours). This is a success condition provided the upper support plate is removed and the liquid from the refueling bay has unrestricted access to the core region (see [5.3], Section 3.2 for discussion of this situation).	81.0 hours
9	Conditions at this state are similar to conditions at POS 7. Decay power is 3.5 MW. Initial liquid temperature is 32 C. The limiting processes are the primary liquid heat-up and boil-off at atmospheric pressure.	5.6 hours
10	Conditions at this state are similar to conditions at POS 6 (mid-loop operation). Decay power is 3.3 MW.	5.6 hours
11	The governing processes in this state are heat-up of the primary and secondary liquid to saturation at atmospheric pressure (1.7 hours) and boil-off through the ruptured tygon tube (14 hours). Decay power is 3.15 MW.	15.7 hours

Table 5.2.2-1 Summary of Loss of Offsite Power and Station Blackout Calculations (Continued)

POS	PROCESS	TIME TO CD
12	1. After refueling. Decay power is 2.6 MW. Two PORVs are capable to relieve pressure. Heat up of the primary and secondary liquid from 93 C to 214 C (7 hours) and boil-off of the primary (25 hours) lead to success on time (more than 24 hours).	32.0 hours
	2. After drained maintenance. Decay power is 10.3 MW (54 hours after shutdown).	8.1 hours
13	1. After refueling. Similar to POS 12. Decay power is 2.4 MW. Initial temperature is higher (177 C).	29.3 hours
	2. After drained maintenance. Decay power is 9.2 MW (78 hours after shutdown).	7.6 hours
14	Initial conditions: nominal temperature and pressure; RHR is isolated; steam generators are used to damp decay heat. The time for high pressure secondary boil-off is 59 hours. However, a seal break develops, so that the limiting process is the primary boil-off (7 hours).	8.0 hours
15	Conditions at this state are similar to conditions at POS 14.	8.0 hours

5.2.3 Reflux Cooling

The reflux cooling [5.3, 5.4] is recognized as an important cooling mode which can establish following a loss of RHR while the plant is in a mid-loop condition: loops are drained to the level of the hot leg mid-plane. At this condition the primary system is sealed and maintained at atmospheric pressure. In this report, it is assumed that the reflux cooling is effective as a means to provide cooling to the core during LOP and SBO accidents.

5.2.4 Gravity Feed from RWST

Gravity feed cooling is considered an option for the POSs starting with the primary at atmospheric pressure and the vessel upper head in place (POSs 5-7 and 9-11). Since the elevation of the liquid discharge on the top of pressurizer is approximately 61.5 ft, the flow through the core driven by gravity head can be maintained only until the RWST water level drops by about 10.5 ft, to the elevation of 46 ft.

The main objective of the gravity feed mode is to remove decay heat while preventing boiling in the core and RCS pressurization. Success in achieving this objective depends on a combination of three system parameters: decay power, core flow rate, and RWST liquid temperature.

Since the geometry and flow resistances of the RWST - vessel path are not presently available, one can only determine the minimum core flow rate required to prevent the core boiling in the most challenging state (the state with the maximum decay power), POS 5 with decay power of 20 MW. Assuming the RWST liquid temperature of 45 F (7 C), the minimum core flow rate at atmospheric pressure is equal to 51 kg/s [20 10⁶ W / (4200 J/kg Deg * 93 Deg)].

Note the estimated value of 0.8 ft for the driving head required to produce a flow rate of about 8.4 kg/s in the Vogtle event [5.4, page 16]. Leaving aside the question of attainability of the core flow rate of 51 kg/s, the time of the RWST depletion to the level of 61.6 ft is about 347500 kg / 51 kg/s = 1.9 hours. Therefore, the time before boiling in the primary is 1.9 hours unless the RWST liquid inventory is replenished with borated water (replenishing the tank with unborated water will raise a concern of recriticality).

Once the RWST water level reaches the 61.5 ft elevation, the flow through the core will drop (gravity head due to density differences still remains), and boiling begin. Note that the starting condition of the boil-off transient is the RCS filled with single-phase water. Liquid or two-phase mixture will be expelled through the PORVs until the liquid inventories of the vessel's upper plenum, cold and hot legs, and pressurizer are depleted and steam reaches the PORV. At this point the RCS pressure will drop, the RWST check valve re-open and gravity feed re-establish.

Degradation of heat transfer in the core (post-CHF regime) is likely to develop at the low flow conditions. The RCS and core conditions after this moment have to be predicted using a time domain thermal hydraulics code.

An alternative way of the RWST gravity feed is by controlling the flow from RWST to vessel in such a way that the water level in the vessel remains at the mid-loop elevation while the water boils-off through the pressurizer. In this case the POS 5 time for heat-up and boil-off at atmospheric pressure for the RWST liquid (387,100 gal) is 56.3 (8.2 + 48.1) hours.

5.2.5 Thimble Tube Seal Failure

The thimble tube seals design pressure is 40 psig. The total flow area available to liquid flow out of vessel in case of a failure of all 50 tube seals is approximately 2.3 in² (close to the area of one PORV). Assuming that this condition develops during a core boil-off situation with the primary pressure around the PORV

setpoint of 365 psig, the coolant loss flow rate should be approximately 700 gpm. If the primary is subsequently depressurized, the thimble tube seal flow will continue until the water level in the vessel drops to the elevation of the thimble tube which is located above the cold leg elevation.

5.2.6 Pressure Relief Capability

Thermal hydraulic analysis of the event trees involving loss of RHR system raised a concern about the capability of the existing primary pressure relief system to prevent over-pressurization at certain plant operating states.

The states of concern are when RHR is operating, PORVs setpoints are lowered to 365 psig, and the secondary side is either not available or can not be depressurized. Since the design pressure of the RHR system is 600 psig, it presents the weakest point in the primary system pressure boundary. The important features of these states are:

- a) Primary safety valves are not available since their setpoints are at 2485 psig: RHR will rupture prior to the valves activation;
- b) Secondary safety (atmospheric dump) valves are not available. Activation of these valves requires heat up and pressurization of the secondary (following primary heat up) to 1085 psig (this pressure is significantly higher than the RHR design pressure);
- c) A condition can develop when boiling in primary is combined with a single (or two-phase) discharge through PORVs.

At the plant states indicated above, the pressure relief capability is provided by two PORVs and the RHR safety valve. The RHR safety valve set point is 600 psig and its capacity is 750 gpm of liquid at this pressure.

The effective flow area of a PORV is about 2.243 in² ($1.45 \cdot 10^{-3}$ m²) and the single phase discharge coefficient (liquid) is 0.461. According to Virginia Power estimates, the PORV capacity is approximately 700 gpm of liquid at 365 psig (900 gpm at 600 psig).

Based on the this information, the total pressure relief capability is $2 \cdot 900 \text{ gpm} + 750 \text{ gpm} = 2550 \text{ gpm}$ of liquid. At the same time, the volumetric vapor generation rate at constant pressure is about $10^6 / (1.7 \cdot 10^6 \text{ J/kg } 20.8 \text{ kg/m}^3) = 436 \text{ gpm}$ for 1 MW of decay power. In actuality, as the system pressure increases, the vapor generation rate drops as a result; therefore, this estimate of volumetric vapor generation rate is conservative.

The estimate above indicates that the primary pressure relief capability may not be sufficient at the conditions described above when the decay power exceeds $2550 / 436 = 5.8 \text{ MW}$ which corresponds to approximately 17 days after shutdown. These conditions can develop for the power operating states ranging from 3 through 8.

It has to be emphasized that the conservative estimates provided in this section should be followed by a detail thermal-hydraulic calculation with the RCS loop model including core, pressurizer, and PORVs as critical components.

5.7.2 References

- 5.1** Andreychek, T.S., et al., "Loss of RHRS Cooling while the RCS is Partially Filled," Revision 0, July 1988, Westinghouse Electric Corporation, WCAP-11916.
- 5.2** NUREG-1410.
- 5.3** Ward, L.W. and Arcieri, W.C. "Evaluation of the Loss of Residual Heat Removal System in Pressurized Water Reactors," INEL, EG&G Idaho, Inc., February 1991.
- 5.4** Fletcher, C.D., McHugh, P.R., Naff, S.A., and G.W. Johnsen, "Thermal-Hydraulic Processes Involved in Loss of Residual Heat Removal During Reduced Inventory Operation," Technical Report, EGG-EAST-9337, February 1991, Revision 1.

Figure S-1 State 1 (0 hours), Low Power and Shutdown

Figure 5-2 State 2 (0 hours), Cooldown with SG

Figure 5-3 State 3 (6 hours), Cooldown with RHR (initial phase)

I-78

Figure 5-4 State 4 (6 hours), Cooldown with RIIR

6/L-1

Figure 5-5 State 5 (6 hours), Draining RCS to MHA-loop

Figure 5-6 State 6 (24 hours) Mid-loop Operation Loops are Isolated

Figure 5-7 State 7 (4 days), Filling for Refuel

Figure 5-8 State 8 (6 days), Refueling

Figure 5-9 State 9 (27 days), Draining to Mid-loop after Refueling

Figure 5-10 State 10 (20 days) Mid-loop Operation

Figure 5-11 State 11 (34 days) Refueling RCS

Figure 5-12 State 13 BCS Heat-up Shutdown and Draw Bubble

Figure 5-13 State 13 (39 days) RCS Heat-up

Figure 5-14 State 14 (90 days) RCS Heat-up with SGs

Figure 5-15 State 15 (30 days) Start-up and Low Power Operation

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Figure 5-16 Boil-Off from Mid-loop Conditions

Appendix I.5 Human Reliability Analysis Approach Used in Phase 1

In this appendix, a chapter, Chapter 7, of the phase 1 report is provided. It documents the human reliability analysis approach used in the phase 1 study.

7. HUMAN RELIABILITY ANALYSIS

7.1 Introduction

This chapter presents the screening human reliability analysis (HRA) methodology used in the initial quantification of human error probabilities (HEPs) for the human actions identified as part of this Surry Unit 1 Low Power and Shutdown PRA.

7.2 Methodology

The HRA used for the initial HEP quantification provides a screening analysis which attempts to assign conservative but not unrealistically high failure probability values (i.e., HEPs) to the low power and shutdown activities associated human actions. The method developed is a workable adaptation of the ASEP (Accident Sequence Evaluation Program) HRA Procedure⁽¹⁾ which has been modified to be usable as a screening HRA in support of the first phase of this PRA.

The screening HRA methodology adopts the more conservative post-accident screening HRA methodology concept of the ASEP HRA Procedure, Table 7-1 and Figure 7-1 to all human actions to be quantified in this PRA while modifying the Procedure (see Section 7.3 for explanation) to the following simplified and workable initial screening HRA procedure.

To start with each human action has an associated HEP within the range of 0 to 1.0 which is composed of two basic parts, HEP_d and HEP_a , such that

$$HEP = HEP_d + HEP_a \leq 1.0, \text{ where}$$

HEP_d = HEP of the control room personnel correctly diagnosing the need for the required action(s) in t_d , the time available to diagnose the action(s), and

HEP_a = HEP of correctly accomplishing the required action(s) in t_a , the time available to perform the action(s), while

$$t_m = t_d + t_a, \text{ where}$$

t_m = estimated time available to correctly diagnose and accomplish the required action(s) before core damage is expected

To determine the t_a and its associated HEP_a , use the Table 7-1 simplified estimates which are based on action(s) being either simple or complex and expected to be performed in either the control room or not.

To determine the t_d and its associated HEP_d , first an estimate of t_m is required. Then using the simplified estimate of t_a , an estimate of t_d is established (since $t_d = t_m - t_a$). The associated HEP_d is determined using Table 7-2, which is a simplified version of the ASEP HRA Procedure Figure 7-1, a time-reliability correlation entitled, "Initial-Screening Model of Estimated HEPs --- for Diagnosis Within Time T of One Abnormal Event by Control Room Personnel."

To calculate the screening HEP, add the simplified estimate of performing each required action(s), HEP_a to the screening estimate of diagnosing the associated action(s), HEP_d associated with the next shorter t_d found in the simplified listing in Table 7-2.

Finally, for those human actions which were also quantified for power operation of Surry in support of NUREG-1150⁽²⁾, use the higher value of HEP either as calculated by this methodology or as obtained from NUREG-1150 for Surry Unit 1.

Table 7-1
Simplified Estimate of Time and HEP Quantification
for Performing Required Actions

<u>Required Action(s)</u>	<u>t_a</u>	<u>HEP_a</u>
Simple*, in Control Room**(CR)	5 min.	0.0001
Complex, in CR	10 min.	0.001
Simple, outside CR	20 min.	0.005
Complex, outside CR	≥30 min.	≥0.01

* Simple - at most a few uncomplicated steps; otherwise assume Complex
 **CR - all steps in CR; if not or not sure, assume outside CR

Table 7-2
Simplified Screening HEP Quantification for
Diagnosing the Need to Perform Required Action

<u>t_a</u>	<u>HEP_a</u>	<u>t_a</u>	<u>HEP_a</u>
<2 min.	1.0	30 min.	0.001
2 min.	0.5	40 min.	0.0004
5 min.	0.2	50 min.	0.0002
10 min.	0.1	≥60 min.	0.0001
20 min.	0.01		

7.3 Methodology Adaptation

The ASEP HRA Procedure (NUREG/CR-4772⁽¹⁾) presents a shortened version of the procedures, models, and data for HRA which are presented in the Handbook of Human Reliability Analysis (NUREG/CR-1278⁽²⁾) and is divided into pre-accident and post-accident task related procedures. It is further divided into procedures for screening HRAs and nominal HRAs. The screening analysis uses deliberately conservative estimates of HEPs while the nominal analysis uses more realistic values. A brief overview of the pre-accident and post-accident screening HRA follows.

The Pre-Accident Screening HRA emphasizes restoration errors which if performed incorrectly could result in the unavailability of necessary systems or components in a nuclear power plant to respond appropriately to an accident and is based on the use of a 0.03 basic HEP (BHEP) with no downward adjustment permitted by the procedure. Nevertheless, the BHEP may be multiplied by one or more 0.1 (non) recovery factors (RFs). The details of the ASEP screening HRA procedural steps for pre-accident tasks are contained in Tables 4-1 through 4-3 of the ASEP HRA Procedure.

The Post-Accident Screening HRA uses the HRA Handbook⁽²⁾ screening diagnosis HEPs and response times. For estimates of screening HEPs for post-diagnosis actions, an HEP of 1.0 is assessed such critical actions as those performed outside the control room or from the control room for which there are no written procedures. Also conservative assumptions are made about response times in the control room, and a 0.05 generic HEP is assessed for all critical post-diagnosis tasks except for memorized post-diagnosis immediate emergency actions constituting skill-based behavior, in which case a generic HEP of 0.01 is assigned. The details of the ASEP screening HRA procedural steps for post-accident tasks are contained in Tables 7-1 through 7-3 of the ASEP HRA Procedure.

Unfortunately, the rigorous application of the ASEP screening HRA procedural steps in support of this Surry Unit 1 Low Power and Shutdown PRA would have provided a conservative but unworkable method. In particular, to assume that there is no chance of performing any critical activity from outside the control room or any critical skill-based or rule based post-accident, post diagnosis actions inside the control room which are not (unambiguously) described in written procedures cannot be applied realistically to a low power and shutdown PRA like this one. The operation of Surry Unit 1 during low power ($\leq 15\%$) and shutdown requires many more manual actions than while the unit is at power ($> 15\%$). If an abnormal or emergency condition develops while the unit is at $< 15\%$ or shutdown, a fair number of conservatively achievable manual actions are realistic options which may be performed from outside the control room or even inside the control room based on written procedures which may be clearly applicable during power operation and/or one or more low power and shutdown plant operational states (POSSs) but not unambiguous during all POSSs. Therefore, a screening HRA described in Section 7.2 was developed as a workable adaptation of the ASEP HRA Procedure to meet the special needs of a low power and shutdown PRA such as this one which the ASEP screening HRA will not handle.

As a final comment about the workability of the ASEP HRA Procedure for screening HRA, according to Chapter 4 of the Procedure, "neither screening procedure was used in the ASEP PRAs."

As noted previously in Section 7.2, the diagnostic part of the HEP quantification (namely, HEP_d) is estimated using a simplified version of the ASEP HRA Procedure Figure 7-1 time-reliability correlation based on an estimate of t_m and assigning a value for t_r . The simplified estimate of t_r is based in part on the ASEP HRA Procedure Table 7-1 which provides post-accident screening values for control room timing for the licensed operator(s) to take required action(s) using appropriate procedures to help assure the correctness of the action(s). The t_r estimates used for actions performed outside the control room are intended to account for time for the operator(s) to perform the action(s) as well as time for them to travel to the known location where the action is to be performed.

Finally, the required action part of the HEP quantification (namely, HEP_a) has been selected after qualitatively assessing the usefulness of all Surry Unit 1 operating procedures (OPs), abnormal procedures (APs), and emergency procedures (EPs) pertaining to one or more of Surry's low power and shutdown plant operational state (POS) related activities, talking to several Surry senior licensed operators and training personnel about the use of the OPs, APs and EPs, and observing actions performed by a licensed shift of operators on the Surry Simulator during a loss of RHR accident scenario. Note that the HEP_a estimates for the control room related required actions are based on the fact that the actions are to be performed by Surry licensed operators (with senior licensed operators overseeing the actions) who are trained periodically on the Surry Simulator (as well as experience on the plant) to anticipate and seek out corroborating feedback to help verify and validate their actions as being performed correctly. This training should also help them quickly recover from incorrect actions. Required actions performed outside the control room are considered to be indirectly (or in some cases directly) monitored by the licensed control room operator(s) since in the absence of control evacuation or loss of all control room instrumentation, all operational status information will be monitored in the control room.

7.4 References:

- (1) Swain, A.D. "Accident Sequence Evaluation Program Human Reliability Analysis Procedure," NUREG/CR-4772, Sandia National Laboratories, February 1987.
- (2) Bertuccio R.C. and Julius, J.A. "Analysis of Core Damage Frequency: Surry, Unit 1 Internal Events," NUREG/CR-4550, Volume 3, Rev. 1, Sandia National Laboratories, April 1990.
- (3) Swain, A.D. and Guttman, H.E. "Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications," NUREG/CR-1278, Sandia National Laboratories, August 1983.

Appendix I.6 Event Tree Analysis of the Phase 1 Study

This appendix is a copy of the chapter 6 of the Phase 1A study.

6. Event Tree Analysis

6.1 Event Tree Approach and Assumptions

Event trees model the responses of the plant and operators to the initiating events. For power operations, various procedures are available to assist the operators. In most cases, many possible scenarios were recognized and analyzed, and methods for mitigating the initiating events are specified in the procedures. The procedures provide good guidance in development of event trees. For shutdown conditions, very few procedures are written specifically for accidents that may happen. For the Surry plant, the only procedure that is specifically written for shutdown conditions is the loss of decay heat removal procedure AP-27. It provides guidance on restoring RHR, primary inventory makeup and alternate decay heat removal. For other initiators, the expected plant and operator responses are not clearly defined.

The event tree development in this study was performed in a talk through format. Typically, a team of 4 members met in the conference room. The members include engineers with background and training in the area of plant operations, PRA, and thermal hydraulics. The plant's initial conditions and the responses to the initiators are discussed. The related procedures and other documents were reviewed and then accident scenarios were developed. The process is very time consuming and agreement among the members is sometimes hard to reach. It was found very difficult to maintain consistency in the assumptions made, the level of detail of the model, and the timing of the scenarios. In general, it was found that more deterministic analysis is needed to better understand the behavior of the plant.

For those POSs that are similar to full power operations, e.g. POSs 1,2,14 and 15, the NUREG-1150 event tree are reviewed, modified if necessary, and used. For those POSs in which the RHR system is initially running, new event trees were developed from scratch.

At this phase of the study, no containment related question is included in the event trees, except those event trees that were taken from the NUREG-1150 study with these questions built in. No sequence cutset level recovery action are modelled.

6.2 General Discussion of Success Criteria

For those POSs similar to full power operations, the success criteria used in the NUREG-1150 study were reviewed and used. For other POSs, engineering judgment and assumptions were made. Some back-of-the-envelope type of calculations were done. Section 5.2 documents some of the deterministic calculations that were done. In some cases, it was felt that more deterministic analysis will be needed to confirm the assumptions made. One example is the use of gravity feed to provide long term core cooling. The other example is the issue of whether mixing will take place in the scenario of starting the reactor coolant pump after dilution in a loss of off site power transient.

As stated in the program scope, a mission time of 24 hours was assumed. It is assumed that after 24 hours into an accident, many alternative methods of maintaining the plant in a safe condition become available.

Some of the assumptions made in the event tree development are the following:

1. If the plant is initially in a cold shutdown condition, the turbine driven auxiliary feed water pump is assumed to be unavailable.
2. If the reactor coolant loops are isolated, no secondary heat removal is taken credit for.

3. If the plant is initially in a cold shutdown condition, it is assumed that the containment sump will have a high probability of being clogged with debris when an accident that requires taking suction from the containment sump occurs.
4. If the reactor coolant system is not vented with a big opening such as removal of the pressurizer safety valves and the secondary side of the steam generators is filled, it is assumed that either natural circulation on the primary side or reflux cooling will take place to remove decay heat from the primary coolant. This process is assumed to be effective in the long term as long as feeding and bleeding the secondary side of the steam generators can be maintained.

6.3 Transient Event Trees

The generic transient event categories considered in NUREG-1150 were reviewed for their applicability to different plant operational states (POSS) at low power and shutdown conditions. In most cases, the generic transient events were applicable to POSSs 1, 2, 14 and 15. The generic transients were grouped according to the event tree models that represent appropriate plant response to mitigate them. The transient event trees (T_2 and T_3) developed for the Surry plant in NUREG/CR-4550, Vol. 3 were reviewed for their applicability to mitigate transients at low power and shutdown conditions. The following is a summary of the modifications to these event trees that are needed for use in the low power and shutdown study.

(a) T_{2a} Event Tree Analysis

The T_{2a} Event Tree with no change in structure can be used for those transients that lead to loss of main feedwater in POSSs 1 and 15. In these POSSs (power <15%), some of the reactor trip signals are blocked at power level less than 10%, e.g., trips of reactor coolant pumps will not lead to a reactor trip. This is allowed because natural circulation in the reactor coolant loops is sufficient to maintain heat removal from the RCS if feedwater supply is available to the steam generators. However, a loss of main feedwater may lead to a reactor trip due to low-low steam generator level conditions that will generate a reactor trip signal. Before such a reactor trip occurs, the operators may have some time to manually shutdown the reactor or recover feedwater supply. Without main feedwater, the operator will have to manually shutdown the reactor, because the auxiliary feedwater system does not have the capability to remove all secondary side heat at 10% of full power. Therefore, reactor trip is a relevant question for such initiating events. The method of shutdown could be manual insertion of the control rods, manual trip of the reactor, and an automatic trip on low-low steam generator level.

(b) T_2 Event Tree

The T_2 event tree excluding the RPS failure event can be used for modelling loss of feedwater in POSSs 2 and 14. In these POSSs, the reactor is already shutdown with at least 1.77% shutdown margin. Therefore, reactor shutdown is not an applicable issue.

(c) T_3 Event Tree

The T_3 event tree excluding the RPS failure event and interchanging the order of MFW and AFW systems can be used for other transients that may occur in POSSs 1, 2, 14, and 15. Some of these transients start with a reactor trip in POS 1 or 15, e.g., spurious trip. Therefore, no RPS success or availability needs to be questioned. For other transients, reactor trip may not happen automatically because it is not needed. For example, loss of one reactor coolant pump in POS 1 does not lead to an automatic trip. The RCPs in the other two RCS loops should be sufficient to accommodate the additional decay heat removal load. However, the operator may choose to shutdown the reactor to perform the needed maintenance. In this case, it is assumed that the operator will successfully shutdown the reactor and the RPS is not an applicable failure event.

(d) T_{3a} Event Tree

The T_1 event tree with interchanging the order of MFW and AFW failure events can be used to model the progression of transient initiators which challenge RPS availability.

The event tree models used to quantify the accident sequences initiated by T_1 , T_2 , T_{3a} , or T_3 events are shown on Figures 6.3.1 and 6.3.2. The results of the quantification of accident sequences initiated by the transient events (T_2 or T_3) are summarized on Tables 10.1a, b, c, and d.

**Figure 6.3-1 Transient with Loss of Main Feedwater
Transient Event Tree, T2 (Modified from NUREG/CR-4550),
Volume 3, Figure 4.4-4 for Modes 1,2,14, and 15)**

**Figure 6.3-2 Transient Without Initial Loss of Main Feedwater
Transient Event Tree, T3 (Modified from NUREG/CR-4550,
Volume 3, Figure 4.4-5 for Modes 1,2,14, and 15)**

6.4 Event Tree Analysis of LOCAs

6.4.1 LOCAs Considered in Full Power Analyses

For the classical LOCAs (large, medium and small) we started from the NUREG-1150 event trees and modified according to the plant operational state we wished to apply the tree to. The success criteria were taken from NUREG-1150 as well, unless simple calculations showed that they could be relaxed; this was in order not to introduce any undue non-conservatism. More detailed calculations will be performed in Phase II.

In the LOCA event trees, operator emergency procedures are followed; in some cases non-proceduralized operator action is considered. In such cases explanation and justification are provided as appropriate.

It should be noted that the most conservative cases of a given LOCA category are considered. As one moves through the spectrum of plant operational states (POS), the boundaries between different LOCA categories may change depending on the POS. These boundaries are not precisely defined at this time (e.g. break size vs. LOCA categories in a given POS); calculations for this will be performed in Phase II. However, in considering LOCAs for a given POS, we move from the most conservative to the less conservative success criteria of the ECCS, thereby defining the different LOCA categories. Therefore, the initiator frequencies for the various LOCA categories may need adjustments when the more detailed LOCA thermal hydraulic calculations are performed in Phase II.

In the following discussion, unless otherwise noted, POS categories are for any type of outage, i.e. the LOCA trees will apply to a particular POS regardless of outage type (some basic event probabilities may be adjusted however). The LOCA event trees are presented in Figures 3.1 through 3.34.

6.4.1.1 Large LOCA (A)

POS 1 (Fig. 6.4.1)

For POS 1 we use the NUREG-1150 event tree, and modify the initiator frequency according to the time spent in the POS. In this mode the RCS is at full pressure and close to operating temperature; the power level can be as high as 15% of full power; the decay heat level can be a substantial fraction of that at full power.

The success criteria are as follows:

For early heat removal: 1/2 LPI (D6) and 2/2 (D5) accumulators;

Late heat removal: 1/2 LPR and switch to hot leg injection @ 16hr (H1).

Containment pressure suppression, early: 1/2 CSS or 1/2 ISR and SWS to associated ISR HX;

Containment pressure suppression, late: 1/2 ISR and SWS to associated HX or 1/2 OSR and SWS to associated HX.

Early and late containment pressure suppression is indicated by event CS.

Note that 1/2 ISR or 1/2 OSR is the success criterion for containment pressure suppression; for sufficient scrubbing of radionuclides any combination of two trains of ISR and/or OSR is needed (this will be discussed further in the Level 2 analysis).

In Surry, the ISR and OSR provide the long term decay heat removal via the SWS cooled heat exchangers; the LPR (and also HPR) have no such facility. Yet, the ISR and the OSR are not directly needed to prevent core damage (core damage may occur indirectly, when the containment fails due to lack of CHR systems, via the CV (core vulnerable) branch of the event tree; this branch leads to core damage in only 2% of the cases where the CHR function is not provided). At containment failure (about 130 psi mean), the water in the

sump will have reached a temperature of 340 - 350 F. Apparently, prior to that point, this temperature is sufficiently low, and enough NPSH is provided for the LPR pumps and enough cooling is provided in the core that no core damage results. When the containment fails, core damage may occur due to one or more of the following phenomena: plugging of the sump, loss of NPSH for the LPR pumps, loss of inventory in the sump, ECCS piping damage due to wall movement or ECCS equipment damage due to missile generation.

Following are the core damage sequences from Fig. 6.4.1:

Sequence 2 has the success of all the early heat removal, inventory control and CHR functions, but failure of the recirculation leads to core damage. Sequence 4 is similar, except the CHR systems fail; however, core damage results from failure of LPR.

Sequence 5 involves success of core cooling prior to containment failure; however, the containment failure (due to failure of the CHR equipment) leads to core damage via one or more of the mechanisms mentioned above.

Sequence 6 is core damage due to failure of low pressure injection, and sequence 7 is core damage due to failure of both accumulators on the intact legs to inject.

POS 2 (Fig. 6.4.2)

In this POS, the pressure, temperature and decay heat level can still be as high as POS 1, so we take these parameters at their most conservative value for this POS. In addition, the SI signal is disabled early in this POS (for other than Hi (3 psig) or Hi-Hi (25 psia) containment pressure). The SI signal is blocked at 2000 psig. It is assumed that at the lower end of this POS (350 psi, 350 F), the containment Hi condition will not be reached in time to initiate SI in order to avoid core damage. Therefore, the operator has to recognize the symptoms of a LOCA and press the SI button; this will then provide the necessary actuation signals to the ECCS, i.e. the LPI (event ISA2 in the event tree). Here, we assume that high containment pressure (17.7 psia) would not be reached in time to prevent core damage -- calculations will have to be performed to confirm this. Also, the accumulators are disabled for part of this mode (below 1000 psig). We assume, according to a Westinghouse study¹ (see below), that accumulators are not needed in this range (i.e. below 1000 psig); otherwise core damage would result as operators would take about 15 minutes to unblock the accumulators. Therefore, the more conservative part of this POS (and the one that's assumed for the whole POS) is when the accumulators are needed above 1000 psig.

The following are core damage sequences from Fig. 6.4.2:

Sequences 2 and 4 are due to failure of the low pressure recirculation (LPR) system; in addition, sequence 4 also includes failure of the CHR function.

Sequence 5 is caused by containment failure propagating into failure of core cooling systems.

Sequence 6 is due to failure of low pressure injection; sequence 8 is due to failure of 2/2 accumulators.

Sequence 7 represents failure of the operator to recognize symptoms or act in a timely manner to initiate safety injection.

POS 3 (Fig. 6.4.3)

In this POS, according to a Westinghouse paper¹, accumulators are not needed so event D5 was removed. In addition, simple calculations show that the RCS internal energy and decay heat level are sufficiently low, so that it would take more than 24 hours to overpressurize the containment without the CHR working. Therefore, events CS and CV were also removed.

