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Improved Reliability of Residual Heat Removal Capability in PWRs as Related to Resolution of Generic Issue 99

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Brookhaven National Laboratory

Prepared for
U.S. Nuclear Regulatory
Commission

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ABSTRACT

This report summarizes a study performed by Brookhaven National Laboratory for the Reactor and Plant Safety Issues Branch, RES of the U.S. Nuclear Regulatory Commission in pursuit of the resolution of NRC Generic Issue 99. Generic Issue 99 focuses on the risk associated with loss of residual heat removal events at PWRs while shut down. Numerous loss of residual heat removal events have occurred at pressurized water reactors (PWRs) in the USA, which were terminated prior to damaging the reactor core. This study estimates the risk from loss of residual heat removal events and investigates ways of lowering this risk.

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EXECUTIVE SUMMARY

This report summarizes a study performed by Brookhaven National Laboratory (BNL) for the Reactor and Plant Safety Issues Branch, Office of Regulatory Research of the U.S. Nuclear Regulatory Commission in support of the resolution of NRC Generic Issue 99. Generic Issue 99 deals with loss of residual heat removal events in pressurized water reactors (PWRs) while they are shut down. For example, on April 10, 1987, a loss of residual heat removal event occurred at Diablo Canyon Unit 2.[1-2] The cause of the loss of RHR was inadvertent draining of the reactor coolant system while the RCS was drained to the hotleg midplane. The RHR capability was not restored until 1 hour and 29 minutes later and the RCS was boiling with the pressure increased to approximately 7 to 10 psig. The Diablo Canyon event occurred after the bulk of this study was already completed, however, it serves to reemphasize the need to develop a resolution of this generic issue. This study attempts to assess what could be done to lower the frequency of occurrence of these events and to mitigate their consequences in the event that one occurs.

The starting point for the BNL study was NSAC-84,[3] "Zion Nuclear Plant Residual Heat Removal PRA." This probabilistic risk assessment was sponsored by the Electric Power Research Institute in cooperation with Commonwealth Edison Company. The benefits derived from using NSAC-84 as a starting point included a shutdown-specific data base, a detailed plant description for accurate modelling and insights into the progression of various accident sequences. NSAC-84 investigated three initiating event categories to provide a broad picture of shutdown risk. The BNL analysis includes two of those three initiating event categories (i.e., loss-of-cooling events and loss of coolant accidents (LOCA)), however, the BNL analyses separates out loss-of-offsite power (LOOP) events from the loss-of-cooling events. The third initiating event category from NSAC-84 (low temperature overpressurization events) was not included as it is being handled separately under NRC Generic Issue 94. Because Generic Issue 99 deals primarily with loss-of-cooling events and loss-of-cooling events dominate the results of this study, the insights derived from this study are focussed upon the loss-of-cooling event results. However, the results for all three initiating event categories are included in this report.

Modifications applied to the NSAC-84 model included redefinition of the phases of an outage, new estimates of the durations of phases (in particular, the duration that a plant stays in the partially drained condition), and the modelling of human errors. In the BNL analysis, generic shutdown data were collected and used to estimate the frequencies of initiating events. The results of the BNL analysis are based upon the Zion plant systems configuration under the assumption that the Zion plant is representative of a majority of the U.S. PWRs.

The BNL estimate of overall core damage frequency resulting during shutdown from this study is 5.22×10^{-5} per year. Loss-of-cooling events were estimated to represent a core damage frequency of 4.28×10^{-5} per year. Core damage frequency during shutdown is typically not included in probabilistic risk assessments (PRAs). Adding approximately 5×10^{-5} per year to a plant's overall core damage frequency would in most cases represent a nontrivial contribution. Therefore, BNL identified three potential improvements that would serve to reduce the risk during reactor shutdown conditions and these have been factored into this study. The three design/procedural improvements are briefly discussed below:

1 - Upgraded instrumentation for RHR pumps with emergency procedures governing shutdown conditions.
The upgraded instrumentation proposed for the residual heat removal (RHR) pumps would provide an alarm to alert the operators that RHR capability has been affected and information that would allow the operators to easily identify the cause of the problem. The attendant proposed emergency procedures would help the operators to determine the corrective actions needed to restore the RHR capability. The benefit of this design improvement is that the operators would be able to respond to loss of RHR events in a timely fashion. The 90% reduction in core damage frequency calculated for this improvement was derived by assuming that given the improvement the operators will always be able to diagnose the situation and initiate proper corrective actions.

2 - Upgraded vessel level indication.
Given reliable vessel level indication, the operators will be able to avoid overdraining in a draindown operation and will be able to maintain proper vessel level when the RCS is drained to the hotleg midplane. The benefit of this upgrade is a reduction of the initiating frequency of loss-of-cooling events and an attendant reduction in core damage frequency. A potential 22% reduction in the frequency of loss-of-cooling events was calculated by setting the probability of overdraining to zero. This upgrade results in an 78% reduction in calculated core damage frequency per year.

3 - Removal of auto-closure interlocks on the RHR suction valves.
With this design change, the frequency of spurious closure of an RHR suction valve would be significantly reduced. This design change reduces the frequency of loss-of-coolant events and reduces the calculated core damage frequency. Due to the large number of already experienced spurious isolation events, this event is an important contributor to the estimated frequency of loss-of-cooling events. The proposed design change results in a 60% reduction in the initiator frequency of loss-of-cooling events. The reduction in calculated core damage frequency based upon implementation of this possible upgrade is 8%.

The results obtained from implementing the above potential improvements are summarized in Table E.1. Also shown in Table E.1 are the results of implementing improvements one and two simultaneously. Upgraded instrumentation for the RHR pumps is the most effective change in terms of reducing the core damage frequency. Both upgraded vessel level instrumentation and removal of auto-closure interlocks (ACI) reduce the frequency of the initiating events. Removal of ACI is very effective in reducing the frequency of loss-of-cooling events, but its reduction in core damage frequency is smaller than that for upgraded vessel level instrumentation, because the spurious isolation of an RHR suction valve may occur any time during a shutdown and is fairly easily recovered while overdraining is postulated to occur only when the RCS is in a partially drained condition and recovery actions are more involved as well as lengthy. This is an important distinction because interruption of the RHR system during most of the time over a given shutdown yields ample time for operator recovery; whereas, loss of level instrumentation during a partially drained condition represents a much more vulnerable scenario.

A containment event tree was developed to assess the integrity of the containment given that a core damage event occurred during shutdown. Due to insufficient data on the top events in the event tree, it could not be fully quantified. Therefore, sensitivity calculations were done to assess the sensitivity of the containment event tree to uncertainty in the top events. Each of the containment event tree end points imply a fission product release path (or release category) from the damaged reactor core to the environment. A range of possible release categories during shutdown were estimated from previous calculations for accidents from power operation. The offsite consequences of the release categories were assessed using the MACCS[4] code. The results of these calculations are presented in the framework of sensitivity study. The sensitivity study addresses the possibilities of 1) having the equipment hatch open and not being able to close it, 2) having a containment penetration open and not being able to seal it, and 3) the potential for reducing the source terms given containment spray availability. The insights derived from the sensitivity study cover potentially beneficial changes to the Technical Specifications. The following insights were derived from the sensitivity study on the containment event tree and are listed by priority:

a. During partially drained conditions within the reactor coolant system (RCS), consideration could be given to a requirement that the equipment hatch either be in place or be in a position such that it could be closed quickly.

b. Also during partially drained conditions, consideration could be given to assuring that the containment penetrations are either closed or in a state in which they could be resealed quickly.

c. During shutdown conditions, consideration could be given to a requirement that a train of containment spray be kept available.

The following is a listing of other insights derived throughout this study concerning Technical Specifications. The purpose of this listing is to simply highlight the observations that have been made.

1. The inclusion of the availability of the safety injection (SI) system during shutdown lowered the calculated core damage frequency by about an order of magnitude. Current Technical Specifications (TS) require the disabling of SI as reactor pressure is reduced on the way to shutdown. Insights offered on this subject are the following: 1) disabling of the SI system should be done in a manner that would allow minimum effort for restoration and 2) simultaneous maintenance on all trains should be avoided.

2. NSAC-84 gives the maintenance unavailability of both charging pump trains as 6% (in Procedure Event Tree 3 of a refueling outage). This represents a relatively high unavailability and consideration could be given to avoiding simultaneous maintenance in this system.

3. Based on items 1 and 2 above, both of which address systems with injection capability, alternative consideration might be given to a Technical Specification requiring at least one train from either system be kept in an operable condition during shutdown. It is recognized that the SI system would be in a bypassed condition and in that case, "operable" would mean free from maintenance activities.

4. In reviewing the Zion Technical Specifications, it was found that, during shutdown conditions, if offsite power is available, both diesels could be in simultaneous maintenance. BNL has been informed that diesel power from the second Zion unit could be transferred to the first if both diesel generators were in maintenance and a loss of offsite power occurred. Although this may not therefore be a concern for Zion, if there are other plants with similar latitude in their Technical Specifications with no similar local sources of power, these plants would exhibit a higher vulnerability to core damage during shutdown.