For this POS, and the ones below it (4, 12, 13, 14, 15) that have lower decay power, and also for other LOCAs, we have considered the possibility that the onset of recirculation can be delayed, perhaps beyond 24 hours by intermittent operation of the LPI. Indeed, the procedure directs the operator to switch off the LPI when subcooling and pressure have reached certain limits and are increasing. Assuming that the RWST holds about 350,000 gallons of borated water (actually the Surry tech specs for POS 1 AND 2 require about 387,000 gal, but the tank will not be completely empty when switchover to recirculation is attempted) and that this water is heated up from 45 F to 212 F (i.e. the RCS depressurizes rather quickly), the total amount of sensible plus latent heat that can be carried off by the RWST water is about $3.51\text{E}+6$ MWsec. About 1/3 of that amount will be wasted by spillage from the break, so about $2.34\text{E}+6$ MWsec is available. The integral decay heat energy between 12 hrs (beginning of POS 3 in an average non-drained maintenance outage with RHR) and 36 hrs (mission time of 24 hours) is about $1.0\text{E}+6$ MWsec. So, at first glance it seems that recirculation would not be needed. This is assuming that credit is taken for latent heat, i.e. there is sufficient pressure drop between the core and the break to avoid boiling in the core. In reality, the procedure instructs the operator to maintain subcooling of at least 30 F at core exit. If we only take credit for sensible heat between 45 F and 180 F, the amount of heat that can be carried off by the RWST water is only $4.2\text{E}+5$ MWsec; spillage out the break reduces this to $2.8\text{E}+5$ MWsec. This is insufficient to take care of the heat load in most POSs and most types of outage. Even in POS 15 in a refueling outage of 30 days, the integral decay heat over a 24 hour period is $2.3\text{E}+5$ MWsec. The above analysis assumes that the RCS pressure has to reach 250 psig before the LPI pumps are no longer supplying RWST water and are shut off per procedure and that the average flow has to match the spillage rate. 350,000 gallons over a 24 hr period is an average flow of only 243 gpm.

Therefore, we will not give credit for intermittent LPI operation in order to avoid LPR. In very small LOCAs, however, credit will be given for throttling or intermittent operation of HPI in certain POSs.

In this POS we also ask a question about the availability of the RWST, since the Surry tech specs require the RWST to have 387,000 gal of borated water only above 350 F and 450 psig, i.e. in our POS 1, 2, 14 and 15. The only other requirement is that the other Unit RWST be kept at at least 14% capacity during filling the refueling cavity; the RWST of the Unit undergoing the fill must be kept at 20% in the event of cavity seal failure. The RWSTs can be cross connected locally or by receipt of a SLB (steam line break) signal from either unit.

Operator action is required in this mode. After diagnosis, the operator will close the breakers on the LPI pumps and actuate SI. This is event ISI3 in the event tree. It should be noted that if the accumulators were required in this mode, the initiator would automatically lead to core damage, since it would take the operator approximately 15 min after diagnosis to unblock the accumulators.

The core damage sequences are as follows:

Sequence 2 involves successful diagnosis and action by the operator, availability of the RWST and successful operation of LPI in the injection mode, but failure of the recirculation mode. Sequence 3 involves failure of injection mode LP.

In sequence 4, core damage is caused by unavailability of the RWST, whereas in sequence 5 the operator did not diagnose the problem and/or act in time to prevent core damage.

POS 4 (Fig. 6.4.4)

This POS is very similar to POS 3. Conservatively, the decay heat level is the same. The initial temperature is lower (200 F), while the initial pressure is the same. This means that in this POS the initial break flow will actually be about 20% higher than in POS 3, because the RCS coolant is subcooled and there will be no

choking. The initial break flow is about 1/2 that in POS 1. At 100-200 psia, when the LHS pumps come on, the break flow in a LOCA initiated in POS 4 can be 3-10 times that of POS 1 LOCA at that pressure. Therefore, POS 4 LOCA will depressurize much faster. On the other hand, at each pressure, the cumulative volume defect will be higher in POS 1 LOCA, as the liquid flashes to steam inside the RCS. We have assumed that the ECCS success criteria stay the same.

The description (but not necessarily the quantification) of POS 4 sequences is the same as for POS 3.

POS 12 and 13 (Fig. 6.4.5 and 6.4.6)

The event tree structure and sequence description is the same as in POS 3 and 4. The decay heat level will be lower (except in N1 type outage), but this does not impact quantification.

POS 14 (Fig. 6.4.7)

This POS is similar to POS 2; however due to much lower decay heat levels, the CHR need not work in the first 24 hours to prevent containment failure, so the CS and the CV events do not appear in this tree. It is assumed that the ISI14 quantification will be the same as ISI2.

The core damage sequences are:

Sequence 2 has successful operation of the accumulators, operator diagnosis and action (pressing of the SI button) and the LPI system, i.e. the event's early phase is successfully mitigated; however failure of recirculation leads to core damage. Sequence 3 involves success of the accumulators and operator action, but failure of the LPI. Sequence 4 is due to failure of the operator to diagnose and/or act in time. In sequence 5, failure of 2/2 accumulators to inject causes core damage.

POS 15 (Fig. 6.4.8)

This POS is similar to POS 1; however due to much lower decay heat levels, the containment will not fail in the first 24 hours regardless of CHR operation. The core damage sequences involve failure of recirculation (sequence 2), LPI (sequence 3) or accumulators (sequence 4).

6.4.1.2 Medium LOCA (S1)

POS 1 (Fig. 6.4.9)

The event tree for this POS will be the same as for full power, except that IE frequency will be different. The top events are: D1 (automatic HPI; success criterion is 1/3 charging pump), D5 (accumulators; success criterion 2/3 accumulators -- this is different than large LOCA success criterion for D5), CS (containment systems; success criterion 1/2 CSS and 1/2 ISR, or 1/2 CSS and 1/2 OSR, or 1/2 ISR in both short term and long term), CV (core vulnerable due to containment failure, a basic event), D6 (LPI; success criterion 1/2 LPI) and H1 (LPR; success criterion 1/2 LPR).

The core damage sequences are:

Sequence 2; success of all systems except recirculation.

Sequence 3; failure of the LPI. Sequence 5; success of the HPI, LPI and the accumulators; the containment pressure suppression fails causing failure of the containment; however, this does not cause failure of the required ECCS equipment; core damage is caused by failure of low pressure recirculation. Sequence 6 is similar, except core damage is caused by LPI failure. Sequence 7 has success of HPI and the accumulators, however due to failure of the CHR the containment fails causing failure of the ECCS.

Sequence 8 has success of the HPI in the early phase, but failure of the accumulators. Finally sequence 9 is caused by failure of the HPI.

POS 2 (Fig. 6.4.10)

This tree is similar to POS 1 tree, except that operator action is required to initiate the ECCS (i.e. the operator has to recognize the event and press the SI button). This is represented by the event ISI2. Failure of the operator to actuate SI in time leads to sequence 10 resulting in core damage. All other sequences are similar to POS 1 sequences. Event D1 (auto actuation of HPI) has been substituted by event D2 (manual actuation of HPI).

POS 3 (Fig. 6.4.11)

Top events CS and CV have been deleted due to lower decay heat levels than in POS 1 and 2 (i.e. containment failure for the first 24 hours is not of concern). Event D5 (accumulators) has been deleted because it's postulated that the accumulators will not be needed in these POS's (see the discussion in large LOCAs above). Event RWST has been added to ask the question on RWST availability, as it is not required to be available by the Technical Specifications. A question is also asked about isolation of the RHR system by the operators. The operators may isolate the RHR in order to try to isolate the LOCA; however, this LOCA does not occur in the RHR system, hence this action would be counterproductive. The RHR isolation is not in the LOCA procedures, but there is some likelihood that it will be attempted. If the RHR system is available, it can provide decay heat removal in the short term (i.e. until the RCS depressurizes to the LPI setpoint), in which case the HPI system would not be needed.

The core damage sequences are as follows:

Sequence 2 involves successful mitigation of the accident in the injection phase, but failure in the recirculation phase. Sequence 3 has success of most events in the injection phase (operator action, RWST availability and HPI) but failure of the LPI. Sequences 5 and 6 are similar to 2 and 3, except the RHR cannot provide early decay heat removal as it is isolated, so HPI successfully performs that function. In sequence 7, core damage is caused by failure of HPI and unavailability of RHR. Sequence 8 involves unavailability of the RWST and sequence 9 is caused by the operators' inability to respond in a timely manner.

POS 4 (Fig. 6.4.12)

The POS 4 S1 ET is similar to the POS 3 S1 ET. While in this POS we have subcooled water flowing out of the break, hence, depressurization is faster, the decay heat level in the worst case (i.e. only 6 hours after shutdown) is high enough so that HPI or RHR is needed.

POS 12 (Fig. 6.4.13)

The decay power is an order of magnitude lower than in POS 4 (for a refueling outage). For a 2" break case, it will take less than an hour to depressurize to the actuation pressure of the LPI (initial break flow of about 2000 gpm), while it will take more than 5 hours to raise the temperature of the RCS from 200 F to 350 F (saturation temperature at the LPI actuation pressure). Therefore, questions are not asked about the early decay heat removal (i.e. HPI or RHR availability). For an N1 outage, which is much shorter, this tree will have the same structure as the one for POS 4.

The following are the core damage sequences:

Sequence 2 involves success of the injection phase and failure of the recirculation phase. Sequence 3 is caused by LPI failure. In sequence 4, RWST is not available, while in sequence 5 the operators do not act in time to prevent core damage.

POS 13 (Fig. 6.4.14)

Due to the elevated temperature in this POS, it will take much less time to reach bulk boiling, and longer to depressurize to LPI actuation, so the tree structure reverts to that of POS 3 (and POS 4), i.e. early decay heat removal is needed.

POS 14 (Fig. 6.4.15)

The structure of this tree is similar to the one in POS 2, except that events CS and CV are omitted due to lower decay power level. Sequences 2-3 correspond to sequences 5-6 in POS 2, and sequences 4-6 correspond to sequences 8-10 in POS 2.

POS 15 (Fig. 6.4.16)

The ET for this POS is similar to POS 1, and the same remark applies as for POS 14 above. Sequence 2 is caused by failure of recirculation phase (success in the injection phase). In sequence 3, failure in the injection phase is due to failure in the LPI (while the HPI and the accumulators are successful). In sequence 4, failure of 2/3 accumulators to inject causes core melt, while in sequence 5, failure of the HPI means there is insufficient heat removal in the early phase of the accident.

6.4.1.3 Small LOCA (S2)

In this LOCA, due to its small size, success criteria for full power include success of the scram function (event K), success of the high pressure means of decay heat removal in the early phase (1/3 HPI (D1) in combination of 1/2 PORVs (P1) or 1/3 AFW (L)), and of the high pressure recirculation (1/3 HPR (H2) and 1/2 LPR (H1)) in the late phase (unless operator depressurization (OD) is provided, in which case low pressure recirculation only (1/2 LPR) is possible). Containment heat removal is desirable, although lack of it does not necessarily lead to core damage. MFW (M) is not credited for early decay heat removal (1/2 MFW with 1/3 HPI) in POS 1, 2, 14 and 15 because the break is of sufficient size to cause containment isolation on receipt of Containment Hi-pressure signal. MFW will be credited in POS 3 and POS 4 because there containment Hi setpoint would not be reached due to the lower temperature and pressure of the RCS coolant. RHR (1/2 RHR) is credited in POS 3, 4, 12 and 13.

POS 1 (Fig. 6.4.17)

The ET for this POS has the same structure as the one for full power. Sequences 2, 4, 5, 7, 10, 11, 13, 14, 16 and 17 have core melt as a result of failure of the recirculation system (high pressure or low pressure) in combination with successes or failures in other functions (but overall success in the early phase of the accident). Failures in the injection phase cause core damage in sequences 19 (caused by failure of AFW and failure in both PORVs to open on demand) and 20 (failure in the HPI). Sequences 8 and 18 are due to containment failure (caused by lack of CHR functions) effecting a failure in the ECCS systems. Sequence 21 transfers to the ATWS event tree as it involves failure to scram (reactor protection system).

POS 2 (Fig. 6.4.18)

The event tree for this POS is similar to the one for POS 1, except the scram question is no longer asked as the reactor is already shut down and the ECCS initiation requires operator action. Lack of timely operator action causes core damage in sequence 21.

POS 3 (Fig. 6.4.19)

In this POS, questions are asked about timely operator action; availability of the RWST; availability of the HPI in conjunction with 1/2 PORV or the AFW or the MFW, or presence of the RHR for early decay heat removal; and questions on long term makeup via the recirculation systems (LPR and HPR). The MFW can also be credited in POS 3 and 4 as the containment pressure will not reach Hi setpoint. It is assumed that the break flow is sufficient (about 2000 gpm initially for a 2" break) to disable the RHR before it can be used for long term heat removal or cooldown. However, in the early phase, RHR can be used for decay heat removal, in conjunction with the HPI, for success of the injection phase. For instance for a 0.5" break, the initial break flow will be about 130 gpm; assuming the HPI is throttled to deliver this flowrate (as the operators are instructed), this is insufficient to remove about 20 MW (worst case decay power) in this POS, without resorting to the PORVs (i.e. using latent heat of water), AFW, MFW, or the RHR. Event ISRH3 questions availability of the RHR in this POS, as the operator may have isolated it in response to this unisolable LOCA. Success of this event means the operator did not isolate the RHR so it is available for early decay heat removal.

Some decay heat removal systems are conservatively not modeled in this Phase (condensate pumps, firemain attachment to the AFW).

The core damage sequences in this POS are:

Sequences 2, 4, 5, 7, 9, 10, 12, 14, 15, 17, 18 have core damage caused by failures in the LPR and/or the HPR. Sequence 19 entails failure of early decay heat removal due to failure of both PORVs, or the AFW, or the MFW to operate or due to the operator isolation of the RHR system. Sequences 20 and 21 are makeup failure in the injection phase due to failure of the HPI or the unavailability of the RWST tank. Finally, sequence 22 is due to lack of timely operator action.

POS 4 (Fig. 6.4.20)

The ET structure in this POS is the same as in POS 3 (however, the quantification may not be necessarily the same). While break flow in this POS is somewhat higher, it is still necessary to have early decay heat removal. Additional decay heat removal equipment may be available in this POS, e.g. the steam generator RT system (used when the RCS temperature is below 200 F); no credit has been taken for this system, as it may not be able to remove all decay heat in the worst case (per attachment 5 of AP 27.00, it can remove decay heat at 35 hours after shutdown if both RT coolers are available).

POS 12 (Fig. 6.4.21)

In this POS, and for refueling and drained maintenance outages (R and D), the decay power is very low. That means that AFW and PORVs are not needed in case that RHR fails in the early phase; i.e., in the early phase, flow through the break and the resulting HPI makeup remove all the decay heat. The RHR question (event W3 in the event tree) is asked only for long term (24 hr) heat removal, because success of RHR to continue to run will obviate the need for LPR or HPR. This can be seen by considering two extremes of S2 LOCA sizes. If the break is 2", the RCS will depressurize to 40 psi (at which point the break flow is balanced by the maximum HPI flow of about 600 gpm) in less than an hour, at which point the coolant level will still be above midloop, so the RHR will presumably be still working. Further depressurization can lead to the RCS pressure reaching atmospheric pressure, at which time the break flow (and required HPI makeup flow) will be minimal. It is expected that this condition will be reached well before the RWST tank has been exhausted (at 600 gpm of maximum HPI flow it takes 10 hours). From then on, long term decay heat removal can proceed on RHR. Then, for a mission time of 24 hours, LPR and HPR are not needed. The other extreme is when the LOCA size is 1/2". At this size, the break flow (150 gpm) is initially within the capacity of the HPI pump (whose flow will be controlled by the operator as he tries to maintain the pressurizer level per procedure), so the pressure

will stay at its initial value (350 - 400 psia). The HPI flow of 150 gpm will also remove all the decay heat produced at this time. In 24 hours this flow will deplete about 216,000 gal from the RWST, leaving it about 44% full at that time. Again, no recirculation function is needed in this case.

Note that heat removal via steam generators (they are in wet layup in this POS) is not credited for decay heat removal for this LOCA. It is assumed that the break flow would interfere with natural circulation for larger break sizes in this category (for these break sizes the operating RCP will also trip due to depressurization).

The core damage sequences are as follows:

Sequences 3 and 4 result from failure of recirculation, when RHR is unavailable (for larger break sizes in this category). Sequences 5 and 6 represent failure in the injection phase, as either the HPI or the RWST is unavailable. Finally, sequence 7 is caused by lack of timely operator action to manually initiate safety equipment.

In an N1 type outage, the POS 4 event tree will be used also for this POS, as the decay power would be much higher than in an R or D outage.

POS 13 (Fig. 6.4.22)

The tree structure in this POS is the same as the one in POS 12. Similar conditions prevail, although in this POS, break flow will be somewhat smaller and depressurization will take longer, probably leading to lower coolant levels in the RCS when the conditions are stabilized.

In an N1 type outage, POS 3 event tree will be used for this POS as well.

POS 14

R and D outage (Fig. 6.4.23)

Due to the lower decay heat levels, this tree was created by deleting the following events from POS 2 ET: CS and CV as containment overpressurization will not occur in the first 24 hours, and P1 as PORVs are not needed for decay heat removal. The AFW system is not needed for decay heat removal either, but is needed for operator depressurization, as MFW will be isolated due to Hi containment pressure.

The core damage sequences are as follows:

Sequences 2, 4, 5, 7 and 8 involve failure of the recirculation phase after successes in the injection phase. In sequence 2, LPR fails after successful operator depressurization. Sequence 9 is failure of HPI and sequence 10 is due to lack of timely operator action to initiate SI (in this case mostly due to lack of timely diagnosis).

The tree structure for this POS in N1, N2 type outages is presented in Fig. 6.4.24. For this type of outage, the decay power is at such a level that containment systems are not needed, but injection phase heat removal is.

The tree structure is similar to the one in POS 2. Events CS and CV were removed out as lower decay heat levels obviate the need for containment protection in the first 24 hours. Sequences 2, 4, 5, 7 and 8 represent core damage due to failure in some aspect of recirculation, as the injection phase is successful. In sequence 2, LPR fails after successful operator depressurization. Sequence 9 is due to failure of decay heat removal due to failures in AFW and PORVs, while sequence 10 is core damage caused by failure of the HPI. Sequence 11 is caused by lack of timely operator action.

POS 15

For R and D type outage (Fig. 6.4.25):

Due to lower decay power, the ET for this POS lacks the following events from the ET of the symmetrical POS 1: CS and CV as there is no need for containment heat removal in the first 24 hours, and P1 as PORVs decay heat removal in the early phase is not needed. AFW needs to be available if credit for operator depressurization is taken.

Sequences 2, 4, 5, 7 and 8 are due to failure in some aspect of recirculation. Sequence 9 represents HPI failure and sequence 10 is transfer to the ATWS ET caused by scram failure.

For N1 and N2 type outages (Fig. 6.4.26):

The decay heat level is still sufficiently high to require early supplemental decay heat removal capability in the form of AFW or feed and bleed via PORVs. Containment systems are not needed in the first 24 hours.

Sequences 2, 4, 5, 7, 8 represent failures in the recirculation phase. Sequence 9 is caused by inadequate decay heat removal in the injection phase due to failure of both the AFW and PORVs. Sequence 10 is due to failure of HPI and sequence 11 is transfer to ATWS due to failure of the RPS.

6.4.1.4 Very Small LOCA (S3)

POS 1 (Fig. 6.4.27)

In this POS, the full power S3 LOCA tree is used. The successful mitigation of the accident requires the scram system, injection via 1/2 HPI in conjunction with early decay heat removal via 1/3 AFW or 1/2 MFW or 2/2 PORV, and late decay heat removal via 1/2 RHR or 1/2 LPR (if operator depressurization is successful), or 1/2 LPR and 1/3 HPR. Containment pressure suppression failure may lead to core damage. MFW system is credited here (unlike S2 LOCA) because the break flow is not sufficient to pressurize the containment to Hi signal that isolates the MFW, in the early time frame.

The top events in the event tree are K (RPS), D1 (HPI), QC (RCS integrity, i.e. whether the PORVs stick open), L (AFW), M (MFW), P (PORVs opening), CS (containment pressure suppression), CV (core vulnerable due to containment failure), OD (operator depressurization), W3 (RHR), H1 (LPR) and H2 (HPR).

The core damage sequences are as follows:

Sequences 3, 5, 6, 9, 11, 12, 14, 15, 17 and 18 involve successes in the injection and early decay heat removal phase but failure in the long term decay heat removal due to failures in the RHR and the LPR and/or the HPR. Sequence 19 has successful operation in the injection and early decay heat removal, but the ECCS ultimately fails due to containment failure. Sequence 20 has successful operation of the HPI but failure in the early decay heat removal, i.e. PORVs, MFW and AFW all fail. Sequence 21 transfers to small LOCA, as it has a PORV sticking open. Sequence 22 is core melt caused by failure of the HPI, and sequence 23 transfers to the ATWS event tree.

POS 2 (Fig. 6.4.28)

The ET for this POS is similar to the one for POS 1, except that event K is deleted (the reactor is shut down) and event ISI2 (operator recognizes symptoms and initiates SI) is added. Then sequence 23 is core damage due to failure of the operator to act timely. All other sequence definitions are the same as in POS 1.

POS 3 (Fig. 6.4.29)

In this POS events CS, CV and OD have been deleted. H1 and H2 have been deleted also, as the break flow rate is sufficiently small that the RWST depletion rate is sufficiently small not to require recirculation within 24 hours (it is expected that CSS will not actuate and thus cause RWST depletion). The HPI will be able to keep the reactor pressurized (to 350 psi) as the initial flow rate (about 125 gpm) is well within the HPI capacity at that pressure. The HPI flow rate is not sufficient to remove decay heat in this POS (in the worst case scenario). The early decay heat removal can be provided by the AFW, MFW, and PORVs, as before. In addition, late decay heat removal can be provided by these systems, as the RCS is pressurized and the steam generator primary can be filled with coolant. In addition, the RHR can be used for early and late decay heat removal. The top event ISR3 asks about operator isolation of the RHR in response to a LOCA. No question is asked of the RWST availability, as the makeup system can provide the initial flow rate via the charging pumps and enough boric acid is assumed to be on site. No question is asked about operator initiation of the SI, as the CVCS control system is still in operation, and the pressurizer level can be maintained by the charging system continuing to operate at a higher flow rate.

The sequences of interest are:

Sequences 5 and 8 transfer to the loss of RHR event tree, as decay heat removal is lost. In sequence 5, the operator does not isolate the RHR, but it (and other decay heat removal options) fails to continue operating, whereas in sequence 8 the RHR is isolated and other DHR options fail. In sequences 6 and 9 one or both pressurizer PORVs stick open, so these transfer to the small LOCA event tree.

POS 4 (Fig. 6.4.30)

This POS ET has the same structure as the POS 3 ET. Note that some modes of DHR have been neglected (condensate pumps, firemain). RT and SG wet layup are insufficient to remove the decay heat at this time (in the worst case scenario).

POS 12 (Fig. 6.4.31)

In this POS, for R and D outage, the decay heat level is so low that there are many modes of DHR. Wet layup of the SGs alone can provide for DHR well beyond the mission time of 24 hours; as before the RWST depletion rate is low so that recirculation makeup is not needed, regardless of the pressure and temperature of the RCS. The only equipment needed is one charging pump to continue to work for 24 hours. No operator action is needed as the CVCS system can continue using the charging pumps to maintain pressurizer level. Sequence 2 is CD due to failure of the HPI.

POS 13 (Fig. 6.4.32)

This tree has the same structure as the POS 12 ET.

POS 14 (Fig. 6.4.33)

This POS ET is similar to the POS 2 ET except that CS and CV events are taken out due to lower decay power. While the decay power is very low in R and D outages, the break flow (for the smaller sizes in this category) is still insufficient to remove decay heat in the injection phase.

Sequences 3, 5, 6, 9, 11, 12, 14, 15 are due to long term DHR failure, either by failure of the RHR or the failure of the LPR and/or HPR. Sequence 16 is caused by failure in the early decay heat removal. Sequence 17 is transferred to S2 as the PORVs stick open. Sequence 18 is caused by the HPI failure, and sequence 19 is due to lack of operator response.

POS 15 Fig. 6.4.34)

The event tree is similar to POS 1 ET, but CS and CV are deleted due to lower decay heat level. Long term decay heat removal is represented by sequences 2, 5, 6, 9, 11, 12, 14 and 15; short term decay heat removal by sequence 16. Sequence 17 is transferred to S2 due to PORV sticking open. Sequence 18 is failure of the HPI, while sequence 19 transfers to ATWS as the RPS fails.

6.4.2 LOCAs Specific to Shutdown

The following LOCAs have been considered that are specific to shutdown:

6.4.2.1 Failure of the Refueling Cavity Seal

This failure will cause the coolant to drain into the containment sump, draining the refueling cavity down to the level of the vessel flange. It is expected that this failure would be precipitated by a station blackout (causing i.e. a random seal failure) rather than an SBO, the probability of core damage would be negligible. The RHR cooling would continue to work, and there would be many additional ways to cool the core. The inventory loss would stop at the vessel flange. The water would be in the containment and available for use via the LPR system. Additional water would be available in the RWST (and the other Unit RWST) for injection via the LPI, the HPI or the RWST recirculation pumps. Gravity feed could be used, as the RCS would be open to the atmosphere. The decay power would be low such that the HPI (including cross-connect from the other Unit) would be able to remove the decay heat. We assume that in this POS all three loops are isolated, so no credit can be taken for steam generator wet layup.

The core damage by this initiator could be caused via two scenarios:

1) cavity seal failure followed by failure of the RHR in the 24 hour mission time followed by failure of many other redundant ways (as enumerated above) of makeup and decay heat removal and 2) cavity seal failure followed by station blackout within the following 24 hours and nonrecovery within 2-3 hours it would take to uncover the core. In the first scenario we use the following expression for rough quantification of its CDF frequency:

$$CDF(sl1) = (RWSTX + (HPI2+HPI1)*GF + RWSTU2) *$$

$$LPR1 * RHR1 * RCSLOCA$$

where

RWSTX = probability of failure to cross-connect the RWST from Unit 2, taken as 1.E-2

HPI1 = HPI from Unit 1, 7.E-4

HPI2 = HPI from Unit 2, 7.E-4

GF = gravity feed, 7.E-2

RWSTU2 = availability of full Unit 2 RWST, 5.E-2
(based on 1 month refueling operation every 18 months)

LPR1 = Unit 1 LPR, one train down for maintenance, 1.8E-2

RHR1 = Unit 1 RHR, 1.4E-2

RCSLOCA = frequency of refueling cavity seal failure, 1.E-2
(2 events, 600 reactor years)

Note that the values used were conservative. No credit was taken for multiple injection paths (LPI, HPI, GF, RWST recirc pumps from Unit 1 RWST (which would have at least 100,000 gal available-- enough for 5 hours at 8 MW decay heat load at 5 days after shutdown). Another 2-3 hours would elapse before core uncover. This time of 7-8 hours would give opportunity for operator recovery action, which is not taken into account here.

The CDF (sl1) frequency is then $1.5E-7/\text{yr}$. With recovery action this would decrease to a few times $E-8$.

The other possibility is seal failure followed (within 24 hours) by an SBO and followed by nonrecovery in 2 hours (time to core uncover assuming all three loops are isolated). The CDF expression is:

$$\text{CDF}(\text{sl2}) = \text{RCSLOCA} * \text{LOSP} * \text{DG} * \text{NR2}$$

where

RCSLOCA = initiator, $1.E-2$ as above

LOSP = $2.85E-5/\text{hr}$ during shutdown or $6.8E-4$ for 24 hours

DG = 0.07 (one DG assumed under maintenance)

NR2 = nonrecovery in 2 hours, 0.285 in shutdown

No credit is given to gravity feed in SBO conditions (successful GF would lead to non-recovery probability of 0.06 as 7 hours would be available). This leads to $\text{CDF}(\text{sl2}) = 1.4E-7$. The total CDF due to random failure of the refueling cavity seal would then be $2-3E-7/\text{yr}$ under conservative assumptions. It should be noted that the Seabrook Shutdown Study² considers the frequency of this initiator to be about 2 orders of magnitude lower (however, the seal design may be different).

6.4.2.2 Cavity Drain Valve Left Open During Fill for Refueling (POS 7)

The consequences of this would be the same as for the refueling cavity seal failure. If we assume the same initiating frequency ($1.E-2$), the same CDF frequency will result, i.e. this initiator will have a medium CDF. It should be noted that the Seabrook Shutdown Study² uses $1.7E-5/\text{yr}$ as the frequency of this initiator, i.e. almost 3 orders of magnitude lower.

6.4.2.3 RCS LOCAs due to Maintenance (K LOCAs)

At pressure, (POSs 1-4 and 12-15) these LOCAs are covered in the discussion of the classical LOCAs. At atmospheric pressure (RCS open), the level would drop to the bottom of the nozzles, if the opening was sufficiently large. This event would be covered in the loss of the RHR event tree. A LOCA would be defined as a leak rate that, if undetected for 12 hours (as RCS inventory balance is required once per shift) would lead to loss of the RHR, i.e. the level would drop to below the midloop. This required leak rate would vary from about 0 to about 100 gpm, depending on the POS and the configuration.

6.4.2.4 LOCAs through the CVCS

LOCAs due to a break in the CVCS (J LOCAs), due to maintenance errors in the CVCS (K LOCAs) and due to recoverable flow diversion via CVCS (H LOCAs) are all covered in the loss of the RHR event trees in POS 3-13.

In POS 1-4 and 12-15, any leaks would be compensated for (to the extent possible) by the automatic control system. There are low level alarms, and the letdown would isolate on low level or SI injection or the operator would isolate it. A LOCA in the charging pump suction line would disable the charging and isolate the letdown. A LOCA in the charging pump injection line would be stopped by a check valve. The letdown in POS 1, 2 and 15 is limited to about 60 gpm; in POS 14 it is about 150 gpm, and in other POSs is much lower (about 20 gpm). These are then the maximum leak rates from the RCS in these POSs. It would take several hours to a few days to uncover the core, or affect the RHR operation in these POSs at these leak rates. If the various control system features failed to isolate the LOCAs, as described above, an operator would

probably discover it before core uncover during RCS balance checks or via low level alarm on the pressurizer.

In the LOCA occurred in an unisolable part of the CVCS or if it wasn't isolated by the operator or the control system, (a small fraction of CVCS LOCAs as explained above), it would be covered in the classical LOCA event trees, in POS 1-4 and 12-15.

6.4.2.5 LOCAs in the RHR: J and K LOCAs

These LOCAs will be considered in POSs 3, 4, 12 and 13, i.e. when the RHR is pressurized. In other POSs, this initiator will lead to a loss of RHR (whether or not it's isolated), which is covered in the loss of RHR event tree.

POS 3, 4, 12, 13 (Fig. 6.4.35-6.4.38)

In all four applicable POSs, the tree structure looks the same, so they will be discussed all at once. The tree asks the questions as to what size LOCA for this POS the initiator is, and whether or not the operator isolates it. The question on isolation is size dependent, as the LOCA size determines the timing (the timing will be to loss of RHR). The timing is also POS dependent.

If the isolation is successful, the sequence transfers to "loss of RHR with no restart" tree (sequences 2, 4, 6 and 8). If no isolation is successful, the sequence transfers to the appropriate LOCA event tree (sequences 3, 5, 7 and 9). In Fig. 6.4.35:

- IELR3 = initiator, LOCA in the RHR (J or K LOCA)
- A3LO = probability that the initiator is NOT a large LOCA in POS 3
- S13LO = probability that the IE is NOT a medium LOCA in POS 3
- S23LO = probability that the IE is NOT a small LOCA in POS 3
- S33LO = probability that the IE is NOT a very small LOCA in POS 3
- ISOA3 = probability of isolating a large LOCA in POS 3
- ISOS13 = probability of isolating a medium LOCA in POS 3
- ISOS23 = probability of isolating a small LOCA in POS 3
- ISOS33 = probability of isolating a very small LOCA in POS 3

In figures 6.4.36 - 6.4.38, the headings have a similar explanation. The "loss of RHR without restart" event tree for POS 3, 4, 12 and 13 is presented in Fig. 6.4.39. This is really a subtree of the loss of RHR event tree. Event RCSV3 asks if the RCS is vented (which enables gravity feed); SGFNB is steam generator feed and bleed cooling, FNBF3 is feed and bleed via charging pumps and FNBG3 is gravity feed cooling.

The core damage sequences are:

Sequence 3 has the RCS vented and failure of feed and bleed via charging and failure of the gravity feed. Sequence 6 has the RCS non-vented, failure of the steam generator feed and bleed and failure of the primary feed and bleed via charging.

6.4.2.6 LOCAs in the RHR: H LOCAs

These LOCAs represent a recoverable diversion of reactor coolant. They would occur if valves MOV-100 (located outside the containment and operable from the control room) and RH-29 (manual valve located inside the containment) were both inadvertently open (the two valves are in series) while the RHR was operating. This would divert the coolant from the RCS to the RWST. The event could be recovered by closing one of the valves and actuating the LPI to transfer the coolant back from the RWST into the RCS.

In this section we consider the H LOCAs through the RHR only in a depressurized condition. When the RHR is pressurized (i.e. POS 3, 4, 12 and 13) this event would lead to overpressurization of the RWST and a possible interfacing LOCA outside the containment (V event). This scenario is covered in the discussion of interfacing LOCAs. Therefore the POSs where this LOCA can be considered are 5-11. In POS 7, the LPI pumps are operating and there would be no net loss of inventory, so this POS is not considered. In POS 9, the RHR is purposely employed to empty the refueling cavity into the RWST, so this POS is not considered, either. In POS 8, the refueling cavity is full of RWST water. Any drop in the cavity water level would be monitored and alarmed, therefore, this POS is not considered. Hence, this is a valid initiator in POS 5, 6, 10 and 11. POS 6 and 10 are midloop configurations.

The event trees are presented in Fig. 6.4.40-6.4.43. The structure of the trees is similar. If an operator detects and terminates the diversion (before loss of RHR), the end result is a success. If this doesn't happen, the RHR pumps will eventually lose suction, which will terminate loss of inventory, but will lead to loss of RHR. Then the sequence transfers to the loss of RHR event tree for the appropriate POS.

Results of Quantification - LOCAs

The results of quantification of LOCA event trees are presented in Table 10-2a in Chapter 10, which shows the results by individual sequences. Some conservative assumptions were taken, and maintenance data for particular POS were used. For isolation of the RHR system in a LOCA, zero was assumed as the HEP, including in RHR originating LOCAs as there is no such action in the procedure. For RWST unavailability, a value of 0.05¹ was conservatively used in POS 3,4,12,13. In POS 2-14, one train of LPI/LPR was assumed unavailable due to maintenance as is allowed by tech specs. In POS 4 and 12, the probability of sump plugging (due to maintenance work) was assumed to be a high value of 0.5. For operator initiation of SI for large LOCAs, it was assumed that the RCS will depressurize to atmospheric pressure and to the reactor vessel flange immediately, and the time was calculated for operator action for the decay heat to uncover the core by boiloff. For POS 2 and 14, the operator action was deemed "simple in the control room" (push the SI button), whereas for POS 3,4,12 and 13 it was deemed "complex in the CR". Based on the conservative estimates of the decay heat in a POS, the following are large LOCA SI actuation HEPs:

POS	HEP
2	0.2
3,12,13	1.2-3
14	2.0-4

For other LOCAs, the same HEPs were conservatively used except for POS 2, where the 1150 value for feed and bleed HEP was used (0.071), as the HPI system would be actuated early in other LOCAs.

The quantification process showed that the relatively high values (i.e. "medium" in our jargon) in POSs 4 and 12 for large and medium LOCAs were due to the assumption of high sump blocking probability. Sequences caused by this event substantially contributed to the total CDF of these LOCAs (which were in the high-medium range). Significant fraction of large LOCA CDF was due to LOCAs initiated in the interfacing systems (accumulator injection lines and the RHR). Before the transfers from the transients, (T2,T2A,T3,T3A) were quantified, the same was true for the S2 event tree, i.e. most of the initiator contribution was from ISL LOCAs (in POS 2 and 14 which gave S2 a "high" total CDF (in POS 2, the main contributors were sequences with failure of the operator to actuate SI, and in POS 14 failure of the LPR). However, the initiator contribution from the transients (due to stuck open PORV) gave S2 a "very high" frequency. This is due to the fact that a high value (2.7-2) was used for top event Q (failure of PORVs to reclose) in the IRRAS data base and the fact that transients have a relatively high frequency in POS 1,2,14 and 15.

The S3 LOCAs have a "medium" total CDF due to contributions from POS 2 (operator failure to activate SI), 4,12,15 (failure of HPI).

H LOCA transfers to loss of RHR event tree with a low frequency (for an initiator) of 6.-7/yr.

As mentioned above, cavity seal and drain valve failure induced CDFs are in the medium range.

6.4.3 Steam Generator Tube Rupture Event Trees

6.4.3.1 Introduction

In this section, the SGTR event trees are developed for the different low power or shutdown operating modes. The SGTR event trees are based on the full power tree contained in NUREG-1150 and are modified according to the requirements of the particular operating mode. The success criteria are also based on the full power operation and was discussed in Section 4.2.2.3.3.

After a SGTR event, the operator must insure that the reactor is in stable condition and long term heat removal is satisfied. The heat removal is accomplished by using the unfaulted steam generators and/or the

RHR system depending on the operating mode. In addition, the reactor must be depressurized to achieve stable conditions and to limit the outflow through the break.

The success criteria for the heat removal function is 1/2 SG and the associated 1/3 auxiliary feedwater pumps and, if used 1/2 trains of the RHR system. The reactor is depressurized by the pressurizer spray or by opening the PORV.

The SGTR event trees are presented in Figure 6.4.44 through 6.4.51. In the following sections more detailed discussions are presented on the event trees with emphasis on the specific features related to the different POS categories.

6.4.3.2 SGTR Event Trees

6.4.3.2.1 POS 1

This POS represents low power operation and the SGTR event tree is basically the same as the full power event tree developed in NUREG-1150. The event tree is presented in Figure 6.4.44. The first top event IESG1 represents the initiating frequency of the SGTR in POS 1. The reactor must be scrammed and safety injection is initiated to provide makeup to the RCS (Top events K and D1).

Heat removal is insured by initiating the auxiliary feedwater system to the steam generator (Top event L3). The operator has to depressurize the reactor to reach stable conditions utilizing the pressurizer spray and relief valves (Top event OD). The integrity of the RCS and/or the secondary system is questioned next (Top event Q and QS) and finally the recirculation functions are listed which insure ultimate heat removal.

Sequence 1 is a success state, where the reactor is depressurized, the faulted SG is isolated. At this point, the operator may initiate a backfill procedure to place the reactor in cold shutdown. A simultaneous loss and subsequent restart of the RCPs may lead to a boron dilution reactivity accident. This is represented by a transfer to an other event tree (SGRCT), which is discussed in Section 6.4.3.2.6.

The sequences are essentially identical to the full power operation sequences and only a brief discussion is given. Sequence 3 is a success even though SG integrity fails, but the leak rate is expected to be relatively small (initially 600 gpm reduced to ~ 200 gpm), at which rate the RWST would not be depleted in the mission time of 24 hours. Similarly, Sequence 4 is success since heat removal is assured through the steam generators using high or low pressure recirculation systems. The reactor may be depressurized to allow the operation of the low pressure recirculation system if required.

The core damage sequences 5, 6 and 7 represent scenarios where the RCS integrity fails and either the recirculation system fails or the SG integrity is not maintained losing primary coolant to the atmosphere bypassing the containment.

Sequence 9 is similar to Sequence 3 with higher flow rates due to the operators inability to depressurize the reactor combined with a loss of SG integrity. Sequence 10 is a safe state, since SG integrity is preserved allowing coolant recirculation from the sump. The core damage sequences 11 and 12 represent failures in the recirculation systems.

In Sequence 13, the integrity of RCS and SG are lost and coolant inventory is diverted to outside the containment with relatively large flow rates due to the failure of depressurization. Sequence 14 represents the loss of AFW and secondary heat removal function. The operation of the feed and bleed method is not assured due to the fluctuating pressure.

Sequence 15 is an early failure of the safety injection system with successful depressurization. In this case the break outflow may be terminated when the RCS pressure is reduced to the secondary side pressure. The

backfill procedure may be utilized by the operator to provide the reactor with makeup and to achieve cold shutdown. The potential for a dilution reactivity accident is accounted for by transferring to the SGRCT event tree.

The core damage sequences 16, 17, 18 and 19 represent failure of the safety injection combined with the loss of SG or RCS integrity. This leads to unrecoverable loss of the inventory without makeup capability. If the operator fails to depressurize, the outflow through the break uncovers the core without any makeup.

Sequence 20 is an ATWS sequence, which is assumed to be core damage due to an initial pressure increase in the RCS that may cause further ruptures increasing the leak rate from the primary system.

6.4.3.2.2 POS 2

The low power SGTR event tree is modified to take into account the mode specific features. The event tree in POS 2 is presented in Figure 6.4.45. The automatic safety injection signal is blocked in this mode, therefore manual actuation is required by the operator (Top event D2).

In addition to the AFW, the operator can supply feedwater to the SG using the main feedwater system, which was included in the event tree as an alternative means to provide secondary side inventory. The other top events are related to the depressurization, RCS and SG integrity and recirculation functions and are the same as in the previous POS.

The core damage sequences are basically the same as for the previous event tree due to failures to maintain RCS and/or SG integrity or failure of the depressurization function (Sequences 5, 6, 7, 9, 11, 12, 13). Sequences 17, 18, 19, 21, 23, 24, and 25 represent similar failure modes with the additional failure of the AFW system, which is functionally replaced by the successful MFW system.

Sequence 26 represents the failure of heat removal through the SG. The remaining core damage sequences, Sequences 28, 29, 30, 32-35 are related to the failure of safety injection function combined with the failure of the depressurization function or in the integrity of the RCS or SG.

Sequences 2, 14, and 27 are successful states, where the operator may initiate a backfill procedure and are transferred over to the SGRCT event tree.

6.4.3.2.3 POS 3, POS 4

The event trees are presented in Figures 6.4.46 and 6.4.47. The structure of the event trees in these modes are identical with the exception of the numerical representation of the failure data. The safety injection system must be manually initiated by the operator (Top event D2). The availability of the RWST system is not defined by the TS, therefore the top event D2 is modified to reflect the potentially increased unavailability of the RWST.

The RHR system may be used in these modes to remove decay heat and is represented by top event W3. Initially, the operator may misdiagnose the SGTR event and may attempt to isolate the RHR system, which is represented by an operator error in top event W3. The remaining top events ask the availability of the AFW or MFW systems, the success of the depressurization function, integrity of the RCS and the SG and the availability of the recirculation functions.

The first group of sequences represent the failure of the RHR system in combination with other failures. The core damage Sequences 7, 8, and 9 represent the failure of the RCS integrity combined with a failure of the recirculation system or SG integrity. Sequences 11, 13, 14, and 15 represent similar core damage scenarios where, in addition, the operator failed to depressurize the reactor.

Sequences 19-21, 23, and 25-27 represent the initial failure of the AFW system, which is functionally replaced by the MFW system. Sequence 28 is the failure of the secondary side heat removal.

The failure of the manual initiation of the safety injection system is represented by the following sequences. Sequence 29 is the failure of the SG integrity leading to coolant loss outside the containment. Sequences 32-34 are failures in the depressurization or in the integrity of the RCS or SG combined with the loss of the RHR system. Sequences 36-39 represent failures involving the makeup capability combined with the total loss of heat removal function provided by either the AFW or RHR system.

Sequences 2, 4, 16, and 29 are successful states, where the operator may initiate backfill procedure to place the reactor in cold shutdown. These end-states are transferred over to the SGRCT event tree that represents the boron dilution reactivity accident.

6.4.3.2.4 POS 12, POS 13

In these modes the decay heat levels may be substantially lower depending on the type of outage. In R and D (Refueling and Drained Maintenance) the decay heat levels are at such levels that the use of AFW system is not required for heat removal and the safety injection flow is capable of removing decay heat even if the RHR system initially fails. The availability of the RHR system is questioned only for long term heat removal.

The event trees are presented in Figures 6.4.48 and 6.4.49. The safety injection signal has to be manually initiated by the operator (D2) to initiate makeup flow to the RCS after an SGTR event. The heat removal function is represented by the RHR (W3) system, but only for the long term. The SG must be isolated to preserve the volume of the RCS (QS).

The primary system is at 300 psig in these modes and is expected to be quickly depressurized to the secondary side pressure, 120 psig, after the tube break. The pressurizer spray is available to perform depressurization, however, the opening of the PORV is not required in these conditions.

Sequence 5 represents the failure of the heat removal function and the failure to isolate the secondary side. This combination of failures leads to coolant loss bypassing the containment and eventual core uncover. Sequence 7, represents the failure of the safety injection combined with SG isolation failure again leading to coolant loss outside the containment and no makeup capability. Sequence 8 is the loss of safety injection capability combined with heat removal failure due to the loss of the RHR system.

In outage type N1 and N2 the structure of the event trees is identical to POS 3 and 4, respectively, and are used to represent these outage types. These event trees are discussed in the previous Section.

6.4.3.2.5 POS 14, POS 15

The SGTR event trees are presented in Figures 6.4.50 and 6.4.51. The structure of the trees are similar to POS 2 for POS 14 and POS 1 for POS 15. The differences are primarily numerical with different initiating frequencies and system unavailabilities. The core damage sequences are also similar as discussed in Sections 6.4.3.2.1 and 6.4.3.2.2.

6.4.3.2.6 Boron Dilution Reactivity Accident Event Tree

There is a potential for a boron dilution reactivity accident following a SGTR event. When the reactor is stabilized by successful depressurization and heat removal, the operator may attempt to place the reactor to cold shutdown by a backfill procedure. In this case, the reactor is further depressurized below the faulted SG pressure and AFW is allowed to drain into the RCS through the ruptured tube by maintaining AFW supply to the SG.

If coincidentally the RCPs are unavailable and the reactor is on natural circulation, unborated AFW may accumulate in the cold leg of the faulted SG due to lack of thermal mixing. A subsequent start up of the RCP may introduce a slug of unborated water into the core leading to a reactivity accident damaging the core.

The associated steam generator reactivity (SGRCT) event tree is presented in Figure 6.4.52. The following preconditions must exist in conjunction with the SGTR event:

- Coincident unavailability of the RCPs
- Backfill is selected as the method of cooldown
- RCPs are unavailable for sufficient time to allow flow stagnation in the RCS loop with the faulted SGTR and unborated AFW drains into the RCS
- Operator restarts the RCP in the loop with the ruptured SG

The first top event, LORCP, questions the availability of the RCP. It is modeled as a coincident random loss of off-site power event during the mission time of 24 hrs. The next top event selects the method of cooldown. Three different methods are available to the operator, but the backfill is the preferred one at Surry. Presently, for the phase I calculation, this is modeled as a definite success that is no other method is likely to be selected.

The next top event, WTRACC, models the rate of water accumulation in the cold leg. The failure state is when sufficient unborated water accumulated in the cold leg that could cause a reactivity accident. The last top event represents operator action that is the operator recognizes the potential for the reactivity accident (or for any other reasons) and will not start the RCP in the ruptured loop. The failure state is when the RCP is restarted. For conservatism, this is presently modeled as a definite failure that is the operator will restart the RCP as soon as off-site power becomes available.

Sequence 4 represents the core damage sequence due to the introduction of an unborated slug of water into the core during the backfill process by restarting the RCP.

6.4.3.3 Quantification and Results

The SGTR event trees were quantified by utilizing the previously described event tree structures with the top events replaced by the appropriate system fault trees. The system fault trees developed for the full power operations in NUREG-1150 were modified to reflect maintenance data applicable to the shutdown modes.

In addition, human error probabilities were developed specifically for the failure to manually initiate safety injection, since in POS 2-14 the automatic SI signal is blocked. In POS 3, 4, and 12, 13, the RHR system is in use and may be utilized to perform heat removal during an SGTR event. However, the operator may misdiagnose the indications and identifies the event as a LOCA and may isolate the RHR system. A human error probability value was developed to represent this particular failure mode base on the complexity of the action and the available time.

The availability of the RWST is not specified by the TS in POS 3, 4, and 12, 13, and therefore and increased RWST unavailability value was introduced and incorporated in the safety injection system fault tree D2.

The quantification was performed with a cut-off value of 1.0E-08 and sequences below this value were neglected. Table 6.4.3.1 presents the summary of the sequences identifying the failure paths and the corresponding core damage frequency groups.

The following SGTR sequences are discussed in some detail:

- SGTR2 Seq. 30 - T7D2ODS - Medium
- SGTR14 Seq. 30 - T7D2ODS - Medium

Both of these sequences represent the same failure modes in different operating mode, POS 2 and POS 14. After an SGTR event (T7) the safety injection function fails combined with the operator inability to depressurize the reactor. The resulting outflow through the ruptured tube bypasses the containment leading to core damage.

The cut set analysis indicates that the sequence is dominated by human failures related to both the HPI and depressurization function.

SGTR14 Seq. 9 - T7ODSQS - Medium

This sequence represents the failure of the operator to depressurize the reactor and consequently to isolate the faulted steam generator. The outflow through the break again bypasses the containment and results in core uncover.

Table 6.4.3.2 presents the core damage frequency results of one particular sequence representing the boron dilution reactivity accident. Even with the rather conservative assumptions (a; the operator will definitely restart the RCP after a LOOP event, b; there is sufficient time to dilute the cold leg of the SG), the total core damage frequency is still in the medium category, which is a relatively small contribution to the CDF due to other boron dilution scenario.

6.4.4 REFERENCES:

1. Gresham, J.A., et al. "Core Damage Risk Associated with Loss of Coolant Accidents During Shutdown Operations", presented at the PSA '89, International Topical Meeting on Probability, Reliability and Safety Assessment, April 2-7, 1989, Pittsburgh, PA.
2. Kiper, K.L., et al. "Seabrook Station Probabilistic Safety Study (Modes 4,5 and 6)", Vol. 1 and 2, May 1988.

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1E-A1	D5	D6	CS	CV	H1	SEQ #	END-STATE-NAMES
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D5: ACCUMULATORS
D6: LPI
CD: CONTAINMENT SYSTEMS
CV: CORE VULNERABLE
H1: LOW PRESSURE RECIRCULATION

Figure 6.4.1 Large LOCA in POS 1

IE-A2	D5	ISA2	D6	CS	CV	H1	SEQ #	END-STATE-NAMES
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Figure 6.4-2 Large LOCA in POS 2

IEA3	ISI3	RWST3	D6	H1	SEQ #	END-STATE-NAMES
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Figures 6.4-3 Large LOCA in POS 3

IEA 4	ISI 4	RWST 4	D6	H1	SEQ #	END-STATE-NAMES
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Figure 6.4-4 Large LOCA in POS 4

IEA12	ISI12	RWST1	D6	H1	SEQ #	END-STATE-NAMES
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Figure 6.4-5 Large LOCA in POS 12.

IEA13	ISI13	RWST1	D6	H1	SEQ #
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END-STATE-NAMES

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Figure 6.4-6 Large LOCA in POS 13

IEA14	D5	ISA14	D6	H1	SEQ #	END-STATE-NAMES
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Figure 6.4-7 Large LOCA in POS 14

1EA15	D5	D6	H1	SEQ #	END-STATE-NAMES
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Figure 6.4-8 Large LOCA in POS 15

IES11	D1	D5	CS	CV	D6	H1	SEQ #	END-STATE-NAMES
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Figure 6.4-9 Medium LOCA in POS 1

IES12	IS12	D2	D5	CS	CV	D6	H1	SEQ #	END-STATE-NAMES
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Figure 6.4-10 Medium LOCA in POS 2

IES13	ISI3	RWST3	ISRH3	D2	D6	H1	SEQ #	END-STATE-NAMES
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Figure 6.4-11 Medium LOCA in POS 3

IES14	ISI4	RWST4	ISRH4	D2	D6	H1	SEQ #	END-STATE-NAMES
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Figure 6.4-12 Medium LOCA in POS 4

IES11	ISI12	RWST1	D6	H1	SEQ #	END-STATE-NAMES
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Figure 6.4-13 Medium LOCA in POS 12

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IES11	ISI13	RWST1	ISRH1	D2	D6	H1		SEQ #	END-STATE-NAMES
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Figure 6.4-14 Medium LOCA in POS 13

IES11	ISI14	D2	D5	D6	H1	SEQ #	END-STATE-NAMES
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Figure 6.4-15 Medium LOCA in POS 14

IES11	D1	D5	D6	H1	SEQ #	END-STATE-NAMES
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Figure 6.4-16 Medium LOCA in POS 15

IES21	K	D1	L	P1	CS	OD	CV	H1	H2	SEQ #	END-STATE
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I-140

Figure 6.4-17 Small LOCA in POS 1

I-141

Figure 6.4-18 Small LOCA in POS 2

IES23	ISI3	RWST3	D2	ISRH3	L	M	P1	OD	H1	H2	SEQ #	END
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Figure 6.4-19 Small LOCA in POS 3

IES24	ISI4	RWST4	D2	ISRH4	L	M	P1	OD	H1	H2	SEQ #	END
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Figure 6.4-20 Small LOCA in POS 4

IES21	ISI12	RWST1	D2	W3	H1	H2	SEQ #	END-STATE-NAMES
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Figure 6.4-21 Small LOCA in POS 12

IES21	ISI13	RWST1	D2	W3	H1	H2	SEQ #	END-STATE-NAMES
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Figure 6.4-22 Small LOCA in POS 13

IES21	ISI14	D2	L	OD	H1	H2	SEQ #	END-STATE-NAMES
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Figure 6.4-23 Small LOCA in POS 14 for Refueling and Drained Maintenance Outage

IES21	ISI14	D2	L	P1	OD	H1	H2	SEQ #	END-STATE-
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Figure 6.4-24 Small LOCA in POS 14 for Non-Drained (N1 Outage)

IES21	K	D1	L	OD	H1	H2	SEQ #	END-STATE-NAMES
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Figure 6.4-25 Small LOCA in POS 15 Refueling and Drained Maintenance Outage Only

Figure 6.4-26 Small LOCA for POS 15, Non-Drained (N1) Outage

IESJ1	K	D1	QC	L	M	P	CS	CV	OD	W3	H1	H2	SEQ #	END-STATE-
-------	---	----	----	---	---	---	----	----	----	----	----	----	-------	------------

I-150

Figure 6.4-27 Very Small LOCA, POS 1

IES32	ISI2	D2	QC	L	M	P	CS	CV	OD	W3	H1	H2	SEQ #	END-STATE
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I-151

Figure 6.4-28 Very Small LOCA, POS 2

IES33	D2	L	M	ISR3	W3	QC	P	SEQ #	END-STATE-NAMES
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I-152

Figure 6.4-29 Very Small LOCA, POS 3

IES34	D2	L	M	ISR4	W3	QC	P	SEQ #	END-STATE-NAMES
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I-153

Figure 6.4-30 Very Small LOCA in POS 4

I-154

IES31	D2	SEQ #	END-STATE-NAMES
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1 2	OK CD
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Figure 6.4-31 Very Small LOCA in POS 12

I-133

IES31	D2	SEQ #	END-STATE-NAMES
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	1	OK
	2	CD

Figure 6.4-32 Very Small LOCA in POS 13

IES31	IS114	D2	QC	L	M	P	OD	W3	H1	H2	SEQ #	END-STATE-NAMES
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951-1

Figure 6.4-33 Very Small LOCA in POS 14

IES31	K	D1	QC	L	M	P	OD	W3	H1	H2	SEQ #	END-STATE-NAMES
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Figure 6.4-34 Very Small LOCA in POS 15

IELR3	A3LO	S13LO	S23LO	S33LO	ISOA3	ISOS1	ISOS2	ISOS3	SEQ #	END-STATE-
-------	------	-------	-------	-------	-------	-------	-------	-------	-------	------------

Figure 6.4-35 J&K LOCA through the RHR System, POS 3

I-159

IELR1	A4LO	S14LO	S24LO	S34LO	ISOA4	ISOS1	ISOS2	ISOS3	SEQ #	END-STATE-
-------	------	-------	-------	-------	-------	-------	-------	-------	-------	------------

Figure 6.4-36 J&K LOCA through the RHR System, POS 4

IELR1	A12LO	S112L	S212L	S312L	ISOA1	ISOS1	ISOS2	ISOS3	SEQ #	END-STATE-
-------	-------	-------	-------	-------	-------	-------	-------	-------	-------	------------

I-160

1	OK
2T	RHRNRST
3T	S312
4T	RHRNRST
5T	S212
6T	RHRNRST
7T	S112
8T	RHRNRST
9T	A12

Figure 6.4-37 J&K LOCA through the RHR system, POS 12

IELR1	A13LO	S113L	S213L	S313L	ISOA1	ISOS1	ISOS2	IS031	SEQ #	END-STATE-
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I-161

Figure 6.4-38 J&K LOCA through the RHR System, POS 13

RHRNR	RCSV3	SGFNB	FNBF3	FNBG3	SEQ #	END-STATE-NAMES
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Figure 6.4-39 Loss of RHR Transfer from J&K-RHR LOCAs

IEHR5	TERM5	SEQ #	END-STATE-NAMES
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1	OK
2T	RHR2A

Figure 6.4-40 H LOCA through RHR, POS 5

IERH6	TERM6	SEQ' #	END-STATE-NAMES
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		1	OK
		2T	RHR2A

I-164

Figure 6.4-41 H LOCA through RHR, POS 6

1EHR1	TERM1	SEQ #	END-STATE-NAMES
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	1	OK
	2T	RHR2A

Figure 6.4-42 H LOCA through RHR, POS 10

IEHR1	TERM1	SEQ. #	END-STATE-NAMES
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1	OK
2T	RHR2A

Figure 6.4-43 H LOCA through RHR, POS 11

SGTR1	IESG1	K	D1	L3	ODS	Q	QS	H1	H2	SEQ #	SEQ-NAMES	END-STATES
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I-167

Figure 6.4-44 SGTR Event Tree POS-1

SGTR2	IESG2	D2	L3	M	ODS	Q	QS	H1	H2	SEQ #	END-STATES
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I-168

Figure 6A-45 SGTR Event Tree POS-2

Figure 6.4-46 SGTR Event Tree POS-3

SGTR4	IESG4	D2	W3	L3	M	ODS	Q	QS	H1	H2	SEQ #	END-STATES
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I-170

Figure 6.4-47 SGTR Event Tree POS-4

SGT12	ISG12	D2	W3	QSIISO	SEQ #	END-STATES
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I-171

Figure 6.4-48 SGTR Event Tree POS-12

SGT13	ISG13	D2	W3	QSISO	SEQ #	END-STATES
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I-172

Figure 6.4-49 SGTR Event Tree POS-13

SGT14	ISG14	D2	L3	M	ODS	Q	QS	H1	H2	SEQ #	END-STATES
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I-173

Figure 6.4-30 SGTR Event Tree POS-14

SGT15	ISG15	K	D1	L3	ODS	Q	QS	H1	H2	SEQ #	SEQ-NAMES	END-STATES
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I-174

Figure 6.4-51 SGTR Event Tree POS-15

SGRCT	LORCP	BCKFL	WTRAC	RSRCP	SEQ #	END-STATES
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Figure 6.4-52 Boron Dilution Reactivity in SGTR Event Tree

Table 6.4.3-1**SGTR CORE DAMAGE FREQUENCY
SUMMARY - SEQUENCE QUANTIFICATION**

EVENT TREE	SEQUENCE	POS	CORE DAMAGE FREQUENCY CATEGORIES
SGTR2	T7D2Q - #29	2	MEDIUM
SGTR2	T7D2ODS - #30	2	MEDIUM
SGTR2	T7ODSQS - # 9	2	MEDIUM
SGTR12	T7D2QS - # 7	12	LOW
SGTR14	T7D2QS - #28	14	LOW
SGTR14	T7D2Q - #29	14	MEDIUM
SGTR14	T7D2ODS - #30	14	MEDIUM
SGTR14	T7ODSQS - # 9	14	MEDIUM
SGTR15	T7L3 - #14	15	LOW
SGTR15	T7D1ODS - #18	15	LOW
SGTR15	T7ODSQS - # 9	15	MEDIUM

TABLE 6.4.3-2**SGTR CORE DAMAGE FREQUENCY
BORON DILUTION SEQUENCES**

EVENT TREE	SEQUENCE #	POS	CORE DAMAGE FREQUENCY CATEGORIE
SGTR2/SGRCT	T7 - # 2/4	2	LOW
SGTR4/SGRCT	T7 - # 2/4	4	LOW
SGTR12/SGRCT	T7 - # 2/4	12	LOW
SGTR14/SGRCT	T7 - # 2/4	14	LOW
SGTR15/SGRCT	T7 - # 2/4	15	LOW
TOTAL			MEDIUM

6.5 Loss of Offsite Power and Station Blackout Event Tree Models

This section describes the event tree models that were developed to investigate high risk accident sequences initiated by a loss of offsite power event during low power/shutdown conditions at the Surry Nuclear Plant. Loss of offsite power (LOSP) will initially de-energize the normal and emergency 4.16 kv AC buses, and this results in de-energization of all associated lower voltage buses. The emergency AC buses are restored when the Emergency Diesel Generators (EDGs) start automatically to supply power; otherwise, there is a station blackout (SBO). The DC buses and the vital AC buses would still be available, unless additional random failures of these buses were postulated. In accordance with the approach used in the NUREG-1150 study, event trees were developed to evaluate the LOSP initiating event at various plant operational states (POS) for three boundary conditions:

- 1) LOSP where only one EDG is available,
- 2) Station blackout (SBO) at Unit 1, and
- 3) Station blackout at both units

In the first case of analysis, EDG 1 is assumed to start successfully and continue operating to supply emergency AC power to vital safety equipment during the duration of the LOSP event. EDG 2 is not available due to failure to start or continue to run, and EDG 3 (swing diesel generator) is under maintenance. For the second case, SBO at Unit 1 occurs when there is no emergency AC power supply to its vital safety equipment. EDG 1 fails to start or continue to run; however, there is one operable DG at Unit 2 to provide AC power to operate cross connectable systems such as the charging and AFW systems. This DG could be EDG 2, supplying power to the 2H bus, or EDG 3 (if it is not under maintenance) supplying power to the 2J bus. A critical event during SBO conditions is battery depletion. Depletion of the vital batteries will result in loss of instrumentation and control power. In the third case of analysis, SBO at both units is caused by the unavailability of all three DGs upon loss of offsite power. This condition arises when EDG 1 and EDG 3 are in failed states, and EDG 2 is under maintenance. Therefore, the unavailability of AC power at Unit 2 results in the unavailability to provide charging flow and AFW supply from Unit 2 systems.