With respect to the applicability of this generic analysis to specific plants, the following items are noted:

1. Some newer plants have double drop lines while Zion has only one. This difference is judged to have a small effect on the frequency of loss of RHR events. Because most of the time during an outage only one RHR train is operating, a spurious isolation signal to the suction valves of the operating train will lead to loss of cooling. The type of situation in which the double drop line helps most is when one suction valve has a mechanical failure and can not be opened by any means. However, the probability of such failure may be small compared with that for other failure modes, e.g., loss of suction due to overdraining. This should be evaluated on a case-by-case basis.

2. A dominant cause of core damage during a shutdown is due to the failure of the operator to respond. Operator performance depends on the information available to him. The analysis in this report assumes that the instrumentation and annunciators available for the operator are those given in Table 2.2 of NSAC-84. Therefore, there is no alarm assumed to be in the Generic Plant control room for low RHR pump suction pressure. In the latter stages of this study BNL was informed that such an alarm is being installed at Zion and this should help the operators to respond to events such as loss of pump suction.

3. The BNL analysis used the Zion plant configuration but did not consider the loss-of-component-cooling-water event as an initiating event, because the dependence of safety systems on component cooling water and service water systems may vary from plant to plant. For Zion, loss of component cooling water may lead directly to core damage. For Byron, loss of service water may lead to core damage. For other plants, some safety systems may not depend on component cooling water, or service water. An NRC survey of PWRs found that approximately 16 plants are potentially vulnerable to such types of dependence on support systems. Such issues can only be addressed on a plant-specific basis.

4. The BNL analysis (as well as the NSAC-84 analysis) has assumed that the progression to cold shutdown conditions will proceed in an orderly and unhurried fashion. This is a major assumption of the study as it factors into the determination of how much time is available for operator actions as a function of the decay heat rate of the core. The most significant benchmark of this time frame would be the modelled assumption that vessel level would not be drained to the hotleg midplane until 83 hours after reactor trip. Technical Specification - allowed cooldown rates, if actually followed, could yield a drained condition in as short a period as one day. Reference 2 identified a scenario that may cause core uncovery to occur sooner, i.e., a loss of cooling event occurs when the reactor coolant system is drained to the midplane of the hot leg with the cold leg opened due to maintenance of a reactor coolant pump, and the system becomes pressurized and forces coolant to flow out of the cold leg opening. The risk associated with any such practice is not bounded by this analysis. Such actions could represent a significant increase in shutdown risk based primarily upon a much more limited time available for operator recovery actions.

Table E.1
Summary of Benefits of the Proposed Improvements for Loss-of-Cooling Events

	Base Case	I1	I2	I3	I1+I2
f(LC) (per year)	3.21×10^{-1}	3.21×10^{-1}	2.49×10^{-1}	1.7×10^{-1}	2.49×10^{-1}
Δf(LC) (per year)	N.A.	~0	7.20×10^{-2}	1.19×10^{-1}	7.20×10^{-2}
% Reduction		0%	22%	60%	22%
Loss-of-cooling CDF (per year)	4.28×10^{-5}	4.27×10^{-6}	9.59×10^{-6}	3.94×10^{-5}	4.07×10^{-6}
Loss-of-cooling ΔCDF (per year)	N.A.	3.85×10^{-5}	3.32×10^{-5}	3.40×10^{-6}	3.87×10^{-5}
% Reduction		90%	78%	8%	90%

I1 = Upgraded instrumentation for RHR pumps.
I2 = Upgraded vessel level indication.
I3 = Removal of auto closure interlock.
f(LC) = Frequency of loss of cooling.
CDF = Core damage frequency.

1. INTRODUCTION

The objectives of this study are to establish the risk during plant shutdown on a generic basis which is representative of the PWR population in the USA and to obtain generic estimates of the risk reduction potential provided by various RHR design/operational changes. The results of this study will be used by the U.S. Nuclear Regulatory Commission in developing a generic PWR perspective of shutdown risk in pursuit of the resolution of NRC Generic Issue 99.