In the quantification of the above 3 types of event trees, the approach that was used in the NUREG-1150 analysis was used. Given a loss of offsite power, the conditional probabilities of one unit blackout and two unit blackout were calculated using the boolean expressions given in Appendix B of NUREG/CR-4550, volume 3, part 2. The expressions were evaluated for each of the plant operational states and are tabulated in table 6.5-1.

The LOSP and SBO event trees were developed by considering the specific actions required of the hardware and operators to recover the plant from the LOSP event from the review of related Surry plant procedures:

- i) AP-10.00: Loss of Offsite Power
- ii) AP-17.00: Auto Start Failure of 1 or 2 EDG
- iii) AP-27.00: Loss of Decay Heat Removal Capability
- iv) ECA-0.0: Loss of All AC Power
- v) 1-FR-H.1: Response to Loss of Secondary Heat Sink (Rev. 3, 12/2/90).

The postulated risk sequences in the event tree models were identified using simple "back-of-envelope" thermal-hydraulic analyses of reactor core cooling and decay heat removal capability requirements to cope with the initiating event. These analyses were also performed to estimate the time available for AC power recovery prior to core damage. The following subsections provide discussion of the accident sequences, initial conditions of plant operational states, and top events of the event tree models for the three cases of LOSP conditions.

6.5.1 Event Tree Models for LOSP where 1 EDG is Available

Two event tree models were constructed to define the accident sequences initiated by a LOSP event where at least one EDG is available for POSs 1 through 15 during low power/shutdown conditions. Upon loss of offsite power, the supply breakers to 4.16 kv AC buses open, EDG 1 starts automatically and its output breakers close to load the DG on to the 1H 4.16 kv AC bus. This 4.16 kv AC bus provides power to the HPI pumps and the stub bus which supplies power to one CCW and RHR pump, and the 480 V AC buses through transformers. The 1H 480 V AC bus provides power to pumps such as the "A" train low pressure injection pump. In this scenario, the 1A 125 V DC bus provides control power to the switchgear for the pumps powered from the 1H bus. Therefore, the LOP12 and LOP3 event trees were developed to represent the plant response to the initiator for (i) low power operation, and (ii) shutdown conditions when the Residual Heat Removal System (RHRS) is in operation.

6.5.1.1 LOP12 Event Tree

The LOP12 event tree represents sequences where at least one EDG is available at Unit 1. This event tree is the same as the T1 event tree developed in NUREG-1150. The LOP12 event tree is used to analyze the high risk sequences at POSs 1 and 2 as well as POSs 14 and 15 because the plant conditions can be assumed to be similar to full-power operation.

6.5.1.1.1 Top Events of the LOP12 Event Tree

The top events of the LOP12 event tree describe the plant response to the initiator and the various means to maintain RCS cooldown after reactor shutdown. The top events are:

RSD	-	This top event represents the operator action to effect manual reactor trip.
RCI	-	This event represents the preservation of RCS integrity where pressurizer PORVs operate to cycle open and reclose after a demand. Failure of this event means a PORV is stuck open.
AFW	-	This event represents the availability of 1/3 AFW pumps to deliver water to 1/3 steam generators.
SL-CL	-	This event represents the availability of 1/3 charging pumps to provide RCP seal injection through 1-CH-MOV-370.
CCW	-	This event represents the availability of CCW flow to the thermal barriers of RCPs.
HPI	-	This event represents the availability of 1/3 HPI pumps to inject borated water from RWST into 1/3 RCS cold legs.
PRV	-	This event represents failure of 2 PORVs to open, and thus, prohibit successful feed and bleed operation of the primary system.
CS	-	This event represents containment heat removal capability which includes availability of the containment spray, inside spray recirculation and outside spray recirculation systems.
CV	-	This event represents probability of core damage for core vulnerable states where containment cooling has failed as the core is being cooled.
LPR	-	This event represents the failure of low pressure injection system in the recirculation mode.
HPR	-	This event represents the failure of charging pump systems in the high pressure recirculation mode.

6.5.1.1.2 Sequence Description

Sequences 1,2,4, and 7 represent stable plant conditions where there is successful mitigation of the initiator as a result of successful secondary side heat removal or core cooling via primary system feed and bleed operation (using one charging pump and opening both PORVs). Sequence 3 represents a RCP seal vulnerable condition when CCW flow to the thermal barriers and seal injection flow have failed. In Appendix D.5 of NUREG-1150 study, expert elicitation indicates that the risk of seal LOCA is averted if AC power is restored within 90 min. to enable safety injection flow.

Sequences 5 and 6 result in core damage through the failure to provide long term feed and bleed cooling in the containment sump recirculation mode. Sequence 5 is due to failure of the high pressure recirculation system, and sequence 6 is due to failure of the low pressure recirculation system.

Sequences 8 through 10 represent the occurrence of a core vulnerable state when containment heat removal fails after feed and bleed is initiated. Sequences 8 and 9 represent containment failure, followed by failure of long term cooling in the recirculation mode. Sequence 10 represents failure of containment heat removal systems which results in gradual containment overpressure conditions that cause failure of ECCS systems. This results in core damage.

Sequences 11 and 12 represent failure of SG heat removal followed by failure to establish primary feed and bleed cooling, either due to PORV failure to open or loss of HPI flow. Sequence 13 represents a transient-induced LOCA caused by a stuck-open PORV. Sequence 14 is an ATWS condition.

6.5.1.2 LOP3 Event Tree

The LOP3 event tree represents sequences in which the RHR system is initially operating during POSs 3 through 13 to remove the decay heat when the LOSP event occurs. After the LOSP occurrence, RHR pumps are tripped. If RCS decay heat removal capability is lost due to inability to restore RHR pumps during POSs 3 and 13, the responsive operator actions are to control RCS temperature by continuing to dump secondary steam to the main condenser via main turbine steam dump valves. Makeup to the SGs is possible when AFW is available, while RCS makeup can be provided by gravity feed from RWST inventory when depressurized or by feed and bleed operation using the charging pump. When the RCS is depressurized, the pressurizer PORVs are opened when pressure reaches 365 psig as a result of the LTOP protection limits. If they fail to open, the RHR relief valve opens at 600 psig.

6.5.1.2.1 Top Events of the LOP3 Event Tree

The top events of the LOP3 event tree describe the plant response to the initiator and the various means to accomplish decay heat removal as a result of RHR pump trip upon loss of power. The top events are:

- RHR -** This top event represents the restoration of the RHR pumps and the return to pre-initiator flow conditions. Upon LOSP, the stub buses that supply power to RHR pumps are shed from the emergency buses and must be manually reconnected to restore power to the RHR pumps.
- RCSV -** This top event represents the plant evolution in which the RCS is vented as a result of removal of one or more pressurizer safety valves or PORVs, pressurizer manway, or the reactor vessel head. This can occur in POSs 5 through 11 when the RCS is not pressurized, and it is possible to use gravity feed of RWST inventory to provide makeup to the RCS.
- SGFNB -** This top event represents heat removal capability from the secondary side of the SGs. If RCS is in mid-loop conditions, and one or more RCS loops are not isolated, reflux cooling ("fall back" cooling) in the primary system would be adequate to accomplish decay heat removal and prevent core damage. If SG tubes are full, natural circulation in the primary system would also be adequate to remove decay heat. Feedwater to the

SGs can be delivered by AFW pumps, AFW booster pumps, or fire main pumps. If the RCS temperature is below 150° F (POs 5 to 11), secondary side heat removal can be accomplished using the SG recirculation and transfer system. When secondary steam is available, it can be dumped via the main condenser to the Turbine Bldg. deck or through the SG PORVs to the atmosphere.

- FNBF -** This event represents forced loop flow using charging pumps or low head injection pumps to provide RCS makeup and bleeding through the primary PORVs or RHR relief valve.
- FNBG -** This event represents gravity feed of RWST inventory through the low head injection flowpath to the RCS when RCS is vented, or maintained in depressurized conditions.
- NRLOP -** This event represents failure to recover offsite power within 6 hours (before the onset of RCS bulk boiling). In the worst case scenario (POS3: Sequence 9), the allowable time for restoration of offsite power is 42 minutes.

6.5.1.2.2 Sequence Description

Sequence 1 represents stable plant conditions when RHR pumps are restored after pump trip, and RHR flow is re-established to pre-event flowrate and temperature conditions to control RCS temperature. Sequence 2 represents the scenario in which RHR capability was not restored, and feed and bleed operation on the primary side is utilized to remove decay heat. In this sequence, the RCS is vented, e.g. when the pressurizer safety valves are removed. Therefore, a bleed path exists, and forced flow to provide RCS makeup can be accomplished by using any available charging pump or low pressure injection pump. In sequence 3, there is no available pump to provide RCS makeup; however, gravity feed of RWST inventory via the LPSI flowpath is used for decay heat removal. The large inventory in the RWST allows sufficient time for the operators to establish other means of makeup for decay heat removal. Sequence 4 represents the scenario where offsite power is recovered in time to restore all necessary equipment for decay heat removal and successfully return the plant to stable conditions. In sequence 5, offsite power is not recovered in time to operate equipment to achieve decay heat removal prior to onset of RCS boiloff; hence, core damage ensues.

Sequence 6 represents the scenario in which RHR is not restored, the RCS is not vented with a large opening; however, natural circulation flow is sufficient to remove decay heat. In sequences 7 and 8, secondary side heat removal is not established and the RCS is closed. Pressure buildup due to heatup in the primary side could result in system rupture. In sequence 7, operators could establish low pressure injection and recirculation to prevent core uncover. In sequence 8, operators fail to accomplish long term cooling; however, offsite power was recovered in time to operate all necessary equipment required to remove decay heat and prevent RCS boiloff. In sequence 9, offsite power was not recovered in time to restore equipment for core cooling and decay heat removal, and core damage results from RCS boiloff. Based on conservative assumptions, thermal-hydraulic analyses indicate that sequence 9 is a "fast" scenario, i.e. time to reach core damage conditions is short.

6.5.2 Event Tree Models for Station Blackout at Unit 1

Several event tree models were constructed to analyze the accident sequences where Unit 1 has no AC power, but Unit 2 has one operable DG. This DG could be EDG2, supplying power to the 2H bus or EDG 3 supplying power to the 2J bus. These event trees are SB12, SB345, SB-ML, SB789, SB13 and SB145 which represent the response to the initiating event at the different POSs. During SBO conditions, loss of instrument air causes the main steam trip valves (MSTVs) to fail closed, the SG PORVs fail closed, and AFW flow distribution to the SGs is lost. A critical event for timing purposes in SBO evaluation is battery depletion. Depletion of the vital batteries will result in the loss of instrumentation or control power throughout the plant. The batteries at the Surry plant are designed for a two hour load discharge; however, battery depletion time could be extended with shedding of non essential loads. The NUREG-1150 assumption of four hours to

expect battery depletion in the SBO sequences was adopted in the event tree models that are discussed in the following subsections.

6.5.2.1 SB12 Event Tree

The SB12 event tree represents accident sequences that may occur during a station blackout at Unit 1 when in POSs 1 and 2 of low power/shutdown operation. Important considerations during a SBO are the preservation of RCS inventory by cross-connecting charging pump flow from Unit 2 and decay heat removal by the supply of auxiliary feedwater to the steam generators, and the extension of battery life. The SB12 event tree was developed from similar considerations used in the NUREG-1150 SBO event tree except for the following:

a) RCS decay heat at low power/shutdown operation is lower than at full power operation, therefore, time to RCS boiloff and steam generator dryout were calculated either at the decay power level corresponding to the time of scram, or to several hours after scram depending on the particular sequence (e.g. 4 hours after U-2 charging flow).

b) Cross-tie from a Unit 2 charging pump is to provide HPI for RCS makeup, and not seal injection flow for RCP seal cooling; therefore, cross-tied charging flow does not help to preserve seal integrity, and

c) Ability to maintain sufficient water inventory in the Intake Canal by remote closing of eight condenser isolation valves which are the total number for both units with any EDG available (DG1,3 or 2).

Prior to establishing the cross-tie of a Unit 2 charging pump to provide RCS makeup to Unit 1 during a SBO event, the emergency operating procedure (Steps 9-10; ECA 0.0, Rev. 3, 12-2-90) at Surry direct the operators to locally isolate the RCP seals by closing the valve 1-CH-MOV-1370 on the seal injection line. This is to reduce the susceptibility of the seals to failure from a sudden thermal shock of cold water on potentially hot seals. The seals eventually become hot as a result of loss of CCW flow to the RCP thermal barriers during SBO conditions. Upon a return to stable plant conditions, normal CCW flow is restored to establish flow of 35 gpm to thermal barrier and reduce seal leakoff temperatures to below 235°F before operators are instructed to introduce seal injection flow to cool the RCP seals (Steps 8-18, AP-9.02, Rev 1, 12-29-89).

The NUREG-1150 SBO event tree model credits the use of cross-tied Unit 2 charging pump or CCW flow for RCP seal cooling during SBO at Unit 1. This assumption is modified in the SB12 event tree model. Because the Surry plant operators are now instructed to locally isolate seal injection and seal leakoff (ECA 0.0, Rev. 3), and not attempt to re-establish seal cooling, RCP seal integrity is always challenged in a SBO event. The assumption in the SB12 event tree model is that, if a seal LOCA occurs, RCS makeup can be provided by Unit 2 charging pump during the first 4 hours. After four hours, it is assumed that the PORVs fail closed due to loss of battery power to open them) and no RCS makeup is viable. If the secondary side is depressurized, the RCS outflow is reduced since RCS pressure is reduced and RCS makeup by one cross-tied charging pump flow is adequate. However, if the secondary side is not depressurized, 750 gpm of RCS makeup is required and flow from one charging pump is insufficient despite reduced RCS pressures. The NUREG-1150 seal LOCA model was used to determine the allowable time for AC recovery before seal failures result in core uncover conditions.

The Unit 2 charging system and CCW depend on service water as a heat sink. The Intake Canal supplies the circulating ("circ") water and service water loads, with the largest loads being the circ water to both unit main condensers (up to 800,000 gpm per unit). Each unit's main condenser has four parallel 96 inch diameter pipes which carry the circ water from the Intake Canal to the discharge tunnel. Each circ water pipe has a condenser inlet and condenser outlet isolation motor-operated valve powered by separate unit emergency buses. In the event of station blackout at both units, there is no power to isolate the circ water flowing through the condensers. This results in an estimate of canal drainage in 30 minutes. For SBO at Unit 1, the recent ability of remote closure of condenser isolation valves (one per pipe) would help prevent drainage of

intake canal to unacceptable levels that would affect the availability of both unit systems. In NUREG-1150, manual isolation of these valves by physical operator actions was identified as a response to maintain sufficient water level in the intake canal.

From discussion with Virginia Power personnel, it was concluded that manual isolation of the condensers to conserve intake canal inventory during SBO is not practically achievable. They estimated that it would take 30 minutes for two operators working together to close one condenser isolation valve. Thus, the assumption of remote closure of the condenser isolation valves per 1-ECA-0.0 is considered in the SB12 event tree model. A review of available power supply to the isolation valves indicates that Unit 2 condenser isolation valves have power to close if either of two Unit 1 emergency buses (1H or 1J) is energized, or in a Unit 1 SBO, if the Unit 2 emergency bus 2J is energized. During a Unit 1 SBO, if the one operable EDG at Unit 2 is EDG 2, the Unit 2 emergency bus 2H is energized and 2J bus is not energized. According to 1-ECA-0.0, cross-tying of the two Unit 2 emergency buses including some local actions is required to establish power supply to Unit 1 condenser isolation valves (1-CW-100 A&C and 1-CW-106 B&D). The conservative assumption is that the 2J bus is not energized during a Unit 1 station blackout because if it was energized, no cross-tying of emergency buses would be necessary.

The above considerations were used to define the top events for the SB12 event tree which were different from the NUREG-1150 models. The NUREG-1150 SBO event tree model does not credit the use of Unit 2 charging pump for RCS makeup to feed RCS in a primary feed and bleed operation. In the SB12 event tree model, this cross-tie of Unit 2 charging pump is considered as a viable means to provide RCS feed for primary feed and bleed operation to maintain RCS cooldown and depressurization. When the secondary side cooling is available, it is assumed that one PORV is adequate for the "bleed" capability. If secondary side cooling is not available, two PORVs are required. The PORVs are assumed to fail closed after 4 hours of battery depletion. This assumption was applied to the case of an initially stuck open PORV. Therefore, successful primary feed and bleed operation to cooldown and depressurize the RCS is only viable during the first four hours of SBO.

6.5.2.1.1 Top Events of the SB12 Event Tree

The top events of the SB12 event tree describe the plant response to the initiator at low power operation (POSS 1 and 2) after descent from at-power operation. The top events are:

NRO.5 -	This event represents the recovery of AC power within 30 minutes before SG inventory is depleted to unacceptable levels if AFW is unavailable.
RCI -	This event represents the preservation of RCS integrity where the pressurizer PORVs operate to cycle open and reclose after a demand. Failure of this event means that a PORV is stuck open.
2-CHG -	This event represents the cross-tie of a Unit 2 charging pump to provide HPI flow for Unit 1 RCS makeup.
NRAC1 -	This event represents the recovery of AC power within one hour before depletion of SG inventory and the needed time to cross-tie Unit 2 charging pumps to provide HPI flow.
SGI -	This event represents secondary side integrity in which steam relief from the steam generators is through the safety valves. One SV on each SG was estimated to open every 20 min beginning within one hour (NUREG-1150). If one SV fails to reclose, an uncontrolled depressurization in one SG will occur.
SGFW -	This event represents the availability of auxiliary feedwater to the SGs through various means (e.g. turbine driven AFW pump in Unit 1, AFW cross-tie from Unit 2, or Fire main system).
O-DPR -	This event represents operator actions to depressurize and cooldown the RCS by a controlled SG depressurization by steaming into the main condenser via bypass of the MSTVs, and eventual rupture of LP turbine hoods to the Turbine Bldg. deck.

- SEALI - This event represents RCP seal integrity where seal leakage of 21 gpm per pump can be tolerated without causing core uncover.
- PRV - This event represents two open PORVs which allow successful bleed for primary feed and bleed operation to cooldown and depressurize the RCS.
- NRACS - This event represents the recovery of AC power prior to a seal LOCA which results in core damage.
- NRACD - This event represents the recovery of AC power prior to core uncover and ensuing core damage.

6.5.2.1.2 Sequence Description

Sequence 1 represents restoration of AC power to the electrical buses within 30 minutes when SG inventory is not depleted to unacceptable levels. Restoration of power within 30 minutes can result in successful mitigation of the SBO, regardless of other failures in that time period.

Sequence 2 represents successful mitigation of a long term station blackout. The pressurizer PORV cycles open and closed, thus maintaining RCS integrity. The SG safety valves reclose, and SG inventory is maintained as AFW supply is available to provide makeup to the SGs. RCS makeup flow from the Unit 2 charging system is provided to cool the core, and operators are successful in RCS depressurization. There is no seal LOCA and AC power is restored within 2.5 hours (prior to core uncover conditions after battery depletion (NRACD = 6.5 hours)). In sequence 3, battery depletion after 4 hours results in an inability to adequately control AFW supply to the SGs and subsequent loss of SG heat removal capability leads to core uncover and ensuing core damage. Sequences 4 and 5 represent outcomes when there is a breach in the RCP seals. Core damage is averted in sequence 4 because restoration of AC power would allow RCS to be depressurized to reduce outflow, whereas failure to recover AC power in 6.3 hours would cause seal LOCA to end in core damage conditions in sequence 5. Sequences 6 through 9 represent operator failure to depressurize by SG depressurization to the condenser after battery depletion in 4 hours. In sequence 6, there is no breach in RCP seals; however, AC power is restored prior to core uncover. In sequence 7, failure to recover AC power within 6.3 hours leads to core uncover due to RCS boiloff and this results in core damage. In sequence 8, seal LOCA occurs; however, AC power is restored in time to allow RCS depressurization and prevent core uncover due to seal LOCA. Sequence 9 represents failure to recover AC power in time (6.4 hours) to avert core damage conditions due to seal LOCA.

Sequences 10 through 17 delineate possible sequence outcomes when AFW supply is not available for makeup to the SGs and thus results in eventual SG dryout. On the primary side, RCS makeup is available from the cross-tied Unit 2 charging system; however, the relief capability of two open PORVs is required to allow successful primary feed and bleed operation to cooldown and depressurize the RCS. In sequence 10, there is no seal LOCA and the two PORVs remain open until battery depletion in 4 hours. AC power is restored prior to core uncover due to PORV failure upon loss of battery power; and therefore, no core damage occurs. Sequence 11 represents failure to restore AC power (in 6.5 hours) prior to core uncover as a result of PORV failures due to battery depletion, and core damage occurs. In sequence 12, the second PORV fails to open; however, AC power is restored in time to enable equipment to cooldown the RCS and prevent core damage. Sequence 13 represents failure to restore AC power in time (0.8 hours) to allow successful RCS feed and bleed operations. Sequences 14 through 17 represent situations when RCP seal integrity is challenged; core damage is averted in sequence 14 due to restoration of AC power prior to seal LOCA after 4 hours of battery depletion (NRACS = 4.1 hours). In sequence 15, failure to recover AC power after battery depletion would cause seal LOCA to end in core damage conditions. In sequence 16, the second PORV fails closed; however, AC power is restored prior to seal LOCA. Sequence 17 represents failure to recover AC power in time (0.8 hours) to prevent seal LOCA and core damage after the second PORV fails to open.

Sequences 18 through 27 delineate sequence outcomes when a SG safety valve is stuck open and RCS makeup is provided from the cross-tied Unit 2 charging system. A SG with the stuck open valve would depressurize that SG but not the other 2 SGs because the MSTV fails closed upon loss of instrument air during the SBO

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event. Flow to the faulted SG from the turbine-driven pump would increase but decrease to the other two higher pressure SGs. In SBO conditions, there is no operator manual control over the flow distribution to each SG because the associated valves on the AFW supply lines inside containment are motor operated (FW-151 A through F). The operator may have to throttle the AFW pump to prevent SG overfill. In sequence 18, adequate AFW makeup is provided to the SGs, and SG inventory is maintained. If AC power is restored within 6.8 hours (prior to core uncover due to primary heatup and pressurization), sequence 18 is a success sequence; otherwise, core damage occurs in sequence 19 due to degraded SG heat removal capability. Sequences 20 through 27 are analogous to sequences 10 through 17, except that SG dryout when the SG has a stuck open safety valve is sooner. The only impact on the sequence progression is that core uncover occurs sooner, and thus the time for successful AC power recovery is smaller, and non-recovery probabilities are higher. Therefore, the value for NRACS in sequence 17 is lower than the value for NRACS in sequence 27.

Sequences 28 through 39 delineate possible sequence outcomes when RCS makeup from Unit 2 charging system is not available. Sequence 28 represents success in maintaining SG inventory after SVs reclose and adequate supply of AFW is provided. Operators are successful in RCS depressurization and seal LOCA is averted. AC power is restored prior to any RCS boiling. In sequence 29, failure to recover AC power in 6.5 hours (before core uncover due to RCS boiling) results in core damage. Sequences 30 and 31 represent the situation when seal LOCA occurs; core damage is averted in sequence 30 because restoration of AC power would allow RCS to be depressurized to reduce outflow, whereas failure to recover AC power in 6.3 hours results in core damage in sequence 31. Sequences 32 through 35 are analogous to sequences 28 through 31, except that operators fail to depressurize the SGs; and therefore, RCS outflow through any breached RCP seals is greater and core uncover occurs sooner. Thus, the value for NRACS in sequence 35 is higher than that for NRACS in sequence 31 because the allowable time to AC power recovery is smaller. In sequence 36, there is no AFW makeup to the SGs and this results in eventual SG dryout. On the primary side, there is no available RCS makeup from the Unit 2 charging system; therefore, core damage results from RCS boiloff. Sequences 37 through 39 represent outcomes when a SG safety valve is stuck open and there is no RCS makeup from Unit 2 charging system. In sequence 37, AFW supply is available and the faulted SG automatically helps to provide RCS cooldown; therefore, the top events for operator actions associated with SG depressurization do not apply. If AC power is restored within 6.8 hours (prior to core uncover due to primary heatup), sequence 37 is a success sequence; otherwise, core damage occurs in sequence 38. In sequence 39, there is no AFW supply to the SGs. SG dryout and slow RCS boiloff results in core damage.

Sequences 40 through 47 delineate sequence outcomes when a pressurizer PORV is stuck open, RCS makeup from Unit 2 charging system is available, SG safety valves reclose to maintain secondary side integrity, and AFW supply is available for SG makeup. In sequence 40, operators are successful in SG depressurization and RCP seal integrity is not challenged. AC power is restored prior to any RCS boiling. In sequence 41, failure to recover AC power in 6.5 hours (before core uncover due to RCS boiloff) results in core damage. In sequence 42, there is a breach in RCP seals; however, AC power is restored to prevent the seal LOCA progressing to core damage conditions. Sequence 43 represents failure to restore AC power which allows the seal LOCA to proceed to core damage. Sequences 44 through 47 are analogous to sequences 40 through 43, except that operators fail to depressurize the SGs. RCS outflow through breached RCP seals is greater and core uncover occurs in a shorter time. Therefore, the value for NRACS in sequence 47 is higher than that for NRACS in sequence 43.

Sequences 48 through 55 delineate sequence outcomes when AFW supply is not available for SG makeup; and on the primary side, the pressurizer PORV is stuck open and RCS makeup is available from Unit 2 charging system. In sequence 48, there is no seal LOCA and the second PORV stays open until battery depletion in 4 hours. AC power is restored prior to core uncover (due to RCS boiling) after battery depletion. Sequence 49 represents failure to recover AC power in time to enable equipment to cooldown the RCS and prevent core damage. Sequence 51 represents failure to restore AC power in time to depressurize and cooldown the RCS, and core damage occurs. Sequences 52 through 55 represent seal LOCA situations where the ability to keep the second PORV open and recovery of AC power would mitigate the consequences. Core damage is averted

in sequence 52 due to restoration of AC power prior to seal LOCA proceeding to core damage conditions after 4 hours of battery depletion. In sequence 53, failure to recover AC power after battery depletion would result in seal LOCA progressing to core damage conditions. In sequence 54, the second PORV fails closed; however, AC power is restored prior to seal LOCA causing core uncover. Sequence 55 represents failure to recover AC power in time (4.3 hours) to prevent core damage from a seal LOCA after the second PORV fails to open.

Sequences 56 through 63 delineate sequence outcomes when the SG safety valve is stuck open, RCS makeup from Unit 2 is available, and the pressurizer PORV is stuck open. In sequence 56, AFW supply is available and the faulted SG helps to maintain RCS depressurization. If AC power is restored within 4.3 hours (prior to core uncover due to primary heatup), sequence 56 is a success sequence; otherwise core damage occurs in sequence 57. In sequences 58 through 63, AFW supply is not available and this results in eventual SG dryout. Sequences 58 through 61 are analogous to sequences 20 through 23, except that the stuck open pressurizer PORV results in faster RCS boiloff times. The only impact on the sequence progression is that core damage occurs sooner, and thus the time to successful AC power recovery is smaller. Therefore, the value for NRACD in sequence 61 is higher than that for NRACD in sequence 23.

Sequences 64 and 65 represent the failure to provide RCS makeup from the cross-tied Unit 2 charging system and a stuck open PORV. In sequence 64, recovery of AC power within one hour would allow the operators to successfully depressurize and cooldown the RCS. Core damage occurs in sequence 65 when AC power is not restored within one hour.

6.5.2.2 SB345 Event Tree

The SB345 event tree represents sequences for Station Blackout at Unit 1 when the RHR system is initially operating to remove the decay heat in POSs 3,4 and 5.

The RHR pumps are tripped when the stub buses supplying power to the pumps are shed from the emergency AC buses, 1H and 1J, upon LOSP. During SBO, there is no AC power supply to the affected buses to enable restoration of RHR pumps. As a result, RHR capability is lost and operator actions are to control RCS temperature by continuing to dump secondary steam to the main condenser via bypass of MSTIVs until SG pressures are reached at 175 psig. SG pressures are maintained at 175 psig to prevent injection of nitrogen from the accumulators into the RCS. Makeup to the SGs is possible when AFW is available, while RCS makeup can be provided by primary feed and bleed operation using the charging pump or gravity feed from RWST inventory when RCS is depressurized. When the RCS is pressurized, the pressurizer PORVs are opened when pressure reaches 365 psig as a result of the LTOP protection limits. If primary PORVs fail to open, the RHR relief valve opens at 600 psig.

Because RCS decay heat is much lower at POSs after refueling, another event tree model was developed to account for the different timing considerations in functional and phenomenological failures in POSs 11,12 and 13. The SB345 event tree was constructed to model sequences that may occur at POSs 3,4, and 5 where relief capability of PORVs and RHR relief valve at two-phase flow conditions during SBO is questioned. The PORVs are assumed to fail at 4 hours into the scenario due to loss of the vital batteries. The second event tree, SB13, was developed to represent sequences during POSs 11, 12, and 13 where RCS decay heat is much lower after a fuel reload. In these sequences, the relief capability of PORVs is not questioned.

6.5.2.2.1 Top Events of the SB345 event Tree

The top events of the SB345 event tree are:

- O-DPR - This event represents operator actions to depressurize and cooldown the RCS by a controlled SG depressurization by steaming into the main condenser via bypass of the MSTVs, and eventual rupture of LP turbine hoods to the turbine Bldg. deck.

- XFNBF -** This event represents RCS feed using Unit 2 cross-tied charging pumps to provide RCS makeup and bleeding through the primary PORVs or RHR relief valve.
- NRACD -** This event represents the recovery of AC power prior to core uncover and ensuing core damage.

6.5.2.2.2 Success Criteria

SBO at POSs 3,4,5,11,12 and 13 results in loss of RHR. RCS integrity requires the need to prevent overpressurization of the RHR system as a result of primary coolant heatup in the reactor vessel. The relief capability of PORVs at two-phase flow conditions is questioned at POSs 3,4,5 rather than at POSs 11,12, and 13 because of higher RCS decay heat loads. Secondary side cooling is unavailable because the SG relief valve setpoints are set at operating conditions, i.e., 1085 psig. The postulated inability of PORVs and RHR relief valve to relieve pressure could result in rupture of the RHR system and primary coolant flashing to steam (LOCA conditions).

6.5.2.2.3 Sequence Description

Sequences 1 through 4 delineate sequence outcomes when operators successfully depressurize RCS by controlled blowdown of steam into the condenser. However, SG feedwater is not available because the motor-operated isolation valves on the AFW supply line inside containment are closed by procedure, and no credit was allowed for entry into containment to open the isolation valves during an SBO. This results in primary side heatup, and operators would have to provide RCS makeup from the cross-tied Unit 2 charging system and bleeding through the PORVs to maintain RCS depressurization. The time to recover AC power to prevent core damage is longer compared to the situations when the operators fail to blowdown steam from the SGs, or are unable to cross-tie Unit 2 charging system. Therefore, sequence 1 represents the success outcome when operators can maintain RCS cooldown and depressurization, and AC power is restored prior to onset of core damage conditions within 10.0 hours. Sequence 2 represents failure to restore AC power within 10.0 hours prior to core damage (POS 3). Sequences 3 and 4 represent failure to provide RCS makeup from Unit 2 cross-tie of charging system. In sequence 3, recovery of AC power within 6.5 hours helps to prevent core damage. Sequence 4 results in core damage because AC power is not restored in time.

Sequences 5 through 8 delineate sequence outcomes when operators fail to depressurize RCS initially via SG blowdown to the condenser. The sequence outcomes correspond to sequences 1 through 4 except that the time to core damage is shorter when AC power is not restored. In sequence 7, the allowable time for recovery of AC power is 42 minutes.

6.5.2.3 SB13 Event Tree

The SB13 event tree for POSs 11,12, and 13 was developed to represent sequences where RCS decay heat is assumed to be lower. This assumption precludes questions asked about the relief capability of PORVs or the RHR relief valve. Therefore, success criteria for this event tree is less stringent than the event tree for POSs 3,4,5.

6.5.2.3.1 Sequence Description

Sequences 1 through 3 delineate outcomes when operators are able to depressurize RCS initially by controlled blowdown of steam into the condenser. In sequence 1, operator actions to provide RCS makeup from the cross-tied Unit 2 charging system and bleeding through the PORVs is sufficient to prevent core damage. No questions are asked about recovery of AC power. Sequences 2 and 3 represent failure to provide RCS makeup from Unit 2 cross-tie of charging system. In sequence 2, recovery of AC power within 24 hours can prevent core damage conditions. In sequence 3, core damage occurs when AC power is not restored in time (10.5 hours in POS 11) after failure to provide RCS makeup in the primary system.

Sequences 4 through 6 delineate outcomes when operators fail to successfully depressurize RCS initially by SG blowdown to the condenser. Sequence 4 represents a success outcome when operators could provide RCS makeup from Unit 2 cross-tie of charging system. Sequences 5 and 6 are similar to sequences 2 and 3 except that the time to core damage is shorter when AC power is not restored.

6.5.2.4 SB-ML Event Tree

The SB-ML event tree represents sequences when SBO occurs during mid-loop operation. When the plant is in mid-loop conditions (POSS 6 and 10), the RCS loops may be isolated and drained to a reference mid-loop level for loop maintenance and SG tube inspection. The SG primary side manways are removed, and the RCS is vented. The RHR system is in service to remove decay heat, and the reactor is maintained shut down to cold shutdown or refueling boron concentration. This configuration is held prior to head stud detensioning and removal of reactor vessel head in preparation for refueling activities. After refueling, the RCS may be returned to mid-loop conditions for the second time to allow uncompleted loop maintenance to be performed.

When SBO occurs, the RHR pumps are tripped and decay heat removal capability is lost. If the RCS is vented, RCS makeup from Unit 2 charging system or gravity feed from RWST inventory can be utilized to maintain RCS water level. If the SGs are in "wet layup" conditions, the SG inventory is a heat sink to help mitigate any primary heatup unless it is isolated on the primary side during refueling operations.

6.5.2.4.1 Top Events of the SB-ML Event Tree

The top events of the SB-ML event tree are:

- RCSV - This top event represents the RCS vented conditions such as when pressurizer safety valves, PORVs, or pressurizer manway, is removed.
- SGFNB - This top event represents heat removal capability from the secondary side of the SGs. Reflux cooling in the primary system is adequate to accomplish decay heat removal when the RCS is in mid-loop conditions. Feedwater to the SGs can be delivered by AFW pumps, AFW booster pumps, or fire main pumps. If RCS temperature is below 150°F, secondary side heat removal can be achieved using the SG recirculation and transfer system. When secondary steam is available, it can be dumped via the condenser to the turbine Bldg. deck or through SG PORVs to the atmosphere.
- 2-CHG - This event represents RCS makeup from cross-tie of Unit 2 charging system.
- FNBG - This event represents gravity feed of RWST inventory through the low head injection flowpath to the RCS when RCS is vented.
- NRACD - This event represents the recovery of AC power prior to core damage conditions.

6.5.2.4.2 Sequence Description

Sequences 1 through 5 delineate sequence outcomes when RCS is vented. In sequence 1, RCS makeup from Unit 2 charging system is sufficient to maintain RCS cooldown when RHR capability is lost as a result of SBO. Sequence 2 represents the unavailability of RCS makeup from Unit 2 charging; however, gravity feed from RWST and recovery of AC power within 24.0 hours would prevent core damage. In sequence 3, failure to restore AC power prior to core uncover results in core damage. Sequence 4 represents unavailability of RCS makeup from either sources, forced flow from Unit 2 charging system or gravity feed from RWST; however, AC power is restored prior to core damage. In sequence 5, failure to restore AC power within 1.5 hours leads to core damage.

Sequences 6 through 9 delineate outcomes when RCS is not vented. In sequence 6, reflux cooling is sufficient to prevent core damage. Sequence 7 represents availability of RCS makeup from Unit 2 charging system to maintain RCS cooldown. In sequence 8, the recovery of AC power within 1.5 hours would prevent primary heatup and subsequent core damage. Sequence 9 represents failure to restore AC power in time to prevent

primary heatup as a result of unavailability of RCS makeup from Unit 2 charging system and loss of secondary side heat removal.

6.5.2.5 SB789 Event Tree

The SB789 event tree represents sequences when SBO occurs during POSs 7,8, and 9. When in POS7, the reactor vessel water level is raised to below the reactor vessel head flange, the refueling cavity seal is installed, and the reactor vessel head is detensioned and removed in preparation for refueling (POS 8). After refueling, the refueling cavity is dewatered to the reactor vessel head flange, the reactor vessel head is lowered and set in place, and the head bolts are retensioned (in POS 9) before RCS draindown to mid-loop conditions is performed. The RHR system is in service during these POSs to control RCS temperature below 140°F. When a SBO occurs, the RHR capability is lost.

6.5.2.5.1 Top Events of the SB789 Event Tree

The top events for the SB789 event tree are:

- CV-SL - This event represents failure of the refueling cavity seal which results in the loss of reactor vessel inventory.
- FNBG - This event represents gravity feed of RWST inventory through the low head injection flowpath to the RCS, and this event is applicable only for POS 7 and 9. In POS 8, most of the RWST water is in the refueling cavity.
- NRACD - This event represents the recovery of AC power prior to core damage conditions.

6.5.2.5.2 Sequence Description

Sequences 1 and 2 represent the outcomes when the refueling cavity seal is intact during SBO. In sequence 1, the recovery of AC power within 24.0 hours would prevent core damage conditions that result from reactor vessel water heatup. Sequence 2 represents the failure to restore AC power in time to maintain long term cooling using the RHR system. Hence, core damage ensues.

Sequences 3 through 5 delineate outcomes when the refueling cavity seal fails and causes a loss of reactor vessel inventory. In sequence 3, gravity feed of RWST inventory through the low head injection flowpath and recovery of AC power in time to maintain RHR capability would prevent core damage conditions. In sequence 4, failure to restore AC power (2.2 hours) to reestablish RHR capability would result in core damage. Sequence 5 represents the failure to provide gravity feed of RWST inventory to replenish the inventory loss due to refueling cavity seal failure.

6.5.2.6 SB145 Event Tree

The SB145 event tree represents sequences when SBO occurs during POSs 14 and 15. These POSs characterize the ascent from RCS temperature of 345°F to when RHR is removed from service, engineered safeguards systems are enabled, and LTOP is disabled. RCS heatup and pressurization is continued with SG steam dump to the main condenser until normal operating RCS temperature and pressure at hot shutdown conditions are reached. Important considerations during an SBO in POSs 14 and 15 are the preservation of RCS inventory, feedwater supply to the SGs, and the extension of battery life. The decay heat load in POSs 14 and 15 is assumed to be lower than in POSs 1 and 2 because of continued shutdown and possibly a refueling of core. The assumption of low decay heat levels allows much longer time to RCS boiloff and steam generator dryout. Therefore, the allowable time for AC power restoration before core damage is much longer.

6.5.2.6.1 Top Events of the SB145 Event Tree

The top events of the SB145 event tree describe the plant response to the initiator during ascent from cold shutdown to low power operation (POSS 14 and 15). These events are 2-CHG, O-DPR, SEALI, NRACS and NRACD, and are similarly described for the SB12 event tree.

6.5.2.6.2 Sequence Description

Sequence 1 is a success sequence where secondary side integrity is maintained, i.e., steam relief is through the SVs, steam generator inventory is replenished by available AFW supply, and Unit 2 charging system is providing RCS makeup to maintain RCS cooldown.

Sequences 2 through 8 delineate outcomes when cross-tie of Unit 2 charging system is not available. In sequence 2, operators are successful in maintaining RCS depressurization by SG blowdown to the condenser, and there is no breach in RCP seals. Sequence 3 represents the recovery of AC power before a RCP seal LOCA results in core uncover. In sequence 4, failure to recover AC power in time allows the RCP seal LOCA to proceed to core damage. Sequence 5 represents operator failure to depressurize RCS by SG blowdown to the condenser, and this results in eventual heatup on the primary system. RCP seal integrity is not challenged, and AC power is restored in time to enable equipment of provide RCS feed to cooldown the primary system. In sequence 6, failure to recover AC power in time such that primary heatup causes RCS boiling to end in core damage conditions. In sequence 7, there is a breach in RCP seals; however, AC power is restored before the RCP seal LOCA results in core uncover. Sequence 8 represents failure to recover AC power in time (9.0 hours) to prevent the RCP seal LOCA to proceed to core damage after operator failure to depressurize the RCS by SG blowdown to the condenser when cross-tie of Unit 2 charging system is not available.

6.5.3 Event Tree Models for Station Blackout at Both Units

Station blackout at both units occurs when all three diesel generators are unavailable upon LOSP. Due to the unavailability of AC power supply at Unit 2, charging pump and AFW flow are not available from Unit 2. Therefore, the event tree models for dual unit SBO differ from single unit SBO event trees only in that charging flow and AFW supply from Unit 2 are not considered.

The event tree models developed for dual unit SBO are the SB212 and SB215 event trees for POSS 1 and 2, and POSS 14 and 15 respectively. The SB23 event tree was developed for POSS 3,4, and 5. The SB26 event tree was developed for POS 6 and 10, while the SB27 event tree represent sequences for POSS 7,8 and 9. Finally, the SB211 event tree was developed for POSS 11,12, and 13.

6.5.4 Accident Sequence Quantification

The accident sequences initiated by LOSP and SBO events during low power/shutdown conditions that result in core damage at the Surry nuclear plant are identified by the event trees in Figures 6.5.1 through 6.5.14. The sequence minimal cutsets were quantified using the point estimates of mean values for event failure probabilities. Conservative estimates of human error probabilities (HEPs) were used in the quantification of individual event tree models to assure that human error sensitive sequences were not screened out in the quantification process. The results of the accident sequence quantification for LOSP and SBO events are summarized on Tables 10.3a,b,c,d,e, and f. The various core damage sequences for each particular POS are characterized accordingly as high, medium or low (H,M,L) core damage frequency, early or late (E,L) core damage sequences, and whether the sequence occurs in a closed or open containment (C,O). The insights obtained from the accident sequence quantification for LOSP and SBO events at each particular POS are summarized below:

6.5.4.1 LOSP Event

POS 1-	The core damage frequencies (CDFs) for all the core damage sequences at this particular POS are very low ($<5E-9$ /yr) because of the short duration of POS 1 in the refueling outage. Sequence 13 (a transient-induced LOCA) and sequence 14 (ATWS) have medium-ranked CDFs.
POS 2-	Sequence 3 (a seal vulnerable condition), and sequences 11 and 12 (failure of SG heat removal) have medium-ranked CDFs as a result of the unavailability of charging pump cooling water (CPC-MDP-MA-SWIOA) and AFW pumps (AFW-TDP-MA-FW2, AFW-MDP-MA-FW3A) that were downed for maintenance.
POS 3-	Sequence 9 has a medium-ranked CDF as a result of these failure events: a) operator misdiagnosis of the accident conditions, b) failure to cross-tie AFW flow from Unit 2 to replenish feedwater to depleted SGs, and c) failure to cross-tie charging pump flow from Unit 2 to accomplish successful primary feed and bleed operation.
POS 4-	Same as POS 3.
POS 5-	Sequence 9 has a medium-ranked CDF as a result of operator misdiagnosis of accident conditions and failure to reconnect the stub bus that provides power to RHR pumps. Operator failure to cross-tie AFW pump flow from Unit 2, and failure to restore RHR pumps also contribute to the accident sequence frequency.
POS 6-	Sequence 5 has a medium-ranked CDF as a result of operator errors: (a) operator misdiagnosis, b) failure to reconnect the stub bus to enable restoration of RHR pumps, (c) failure to provide RCS makeup by gravity feed of RWST inventory via the LPSI flowpath to help decay heat removal, (d) failure to cross-tie charging pump flow from Unit 2. Sequence 9 has high CDF due to operator errors described in sequence 5, CPC service water pump under maintenance, and the fraction of time in which the RCS loops are isolated.
POS 7-	Same as POS 6; except that sequence 9 has a much higher CDF as a result of RCS loops being completely isolated from the secondary side, and this prevents reflux cooling to help accomplish successful decay heat removal.
POS 8-	Sequence 5 has a medium-ranked CDF due to operator failure to reconnect the stub bus to enable restoration of RHR pumps, and the unavailability of several pumps that were under maintenance.
POS 9-	Sequence 5 has a low CDF as a result of longer allowable time for recovery of AC power. This is due to lower decay heat level in the RCS because of the fresh fuel load after refueling. Sequence 9 has a high CDF due to the RCS loops being completely isolated.
POS 10-	Same as POS 9.
POS 11-	Sequence 5 has low CDF because the RCS is in vented conditions, i.e. pressurized safety valves are removed, and this provides a "bleed" path for decay heat removal. Sequence 9 has a medium-ranked CDF as a result of operator misdiagnosis, failure to reconnect the stub bus to enable restoration of RHR pumps, and failures to cross-tie AFW and charging pump flow.
POS 12-	Same as POS 11.
POS 13-	Sequence 9 has low CDF as a result of the longer allowable time for recovery of AC power.
POS 14-	Same as POS 2.
POS 15-	Same as POS 1.

6.5.4.2 Station Blackout at Unit 1

POS 1-	All core damage sequences at this particular POS have low CDFs because of the short duration of POS 1 at low power operation.
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- POS 2- Sequences 3 and 5 have medium-ranked CDFs as a result of the longer duration of POS after the descent from at-power operation to cold shutdown. All other sequences in the event tree model for this POS have low CDFs.
- POS 3- All core damage sequences at this particular POS have low CDFs ($<5E-8/\text{yr}$) because of the short duration of POS 3.
- POS 4- All core damage sequences at this particular POS have medium-ranked CDFs due to these failure events: (a) operator failure to cross-tie Unit 2 charging pump flow to Unit 1 to achieve successful primary feed and bleed operation, (b) the inability to close the condenser isolation valves, and (c) the unavailability of primary PORVs due to maintenance. Operator failure to achieve SG depressurization also contribute to the medium-ranked accident sequence frequencies.
- POS 5- Same as POS 4.
- POS 6- Sequence has a medium-ranked CDF as a result of these failure events: - (a) operator failure to cross-tie Unit 2 charging system to Unit 1, and the inability to close the condenser isolation valves. Sequence 5 has a low CDF because the RCS is in vented conditions (e.g. pressurizer safety valves were removed) to allow a "bleed" path to exist. Therefore, gravity feed of water inventory to the RCS from RWST helps to achieve successful decay heat removal. Sequence 9 has a high CDF as a result of these events: (a) operator failure to cross-tie Unit 2 charging system to Unit 1 to provide RCS makeup, (b) the inability to close the condenser isolation valves, and (c) the fraction of time in which RCS loops are isolated. The impact of isolated RCS loops is a loss of secondary heat removal capability.
- POS 7- Sequence 4 has a medium-ranked CDF as a result of these events: (a) refueling cavity seal failure which depletes the reactor vessel inventory, and (b) operator failure to open the instrument air supply isolation valves to the refueling cavity seal. Sequence 5 has a low CDF because gravity feed of water inventory from RWST helps to replenish the depleted reactor vessel inventory before AC power is restored to prevent core uncover.
- POS 8- All sequences at this particular POS have medium-ranked CDFs because of the longer duration of POS 8.
- POS 9- Same as POS 7.
- POS 10- Same as POS 6; except that sequence 9 has a medium-ranked CDF because of the longer allowable time for AC power recovery. The decay heat levels are lower in POS 10 than in POS 6.
- POS 11- Sequence 4 has a low CDF as a result of longer allowable time for recovery of AC power.
- POS 12- Sequence 12 has a medium-ranked CDF due to these failure events: (a) operator failure to cross-tie Unit 2 charging system to Unit 1 to provide RCS makeup, (b) operator failure to achieve SG depressurization, (c) the inability to close the condenser isolation valves, and (d) the unavailability of CPC service water pumps due to maintenance.
- POS 13- Same as POS 11.
- POS 14- Sequence 4 has a medium-ranked CDF while sequences 6 and 8 have low CDFs. In sequence 4, the important failure events are: (a) operator failure to cross-tie Unit 2 charging system to Unit 1 to provide RCS makeup, and (b) the inability to close the condenser isolation valves. Successful operator actions to achieve SG depressurization in sequences 6 and 8 help to decrease the CDFs.
- POS 15- Same as POS 14.

6.5.4.3 Station Blackout at Both Units.

- POS 1- All core damage sequences at this particular POS have low CDFs ($<1E-10/\text{yr}$) because of the short duration of POS 1 at low power operation.

POS 2-	All core damage sequences at this particular POS have low CDFs ($<1E-8/\text{yr}$) as a result of low conditional probabilities of all three diesel generators being unavailable.
POS 3-	Sequence 2 has a medium ranked CDF because the conditional probability of all three diesel generators being unavailable is higher.
POS 5-	Same as POS 3.
POS 6-	Sequences 2 and 4 have low CDFs because the RCS is in vented conditions and gravity feed of RWST inventory to provide RCS makeup helps to achieve successful decay heat removal. Sequence 7 has a medium-ranked CDF due to operator failure to achieve SG depressurization for successful decay heat removal.
POS 7-	Sequence 4 has a medium-ranked CDF as a result of these events: (a) refueling cavity seal failure which depletes the reactor vessel inventory and (b) operator failure to open the instrument air supply isolation valves to the refueling cavity seal. Sequence 5 has a low CDF because gravity feed of RWST inventory helps to replenish the depleted reactor vessel inventory prior to recovery of AC power.
POS 8-	Sequences 4 and 5 have low CDFs because of the longer allowable time for recovery of AC power and the low conditional probability of all 3 diesel generators being unavailable.
POS 9-	Same as POS 8.
POS 10-	Same as POS 9.
POS 11-	Sequence 3 has a low CDF because of low conditional probability of all 3 diesel generators being unavailable.
POS 12-	Same as POS 10.
POS 13-	Same as POS 11
POS 14-	Same as POS 2.
POS 15-	Same as POS 1.

LOP12	RSD	RCI	AFW	SL-CL	CCW	HPI	PRV	CS	CV	LPR	HPR	SEQ #	END-STATE-NAMES
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Figure 6.5-1 Event Tree for Loss of Offsite Power-POs 1,2,14 and 15

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SB12	NR0.5	DCI	E-CMO	NRAC1	SGI	SGFW	D-DPR	SEALI	PRV	NRACS	NRACD	REQ /	END-STATE-NAMES
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Figure 6.5-3 Event Tree for Unit 1 Station Blackout - POS 1 and 2

SB345	O-DPR	XFNBF	NRACD	SEQ #	END-STATE-NAMES
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1. SEALI is not important; it provides an additional path for bleeding the primary system.
2. SGFW is not considered; AFW valves inside containment are closed and no entry is assumed because a potential for PRT rupture exists.
3. Relief capability of PORVs and RHR RV at 2-phase flow during SBO is questioned. PORVs fail at 4 hours.
4. No automatic relief capability for the secondary side; relief setpoints correspond to operating conditions.
5. Covers POSs 3, 4, and 5.

Figure 6.5-4 Event Tree for Unit 1 Station Blackout - POSs 3,4 and 5

SB-ML	RCSV	SGFNB	2-CHG	FNBG	NRACD	SEQ #	END-STATE-NAMES
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Figure 6.5-5 Even Tree for Unit 1 Station Blackout-POs 6 and 10

SB789	CV-SL	FNBG	NRACD	SEQ #	END-STATE-NAMES
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Figure 6.5-6 Even Tree for Unit 1 Station Blackout-POs 7, 8 and 9

SB13	O-DPR	XFNBF	NRACD	SEQ #	END-STATE-NAMES
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1. SEALI is not important; it provides an additional path for bleeding the primary system.
2. No automatic relief capability for the secondary side; relief setpoints corresponds to operating conditions.
3. Covers POSs 11, 12, and 13.

Figure 6.5-7 Event Tree for Unit 1 Station Blackout-POs 11,12 and 13

SB145	2-CHG	O-DPR	SEALI	NRACS	NRACD	SEQ #	END-STATE-NAMES
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I-201

Figure 6.5-8 Event Tree for Unit 1 Station Blackout-POs 14 and 15

SB212	NR0.5	RCI	NRAC1	SGI	O-DPR	SEALI	NRACS	NRACD	SEQ #	END-STATE-NA
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I-202

Figure 6.5-9 Event Tree for Unit 2 Unit Station Blackout-POs 1 and 2

SB23	OD	NRAC6	SEQ #	END-STATE-NAMES
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I-203

1. SEALI is not important; it provides an additional path for bleeding the primary system.
2. SGFW is not considered; AFW valves inside containment are closed and no entry is assumed because a potential for PRT rupture exists.
3. Relief capability of PORVs and RHR RV at 2-phase flow during SBO is questioned. PORVs fail at 4 hours.
4. No automatic relief capability for the secondary side; relief setpoints correspond to operating conditions.
5. Covers POSs 3, 4, and 5.

Figure 6.5-10 Event Tree for Unit 2 Station Blackout-POSs 3,4 and 5

SB26	RCSV6	OD	FNBG2	NRA24	SEQ #	END-STATE-NAMES
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Figure 6.5-11 Event Tree for Unit 2 Station Blackout-POs 6 and 10

SB27	CVSL7	FNBG2	NRA24	SEQ #	END-STATE-NAMES
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Figure 6.5-12 Event Tree for Unit 2 Station Blackout-POs 7,8 and 9

SB211	OD	NRAC1	SEQ #	END-STATE-NAMES
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1. SEALI is not important; it provides an additional path for bleeding the primary system.
2. No automatic relief capability for the secondary side; relief setpoints corresponds to operating conditions.
3. Covers POSs 11, 12, and 13.

Figure 6.5-13 Event Tree for Unit 2 Station Blackout-POs 11,12, and 13

SB215	O-DPR	SEALI	NRACS	NRACD	SEQ #	END-STATE-NAMES
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Figure 6.5-14 Event Tre for Unit 2 Station Blackout-POs 14 and 14

Table 6.5-1
Conditional Probability of Station Blackout

STATE	SBO-U1	SBO-U2
1	3.695E-3	4.857E-4
2	3.695E-3	4.857E-4
3	3.695E-3	4.857E-4
4	1.400E-2	7.685E-4
5	9.027E-3	6.317E-4
6	3.923E-2	1.469E-3
7	1.224E-2	7.201E-4
8	2.339E-2	1.530E-3
9	1.525E-2	1.133E-3
10	2.619E-2	2.267E-3
11	1.000E-2	7.775E-4
12	2.137E-2	1.398E-3
13	3.695E-3	4.857E-4
14	3.695E-3	4.857E-4
15	3.695E-3	4.857E-4

6.6 Loss of Residual Heat Removal Event Trees

6.6.1 Introduction

The loss of RHR event tree in figure 6.6-1 was developed based on abnormal procedure 1-AP-27.00 for loss of decay heat removal capability. This procedure is the only abnormal procedure written specifically for shutdown conditions. It is applicable to any loss of decay heat removal condition especially the mid-loop condition. The event tree is a translation of the abnormal procedure into a logical format. It is a generic tree that is specialized for different loss of RHR initiating event categories and different plant operational states. It is applicable to POSs 3 to 13 in which RHR is in operation.

The RHR system is placed in service when the primary system temperature is approximately 350 degree F, and pressure is approximately 300 psig. Normally, one RHR pump is in operation with the other pump on standby. The following alarms may be available upon loss of RHR:

- a. RHR HX outlet high temperature alarm at 340 degree F,
- b. RHR HX low flow alarm at 3200 gpm,
- c. RHR pump OL trip alarm,
- d. Shutdown cooling low level alarm,
- e. Containment CC outlet header low flow alarm,
- f. RHR pumps discharge high pressure MPT alert alarm at 480 psig.

Upon entry to the loss of RHR procedure, the operator will first check the RCS inventory. If the inventory is decreasing, the operator will try to stop inventory loss by isolating any known draining, the RHR letdown line (valve 1-RH-HCV-1142), and the RCS loop drains. If the inventory loss is stopped, or no inventory loss occurred, attempt will be made to restore the RHR system. This includes possible inventory makeup, restarting RHR pump, and establishing component cooling water to the RHR heat exchanger. If RHR can not be restored, the operator will try to establish feed to the steam generators, and dump steam to the main condenser or through the SG PORVs. Continuing attempt will be made to restore the RHR system. If time to boiling is less than 2 hours, containment closure will be initiated. The following alternate decay heat removal methods can possibly be used:

1. Feed and bleed the RCS- Feeding can be performed with one of the charging pumps, low head safety injection pumps, and possibly the unit 2 charging pumps by cross tying. Bleeding can be done by opening a PORV.
2. Gravity feed from RWST- This is done through the low head safety injection flow path. It is possible only if the RCS is vented with removal of the pressurizer safety valves, pressurizer manway, steam generator manway, or vessel head.
3. Secondary heat removal- This is done by feeding the steam generators with auxiliary feedwater pumps, auxiliary feedwater booster pumps, or the firemain, and dumping the steam to condenser or through the SG PORVs. The primary side can be in natural circulation or reflux cooling (fall back cooling).
4. Steam Generator Recirculation and Transfer System- If the RCS temperature is below 200 degrees F, the SG recirculation and transfer system can be used to remove heat from the secondary side of the steam generators. Two SG recirculation and transfer coolers are effective in removing decay heat 35 hours after shutdown. One cooler will be effective after 200 hours.

6.6.2 Top Events of the Generic Loss of RHR Event Tree

RCSIN- Reactor Coolant Inventory

This event applies to mid-loop condition(POSS 6 and 10), in which a small inventory loss may lead to loss of the running RHR pump. The lower branch represents a loss of inventory that was the cause of loss of RHR. The upper branch represents no initial inventory problem associated with the initiating event. For other POSS, the operability of the RHR system is not sensitive to the RCS inventory. Therefore, this question is not asked.

R C SMU-Reactor Coolant System Makeup

If a loss inventory occurs and causes a loss of RHR when the RCS is at mid-loop, this top event is used to model the RCS makeup that is needed to restore inventory such that RHR can be restored. This can be done by using a charging pump, a low head injection pump, gravity feed or volume control tank overpressure feed.

RSTRH- Restore RHR

This top event represents restoration of the RHR system. Depending on the specific loss of RHR category and the POSS, the restoration of RHR is modelled accordingly.

RCSV- RCS Vented

This top event represents the situation in which the RCS is vented, by removing the pressurizer safety valves, pressurizer manway, steam generator manway, or vessel head. This could happen when the RCS is not pressurized, i.e., POSS 5,6,7,8,9,10, and 11.

With such vent, it is possible to use gravity feed to provide makeup to the RCS.

SGFNB- Steam Generator Feed and Bleed

This top event represents heat removal from the secondary side of the steam generators. Primary side can undergo natural circulation, or reflux cooling(fall back cooling) if the RCS is in mid-loop condition. This would be sufficient to remove the decay heat from the core. Feed to the SGs can be done using auxiliary feedwater pumps, auxiliary feedwater booster pumps, main feedwater pumps, condensate pumps, and fire main. Prior to initiating RHR operation, MOVs FW151A-F in the auxiliary feedwater lines are closed. Therefore, the operator will have to manually reopen these valves to establish the auxiliary feed water flow paths needed. If the RCS temperature is below 150 degree F(POSS 5-11), the secondary heat removal can also be done using the SG recirculation and transfer system. Otherwise, secondary steam can be dumped to the condenser or through the secondary PORVs to the atmosphere.

FNBF- Feed and Bleed by Force

This event represents feeding RCS using charging pumps, or low head injection pumps and bleeding through PORVs or RHR relief valve. In case of overpressurization of the RHR system causing a rupture, this top event represents manual actuation of low head injection, and low head recirculation.

FNBG- Feed and Bleed by Gravity

This event represents feeding the RCS from RWST by gravity through the low head injection flow path. It can be used when the RCS is vented(RCSV) for long term (24 hours mission time) decay heat removal.

6.6.3 Sequence Description

Sequence 1 is a success sequence, in which no inventory loss occurs and the RHR is restored. Sequence 2 represents the scenario in which RHR is not restored and primary feed and bleed is used to remove decay heat. In this sequence, the RCS is vented with a large vent, e.g. safety valves removed. Therefore, bleed path

always exists, and feeding with any available pump is sufficient. In sequence 3, no makeup pump is available, and gravity feed from the RWST is used. Due to large inventory in the RWST, it is assumed that this will allow sufficient time for the operators to establish other method of makeup or decay heat removal. As is discussed in section 5.2.2, a more detailed thermal hydraulic analysis may be needed to verify the feasibility of this mode of long term decay heat removal. Sequence 4 represents the scenario in which the RCS is vented, no makeup to the RCS is established, and the RCS boil off until core uncover occurs. In sequence 5, RHR is lost and not restored, the RCS is not vented with a large opening, and secondary cooling is established to remove decay heat. In sequences 6 and 7, no secondary heat removal is established or maintained and RCS is closed. This has the potential to lead to an overpressurization of the RHR system, and a resulting large LOCA. In sequence 6, operator establishes low pressure injection and recirculation to prevent core uncover. In sequence 7, the operator fails to do so.

Sequences 8 to 14 start with an inventory loss with the plant at mid-loop condition. The operator isolates the loss and restores RCS inventory. The rest of these sequences are similar to sequences 1 to 7. In sequence 12, reflood cooling(fall back cooling) is established to remove decay heat. In sequence 14, RCS boil off takes place that leads to core uncover due to loss of inventory makeup and secondary heat removal. In this sequence, PORVs will be bleeding steam and maintain system pressure below 365 psig and prevent overpressurization of the RHR system.

Sequences 15 to 18 represent scenarios in which a loss of inventory with the plant at mid-loop occurs, and the operator fails to isolate the leak. The inventory loss continues until the RCS level drops below the vessel nozzles. At this point, reflux cooling can be established if secondary heat removal is available and the RCS is not vented(sequence 17). If the RCS is vented, gravity feed is assumed to be sufficient for heat removal for 24 hours (sequence 15).