Pressurized Water Reactors (PWRs) in the USA have experienced numerous loss of residual heat removal (RHR) events. Of particular significance is the loss of the RHR system due to the inadvertent closure of the RHR suction valves or the lowering of the water level in the reactor vessel during drained RCS operations. For example, on April 10, 1987, a loss of residual heat removal event occurred at Diablo Canyon Unit 2.[1-2] The cause of the loss of RHR was inadvertent draining of the reactor coolant system while the RCS was drained to the hotleg midplane. The RHR capability was not restored until 1 hour and 29 minutes later and the RCS was boiling with the pressure increased to approximately 7 to 10 psig. This event occurred after the bulk of this study was already completed, however, it serves to reemphasize the need to develop a resolution of this generic issue. As a followup to the Diablo Canyon event, the NRC isssued Generic Letter 87-12[5] which serves to brief licensees on this event as well as request pertinent shutdown information from the licensees.

A plant in the shutdown state differs from an operating plant in many respects. During an outage, actuation of safety systems is in most cases not automatic, some safety systems may be intentionally disabled (e.g., accumulators and the safety injection system), and other safety systems may be unavailable due to maintenance. Also, during an outage, the response time available following an initiating event is longer than at power. Response time following a large loss of coolant accident (LOCA) at full power is measured in minutes whereas the response time to a loss-of-cooling event late in a shutdown could be measured in hours.

NSAC-84[3] is a pioneer study on the risk of a plant during shutdown. The study was prepared by Pickard, Lowe & Garrick, Inc., in cooperation withCommonwealth Edison Company and the Nuclear Safety Analysis Center and focused specifically on the Zion Nuclear Power Station. NSAC-84 was a useful reference document from which BNL was able to draw insights when developing the generic PWR shutdown model. The derivation of the BNL shutdown model from NSAC-84 is discussed in Section 2. The BNL model includes two of the three initiating event categories included in the NSAC-84 work (i.e., loss-of-cooling and LOCA). In the BNL model, loss of offsite power events are separated from the loss of cooling events yielding a third initiating event category. (The third initiating event category from NSAC-84 was low temperature overpressurization events. These events are undergoing independent study under NRC Generic Issue 94.) Inclusion of the three initiating event categories within this study gives a broad picture of shutdown risk. However, since Generic Issue 99 deals with loss-of-cooling events and these dominate the calculated core damage frequency, the conclusions and insights presented in this study focus on the loss-of-cooling events.

The BNL shutdown model is based upon the assumption that the Zion plant configuration is representative of a majority of U.S. PWRs and utilizes actual operational experience of PWRs in the period of 1976-1986 to estimate the frequencies of various initiating events. Generic component failure data was used in the quantification of the core damage frequencies.

Section 3 presents an analysis for a generic plant using the BNL shutdown risk model. In addition, three potential design/procedural improvements have been identified. Estimates for the reduction in the frequency of loss-of-RHR events and the reduction in core damage frequency are also included for the three potential improvements. Section 4 presents a sensitivity study of the consequences and risks associated with an accident at a generic PWR during a shutdown. Section 5 presents a summary of the work, the insights and conclusions, a commentary on plant-specific applications of the results of this report, and some observations on Technical Specifications.

2. SHUTDOWN RISK MODEL

2.1 Approach

NSAC-84 used the large event tree approach in their study of the Zion plant, while the BNL analysis of the generic plant used the large fault tree approach. In principle, the two approaches are equivalent, and should yield similar results. NSAC-84's approach differs from the typical large event tree approach, such as that used in the Zion Probabilistic Safety Study (PSS),[6] in that event trees are developed to assess the frequency of initiating events. These event trees are called procedural event trees and represent different phases during a shutdown or refueling. Also in NSAC-84, three accident event trees for 1) loss of cooling, 2) LOCA, and 3) low temperature overpressurization are used to model the progression of the accidents. Figure 2.1 is an event tree map showing the six procedural event trees and the three accident event trees that defined the scope of the NSAC-84 study. BNL did not use the procedural event tree approach. Instead, the initiating event frequencies were estimated using generic PWR experience. The initiating events considered in the BNL analysis are 1) loss of cooling, 2) LOCA, and 3) loss of offsite power. The event trees of the BNL analysis are derived from the accident event trees of NSAC-84.

2.2 Definition of Loss of Cooling Initiating Event

Loss of cooling with failure of operators to respond is identified in NSAC-84 as the dominant contributor to core damage during shutdown. The top event RM (used in all procedural event trees in NSAC-84) is defined to be failure of both trains of RHR, i.e., failure of just the operating train is not considered loss of RHR, the standby train must also fail. However, the standby train does not start automatically, and operator actions are required. NSAC-84 models operator responses to loss of RHR in the loss of cooling event tree. Therefore, loss of the operating train with operator failure to recognize the need to restore cooling is not considered in NSAC-84. In the BNL analysis, the initiating event of loss of cooling is defined to be loss of the operating train and the loss of the operating train coupled with operator failure to recognize the need to restore cooling is explicitly included.