6.6.4 Success Criteria for Each Plant Operational State

Table 6.6-1 lists the success criteria of the loss of RHR event tree for all applicable POSs in all types of outages. Most of the headings in the table correspond to the top events in the event tree. The RCS integrity heading in the table represents the need to prevent overpressurization of the RHR system as a result of boiling in the vessel and the inability of the PORVs and RHR RV to relief pressure. This may happen when secondary cooling is lost along with loss of RHR. Each of the PORVs has a capacity of 700 gpm of liquid flow at 365 psig and 900 gpm at 600 psig. The RHR relief valve has a capacity of 750 gpm at 600 psig. The combined capacity of these valves is equivalent to the steam generated with 5.8 Mw of decay heat.(17 days after shutdown) Therefore, before 17 days after shutdown, such scenario may lead to a rupture of the RHR system, and resulting flashing of the primary coolant. After 17 days, the scenario becomes that of a boil off of the RCS inventory, if the relief valves work properly.

It is assumed that the turbine driven auxiliary feedwater pump is operable in those POSs in which the RCS temperature is above 200 degrees F, i.e. POSs 3 and 13. In other POSs, it is not operable. In POSs 7 to 10, the RCS loops are isolated. Therefore, the steam generators are not available. This is modelled in the fault trees for SGFNB.

In those POSs in which the RCS temperature is below 200 degrees F, i.e., POSs 4 to 12, no containment integrity is required and activities inside the containment are expected. Therefore, it is assumed that if recirculation from the containment sump is needed, the probability that the sump is plugged is high, due to debris and transient material that are brought inside the containment. A probability of 0.1 was used for such failure to recirculate.

Specialization of the Loss of RHR Event Tree

The specialization of the generic loss of RHR event tree is discussed in the following:

a. The RCSINV question applies only to RHR2A and RHR2B categories with the plant at mid-loop conditions (POS 6 and 10). For other loss of RHR categories and other POSs, only sequences 1 to 7 are used. The RCSMU question is applicable only if RCSINV is failed.

b. In those POSs in which the RCS is pressurized, POSs 3, 4, 12 and 13, the RCS can not be vented. Therefore, sequences 2 to 4 are not applicable and only one core damage sequence, sequence 8, is applicable.

c. In POS 8, time to core uncover due to loss of RHR is more than 1 days. Therefore, no loss of RHR scenario need to be analyzed.

Application of the above resulted in the following scenarios and sequences for the plant operational states:

Refueling Outage

POS 3- In this POS, one reactor coolant pump is normally running, and the main steam trip valves are open with steaming to the condenser. Upon loss of RHR, with no other failure and no operator actions, steaming to the condenser should continue and take over the heat load from the RHR system. It is estimated that this mode of heat removal can continue for approximately 6 hours, before steam generator dry out occurs. This allow sufficient time for the operator to establish auxiliary feedwater to the steam generator or feed and bleed to the reactor coolant system.

POS 4, POS 5- In these POSs, the main steam trip valves are normally closed. Upon loss of RHR, without operator action, the primary coolant will heat up and bring the system solid or saturated in less than an hour and challenge the PORVs and RHR relief valve. Due to the high decay heat, the valves are not expected to relieve the pressure. It is assumed that a rupture of the RHR will occur and require timely actuation of the low head safety injection system and subsequently the low pressure recirculation system.

POS 6, POS 10- In these POSs, the reactor coolant system is at mid-loop. Upon loss of RHR, core uncover will occur in 1.5 and 5.6 hours respectively, due to boil off of the coolant. The descriptions of the generic loss of RHR event tree apply. Due to the use of tygon tube that has a low design pressure, and the fact that the RCS is either vented through the pressurizer relief tank or pressurized with nitrogen blanket with the PORVs open, it is assumed that the RCS pressure will not challenge the RHR system integrity.

POS 7- In this POS, it is assumed that the vessel level is at the vessel flange. Besides, higher vessel level, this POS is similar to POS 6. Upon loss of RHR, it takes approximately 2.2 hours to reach core uncover, due to boil off.

POS 8- In this POS, the reactor cavity is filled, and the time to core uncover due to boil off is approximately 81 hours. Therefore, it was assumed that sufficient time is available for mitigating the initiating event.

POS 9, POS 11- These POSs are similar to POS 7 except decay heat is much lower. It is assumed that the level is initially at the vessel flange. The time to core uncover is estimated to be 5.6, and 6 hours respectively.

POS 12- In this POS, one reactor coolant pump is running, with RCS still under 200 degrees F. The secondary side is isolated from the condenser. Therefore, upon loss of RHR with no operator action, the primary and secondary will heat up at the same rate. It is assumed that the RCS becomes solid in approximately 2.5 hours. In sequence 8 of the event tree, the possibility that PORVs and RHR RV fail to relieve pressure was modelled. It is assumed that 2 out of the 3 relief valves are needed to prevent over pressurization.

POS 13- In this POS, 3 RCPs are running with the secondary side steaming to the condenser. Upon loss of RHR, the steam generators should be able to take over the heat load. It is assumed that the RCPs continue

running contributing to a total heat load of 10 Mw. Therefore, the secondary side will dry out in 12 hours, if no makeup is established. On the primary side, feed and bleed can also prevent core uncover.

Drained Maintenance Outage

POSS 1-6 - Same as Refueling outage.

POS 11 -In this POS, the reactor is shutdown for only as short as 34 hours. The decay heat is 11 Mw, much higher than the POS 11 of a refueling outage. The secondary side of the steam generator should be filled and the RCS loops unisolated. It is assumed that the vessel level is at the vessel flange, and the time to core uncover due to heat up and boil off of the primary side is approximately 1.8 hours.

POS 12-This POS is similar to POS 12 of refueling except that the decay heat is much higher, i.e. 10.3 Mw. It is estimated that in 1.4 hours the RCS becomes solid and challenge to the relief valves occur. At this decay heat level, the relief valves are not capable of relieving pressure. The operator will have to establish secondary heat removal by opening the secondary relief valves prior to primary pressure build up.

POS 13- This POS is similar to POS 13 of a refueling outage, except that the decay heat is higher, i.e.,9.2Mw. Assuming that the RCPs are running with 8 Mw of power, a total of 17.2 Mw needs to be removed. It was estimated that in 8.2 hours the steam generators will dry out. In another 1.8 hours, core uncover will occur due to primary boil off with 9.2 Mw of decay heat.

Non-Drained Maintenance with RHR

POS 3 and 4- Same as refueling.

6.6.5 Quantification of Loss of RHR Event Tree

Re-Grouping of loss of RHR Categories

The loss of RHR event tree is quantified for each POS accounting for the specific plant configuration and maintenance unavailability. Table 3-6 lists the applicable loss of RHR categories of the POSS. The loss of RHR categories of table 3-7 are further grouped based on their impact on the systems. Table 6.6-2 lists the further grouped categories. Table 6.6-3 shows their applicability to the POSS. The impact of the RHR categories are the following:

RHR2A-This event occurs at the beginning of POSS 6 and 10, due to failure of over draining. It is applicable only to POS 6 and 10. The initiating event causes a loss of RHR with possibility of pump cavitation. It takes the scenario to sequences 8 to 18 of the generic loss of RHR event tree.

RHR2B-This event may occur any time when the RCS is at mid-loop, e.g. POSS 6 and 10, due to failure to maintain inventory. Similar to RHR2A, this event is only applicable to POSS 6 and 10.

RHR3B-This event represents diversion of flow through the RHR system. It is covered in LOCA and interfacing system LOCA analysis, and it not further analyzed here.

RHR4-This category represents a loss of RHR event that can not be recovered.

RHRT1-This category represents failure of the running RHR pump. The failure is not recoverable.

RHR6B-This category represents tripping of the running RHR pump. It is assumed that the interruption of the RHR can be easily recovered.

RHR8-This event occurs at the beginning of POS 3. Due to hardware failure of the RHR system, RHR can not be initiated. The plant can simply continue heat removal through the steam generators.

RHR12-This category represents the scenario in which RCP and SG were used to remove decay heat instead of RHR system, and the RCP fails to continue running. In POSs 3,4,12 and 13, at least one RCP is running. These are the applicable POSs for this category. It is assumed the RCP and SG are used because the RHR system is not available.

Human Reliability Analysis (HRA)

The approach in HRA is discussed in Chapter 7. Its application to loss of RHR event tree is summarized below:

In the loss of RHR event tree, top events RCSMU, RSTRH, SGFNB, FNBF, and FNBG require operator actions. Two types of operator errors are modeled, mis-diagnosis and failure to carry out the needed actions. For each core damage sequence, a single diagnosis error is used. This failure is assumed to result in failure of all the operator actions in the sequence. Within the fault trees of each event tree top event, operator failure to performed the needed actions is modeled. It is assumed that failures to perform actions in the sequence are independent.

Quantification of the human error probabilities (HEPs) is performed specifically to account for the time available. First, the total time available for operator actions is determined. This is usually the time to core uncover in the sequence. The time needed to perform the needed actions is estimated and subtracted from the total time available. The remaining time is used to determine the probability of failure to diagnose using the time curves for diagnosis. Quantification of the HEPs associated with taking needed actions is done based on whether or not the action is performed in the control room and whether or not the action is complex or simple. Tables 7-1 and 7-2 list the HEPs for these classes of operator actions as well as the HEPs associated with diagnosis.

Table 6.6-4 lists the estimated time to core uncover, time needed to take action, and time to diagnosis, and their associated HEPs.

6.6.6 Results

This section summarizes the results of the loss of RHR event trees for each POS is a refueling outage. The quantified sequences are summarized in tables 10-4a to 10-4c.

POS 3 -Due to relative short duration, low maintenance unavailability, and the fact that the steam generators are readily available to pick up the decay heat load, the estimated core damage frequency is low.

POS 4 -In this POS, the secondary side of the steam generators is isolated, and timely operator actions, in 42 minutes, are needed to prevent potential over pressurization of the RHR system. Operators can easily open the secondary side by opening the atmospheric dump valve, or close the RHR suction valves to isolate the RHR system. Operator failure to respond to the loss of RHR initiator is the dominant cause of core damage. Upon failure of the RHR system due to over pressurization, a LOCA is assumed to occur. Due to the use of a high probability for containment sump plugging, the low pressure recirculation is not very reliable. The estimated core damage frequency is medium.

POS 5 -This POS is similar to POS 4. It is assumed that the initiating event occurs when the system is filled, with pressurizer near solid. The initial heat up will cause the PORVs to open. They are capable of relieving the liquid during the phase the reactor coolant is expanding thermally. When the system temperature reaches

saturation, the voiding in the vessel causes challenge to the relief valves, and over pressurization of the RHR system. The estimated core damage frequency is medium.

POS 6 - Due to the small inventory in the system and the isolation of the reactor coolant loops, the only way to mitigate the loss of RHR event is to feed and bleed the reactor coolant system. Operator failure to do so, is the dominant cause for core damage. The estimated core damage frequency is high.

POS 7 -This POS is similar to POS 6 except that the inventory is assumed to be at the vessel flange. The estimated core damage frequency is high.

POS 10 -This POS is similar to POS except that the decay heat is lower and maintenance unavailability is higher. The estimated core damage frequency is high.

POS 12 -In this POS, the decay heat is low, the duration is long, and the maintenance unavailability is high. As a result, the probability of loss of RHR is high, and the estimated core damage frequency is high.

POS 13 -Due to low decay heat, low maintenance unavailability, and the fact that the secondary is readily available to take over the decay heat load, the estimated core damage frequency is low.

IE-RH	RCSIN	RCSMU	RSTRH	RCSV	SGFNB	FNBF	FNBG	SEQ #	END-STATE-NAMES
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Figure 6.6-1 Loss of RHR Event Tree - Generic

**Table 6.6-1 Success Criteria for Loss of RHR Event Tree
Refueling Outage**

	RCSMU	RCS Integrity	SGFNB	FNBF	FNBG
POS3	N.A.	secondary cooling	1/3 AFW pumps	1/3 charging pumps AND 1/2 PORVs OR RHR RV	N.A.
POS4	N.A.	secondary cooling	1/2 AFW pumps	1/2 LPI AND 1/2 LPR	N.A.
POS5	N.A.	secondary cooling	1/2 AFW pumps	1/2 LPI AND 1/2 LPR	N.A.
POS6	1/3 charging pumps OR 1/2 LPI pumps OR gravity feed	N.A.	1/2 AFW pumps	1/3 charging pumps OR 1/2 LPI	Yes
POS7	N.A.	N.A.	N.A. (loops isolated)	1/3 charging pumps OR 1/2 LPI	Yes
	RCSMU	RCS Integrity	SGFNB	FNBF	FNBG
POS8	N.A.	N.A.	N.A.	1/3 charging pumps OR 1/2 LPI	N.A.
POS9	N.A.	N.A.	N.A.	1/3 charging pumps OR 1/2 LPI	Yes
POS10	1/3 charging	N.A.	N.A.	1/3 charging	Yes

Table 6.6-1 Success Criteria for Loss of RHR Event Tree (Cont'd.)

Refueling Outage

	pumps OR 1/2 LPI pumps OR gravity feed			pumps OR 1/2 LPI	
POS11	N.A.	N.A.	1/2 AFW pumps	1/3 charging pumps OR 1/2 LPI	Yes

Table 6.6-1 Success Criteria for Loss of RHR Event Tree (Cont'd.)

<u>Refueling Outage</u>					
	RCSMU	RCS Integrity	SGFNB	FNBF	FNBG
POS12	N.A.	secondary cooling	1/2 AFW pumps	1/2 LPI AND 1/2 LPR	N.A.
POS13	N.A.	secondary cooling	1/3 AFW pumps	1/3 charging pumps AND 1/2 PORVs OR RHR RV	N.A.
<u>Drained Maintenance</u>					
POS3	N.A.	secondary cooling	1/3 AFW pumps	1/3 charging pumps AND 1/2 PORVs OR RHR RV	N.A.
POS4	N.A.	secondary cooling	1/2 AFW pumps	1/2 LPI AND 1/2 LPR	N.A.
POS5	N.A.	secondary cooling	1/2 AFW pumps	1/2 LPI AND 1/2 LPR	N.A.
	RCSMU	RCS Integrity	SGFNB	FNBF	FNBG
POS6	1/3 charging pumps OR 1/2 LPI pumps OR gravity feed	N.A. pumps	1/2 AFW pumps	1/3 charging OR 1/2 LPI	Yes
POS11	N.A.	N.A.	1/2 AFW pumps	1/3 charging pumps OR 1/2 LPI	Yes
POS12	N.A.	secondary	1/2 AFW	1/2 LPI	N.A.

Table 6.6-1 Success Criteria for Loss of RHR Event Tree (Cont'd.)

<u>Refueling Outage</u>					
		cooling	pumps	AND 1/2 LPR	
POS13	N.A.	secondary cooling	1/3 AFW pumps AND 1/2 PORVs OR RHR RV	1/3 charging pump	N.A.
<u>Non-Drained Maintenance</u>					
POS3	N.A.	secondary cooling	1/3 AFW pumps	1/3 charging pumps AND 1/2 PORVs OR RHR RV	N.A.
	RCSMU	RCS Integrity	SGFNB	FNBF	FNBG
POS4	N.A.	secondary cooling	1/2 AFW pumps	1/2 LPI AND 1/2 LPR	N.A.

Table 6.6-2 Loss of RHR Categories

RHR2A: Over-Draining during drain down to midloop	1.23E-02/Demand
RHR2B: Failure to Maintain Level at Midloop	8.40E-06/hr
RHR4 : Loss of Both RHR Trains-Non Recoverable	6.61E-05/hr
RHRT1: Loss of One RHR Train-Non Recoverable	1.34E-05/hr
RHR6B: Loss of Running RHR Train-Recoverable	2.52E-02/hr
RHR8 : Failure to Initiate RHR in POS 3	5.23E-03/demand
RHR12: Loss of RCP while at Cold Shutdown	2.81E-05/hr

Table 6.6-3
Applicable Loss of RHR Categories Plant Operational St

Plant Operational States											
	3	4	5	6	7	8	9	10	11	12	13
RHR2A				6				10			
RHR2B				6				10			
RHR4	3	4	5	6	7	8	9	10	11	12	13
RHRT1	3	4	5	6	7	8	9	10	11	12	13
RHR6B	3	4	5	6	7	8	9	10	11	12	13
RHR8	3										
RHR12	3	4								12	13

Table 6.6-4 HEPs Used in Loss of RHR Event Tree

POS 3. Refueling-Time to Core Uncovery=6 hours

RHR4:					
POS-Seq	Action	HEP_a	T_a	T_d	HEP_d
POS3-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	>60min.	0.0001
	FNBF	0.001	10.min.		
RHRT1:					
POS3-7	RSTRHR	0.001	10.min.		
	SGFNF	0.001	10.min.	>60min.	0.0001
	FNBF	0.001	10.min.		
RHR6B:					
POS3-7	RSTRHR	0.0001	5.min.		
	SGFNF	0.001	10.min	>60min.	0.0001
	FNBF	0.001	10.min.		
RHR8:					
POS3-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	>60min.	0.0001
	FNBF	0.001	10.min.		
RHR12:					
POS3-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	>60min.	0.0001
	FNBF	0.001	10.min.		

POS 4. Refueling-Time to Core Uncovery = 42 minutes

RHR4:					
POS-Seq	Action	HEP_a	T_a	T_d	HEP_d
POS4-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	32min.	0.001
	FNBF	0.001	10.min.		
RHRT1:					
POS4-7	RSTRHR	0.001	10.min.		
	SGFNF	0.001	10.min.	32min.	0.001
	FNBF	0.001	10.min.		
RHR6B:					
POS4-7	RSTRHR	0.0001	5.min.		
	SGFNF	0.001	10.min	32min.	0.001
	FNBF	0.001	10.min.		
RHR12:					
POS4-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	32min.	0.001
	FNBF	0.001	10.min.		

POS 5. Refueling-Time to Core Uncovery=60 minutes

RHR4:					
POS-Seq	Action	HEP_a	T_a	T_d	HEP_d
POS5-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	50min.	0.0002
	FNBF	0.001	10.min.		
RHRT1:					
POS5-7	RSTRHR	0.001	10.min.		
	SGFNF	0.001	10.min.	50min.	0.0002
	FNBF	0.001	10.min.		
RHR6B:					
POS5-7	RSTRHR	0.0001	5.min.		
	SGFNF	0.001	10.min	50min.	0.0002
	FNBF	0.001	10.min.		

POS 6, Refueling-Time to Core Uncovery=1.5 hour**RHR2A,RHR2B:**

POS-Seq	Action	HEP _a	T _a	T _d	HEP _d
POS6-11	RSTRHR	0.01	60.min.	30min.	0.001
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.		
POS6-14	RSTRHR	0.01	60.min.	30.min.	0.001
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.		
POS6-16	RSTRHR	0.01	60.min.	30min.	0.001
	SGFNB	0.001	10.min.		
	FNBF	0.001	10.min.		
POS6-18	RCSMU	0.001	10.min.	30min.	0.001
	RSTRHR	0.01	60.min.		
	SGFNB	0.001	10.min		
RHR4:					
POS6-4	RSTRHR	1.00	-	> 60.min.	0.0001
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.		
POS6-7	RSTRHR	1.00	-	> 60.min	0.0001
	SGFNB	0.001	10.min.		
	FNBF	0.001	10.min.		
RHRT1:					
POS6-4	RSTRHR	0.001	10.min.	> 60.min.	0.0001
	FNBF	0.001	10.min.		
	FNBG	0.005	10.min.		
POS6-7	RSTRHR	0.001	10.min.	> 60min.	0.0001
	SGFNB	0.001	10.min.		
	FNBF	0.001	10.min.		
RHR6B: POS6-4					
	RSTRHR	0.0001	5.min.	> 60.min.	0.0001
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.		
POS6-7	RESRHR	0.0001	5.min.	> 60.min	0.0001
	SGFNB	0.001	10.min.		
	FNBF	0.001	10.min.		

POS 7, Refueling-Time to Core Uncovery=2.2 hour**RHR4:**

POS-Seq	Action	HEP _a	T _a	T _d	HEP _d
POS7-4	RSTRHR	1.00	-	> 60.min.	0.0001
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.		
POS7-7	RSTRHR	1.00			

	FNBF	0.001	10.min.	> 60.min	0.0001
RHRT1:					
POS7-4	RSTRHR	0.001	10.min.		
	FNBF	0.001	10.min.		
	FNBG	0.005	10.min.	> 60.min.	0.0001
POS7-7	RSTRHR	0.001	10.min.	> 60min.	0.0001
	FNBF	0.001	10.min.		
RHR6B:					
POS7-4	RSTRHR	0.0001	5.min.		
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.	> 60.min.	0.0001
POS7-7	RESRHR	0.0001	5.min.	> 60.min	0.0001
	FNBF	0.001	10.min.		

POS 9. Refueling-Time to Core Uncovery = 5.6 hour

RHR4:

POS-Seq	Action	HEP_a	T_a	T_d	HEP_d
POS9-4	RSTRHR	1.00	-		
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.	>60.min.	0.0001
POS9-7	RSTRHR	1.00	-		
	FNBF	0.001	10.min.	>60.min	0.0001
RHRT1:					
POS9-4	RSTRHR	0.001	10.min.		
	FNBF		0.001	10.min.	
	FNBG		0.005	10.min.	>60.min.0.0001
POS9-7	RSTRHR	0.001	10.min.	>60min.	0.0001
	FNBF			0.001	10.min.
RHR6B:					
POS9-4	RSTRHR	0.0001	5.min.		
	FNBF		0.001	10.min.	
	FNBG		0.005	20.min.	>60.min.0.0001
POS9-7	RESRHR	0.0001	5.min.	>60.min	0.0001
	FNBF	0.001	10.min.		

POS 10, Refueling-Time to Core Uncovery = 5.6 hours**RHR2A,RHR2B:**

POS-Seq	Action	HEP _a	T _a	T _d	HEP _d
POS10-11	RSTRHR	0.01	60.min.	>60.min.	0.0001
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.		
POS10-14	RSTRHR	0.01	60.min.	>60.min.	0.0001
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.		
POS10-16	RSTRHR	0.01	60.min.	>60min.	0.0001
	FNBF	0.001	10.min.		
POS10-18	RCSMU	0.001	10.min.	>60min.	0.0001
	RSTRHR	0.01	60.min.		

RHR4:

POS10-4	RSTRHR	1.00	-	>60.min.	0.0001
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.		

POS10-7	RSTRHR	1.00	-	>60.min	0.0001
	FNBF	0.001	10.min.		

RHRT1:

POS10-4	RSTRHR	0.001	10.min.	>60.min.	0.0001
	FNBF	0.001	10.min.		
	FNBG	0.005	10.min.		

POS10-7	RSTRHR	0.001	10.min.	>60min.	0.0001
	FNBF	0.001	10.min.		

RHR6B:

POS10-4	RSTRHR	0.0001	5.min.	>60.min.	0.0001
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.		

POS10-7	RESRHR	0.0001	5.min.	>60.min	0.0001
	FNBF	0.001	10.min.		

POS 11, Refueling-Time to Core Uncovery = 6.0 hour**RHR4:**

POS-Seq	Action	HEP _a	T _a	T _d	HEP _d
POS11-4	RSTRHR	1.00	-	>60.min.	0.0001
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.		

POS11-7	RSTRHR	1.00			
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	FNBF	0.001	10.min.	> 60.min	0.0001
RHRT1:					
POS11-4	RSTRHR	0.001	10.min.		
	FNBF	0.001	10.min.		
	FNBG	0.005	10.min.	> 60.min.	0.0001
POS11-7	RSTRHR	0.001	10.min.	> 60min.	0.0001
	FNBF	0.001	10.min.		
RHR6B:					
POS11-4	RSTRHR	0.0001	5.min.		
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.	> 60.min.	0.0001
POS11-7	RESRHR	0.0001	5.min.	> 60.min	0.0001
	FNBF	0.001	10.min.		

6.0 hours was the estimated time to core uncover due to primary boil off with decay heat of 3.15 Mw.

POS 12. Refueling-Time to Core Uncovery = 2.5 hours

RHR4:					
POS-Seq	Action	HEP_a	T_a	T_d	HEP_d
POS12-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	> 60min.	0.0001
	FNBF	0.001	10.min.		
RHRT1:					
POS12-7	RSTRHR	0.001	10.min.		
	SGFNF	0.001	10.min.	> 60min.	0.0001
	FNBF	0.001	10.min.		
RHR6B:					
POS12-7	RSTRHR	0.0001	5.min.		
	SGFNF	0.001	10.min	> 60min.	0.0001
	FNBF	0.001	10.min.		
RHR12:					
POS12-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	> 60min.	0.0001
	FNBF	0.001	10.min.		

3.5 hr is the estimated time to over pressurization if secondary side is not vented and 2 PORVs failed.

POS 13. Refueling-Time to Core Uncovery = 12 hours**RHR4:**

POS-Seq	Action	HEP _a	T _a	T _d	HEP _d
POS13-7	RSTRHR	1.00	-		
	SGFNF	0.001	0.001	10.min	> 60min.0.0001
	FNBF	0.001	0.001	10.min.	

RHRT1:

POS13-7	RSTRHR	0.001	10.min.		
	SGFNF	0.001	10.min.	> 60min.	0.0001
	FNBF	0.001	10.min.		

RHR6B:

POS13-7	RSTRHR	0.0001	5.min.		
	SGFNF	0.001	10.min	> 60min.	0.0001
	FNBF	0.001	10.min.		

RHR8:

POS13-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	> 60min.	0.0001
	FNBF	0.001	10.min.		

RHR12:

POS13-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	> 60min.	0.0001
	FNBF	0.001	10.min.		

It is assumed that 3 RCPs are running contributing to a total heat load of 10 Mw. The SGs should be able to remove decay heat for 12 hours prior to dry out.

POS 3, Drained Maintenance-Time to Core Uncovery = 6 hours**RHR4:**

POS-Seq	Action	HEP _a	T _a	T _d	HEP _d
POS3-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	> 60min.	0.0001
	FNBF	0.001	10.min.		

RHRT1:

POS3-7	RSTRHR	0.001	10.min.		
	SGFNF	0.001	10.min.	> 60min.	0.0001
	FNBF	0.001	10.min.		

RHR6B:

POS3-7	RSTRHR	0.0001	5.min.		
	SGFNF	0.001	10.min	> 60min.	0.0001
	FNBF	0.001	10.min.		

RHR8:

POS3-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	> 60min.	0.0001
	FNBF	0.001	10.min.		

RHR12:

POS3-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	> 60min.	0.0001
	FNBF	0.001	10.min.		

POS 4, Drained Maintenance-Time to Core Uncovery=42minutes**RHR4:**

POS-Seq	Action	HEP _a	T _a	T _d	HEP _d
POS4-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	32min.	0.001
	FNBF	0.001	10.min.		

RHRT1:

POS4-7	RSTRHR	0.001	10.min.		
	SGFNF	0.001	10.min.	32min.	0.001
	FNBF	0.001	10.min.		

RHR6B:

POS4-7	RSTRHR	0.0001	5.min.		
	SGFNF	0.001	10.min	32min.	0.001
	FNBF	0.001	10.min.		

RHR12:

POS4-7	RSTRHR	1.00		-	
	SGFNF	0.001	10.min	32min.	0.001
	FNBF	0.001	10.min.		

POS 5, Drained Maintenance-Time to Core Uncovery=60 minutes

RHR4:

POS-Seq	Action	HEP_a	T_a	T_d	HEP_d
POS5-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	50min.	0.0002
	FNBF	0.001	10.min.		

RHRT1:

POS5-7	RSTRHR	0.001	10.min.		
	SGFNF	0.001	10.min.	50min.	0.0002
	FNBF	0.001	10.min.		

RHR6B:

POS5-7	RSTRHR	0.0001	5.min.		
	SGFNF	0.001	10.min	50min.	0.0002
	FNBF	0.001	10.min.		

POS 6. Drained Maintenance-Time to Core Uncovery=1.5 hour

RHR2A,RHR2B:

POS-Seq	Action	HEP _a	T _a	T _d	HEP _d
POS6-11	RSTRHR	0.01	60.min.	30min.	0.001
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.		
POS6-14	RSTRHR	0.01	60.min.	30.min.	0.001
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.		
POS6-16	RSTRHR	0.01	60.min.	30min.	0.001
	SGFNB	0.001	10.min.		
	FNBF	0.001	10.min.		
POS6-18	RCSMU	0.001	10.min.		
	RSTRHR	0.01	60.min.	30min.	0.001
	SGFNB	0.001	10.min.		
RHR4:					
POS6-4	RSTRHR	1.00	-		
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.	> 60.min.	0.0001
POS6-7	RSTRHR	1.00	-		
	SGFNB	0.001	10.min.	> 60.min	0.0001
	FNBF	0.001	10.min.		
RHRT1:					
POS6-4	RSTRHR	0.001	10.min.		
	FNBF	0.001	10.min.		
	FNBG	0.005	10.min.	> 60.min.	0.0001
POS6-7	RSTRHR	0.001	10.min.	> 60min.	0.0001
	SGFNB	0.001	10.min.		
	FNBF	0.001	10.min.		
RHR6B:					
POS6-4	RSTRHR	0.0001	5.min.		
	FNBF	0.001	10.min.		
	FNBG	0.005	20.min.	> 60.min.	0.0001
POS6-7	RESRHR	0.0001	5.min.	> 60.min	0.0001
	SGFNB	0.001	10.min.		
	FNBF	0.001	10.min.		

POS 11. Drained Maintenance-Time to Core Uncovery=1.8 hour**RHR4:**

POS-Seq	Action	HEP _a	T _a	T _d	HEP _d
POS11-4	RSTRHR	1.00	-		
	FNBF	0.001	10.min.		
	FNBF	0.005	20.min.	> 60.min.	0.0001
POS11-7	RSTRHR	1.00	-		
	SGFNB	0.001	10.min.		
	FNBF	0.001	10.min.	> 60.min	0.0001
RHRT1:					
POS11-4	RSTRHR	0.001	10.min.		
	FNBF	0.001	10.min.		
	FNBF	0.005	10.min.	> 60.min.	0.0001
POS11-7	RSTRHR	0.001	10.min.	> 60min.	0.0001
	SGFNB	0.001	10.min.		
	FNBF	0.001	10.min.		
RHR6B:					
POS11-4	RSTRHR	0.0001	5.min.		
	FNBF	0.001	10.min.		
	FNBF	0.005	20.min.	> 60.min.	0.0001
POS11-7	RESRHR	0.0001	5.min.	> 60.min	0.0001
	SGFNB	0.001	10.min.		
	FNBF	0.001	10.min.		

POS 12. Drained Maintenance-Time to Core Uncovery=1.4 hours**RHR4:**

POS-Seq	Action	HEP _a	T _a	T _d	HEP _d
POS12-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	> 60min.	0.0001
	FNBF	0.001	10.min.		
RHRT1:					
POS12-7	RSTRHR	0.001	10.min.		
	SGFNF	0.001	10.min.	> 60min.	0.0001
	FNBF	0.001	10.min.		
RHR6B:					
POS12-7	RSTRHR	0.0001	5.min.		
	SGFNF	0.001	10.min	> 60min.	0.0001
	FNBF	0.001	10.min.		
RHR12:					
POS12-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	> 60min.	0.0001

FNBF

0.001

10.min.

POS 13. Drained Maintenance-Time to Core Uncovery=10 hours

RHR4:

POS-Seq	Action	HEP _a	T _a	T _d	HEP _d
POS13-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	> 60min.	0.0001
	FNBF	0.001	10.min.		

RHRT1:

POS13-7	RSTRHR	0.001	10.min.		
	SGFNF	0.001	10.min.	> 60min.	0.0001
	FNBF	0.001	10.min.		

RHR6B:

POS13-7	RSTRHR	0.0001	5.min.		
	SGFNF	0.001	10.min	> 60min.	0.0001
	FNBF	0.001	10.min.		

RHR8:

POS13-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	> 60min.	0.0001
	FNBF	0.001	10.min.		

RHR12:

POS13-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	> 60min.	0.0001
	FNBF	0.001	10.min.		

POS 3. Non-Drained Maintenance-Time to Core Uncovery=6 hours**RHR4:**

POS-Seq	Action	HEP_s	T_s	T_d	HEP_d
POS3-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	>60min.	0.0001
	FNBF	0.001	10.min.		

RHRT1:

POS3-7	RSTRHR	0.001	10.min.		
	SGFNF	0.001	10.min.	>60min.	0.0001
	FNBF	0.001	10.min.		

RHR6B:

POS3-7	RSTRHR	0.0001	5.min.		
	SGFNF	0.001	10.min	>60min.	0.0001
	FNBF	0.001	10.min.		

RHR8:

POS3-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	>60min.	0.0001
	FNBF	0.001	10.min.		

RHR12:

POS3-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	>60min.	0.0001
	FNBF	0.001	10.min.		

POS 4. Non-Drained Maintenance-Time to Core Uncovery = 42minutes

RHR4: POS-Seq POS4-7	Action	HEP_a	T_a	T_d	HEP_d
	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	32min.	0.001
	FNBF	0.001	10.min.		
RHRT1: POS4-7	RSTRHR	0.001	10.min.		
	SGFNF	0.001	10.min.	32min.	0.001
	FNBF	0.001	10.min.		
RHR6B: POS4-7	RSTRHR	0.0001	5.min.		
	SGFNF	0.001	10.min	32min.	0.001
	FNBF	0.001	10.min.		
RHR12: POS4-7	RSTRHR	1.00	-		
	SGFNF	0.001	10.min	32min.	0.001
	FNBF	0.001	10.min.		

6.7 Reactivity Accident Event Trees

In Section 4.2.7, several categories of potential reactivity events during shutdown were discussed. They include the following:

1. Addition of diluted accumulator water.
2. Addition of diluted RWST water.
3. Boron dilution due to maintenance problems.
4. Uncontrolled boron dilution from CVCS.
5. Boron dilution via the RHR.
6. Startup of RCP after improper boron dilution.
7. Rod ejection accident.
8. Misloading of fuel assemblies.
9. Uncontrolled bank withdrawal.

After examining each of these categories, it was determined that categories 1, 2, 4, 6 and 8 are relatively risk significant requiring in-depth studies. Probabilistic assessment of various reactivity events during shutdown for Surry, therefore, was focused mainly on the accident sequences belonging to these five categories. The essential details of these analyses are presented in the following sections.

6.7.1 Addition of Diluted Accumulator Water During Shutdown

6.7.1.1 Description of Event

The Surry accumulator system is one of the three major components of the Surry Safety Injection (SI) System (the other two components are high-head safety injection pumps and low-head safety injection pumps). The primary purpose of the accumulators is to provide rapid emergency makeup cooling water to the core in the event of a large break in the reactor coolant system (RCS).

The Surry accumulator system is composed of three independent pressure vessels independently connected to each of the three cold legs. Each accumulator vessel has a volume of 1450 cubic feet and is partially filled with at least 975 cubic feet of 2000 ppm borated water and pressurized with nitrogen gas to 600 to 650 psig. Remote accumulator pressure and level indications are provided in the main control room. A simplified system diagram of one of the three accumulators is shown in Figure 6.7-1. During normal operation, each accumulator is isolated from the RCS by two swing check valves in series (1-SI-107 and 1-SI-109), while the motor-operated isolation valve, MOV-1865A, is kept open. Should the RCS pressure fall below the accumulator pressure due to, for example, a LOCA, the check valves swing open and borated water is forced into the RCS. This accumulator discharge function is considered passive since mechanical operation of the swing disc check valves is the only action required to open the injection path from the accumulators to the core via the cold legs. Following a LOCA (assumed to occur at one of the cold legs), two of the three accumulators refill the reactor inlet plenum, downcomer, lower core basket, and half of the active core with borated water. The third accumulator is assumed to be dumped out of the break.

The analysis of reactivity accidents involving diluted accumulators presupposes that the boron concentration in the accumulators has been critically diluted. If the critically diluted accumulator water is discharged into the reactor core when the RCS has been depressurised to below the pressure of the accumulators, reactivity excursion could ensue. For instance, a large LOCA during POSs 1, 2, 14, or 15, followed by a discharge of critically diluted accumulator water into the vessel could result in such a reactivity transient. The diluted accumulator water can also get into the RCS while the plant is under other shutdown operating states. Note that, in bringing the reactor down to subcritical, the motor-operated isolation valves, (for example, MOV-1865A in Figure 6.7-1), are lock-closed as soon as the RCS pressure falls below about 1000 psig. This is to

prevent any flow from the accumulators into the vessel when the vessel pressure drops below that of the accumulators. If there is a leakage from the isolation MOVs, or if an inadvertent opening of the MOVs occurs subsequent to operator's failure to lock-close the MOVs, the diluted accumulator water can still get into the reactor vessel. These sequences are probabilistically analyzed in this section.

It should be remarked that the amount of accumulator water that can be discharged into the vessel depends on the pressure differential and the size of free volume available in the RCS. If the vessel head is off, the entire accumulator water volume could enter the vessel. Even with the head in place, there is usually sufficient volume in the RCS to accommodate the accumulator water volume because of the relatively low pressurizer water level during reactor shutdown.

6.7.1.2 Probabilistic Analysis of Addition of Diluted Accumulator Water

During the period, August 1975 through September 1982, there were five reported incidents of accumulator boron dilution at Surry, all of them caused by check valve leakages. The various ways in which the accumulator boron concentration can be diluted has been discussed in Section 4.2.7. In the probabilistic analysis of Westinghouse 4-loop plant¹, the probability of accumulator boron concentration being critically diluted and not detected was estimated, with the aid of a fault tree analysis, to be $9.7E-5$. It was determined that this probability is applicable to Surry without modification, because the only significantly different assumption made in the previous analysis was the frequency of refilling the accumulators, which has very little effect on the outcome of the fault tree quantification. The most dominant contributor to the event of diluting the accumulator boron concentration was found to be back-leakage of RCS water into the accumulators through the check valves. The degree of dilution can become particularly significant at the end of the fuel cycle when the RCS boron concentration is low and the time until a refueling outage is short.

The first accident sequence to be analyzed involves an inadvertent blowdown of an accumulator during reactor shutdown. The event tree depicting this sequence is given in Figure 6.7-2. The plant operating states applicable to this sequence are POSs 5 through 12. In shutting down the reactor, the motor-operated accumulator isolation valves (MOV-1865A, MOV-1865B and MOV-1865C) are required to be closed and locked shut when RCS pressure falls below 1000 psig to prevent the accumulators from accidentally discharging into the reactor vessel on a spurious signal. If the operator fails to de-energize the MOVs, they can inadvertently open due to the failure of the valve control. It is assumed that discharging critically diluted water from any one of the three accumulators into the reactor vessel is enough to induce the reactivity accident. The human-error probability of failing to de-energize the MOVs was taken to be 0.01, and the mean rate for MOVs to transfer open was assumed to be $1.2E-7/\text{hr}$. Quantification of the event tree yields a core damage frequency of $9.5E-10/\text{yr}$ for this sequence.

In the second accident scenario, leakings of the accumulator isolation MOVs are assumed to cause the critically diluted accumulator water to gradually seep into the reactor vessel. The event tree for this sequence is shown in Figure 6.7-3. The top event "sufficient core flow" refers to the fact that if there is sufficient circulation in the vessel, then the water which has leaked from the accumulator will mix with the reactor coolant and will not remain stratified in the lower plenum. Any return to criticality, then would be very slow and much larger accumulator dilution would be required. At Surry, the RHRS is required to be in operation during POS 8 through 12. The possibility of not having sufficient core flow during these plant operating states, therefore, is believed to be quite remote. The frequency of this sequence is, therefore, negligibly small.

The last reactivity event scenario involves adding critically diluted accumulator water to the core during the early phase of a large LOCA. If a large LOCA occurs during POSs 1, 2, 14 or 15, the diluted accumulators will automatically discharge as the vessel pressure falls below that of the accumulators. This accident sequence can be delineated by modifying the large LOCA event trees as shown in Figures 6.7-4 through 6.7-7. For POS 1 and 2, both accumulators have to be critically diluted in order for the reactivity excursion to occur. The probability of both accumulators being critically diluted was calculated by assuming a common mode beta-

factor of 0.2. For POS 14 and 15, on the other hand, only one diluted accumulator is enough to cause a reactivity excursion. Partly because of the very low initiating event frequencies for a large LOCA during these POSs, these sequences yield a small core damage frequency of $8.0\text{E-}10/\text{yr}$.

6.7.2 Addition of Diluted RWST Water During Shutdown

6.7.2.1 Description of Event

The refueling water storage tank (RWST) at Surry is a vertical, cylindrical tank having a usable capacity of 398,000 gallons. It is required to contain at least 387,100 gallons of 2000 to 2200 ppm borated water during normal plant operations. The proper boron concentration is maintained by the chemical and volume control system (CVCS). The major functions performed by the RWST include: 1) providing borated water to the safety injection pumps (HHSI and LHSI) and containment spray pumps; 2) providing alternate source of water to the high-head safety injection pumps during abnormal operations; and 3) providing storage water for the refueling cavity.

The accident scenarios considered in this section involve adding diluted RWST water into the reactor vessel during one of the shutdown plant operating states. Three types of accident sequences are considered, including inadvertent actuation of safety injection, inadvertent leakage of RWST water into the vessel and a large LOCA with injection of diluted RWST water.

6.7.2.2 Probabilistic Analysis of Addition of Diluted RWST Water

As compared to the accumulators, the RWST has a much larger volume so that a much larger amount of water would be needed to dilute the contents to a critically low level. Several contributory causes to the dilution of RWST water were explored in the previous study of a Westinghouse 4-loop plant¹ and the probability of RWST water being critically diluted was estimated to be $2.8\text{E-}5$. This value can be conservatively used for Surry, since the only major deviation from the previous analysis was the frequency of refilling the RWST tank, which has negligible impact on the estimated probability.

The RWST is the primary source of borated water for both high-head and low-head safety injection pumps. If the boron concentration in the tank has been diluted to a critical level and the safety injection system is inadvertently actuated, a large amount of diluted water can be forced into the reactor vessel, causing a serious power excursion. Based on Licensee Event Reports² (LER), an incident of inadvertent actuation of safety injection during shutdown actually occurred at Surry 1 in April, 1980. The event tree for this sequence is given in Figure 6.7-8. This accident sequence can only occur when the engineered safeguards are in service, i.e., during POSs 1, 2, 14, or 15. When the safeguards are placed in service, electric power to the safety actuation circuitry is reactivated. Although the intent is to allow the plant to respond automatically if safety injection is needed, it also adds vulnerability to inadvertent injection. The operator error for failing to place the safeguards in service as required was taken to be $5.0\text{E-}4$. The probability of an inadvertent safety injection was derived based on the Surry's actual experience mentioned above. The core damage frequency due to this sequence was calculated to be $6.2\text{E-}8/\text{yr}$.

During plant shutdown, the RWST water can also accidentally leak into the vessel under gravity flow due to, for example, incorrect opening/closing of valves. In fact, according to LER², such an event also occurred at Surry 1 in April 1980, when the unit was at cold shutdown maintenance. As a result, the primary system boron concentration was diluted from 3217 ppm to 2567 ppm. The event was attributed to human errors of not taking precaution against gravity flow from the RWST to the core during valve cycling.

The event tree developed for this sequence is shown in Figure 6.7-9. Since the water would be entering the vessel at relatively slow pace, the major concern is whether the water could accumulate in the lower plenum

in sufficient amount as to cause criticality at the bottom of the core. If the RWST water mixes well with the water in the vessel, then the scenario might just lead to a slow return to power. Since the RHRS is required to be in service during these plant operating states (POSS 9-11), the likelihood of not having sufficient core flow is expected to be extremely small. The frequency of this sequence, therefore, needs no further quantification.

If a large LOCA occurs during plant operating states, 1 through 4 or 12 through 15, low pressure injection system, which takes suction from the RWST, will be needed to provide makeup water to the vessel. For POSS 1, 2, 14 and 15, this injection become important after the accumulator water has flooded the core. According to VEPCO, boron injection tanks (BITs) are no longer in use at Surry. If the RWST has been diluted to a critically low boron concentration, the flow of its water into the vessel may lead to a significant reactivity excursion. The event trees for this scenario can be constructed by modifying the large LOCA event trees, as shown in Figures 6.7-10 through 6.7-13. Due to the very small initiating event frequencies for a large LOCA during these POSSs, quantification of the event trees yields a combined frequency of $2.5\text{E-}10/\text{yr}$ for these accident sequences.

6.7.3 Uncontrolled Boron Dilution from CVCS

6.7.3.1 Introduction

The Surry Chemical and Volume Control System (CVCS) is composed of two major subsystems, charging and letdown, and makeup. The operation of the two subsystems is coordinated to perform the various functions of the CVCS, the primary purpose of which include the following: 1) Adjusts boron concentration in the RCS for reactivity control; 2) Maintains proper water inventory in the RCS; 3) Provide high pressure flow to the RCS upon initiation of safety injection (SI); 4) Maintains proper concentration of corrosion-inhibiting chemicals in the RCS; and 5) Injects seal water into the reactor coolant pumps (RCPs).

Uncontrolled boron dilution is defined as the reduction in the RCS boron concentration caused by inadvertent addition of unborated water. An uncontrolled boron dilution event can occur as a result of operator errors or CVCS equipment failures, such as failure in the blending system or failure in the boric acid/demineralized water makeup flow path to the suction of the charging pump.

If allowed to continue, uncontrolled boron dilution could add enough positive reactivity to cause recriticality during reactor shutdown leading eventually to core damage, if no preventive action is taken. Although inadvertent boron dilution can occur during any plant operating state (POS), the problem can become particularly acute if it occurs when the RCS is in mid-loop operation (POSS 6 and 10). Due to the relatively small amount of water remaining in the RCS, the effect of boron dilution can be more easily felt and transients can proceed at faster pace. Furthermore, since the RCS is operating at near atmospheric pressure, addition of neutronic power to the decay heat can easily cause reactor coolant to heat up quickly and eventually boil, inducing RHRS failure. For Surry, the probabilistic analysis of reactivity accidents due to boron dilution from CVCS, therefore, was centered on those which are assumed to occur when the RCS is in mid-loop operation.

6.7.3.2 Probabilistic Analysis of Boron Dilution Events During RCS Mid-Loop Operation

During the period, 1973 through 1985, there were six reported incidents of uncontrolled boron dilution at Surry Units 1 and 2, five of which occurred during cold shutdown and one during hot shutdown. A half of these inadvertent boron dilution events were attributed to equipment failures and the other half to human

errors. A brief description of the five dilution events which occurred during cold shutdown is given in Table 6.7-1.

To estimate the core damage frequency associated with boron dilution events during RCS mid-loop operations, an event tree was developed as shown in Figure 6.7-14. Although the Surry boron dilution events mentioned above did not necessarily occur during mid-loop operations, they were conservatively included in the estimation of the initiating event frequency. Taking the reactor year during the period indicated above to be 25 years, the initiating event frequency can be calculated as 0.2 event/year.

When an inadvertent boron dilution from CVCS occurs, the first positive indication of low boron concentration in the RCS could come from a high neutron flux alarm. Upon receiving the alarm, the operator is instructed to look for the cause of dilution and terminate the dilution by, for example, stopping the charging pump and isolating letdown. Depending upon the dilution flow rate, he has about 20 to 30 minutes to take these mitigative actions before criticality is reached. Once recriticality occurs, the reactor power will cause the reactor coolant to heat up rapidly and eventually to boil. This will induce cavitation of RHRS pumps and, hence, failure of RHRS. To terminate the boron dilution event at this point would require the operator to actuate emergency boration by injecting borated water into the RCS via the CVCS or the low-head safety injection (LHSI) lines. Failure to initiate the emergency boration and supply makeup to the RCS will lead to core damage.

The branch-point probability assigned to the fourth event-tree top event (No makeup water to RCS) takes into account both operator's failure to initiate the makeup flow and failure of the low-pressure injection system. The second top event models the possible failure of the high neutron flux alarm. The failure probability of the alarm was calculated based on an alarm failure rate of $6.06\text{E-}6$ per hour³ and an average mid-loop operation duration time of 246 hours at Surry. Even the high flux alarm fails, the operator could still be able to realize the abnormal situation through other symptoms of contingency, such as increase in coolant temperature or failure of RHRS pumps. If the operator becomes cognizant of the adverse circumstance after RHRS is lost, he would have to initiate emergency boration within 15 to 20 minutes to prevent core damage. Quantification of the event tree yields a core damage frequency of $9.0\text{E-}6/\text{yr}$ for boron dilution events during RCS mid-loop operations at Surry.

6.7.4 Startup of RCP after Improper Boron Dilution

In Section 4.2.7, several dilution sequences involving accumulation of diluted water in the primary system, followed by startup of an RCP were discussed. Of these, the sequence suggested by French workers⁴ that involves improper dilution during reactor startup and the sequence in which a steam generator tube rupture (SGTR) is followed by dilution from the secondary-side and startup of the RCP in the affected loop were found to require particular attention. This section only deals with the former dilution sequence; the latter dilution sequence is treated and discussed separately in Section 6.4.10.2.6⁷ under SGTR. The applicability of the dilution accident scenario suggested by French workers to Surry was given a close scrutiny. In addition, a screening analysis was performed, with the aid of simple deterministic calculations, to estimate the frequency of occurrence of such an accident. The results and main findings are discussed in this section.

6.7.4.1 Description of Event

Boron concentration in the RCS must be reduced from shutdown concentration before bringing the reactor to criticality. If, for some reason, reactor coolant pumps (RCPs) are stopped and there are not enough decay heat and steam generator heat removal to induce natural circulation, the primary grade water introduced into

the RCS by the charging pump to dilute the RCS may become stratified and collect in one region of the primary-side of the reactor, such as cold-legs, downcomer or the bottom of the lower plenum. A subsequent startup of one or more reactor coolant pumps (RCPs) could then send a slug of this diluted water into the core, causing serious power excursion. Besides the absence of natural circulation in the primary loop, which enables accumulation of diluted water in the primary system, the plant must be in an operational mode which allows for the startup of an RCP, in order for this type of accident to take place. The plant operating state during which this accident sequence could occur is POS 14.

The accident scenario suggested by French workers⁴ involves a loss of offsite power as an initiator during the deboration of the RCS in preparation for attaining criticality. To start up a reactor from cold shutdown condition, a charging pump is initially used to fill the RCS. After the RCS is filled, the charging pump is used to increase the system pressure so that RCP operation can be supported. One of the RCP is then started and the pressurizer heaters energized to heat up the RCS until a bubble is drawn in the pressurizer. After isolating the RHR system and enabling the engineered safeguards systems, the heatup is continued, with the aid of the other two RCPs, until the reactor coolant temperature and pressure reach normal operating values of 543°F and 2235 psig respectively. At this point, the reactor is maintained at cold shutdown boron concentration of 2300 ppm and with shutdown banks withdrawn.

The RCS boron concentration is then reduced from 2300 ppm to 1500 ppm by dilution using water from the Primary Grade Water System so that the control rods can be withdrawn to achieve criticality. For Surry, this boron dilution process was estimated to take about four hours. A schematic diagram of Surry RCS boron dilution using the Chemical and Volume Control System (CVCS) is shown in Figure 6.7-15.

To carry out the RCS deboration, the quantity of primary grade water required is first determined from tables and graphs and set on the BATCH INTEGRATOR control. When boron dilution is initiated, both the primary grade water flow control valve (FCV-1114A) and the primary grade makeup stop valve FCV-1114B open to establish the flow to the Volume Control Tank (VCT), with the primary water supply pump running. Note that the boric acid flow control valve (FCV-1113A) is closed so that only primary grade water can enter the VCT through the VCT spray nozzle. The primary makeup stop valve (FCV-1113B) is also closed to prevent the primary grade water from flowing directly into the charging pump suction header. This is a precaution against a sudden increase in reactivity. When the amount of injected primary grade water reaches the value set on the BATCH INTEGRATOR, the makeup valves shut automatically.

The Volume Control Tank (VCT) of Surry has an internal volume of 300 cubic feet and normal operating pressure and temperature of 15 psig and 105°F respectively. The spray nozzle flow is normally about 120 gallons per minute (gpm). The VCT level control valve (LCV-1115A), located upstream of the VCT, is a solenoid-operated control valve, which is positioned by instrument air to maintain the VCT level at less than 85%. When VCT level reaches a preset value, the VCT level control valve will begin to direct letdown flow to the Boron Recovery System. At a VCT level of 85%, all letdown flow is diverted to the Boron Recovery System. In the event that the VCT level falls to 13%, the charging pump will automatically shift its suction from VCT to RWST (Refueling Water Storage Tank).

During the RCS de-boration prior to withdrawing the control rods, the primary grade water exiting from the VCT is injected into the RCS, through cold-leg B, by a charging pump at a rate of approximately 120 gpm. About 20% of this flow is diverted to the RCPs for use as seal water. The letdown flow from cold leg A, in the meantime, is totally diverted to the Boron Recovery System. As explained above, in order to divert all the letdown flow to the Boron Recovery System, the water level in the VCT must be maintained at 85% or higher. This implies that, during the period of about four hours required for the RCS de-boration at Surry, about 85% of the VCT volume is filled with primary grade water.

The scenario envisioned here assumes an occurrence of loss of offsite power during the RCS de-boration. Upon loss of offsite power, the charging pump, the primary water supply pump and all the RCPs will coast

down and eventually stop completely. If the automatic startup of emergency diesel generator succeeds, power to the charging pump can be restored rather quickly, while the RCPs and the primary water supply pump cannot be restarted until offsite power is recovered. When the charging pump is restarted by the emergency diesel power, it will begin to drain the primary grade water left in the VCT and discharge it into the RCS through cold leg B. With none of the RCPs running and virtually no natural circulation present (due to very low or no decay heat), the primary grade water can gradually make its way to the reactor vessel and settles at the bottom of the vessel. If offsite power is recovered and one of the RCPs is restarted a few moments later, this will send a slug of primary grade water into the core, causing a serious power excursion.

6.7.4.2 Deterministic Considerations

During the RCS deboration, the volume of primary grade water contained in the VCT is approximately 1900 gallons based on 85% water level. Prior to loss of offsite power, the letdown flow from cold leg A is diverted completely to the Boron Recovery System, while the charging pump provides charging flow to the RCS taking suction from the VCT. When the charging pump stops due to loss of offsite, however, the letdown orifice isolation valve (HCV-1200B) shown in Figure 6.7-16 will close automatically, isolating the letdown flow. Unless the operator reopen the valve by resetting the handswitch to OPEN position, it will remain closed even after the charging pump is re-started by emergency diesel. At the moment the offsite power is lost, it can be assumed that the VCT still contains about 1900 gallons of primary grade water. Since power to the primary water supply pump is also lost, no more primary grade water is pumped into the VCT. From this point on, two somewhat different scenarios are conceivable depending upon whether the letdown flow is quickly restored following the restart of the charging pump.

If the letdown orifice isolation valve (HCV-1200B) is not quickly reopened, the relatively cold primary grade water (about 120°F) will be injected into the RCS by the charging pump without being heated properly at the regenerative heat exchanger due to absence of flow on one-side of the heat exchanger. As the water level in the VCT decreases, the VCT level control valve (LCV-1115A) will gradually realign to allow any letdown flow to enter the VCT. Since no letdown flow is entering the VCT, the VCT water level will continue to fall until it reaches the level of 13%, at which point the suction of the charging pump will automatically switch from the VCT to the RWST. With a flow rate of 120 gpm, it will take about 13.5 minutes for the VCT level to drop from 85% level to 13% level. The amount of primary grade water discharged from the VCT into the RCS during this period is about 1600 gallons. If the charging pump continues to run, this volume of primary grade water will be followed by intake from the RWST which contains about 2000 ppm of boron. It is likely, however, that the letdown flow will be reestablished before the VCT level falls to the 13% level. Since the letdown flow is isolated, the charging flow introduced into the RCS will cause the pressurizer level to gradually increase. The charging flow control valve (FCV-1122), which is controlled by a signal from the pressurizer level instrumentation to maintain a prescribed pressurizer water level, will then automatically close if the level set point is reached. If this happens, the operator is likely to quickly reestablish the letdown flow so that the charging flow can be maintained. The subsequent scenario will be similar to that which will be described in the following for another conceivable progression of a series of events.

Assuming that the letdown orifice isolation valve (HCV-1200) is reopened by the operator soon after the restart of the charging pump, the VCT water level will still fall initially, causing the VCT level control valve (LCV-1115A) to adjust its position so as to admit the letdown flow to the VCT. The boron concentration in the letdown flow at this point could range from 1500 to 2300 ppm, depending upon at what stage of the RCS deboration did the loss of offsite power occur. The letdown flow, in this case, will be preceded by about 1900 gallons of primary grade (P.G.) water, since all the P.G. water remaining in the VCT can be drained by the charging pump. It will take about 16 minutes for the charging pump to inject the 1900 gallons of P.G. water into the RCS.

In this analysis, it is assumed that this volume of P.G. water is injected into cold leg B by the restarted charging pump, after the RCPs have completely stopped. Also, the decay heat is assumed to be so small that

there is virtually no natural circulation within the RCS. The fluid injection velocity through the charging pipe (3 inch diameter) was estimated to be roughly 4.5 ft/sec assuming 20% of the charging flow (120 gpm) is diverted to the RCPs as seal flow. As the charging flow is first introduced into the cold-leg, which has a much larger cross-sectional area of about 3.7 square feet, it is likely to cause some local mixing. If the charging flow continues for a while, however, the injected P.G. water could become stratified and make its way to the downcomer, which is only about 8 or 9 feet away from the injection point. Note that the section of cold-leg from the injection point to the upper downcomer has a volume of about 32 cubic-feet (0.9 cubic meters), while the volume of P.G. water from the VCT that can be discharged into the cold leg is several times larger (about 203 cubic feet or 5.8 cubic meters). Since the P.G. water is colder than the RCS coolant, the stratified stream of P.G. water may travel down the downcomer and settle at the bottom of the lower plenum, which has a volume of about 917 cubic feet (26 cubic meters). It was estimated that a slug of 5 cubic meters P.G. water, if swept through the core (which has a volume of 15.8 cubic meters), is sufficient to cause a serious power excursion.

It is apparent from the forgoing discussions that, in addition to the accumulation of a slug of P.G. water in the primary system, the offsite power has to be recovered and one RCP has to be started up in order for this type of reactivity accident to happen. The times at which these causative events occur relative to each other appear to play an important role in deciding the possibility and severity of such an accident. As noted previously, it takes about 13 to 16 minutes to inject all the P.G. water left in the VCT into the cold leg. If the offsite power is recovered about 15 minutes after it is lost, and that one of the RCPs is restarted immediately, there is a high likelihood that the accident will occur. Since the injection of P.G. water is followed by that of either RWST water (which contains 2000 ppm of boron) or letdown water (which contains at least 1500 ppm of boron), the slug of P.G. water which has settled at the bottom of the lower plenum will eventually mix with the borated water as a result of molecular diffusion or thermal convection due to heating by the lower plenum metal. Assuming that the letdown water contains 1700 ppm of boron when it is injected into the RCS following the exhaustion of P.G. water in the VCT and that a complete mixing will occur in the lower plenum, about 25 minutes is sufficient to raise the boron concentration in the lower plenum to the level that it is no longer a threat to criticality. In other words, if offsite power is not recovered for more than 40 minutes and that, during this period, the charging pump (which was restarted by diesel power soon after loss of offsite power) continues to inject water from either the VCT (P.G. water followed by letdown water) or the RWST, the chance of occurrence of such an accident will become increasingly small.

6.7.4.3 Probabilistic Screening Analysis

To perform a probabilistic screening analysis for this accident scenario, an event tree was constructed as shown in Figure 6.7-17.

The initiating event frequencies were calculated based on Surry's plant specific data for refueling outage frequency (0.6/yr), and the frequencies of drained maintenance outage (1.2/yr) and non-drained maintenance outage with RHR (0.8/yr). For plant operating states (POSS) 2 through 14, the failure rate of offsite power at Surry is about $2.85E-5$ /hr, and it was assumed that the RCS deboration requires about four hours. In the event tree, the values shown with a bracket only apply to the two maintenance outages. The distinction between refueling outage and the other two maintenance outages was made mainly because, for the former, decay heat is much lower, and, hence, less chance of bringing on natural circulation to cause mixing of injected P.G. water with RCS coolant.

The second event-tree top event inquires about whether charging flow is established by emergency diesels. The failure probability takes into consideration failure of both diesels A and B to start as well as failure of the charging pump to start on demand. The third top event asks whether offsite power is recovered within 15 minutes. As mentioned previously, this type of reactivity accident is most likely to occur if offsite power is recovered within 15 minutes and that one of the RCPs is restarted immediately. The probability of not recovering offsite power in a given time shown in the event tree is also Surry plant specific. The branch point

probability for the fourth event-tree top event represents the conditional probability of not recovering offsite power in 40 minutes, given offsite power is not recovered in 15 minutes. Since the probability of not recovering offsite power in 40 minutes is 0.5756, the conditional probability becomes $0.5756/0.743 = 0.775$. The fifth event-tree top event denotes the probability that a slug of P.G. water could accumulate in the primary loop. Note that the probability of this event is much larger if offsite power is recovered within 15 minutes. The probability decreases if offsite power is recovered between 15 and 40 minutes, and becomes negligibly small if offsite power recovery occurs after 40 minutes. For the two maintenance outages, considerably smaller probabilities were assigned because of the higher decay heat level, which is likely to cause some natural circulation. The last top event represents the probability of restarting one RCP before the slug of P.G. water mixes well with the RCS coolant. It is apparent that if the restarting of RCP is delayed, there will be more time available for the RCS coolant to homogenize and, hence, less chance of causing reactivity excursion.