2.3 Model of Electric Power System

NSAC-84 defines eight power states in terms of the availability of the 4 kV buses. Appendix C of NSAC-84 provides an analysis of the electric power system. The probabilities of the power states, given a loss of offsite power, were calculated taking into account hardware failures and maintenance of the diesel generators and buses and are tabulated in Table 2.1 (reproduced from the table on page C-63 of NSAC-84). The frequency of loss of offsite power for three types of outages (refueling, drained maintenance, and nondrained maintenance) were also calculated in NSAC-84 and they are 1.66×10^{-2}, 5.09×10^{-3}, and 7.04×10^{-3} per year, respectively. The frequency that a loss of offsite power occurs and the electric power system is in a given power state was calculated in NSAC-84 as the product of the frequency of loss of offsite power and the probability of the power state given a loss of offsite power. The NSAC-84 results for three types of outages are tabulated in Table 2.2 (reproduced from Table C.27 of NSAC-84).

The last row of Table 2.2 represents the frequency of station blackout and if offsite power is not restored, core damage was assumed to occur. NSAC-84 did not document how recovery of offsite power is modelled. NSAC-84 estimated that loss of offsite power only contributes 2.3×10^{-7} per year to the core damage frequency during shutdown. BNL modelled such core damage scenarios using the large fault tree approach and recovery of ac power was included. This approach is described in Section 3.3.

2.4 Frequency of Spurious Signals That Auto-Close the RHR Suction Valves

NSAC-84 estimated the frequency of spurious auto-closure signals for Zion using a Bayesian approach. The evidence used was three spurious isolation events caused by unidentified causes in 27,888 hours. The prior distribution used (with a mean of 2.33×10^{-8} per hour) was a distribution that applies primarily to mechanical failure of valves and does not include the effects from unique sources of spurious control signals. Use of this particular prior distribution artificially reduced the calculated frequency of spurious isolation signals to 1.38×10^{-5} per hour.

Strictly using the evidence, BNL recalculated the frequency to be 3/27,888 hours = 1.08×10^{-4} per hour, which is a factor of eight higher than the NSAC-84 value. To further verify the calculation, BNL performed a two stage Bayesian analysis. The experience for the generic PWR population was used as the evidence for the first stage and the Zion experience of three events in 27,888 hours was used as the evidence for the second stage. The mean of the BNL posterior distribution then became 9.34×10^{-5} per hour. This estimated frequency is applicable to the Zion plant and was not used in the BNL analysis.

In the BNL survey of operational experience for PWRs (discussed in Section 3.1), 64 events of spurious closure of the valves that isolate the RHR from the reactor coolant system were identified. In Section 3.1.1 these events are used to estimate the frequency of spurious isolation signals.

2.5 Definitions of Phases Within Outages

The phases of an outage are defined in terms of the time at which a phase starts and the time at which a phase ends, and are characterized by the conditions of the plant such as to whether or not the RCS is drained and whether or not the RCS is open. The plant conditions are then used to determine the time available for operator actions and the human error probabilities. Table 2.3 summarizes the BNL definitions of the phases for the three types of outages. The sum of the durations is equal to the mean duration of that type of outage as given in NSAC-84.

Different phases of an outage occur sequentially. Therefore, decay heat is lower for later phases, and the time available for operator actions, given a loss-of-cooling event, tends to be longer. The most vulnerable condition of a plant during shutdown is when the RCS is drained. NSAC-84 estimated that for a drained maintenance outage at Zion, it takes 83 hours on the average to bring the RCS to the drained condition. The simple thermal model of Appendix A estimates that less than three hours is available for recovery before the core becomes uncovered if a loss-of-cooling event occurs when the RCS is drained at 83 hours after shutdown.

Refueling Outage

Phase 1 - This phase starts with the initiation of RHR cooling in a refueling outage and ends when the RCS is drained to hotleg midplane. NSAC-84 estimated that the mean time at which the RHR system is initiated in a refueling outage is 54 hours after shutdown. This represents the starting time of phase one. NSAC-84 also estimated that RCS draining is initiated at 118 hours after shutdown and takes 49 hours to complete draining. Therefore, drained conditions are reached at 118 + 49 = 167 hours. This is the ending time of phase one. The duration of phase one is therefore 167 - 54 = 113 hours.

In the BNL analysis, Phase 1 is characterized as a phase with the RCS filled and the decay heat is relatively high. Table 2-6 of NSAC-84 estimated that 3.8 hours will be available for recovery actions prior to the onset of core damage if loss-of-cooling occurs at 6 hours after shutdown with the RCS at 425 psig, 350°F and a bubble in the pressurizer. BNL used this 3.8 hour value as input to the human error probability calculation performed for this event. (The HCR model and this calculation are described in Section 2.6.)

Phase 2 - In this phase, the RCS is drained to the hotleg midplane, so that tests and maintenance can be performed on the steam generators and other components of the primary coolant system. Information from a few plants indicates that a plant may spend approximately 2 to 3 weeks per refueling outage in the drained condition.[11-12] For the subsequent analysis, 2.5 weeks is used as the duration of this phase, i.e., phase 2 starts at 167 hours after shutdown and ends at 587 hours. With the RCS partially drained, the time to core uncovery for a loss-of-cooling event can be determined using the model in Appendix A.

Phase 3 - In this phase, the refueling cavity is filled and actual fuel shuffling takes place. NSAC-84 estimated that the time with the vessel head off is typically 500 hours. This is used as the duration of this phase. Therefore, phase 3 is assumed to extend from 587 hours and is assumed to end at 1087 hours. With the refueling cavity filled, several days may be available for the operator to recover the decay heat cooling capability. Under such conditions, the probability of human error for failure to diagnose and recover can be expected to be negligibly low. As discussed in Section 2.6, the HCR model places a limiting human error probability of 10^{-6} on any human action and this value is used in determining the failure probability under these circumstances. At the end of this phase, the refueling process is completed, the vessel head is back on, and the RCS is full.

Phase 4 - In this phase, test and maintenance after refueling is performed. The RCS is filled and one-third of the fuel is fresh. Again, several days are available for operator actions to respond to loss-of-RHR cooling. The duration of this last phase of a refueling outage is determined such that the total duration of the outage is 1996 hours as estimated in NSAC-84 (i.e., 1996-1087 = 909).

Drained Maintenance Outage

Phase 1 - This phase is similar to Phase 1 of a refueling outage. It starts when the RHR system is initiated and ends when the RCS is drained. Table 3-4 of NSAC-84 lists the times to RHR initiation for maintenance outages at Zion. The mean time is approximately 21 hours. This value has been used as the time at which Phase 1 starts. Figure 3-4 of NSAC-84 estimated that the draining of the RCS is started at 54 hours and the task takes 29 hours to complete. Therefore, the phase ends at 54 + 29 = 83 hours.

Phase 2 - This phase is similar to Phase 2 of a refueling outage, except that the RCS is drained sooner and the duration of the phase is shorter. The phase starts at 83 hours after shutdown. The decay heat is relatively high at this time. With minimal amount of coolant inventory in the system, the time available for the operators to respond to any abnormal event is relatively short. Appendix A estimates that approximately 2.7 hours will be available before core uncovery occurs, if a loss-of-cooling event occurs at the beginning of this phase. The duration of this phase is estimated to be 4 days based on information obtained from Oconee.[8]

Phase 3 - This phase starts when the maintenance activities that require the RCS to be drained are completed at 179 hours, and the duration is estimated so that the sum of the durations of the three phases is equal to the duration of the drained maintenance. Based on the data obtained from Zion drained maintenance outages, NSAC-84 obtained a mean value for duration of a drained maintenance as 982 hours. Thus, the duration of phase 3 is 982 - 179 = 803 hours. During phase 3, the RCS is filled while test and maintenance activities are being performed, and the time available for recovery can be expected to be significantly greater than under conditions when the water is drained to the mid-loop of the nozzles.

Nondrained Maintenance

Only one phase is used to model this type of outage. It is similar to Phase 1 of a drained maintenance outage. For the nondrained maintenance outage RHR cooling is initiated at approximately 21 hours after a shutdown. The duration of this phase is taken to be 125 hours.

2.6 Human Error Probabilities for Loss of Cooling Events

Loss of cooling events are grouped into two types, overdraining events and other loss of cooling events. Overdraining events are modelled separately because the recovery actions tend to take more time and they can only occur when the reactor coolant system is partially drained. Other loss of cooling events include events such as failure of the running RHR pump and spurious closure of the RHR suction valves. There are two human actions for which human error probabilities (HEPs) are needed to be assessed for each of the two types of loss-of-cooling events. These are the failure of the operating crew to diagnose that core cooling has been lost and the failure of the crew to restore cooling (decay heat removal). A discussion of each follows.