Quantification of the event tree yielded a core damage frequency of $1.39\text{E-}5/\text{yr}$ for refueling outage and $1.38\text{E-}5/\text{yr}$ for the two combined maintenance outages respectively. The total core damage frequency, therefore, is about $2.8\text{E-}5/\text{yr}$ for this accident sequence.

6.7.5 Inadvertent Criticality Due to Misloading of Fuel Assemblies

6.7.5.1 Introduction

Fresh fuel assemblies which are not equipped with control rods have relatively high reactivity worth. The fresh fuel assemblies of Surry 1 (cycle 11) are of this type. During refueling operations (POS 8), if these high worth assemblies are incorrectly loaded into the core so that they are clustered together without control rods, then there is a possibility of a reactivity excursion. If a cluster was just subcritical because of a misloading and no preventive action was taken when the count rate increased, then the loading of the next adjacent assembly would cause a rapid insertion of reactivity which would likely result in localized core damage.

To understand the potential for an inadvertent criticality during refueling, a probabilistic assessment was performed based on the loading procedures for Surry 1 (Cycle 11), which bear close similarity to those for Calvert Cliffs Unit 2 (October 1990). For the latter plant, both deterministic and probabilistic analyses were performed recently⁵ to assess the potential for losing the required shutdown margin and, in the extreme, for an inadvertent criticality during refueling. The probabilistic analysis carried out for Surry 1 essentially follows the same approach.

For Surry 1, refueling is done by first off-loading all the fuel assemblies from the core into the suppression pool. Repositioning of the control rods, if required, is done in the spent fuel pool and then the fresh and burned fuel assemblies are returned to the reactor core where they are loaded according to a pattern which allows for proper neutron flux monitoring.

Figure 6.7-18 shows the final layout for Cycle 11 of Surry 1, indicating the positions of both fresh and burned fuel assemblies. The core contains a total of 157 fuel assemblies, of which 52 are fresh fuel assemblies. None of the fresh assembly contains a control rod assembly. All the control rod assemblies are attached to burned fuel assemblies. This makes refueling operation at Surry more vulnerable to the type of loading errors being considered. In this study, the fresh assemblies are grouped into three types, A type (high reactivity worth), B type (medium reactivity worth) and C type (low reactivity worth), according to their fuel enrichment and the number of burnable poison rods they contain. They are denoted as FA, FB and FC respectively in Figure 6.7-18. There are 16 each of FA and FB types and 20 FC types, making up a total of 52 fresh assemblies. Since all the fresh assemblies are uncontrolled, improper arrangement of the fresh assemblies as a cluster during refueling can result in loss of shutdown margin or even inadvertent criticality. The figure also indicates the order in which the core is assumed to be loaded. This is represented by the numbering of assembly core

location. There are other loading sequences that can be used after an offload, but they are not expected to produce significantly different results.

Surry has a shutdown margin of about 8%, which is significantly lower than the margin of 13% used in Calvert Cliff. Based on the detailed shutdown margin calculations performed for Calvert Cliff⁵, it was determined that inadvertent criticality could ensue if a group of five fresh assemblies of A type are arranged in a cross-shaped configuration, or a group of six fresh assemblies of A or B types are arranged in a 2x3 array. These clusters could cause inadvertent criticality if they are placed anywhere in the core except at locations adjacent to the baffle where neutron leakage would mitigate the reactivity.

6.7.5.2 General Approach to Probabilistic Analysis

To facilitate the analysis, it was assumed that : (1) Errors associated with loading or unloading of a fuel assembly in the core or in the spent fuel pool occur randomly, independent of the order of the procedures or the position of the assembly; (2) Detection of loading or unloading errors by uncovering inconsistencies in the procedures are neglected; and (3) No common-cause errors, human or non-human, are considered except for calibration errors for the refueling machine. Of these, the second assumption adds considerable conservatism to the analysis.

The general approach taken in the probabilistic analysis involves the following steps. Description is given based on estimating the probability of forming a cluster of five fresh assemblies of A-type (placed in a cross-shaped configuration) anywhere in the core except at locations adjacent to the baffle. Similar steps were taken to evaluate the probability of placing a cluster of six fresh assemblies of A- or B-type in a 2x3 array.

(Step 1)

For each core location, estimate the probability that it will be loaded with a fresh assembly of A-type, correctly or mistakenly. It is obvious that, for a core location designated to be loaded with a fresh assembly of A-type, this probability is essentially one. For a core location specified to be loaded with an assembly of types other than A, a misloading must have taken place in order for it to be loaded with a fresh assembly of A-type. To calculate the probability of mistakenly loading a given core location with a fresh assembly of A-type, fault trees were developed, as shown in Figure 6.7-19, to logically explore various contributory causes to such a loading mistake. The probability of the fault-tree top event was calculated by quantifying the fault trees using the SAICUT code⁶. More detailed discussions on the fault trees will be given later. Note that, for Surry, there are 36 core locations (see shaded area in Figure 6.7-18) which are adjacent to the baffle and, hence, can be excluded from the analysis. The number of core locations which need to be considered is, therefore, $157 - 36 = 121$. For the case where a cluster of six fresh assemblies of A- or B-type are placed in a 2x3 array, it is necessary to estimate the probability that each of the 121 assembly core locations will be loaded with a fresh assembly of A- or B-type, correctly or mistakenly. The same fault trees can be used by merely changing the probability of the basic event, TYPEFA, so that it will represent the probability of being A- or B-type rather than that of being A-type.

(Step 2)

Based on the final loading pattern shown in Figure 6.7-18, find out all the possible combinations of core locations to form a cluster of five in a cross-shaped configuration. It was found that there are 88 such combinations possible. For example, a combination of core locations, 6-13-14-15-24, or 22-33-34-35-47, will form such a configuration. For the case where a group of six fresh assemblies is placed in a 2x3 array, it is necessary to find out all the possible combinations of core locations to form such an array. A total of 168 such combinations was found possible in this study. An example is the combination of core locations, 16-17-18-26-27-28.

(Step 3)

The probability of forming a cluster of fresh assemblies of a specified type is obtained by multiplying together the probabilities of loading each of the core locations represented by a combination (found in Step 2) individually with a fresh assembly of the specified type. For example, the probability of forming a cluster of five fresh assemblies of type-A, represented by a combination, 6-13-14-15-24, is calculated by multiplying together the probabilities of loading each of the core locations 6, 13, 14, 15 and 24, individually with a fresh assembly of A-type. Note that the probabilities associated with each of the 112 core locations have been calculated in Step 1. Similarly, the probability of forming a cluster of six assemblies of A- or B-type, represented by a combination of core locations, 16-17-18-26-27-28 is obtained by multiplying together the probabilities of loading each of the core locations, 16, 17, 18, 26, 27, and 28, individually with an assembly of A- or B- type. This approach is consistent with the assumption that each of the loading event occur independently.

(Step 4)

Since each combination of core locations to form a problematic cluster is unique and mutually exclusive, the probability of forming a cluster of 5 (of type A, in a cross configuration) or 6 (of type A or B, in a 2x3 array) fresh fuel assemblies anywhere in the core so as to cause inadvertent criticality can be obtained by summing up the probabilities of all the possible combinations. This logic is delineated in Figure 6.7-20. Note that the event, CLUS5, appearing in the fault tree is an abbreviated expression used to represent all of the 88 possible combinations (found in Step 2) to form a cluster of five assemblies of A-type, with the core location indices, "a", "b", "c", "d" and "e" specified to correspond to the core locations for each of the combinations. Similarly, the event, CLUS6, is an abbreviated expression used to encompass all of the 168 possible combinations to form a cluster of 6 fuel assemblies of A- or B- type, with the core location indices, "r", "s", "t", "u", "v" and "w" specified to correspond to the core locations for each of the combinations.

6.7.5.3 Fault Trees for Misloading a Fuel Assembly Location

The fault trees shown in Figure 6.7-19 were developed based on the following core unloading and loading scenarios: Initially, the core is entirely filled with burned fuel assemblies. The core is unloaded by removing all the burned fuel assemblies and transporting them to the spent fuel pool, where they are placed in proper pool locations. Repositioning of control rod assemblies among the burned fuel assemblies is then carried out at the spent fuel pool. After the CRA repositioning is completed, the empty core is loaded, one FA at a time, following the loading pattern indicated by Figure 6.7-18. Note that, since there are a total of 157 core locations to be loaded, the loading consists of 157 loading steps, each of which involves picking up a fuel assembly from a specified spent fuel pool location, moving it to the reactor building, and loading it into a specified core location.

The fault trees take into consideration various human errors that could contribute to misloading of a fuel assembly. They include, for instance, communication errors between the refueling machine operators and the control room operator during the process of loading or unloading the core. Table 6.7-2 lists all the human errors modeled in the fault trees and the probabilities assigned to them in the base case calculation. Essentially all of the human error probabilities were taken from Swain and Guttman⁷, by further applying subjective judgement.

6.7.5.4 Quantification of the Fault Trees

It should be remarked that the fault trees shown in Figure 6.7-19 contain three basic events, TYPEFA, ALOC and SPLOC, whose probabilities depend on the order of loading the fuel assemblies. Given there is a misloading, TYPEFA represents the event that the mistakenly loaded fuel is of the specified type (i.e., type-A

for a cluster of 5 and either type-A or type-B for a cluster of 6). Initially, there are a total of 157 fuel assemblies stored in the spent fuel pool, 16 each of which are of A-type and B-type. As the loading proceeds, the number of A-type or B-type fuel assemblies, as well as the total number of fuel assemblies remaining in the spent fuel pool change accordingly. The probability of randomly picking up a wrong fuel assembly from the spent fuel pool, therefore, varies at each refueling step. In filling a given core location with a specified fuel assembly, the fuel assembly intended for the core location could be mistakenly loaded into another core location. The fault tree event, ALOC, represents the probability that a fuel assembly intended for other core location is mistakenly loaded into the location of interest. This probability also changes with loading steps, since the number of empty core locations available for such a misloading without being detected varies at each loading step. The event, SPLOC, represents the similar probability of storing a wrong fuel assembly in a specified spent fuel pool location during core unloading.

Since the probabilities of these three events vary with loading steps and, hence, with core locations, each core location has a somewhat different probability of being loaded with a wrong fuel assembly. Rigorously speaking, therefore, the fault trees shown in Figure 6.7-19 must be quantified for each of the 121 core locations of interest. To lessen the burden of quantifying the fault trees, however, the probability that was averaged over the entire 157 core locations was used for each of these three events. In a similar study⁵, this simpler approach was found to yield somewhat conservative results. Since no detailed unloading procedures are specified in this analysis, the averaged probability for SPLOC was taken to be the same as that of ALOC. This simplified and conservative approach has one important implication that should be specifically noted, namely, the probability of being misloaded with a fresh assembly of a specified type is the same for all core locations which are not designated to be loaded with the specified fresh assembly. This can be illustrated by referring to the fault tree shown in Figure 6.7-20. As shown, the combination of core locations 6-13-14-15-24 forms a cluster of 5 in a cross configuration. Since core location 6 is designated to be loaded with a fresh assembly of A-type, the probability of loading this location with a fresh assembly of A-type is essentially one. For the remaining 4 core locations, which are not designated to be loaded with a fresh fuel assembly of A-type, they will each have a same probability (say, A) of being misloaded with a fresh assembly of A-type. The probability of forming this particular clutter is, therefore, A^4 .

6.7.5.5 Discussions of Results

The fault trees shown in Figure 6.7-19 were quantified using the SAICUT⁶ code. The probabilities of all the basic events used in the base-case calculation are summarized in Tables 6.7-2 and 6.7-3. For a given core location, the probability of being misloaded either with a fresh fuel assembly of A-type or with a fresh fuel assembly of A- or B-type was found to be $3.23E-3$ and $6.1E-3$ respectively. Substituting these values into the expressions of all the possible combinations (refer to Figure 6.7-20), the probability of forming a cluster of 5 fresh assemblies of A-type (in cross configuration) or a cluster of 6 fresh fuel assemblies of A- or B-type (in a 2×3 array) anywhere in the core so as to cause inadvertent criticality was computed to be $1.8E-9$ and $3.6E-6$ respectively.

Applying the Surry refueling outage frequency of 0.6/yr, the expected frequency of inadvertent criticality was calculated to be $2.2E-6$ /yr. To estimate the frequency of core damage that could ensue from misloading of fuel assemblies, an event tree was constructed as shown in Figure 6.7-21. During refueling, ex-core source range neutron flux monitors are used to monitor the subcritical neutron multiplication. They could normally be capable of detecting an increase in count rate before criticality was reached. The operator would then be able to take preventive measures, such as stopping refueling or initiating boron addition to the RCS. The event tree models possible failure of the neutron flux monitors and failure of the operator to respond timely to prevent core damage. Quantification of the event tree yielded a core damage frequency attributable to fuel misloading of $1.2E-7$ /yr.

6.7.5.6 A Sensitivity Study

About 80% of the basic events modeled in the fault trees are human-error related events. Since the human error probabilities (HEPs) assigned to those basic events are subject to a large uncertainty, a sensitivity study was performed to assess the impact of HEPs on the results of the probabilistic analysis. For this purpose, all the HEPs were multiplied by a constant factor of 0.5, 2 and 5 respectively without altering the probabilities of the basic events unrelated to human errors. The results of this sensitivity study, summarized in Table 6.7-4, indicate that the calculated core damage frequency is rather sensitive to the HEPs used. When all the HEPs are increased or decreased by a factor of 2, the core damage frequency correspondingly increases or decreases by about an order of magnitude respectively. When all the HEPs are increased by a factor of 5, the core damage frequency further increases significantly to $3.2\text{E-}5/\text{yr}$.

As a concluding remark, it should be pointed out that the probabilistic analysis of fuel misloading presented in this section is based on the implicit assumption that any intermediate positioning of fuel assemblies, if permitted by refueling procedures, are analyzed in advance and that other plant procedures are followed as well.

6.7.6 References

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Figure 6.7-1 Simplified Sketch of One of Three Surry Accumulators

Reactor In POSS-12	Accumulator MOVs Deenergized	1/3 Accumulator MOV Opens	Accumulator Boron OK
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graph TD
    A["1) 0.6/yr  
2) 1.2/yr"] --- B1(( ))
    B1 --- B2["1.0E-2"]
    B1 --- B3["1) 1.31E-3  
2) 1.59E-4"]
    B2 --- C1(( ))
    B3 --- C1
    C1 --- D1(( ))
    D1 --- D2["9.7E-5"]
    D2 --- E["1) 7.52E-10  
2) 1.85E-10"]
    E --- F["POTENTIAL REACTIVITY EVENT: 9.5E-10/yr"]
  
```

1) Refueling
2) Drained Maintenance

POTENTIAL REACTIVITY EVENT: 9.5E-10/yr

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EVENT CAUSED BY LEAKING ACCUMULATOR VALVE

Figure 6.7-3 Event Tree for Leaking Accumulator

Large LOCA POS1	Accumulators Inject	Boron Conc. in both accum. low	LPI	Boron conc. in RWST low	Contain- ment Systems	Core Vulnerable	LPR
IE-A1	DS	ACLOW2	DS	RWSTLOW	CS	CV	H1

Figure 6.7-4 Reactivity Accident from Large LOCA Event Tree (POS1)

Large LOCA - POS2	Accumulators Inject	Boron Conc. in both accum. low	Operator Actuates SI	LPI	Boron conc. in RWST low	Contain- ment Systems	Core Vulnerable	LPR
IE-A2	D5	ACLOW2	ISA2	D6	RWSTLOW	CS	CV	H1

Figure 6.7-5 Reactivity Accident from Large LOCA Event Tree (POS2)

Large LOCA POS14	Accumulators Inject	Boron conc. in either accum. low	Operator Actuates SI	LPI	Boron conc. in RWST low	LPR
IEA14	D5	ACLOW1	ISA14	D6	RWSTLOW	H1

Figure 6.7-6 Reactivity Accident from Large LOCA Event Tree (POS14)

Large LOCA POS15	Accumulators Inject	Boron conc. in either accum. low	LPI	Boron conc. in RWST low	LPR
IEA15	DS	ACLOW1	DS	RWSTLOW	H1

Figure 6.7-7 Reactivity Accident from Large LOCA Event Tree (POS15)

EVENT CAUSED BY INADVERTENT SAFETY INJECTION

Figure 6.7-8 Event Tree for Sequence with RWST Diluted

EVENT CAUSED BY IMPROPER VALVE ALIGNMENT

Figure 6.7-9 Event Tree for "Leaking" RWST
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Large LOCA POS3	Operator Actuates SI	RWST available	LPI	Boron conc. in RWST low	LPR
IEA3	ISI3	RWST3	O6	RWSTLOW	H1

Figure 6.7-10 Reactivity Accident from Large LOCA Event Tree (POS3)

Large LOCA POS4	Operator Actuates SI	RWST available	LPI	Boron conc. in RWST low	LPR
IEA4	ISI4	RWST4	DS	RWSTLOW	H1

Figure 6.7-11 Reactivity Accident from Large LOCA Event Tree (POS4)

Large LOCA POS12	Operator Actuates SI	RWST Available	LPI	Boron conc. In RWST low	LPR
IEA12	ISI12	RWST1	D6	RWSTLOW	H1

Figure 6.7-12 Reactivity Accident from Large LOCA Event Tree (POS12)

Large LOCA POS13	Operator Actuates SI	RWST Available	LPI	Boron conc. In RWST low	LPR
IEA13	ISI13	RWST1	D6	RWSTLOW	H1

Figure 6.7-13 Large LOCA Event Tree (POS13)

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Figure 6.7-14 Surry Reactivity Accident Due to Unplanned Boron Dilution During RCS Mid-Loop Operation

Figure 6.7-15 A Sketch of Surry RCS Boron Diffusion During Reactor Startup

Figure 6.7-16 Sherry Charging and Letdown Subsystem

Loss of offsite power during RCS deboration	Charging flow re-established by emergency D.G.	Offsite power recovered in 15 minutes	Offsite power recovered in 40 minutes	A slug of P.G. water collects in one region of primary loop	RCP restarted before the slug of P.G. water mixed well with RCS coolant
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Figure 6.7-17 Event Tree for Surry Startup of RCP after Improper RCS Deboration

Figure 6.7-19 Fault Trees for Misloading a Core Location

Figure 6.7-19 (Continuation 1)

Figure 6.7-19 (Continuation 2)

Figure 6.7-19 (Continuation 3)

Figure 6.7-19 (Continuation 4)

Figure 6.7-20 Fault Tree for Inadvertent Criticality

Refueling Outage	Fresh Fuel of A or B Type Loaded in Cluster Worth More than SDM	Nuclear Monitors Working	Operator Responds Timely
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Figure 6.7-21 Event Tree for Sarry 1 Inadvertent Criticality During Fuel Loading

Table 6.7-1
Summary of Inadvertent RCS Boron Dilution Which Occurred
At Surry Plants During Cold Shutdown

Unit	Date	Brief Description of Event
1	Feb. 24, 1975	Over a 4-day period, too great a flow of primary-grade water to the blender produced a boron concentration of 1312 ppm, which diluted RCS from the required 2000 ppm to 1842 ppm.
2	July 30, 1976	Leakage from three tubes that had been cut during the removal of a section of the seventh tube support plate on the secondary side of the steam generator diluted the RCS boron concentration from 2356 to 1836 ppm.
2	April 6, 1978	A failed flow controller caused the primary-grade water valve to overfeed during blend operation. The RCS boron concentration decreased from 1372 to 1259 ppm in 13 hours. The failure also affected the primary-grade water flow deviation.
1	May 12, 1980	Inadvertent deboration of the RCS occurred when a mixed-bed demineralizer was placed in service without verifying that boron concentration of the effluent was equalized with that of the RCS.
2	October, 1985	RCS boron dilution occurred at cold shutdown. The dilution was caused by problems in the boric acid/P.G. water blend system, which was used to control boric acid concentration in the RCS. Inaccurate operation of boric acid controller.

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Inches

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Table 6.7-2
Human Errors Modeled in the Fault Trees for
Surry Reactivity Transients During Refueling

Event Name	Description of Basic Event	Probability or Unavailability
ANEWOP1	Spent fuel machine operator read the bridge and trolley locations incorrectly when storing new assemblies.	8.0E-3
ANEWOP2	Spent fuel machine is incorrectly aligned to letter or number position markers when storing new assemblies.	1.0E-2
ANEWPR	Errors in the procedure for storing new assemblies.	1.0E-3
CALIBER	Refueling machine position is calibrated incorrectly.	3.0E-3
CALINDX	Index point calibration check is not done by the operator.	1.0E-2
COINTM	A FA with wrong I.D. number is loaded into location being considered due to errors introduced by intermediate positioning etc.	8.0E-3
GRID	Refueling machine operator misunderstood core grid location due to communication errors.	3.0E-3
LUNLDPR	Errors in loading (or unloading) procedures.	1.0E-3
LUNLD1	Control room operator updated the procedure incorrectly leading to specification errors during loading or unloading.	8.0E-3
LUNLD2	Control room operator read a wrong core grid location during loading or unloading.	8.0E-3
LUNLD3	Tag board was updated incorrectly leading to specification errors during loading or unloading.	8.0E-3
LUNLD4	Control room operator misread the procedure due to changes in the order of portion of the procedures during loading or unloading.	1.0E-2

Table 6.7-2 (continued)
Human Errors Modeled in the Fault Trees for
Surry Reactivity Transients During Refueling

Event Name	Description of Basic Event	Probability or Unavailability
RMOP1	Refueling machine operator reads index value from a wrong line.	5.0E-3
RMOP2	Refueling machine operator transposes bridge and trolley locations.	5.0E-3
SPICK1	Spent fuel machine operator read the bridge and trolley locations incorrectly (during core loading).	8.0E-3
SPICK2	Spent fuel machine is incorrectly aligned to letter or number position markers (during core loading).	1.0E-2
SPICKPR	Errors in refueling procedures (for core loading).	1.0E-3
SPINTM	A FA with wrong ID number is stored in the specified spent fuel pool location due to errors introduced by intermediate positioning etc.	5.0E-3
SPSTOR1	Spent fuel machine operator read the bridge and trolley locations incorrectly during core unloading.	8.0E-3
SPSTOR2	Spent fuel machine is incorrectly aligned to letter or number position markers during core unloading.	1.0E-2
SUBSEQ	Shufflings subsequent to core unloading causes a wrong FA to be stored in the specified spent fuel pool location.	5.0E-3
VERIFA1	No verification of FA position is done before fuel loading operation.	1.0E-2
VERIFA2	Errors committed during verification of FA position using remote video camera	2.0E-2
VERISE1	No verification of machine position is done by a second machine operator.	8.0E-1
VERISE2	Verification of machine position incorrectly done.	5.0E-3

Table 6.7-3
Basic Events (Not Related to Human Errors)
Modeled in the Fault Trees for Surry Reactivity Transients During Refueling

Event Name	Description of Basic Event	Probability
ALOC	The wrong location is location being considered.	1.0E-2
SMOOTH1	A FA can be loaded or unloaded smoothly at the wrong core index location.	2.0E-1
SMOOTH2	Mispositioning of refueling machine corresponds to a FA location.	1.0E-1
SPLOC	The wrong spent fuel pool location is the specified pool location.	1.0E-2
TRANSD	Malfunction of position transducer causes the machine to be mispositioned.	2.2E-3
TYPEFA	The loaded wrong FA is fresh.	2.1E-1

Table 6.7-4
Surry 1 (Cycle 11) Reactivity Accident During Refueling
Results of Sensitivity Studies on Human Error Probability (HEP)

	Probability of Forming a Cluster of 5 (2x3 array) of A-or B-type FA	Probability of Forming a Cluster of 5 (cross configuration) of A-type FA	Frequency of Inadvertent Criticality (/yr)	Frequency of Sequence IC-1 (/yr)	Frequency of Sequence IC-2 (/yr)	Total Core Damage Frequency (/yr)
Base Case	3.60E-6	1.77E-9	2.16E-6	1.07E-7	1.68E-8	1.24E-7
HEP x 0.5	4.05E-7	9.52E-11	2.43E-7	1.21E-8	1.89E-9	1.39E-8
HEP x 2	3.50E-5	3.72E-8	2.10E-5	1.04E-6	1.64E-7	1.21E-6
HEP x 5	9.15E-4	3.03E-6	5.51E-4	2.73E-5	4.30E-6	3.16E-5

6.8 Special Initiators

6.8.1 Overpressurization of Temporary Seals of the Thimble Tubes

Section 8.5.4 estimated that in each refueling outage, the RCS is closed for approximately ten days with temporary seals installed at the seal table. This happens in POS 7 prior to vessel head removal, and in POS 9 after refueling. In POSs 7 and 9, the reactor coolant loops are expected to be isolated. Therefore, steam generators are not available for heat removal. The seals can only withstand 40 psia of pressure. Therefore, in a loss of RHR event, upon bulk boiling of the coolant, the RCS pressure is expected to increase above this pressure and cause failure of the seals. Further pressurization may cause a rapid core uncover. Based on the drawing obtained from Virginia Power, it was estimated that the total flow area through the space between the guide tubes and thimble tubes is approximately 2.4 square inches. This is close to the area of a PORV. At 365 psig, a PORV is able to release 700 gallon per minute of liquid. At this flow rate, assuming that the initial vessel level is at the vessel flange, core uncover can occur in 12 minutes.

In order to assess the significance of this issue, a station blackout scenario is used. In this scenario, a blackout occurs when the RCS is closed with the temporary seals in place. The blackout lasts long enough, say 1 hour, to cause the RCS to reach bulk boiling. It is assumed that once bulk boiling occurs, the core will be uncovered (in 12 minutes). The frequency that a loss of offsite power occurs within these ten days is $0.6 \text{ refueling/year} * 2.85\text{E-}05 \text{ loss of offsite power/hr} * 10 \text{ days} * 24 \text{ hours/day} = 4.1\text{E-}03 \text{ /year}$.

The probability that a blackout occurs can be estimated by estimating the maintenance unavailability of one diesel generator (0.136 for EDG 1 in POS 7) and the probability that the second generator fails to start ($2.0\text{E-}02$). The probability that offsite power is not recovered in 1 hour is 0.49.

Therefore, the frequency of the scenario leading to rapid core uncover is $4.1\text{E-}03 \text{ /yr} * 0.136 * 2.0\text{E-}02 * 0.49 = 5.47\text{E-}06 \text{ /year}$.

6.8.2 Low Temperature Over Pressurization (LTOP)

In this section, the scenario in which an inadvertent safety injection signal causing a (low temperature) over pressure transient is analyzed. Such scenario may occur in POSs 3,4,12 and 13 of a refueling outage or drained maintenance outage. It may occur in POSs 3 and 4 of a non-drained maintenance in which RHR was used. In these POSs, the RCS is closed with the level in the pressurizer maintained at 33% or below. With the SI signal, the normally running charging pump delivers approximately 150 gpm to the RCS, and can fill the pressurizer in approximately 10 minutes. Once the pressurizer is solid, the PORVs, RHR relief valve, and RV 1203 in the letdown system can potentially relieve the pressure. If the relief valves fail to open or are not available, the RHR system may become over pressurized, and a rupture can happen.

Figure 6.8.4-1 is a fault tree for calculating the frequency of such scenario. The frequency of inadvertent safety injection has been estimated in section 4.2.6 to be $6.72\text{E-}05$ per hour. In reviewing the maintenance unavailability of PORVs, it was found that both PORVs are inoperable in one of the three refueling outages, for extended period of time including POS 3 and 4. It is conservatively assumed that a probability of 0.3 can be used for the PORVs in all POS in which this scenario may occur. It is also assumed that relief valve RV 1203 is capable of relieving pressure. The failure probability of the RHR relief valve and RV 1203 was assumed to be the same as that for a PORV. Given that 10 minutes is available for operator to terminate the transient, the HRA approach described in chapter 7 estimated a HEP_a of 0.0001 and a HEP_d of 0.2. These estimated probabilities are used in the quantification of the LTOP fault tree, and a frequency $3.0\text{E-}08$ per year of was obtained.

Figure 6.8.4-1 Low Temperature Overpressurization

6.9 ATWS Event Trees

The ATWS event trees are presented in Figures 6.9.1 and 6.9.2. They are the low power part of the NUREG-1150 ATWS event tree with additional credit taken for very negative moderator temperature coefficient (in the NUREG-1150 this credit is taken for the high power part of the tree but not the low power part).

Questions are asked about the manual scram (R), very negative moderator coefficient (Z1), relief valve operation (P2, 3 SRVs or 2 SRVs and 2 PORVs), AFW system (L2, 2 AFWMDP or 1 AFWTDP), RCS integrity (Q, all SRVs and PORVs must reclose) and reactor subcriticality by emergency boration if the operator scram is unsuccessful (D4, 1 charging pump taking suction from the Boric Acid Tank, discharging through the charging line and remaining at elevated temperature). D2 (manual actuation of HPI, taking suction from the RWST) is used instead of D4 if the PORVs/SRVs fail to reclose. The ATWS event tree will have transfers from S2, S3, T2A and T3A event trees in POS1 and 15. These transfers will determine the initiating frequency (6.7-6/yr in POS1 and 1.6-5/yr in POS15).

Following are sequences of interest in the two event trees:

Sequence 3 leads to core damage due to failure of manual scram followed by failure of emergency boration. Sequence 4 is failure of the scram, success of emergency boration but the PORVs and/or the SRVs fail to reclose leading to a small LOCA transfer. Sequence 5 is similar to sequence 3 (failure of the subcriticality function in addition to a small LOCA). Sequence 6 is core damage due to failure of early heat removal from failure of the AFW to provide sufficient flow). Sequence 7 is due to failure to provide pressure relief which is required as the moderator coefficient is not sufficiently negative. Sequences 9-12 are analogous to sequences 3-6; the difference is that in sequences 9-12 the moderator coefficient is very negative.

Results of Quantification - ATWS

For ATWS, 1150 data were used for basic events, except where any Surry-specific data existed for maintenance in POS 1 and POS 15. POS 15 ATWS CDF frequency was much higher (~ 2 orders of magnitude) than POS 1 CDF. Sequences 3,5,6,9, and 12 contributed "evenly", so the CDF was mostly due to the failure of emergency boration/HPI or failure of the AFW. The total ATWS CDF is in the "low-medium" range.

IE-TK	R	Z1	P2	L2	Q	D4	SEQ #	END-STATE-NAMES
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Figure 6.9.1 Low Power - Pos 1 Anticipated Transient Without Scram

IE-TK	R	Z1	P2	L2	Q	D4	SEQ #	END-STATE-NAMES
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Figure 6.9.2 Low Power - Pos 15 Anticipated Transient Without Scram

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Evaluation of Potential Severe Accidents During Low
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Analysis of Core Damage Frequency from Internal
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11. ABSTRACT (200 words or less)

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. The objectives of the program are to assess the risks of severe accidents 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 objective of this report is to document the approach utilized in the Surry plant and discuss the results obtained. A parallel report for the Grand Gulf plant is prepared by SNL. This study shows that the core-damage frequency during mid-loop operation at the Surry plant is comparable to that of power operation. We recognize that there is very large uncertainty in the human error probabilities in this study. This study identified that only a few procedures are available for mitigating accidents that may occur during shutdown. Procedures written specifically for shutdown accidents would be useful.

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