2.6.1 Failure to Diagnose Loss of Cooling

In the BNL analysis, the Human Cognitive Reliability (HCR) model of References 9 and 10 was used for diagnosis of loss of cooling. This model utilizes the time available for operator diagnosis as a parameter in determining the HEP. BNL has used this model to quantify the event trees for loss-of-cooling events and LOCAs. By contrast, the Zion shutdown risk model in NSAC-84 uses a fixed loss-of-cooling event tree) without regard to the time available before the core uncovers. A brief description of HCR model follows along with a comparison with the model given in the Handbook for Human Reliability Analysis.[11]

The HCR model is described by:

$$P(t) = \exp -\left\{ \frac{t/T_{1/2} - C_{\Gamma_i}}{C_{\eta_i}} \right\}^{\beta_i} \qquad (2.1)$$

where,

t = time available for the operating crew to diagnose,

$T_{1/2}$ = estimated median time taken by the operating crew to diagnose,

C_{Γ_i}, C_{η_i}, β_i = correlation coefficients associated with the i-th type of mental processing, e.g., skill, rule or knowledge which can be calibrated with simulator data, and

$P(t)$ = the crew non-diagnosis probability for a given time t.

The specific application of this model within the BNL analysis is described as follows. The HEP for operator response to each of the initiating events has been calculated using Equation 2.1 where HEP(t) = P(t). Section 3.1.3 reviews operational experience of loss of decay heat removal events, and estimates that on the average it took approximately 15 minutes for the operator to diagnose and recover from events such as spurious closure of the RHR suction valves and spurious RHR pump trips. This mean value of 15 minutes actually represents diagnosis plus recovery in the data base. Because diagnosis and recovery portions of this mean are not discernible from the data, BNL has conservatively taken the entire 15 minute mean time interval and input this value as representing diagnosis time $T_{1/2}$ in the HCR model. If a shorter diagnosis time were input, a lower HEP would result.

The variable "t" in Equation 2.1 represents the time that is available for operating crew diagnosis. This time is estimated based upon the condition of the reactor coolant system (RCS) when the initiating event occurs. For example, Appendix A describes a simple thermal model that has been used to determine the time to core uncovery with the reactor in a partially drained condition as a function of time after shutdown (i.e., decay power level). Table 2.4 lists the time to core uncovery for a 3400 MWt reactor as a function of the time after shutdown at which loss of cooling occurs assuming that the initial RCS level is at the midplane of the reactor vessel nozzles and the RCS hotleg is vented to the containment. The time to core uncovery is used as the time available for restoring decay heat removal, including diagnosing the situation, deciding on the actions to take, and actually carrying out the actions. Therefore, the time available for diagnosis is equal to the time to core uncovery minus the time needed to carryout the actions. The time needed to implement the actions depends on the specific cause of the loss of cooling event. Section 3.13, using operational experience, estimated that on the average 49 minutes are needed to restore decay heat removal in an overdraining event and 15 minutes are needed for other loss of cooling events. These estimated times are the actual time needed for diagnosis and recovery. They were conservatively useu as the time needed to carryout the recovery action.

It is assumed that adequate abnormal or emergency procedures are not available and that the operating crew are not adequately trained to respond to a loss-of-cooling event during an outage. Therefore, the cognitive task is considered to be "knowledge based" within the context of the HCR model.[9-10] The following model parameters were obtained directly from Table 2 of Reference 10 for this type of task using the small scale test data:

$\beta_i = 0.81$, $C_{r_i} = 0.527$, and $C_{\eta_i} = 0.744$.

These parameters were used to calculate the HEPs also listed in Table 2.4.

Figure 2.2 is a comparison between the HCR model using Equation (2.1) and the model taken from Figure 12-3 of the Handbook of Human Reliability Analysis.[11] The curve marked with crosses was calculated using Equation (2.1) for t>60 minutes, and it decreases rapidly over several orders of magnitude with available time. For t>300 minutes the HCR model arbitrarily assumes a minimum HEP of 10^{-6} to hold. The other curves in Figure 2.2 were taken from the Handbook.[11] The solid curve from the Handbook represents the median curve. The upper bound and the lower bound were obtained by assuming an error factor of 30. The mean curve is a factor of 8.5 higher than the median curve. It can be seen that the HCR model curve decreases more rapidly with time yet falls within the uncertainty bounds of the model taken from the Handbook.

2.6.2 Failure to Restore Cooling

Two types of loss of cooling events were considered separately. Overdraining events are more difficult to recover from, because operator actions outside the control room such as venting the vapor bound RHR pumps may be required. Other failures of the RHR system are easier to recover from. For example, if the operating pump fails to continue running, the operator can manually start the standby pump from the control room. The degree of difficulty in recovering RHR is reflected in the experienced RHR recovery time, i.e., 49 minutes for overdraining events and 15 minutes for other failures.

Overdraining Events

To date there has been no in-depth human reliability analysis (HRA) undertaken to model the human action to restore decay heat removal cooling which specifically quantifies the probability of such an event. As a viable alternative to such a major undertaking, BNL has used its Human Error Reliability Analysis (HERA) data base to search for quantified human error events documented in existing PRAs which provide appropriately similar human reliability situations as the failure to restore decay heat removal cooling event.

The HERA computerized data base has been developed to support an NRC program aimed at improving the usefulness of PRA results in addressing human risk issues. It contains all published human error events (which totals 1976) as extracted from 65 volumes of 19 different PRAs. The data base is summarized and documented in NUREG/CR-4103.[12] From the HERA data base, two well documented human error events emerge from the search which are judged to provide acceptable overall operator/event similarity to the failure to restore decay heat removal cooling event.

Both of the two similar events were developed to support the HRA needs of a small break LOCA accident for the Arkansas Nuclear One, Unit 1 (B&W-PWR) nuclear power plant as part of the NRC sponsored IREP.[13] The first involves the failure to initiate high pressure injection (HPI) while the other deals with the failure to switch over safety injection suction from the borated (refueling) water storage tank to the containment sump. Both events (as analyzed in Appendix A15 of Reference 13) assume that each particular situation has been diagnosed correctly and that the operating crew is under "moderately high stress." The level of complexity has been analyzed and judged to be qualitatively comparable (especially for the switchover of the safety injection suction line) to the event of interest here. The HEP established by detailed HRA for each of these events is documented as 1×10^{-4}. Therefore, without an extensive HRA performed specifically for the failure to restore decay heat removal cooling event, the HEP value of 1×10^{-4} will be used based on the above similarity of events.

Note that while some plants may not have significantly developed procedures for restoring decay heat removal cooling, licensed operators are periodically trained on a reactor plant simulator to respond to inadequate core cooling events as part of their emergency operating training. This training should provide the instinctive response to restore core cooling during shutdown.

Other Loss of Cooling Events

Since the recovery action for other loss of cooling events is much simpler than that of overdraining events, a HEP of 10^{-5} will be used for other failure events, i.e., a factor of ten reduction in HEP.

2.7 Support Systems Model

The NSAC-84 analysis only considered the service water (SW) system and the component cooling water (CCW) system as support systems for the RHR system. The dependence of the charging system and the auxiliary feedwater system on these support systems was not modelled. BNL modelled dc power, SW and CCW as support systems for all systems that may be used to mitigate the postulated accidents. A description of the BNL analysis of these support systems is given in the following subsections.

Component Cooling Water System - If an initiating event such as loss of RHR or loss of offsite power occurs, and CCW subsequently fails, core damage may occur because the RHR pumps and charging pumps will lose their cooling. NSAC-84 only modelled the dependence of the RHR system on CCW. In the BNL analysis, CCW is included in the fault trees for the RHR; Chemical and Volume Control System (CVCS), and safety injection systems.

In the Sandia review[14] of ZPSS, the frequency that a pipe rupture would lead to a loss of CCW was estimated to be 2.3×10^{-4} per year and the frequency of other failures leading to loss of CCW was estimated to be 7.1×10^{-4} per year. After the Sandia review, Commonwealth Edison submitted a revised evaluation which indicated that only about 6% of the $2.3x10^{-4}$ per year value was due to rupture and that the remaining 94% represented other leakage. Therefore, the frequency of pipe rupture leading to loss of CCW was:

$2.3x10^{-4}$ per year * 6% = $1.38x10^{-5}$ per year.

Loss of CCW was modelled as a basic event with a frequency equal to the frequency of the initiating event of loss of CCW. Consequently, from the above considerations the frequency of loss of CCW is:

$1.38x10^{-5}$ per year + $7.1*10^{-4}$ per year = $8.26*10^{-8}$ per hour.

A mission time of 24 hours was used.

Service Water System

The service water (SW) system is modelled as a basic event in the fault trees for the RHR, Auxiliary Feedwater (AFW), CVCS, and safety injection systems. The frequency of loss of SW was estimated to be $9.4*10^{-4}$ (from ZPSS) per year and a mission time of 24 hours was used.

DC Power System

Failure of a dc bus may cause the failure of a pump or a diesel generator to start. Such dependence was modelled in the fault trees of the RHR, AFW, CVCS, and safety injection systems. The failure rate ($3.6*10^{-6}$ per hour) of a bus was taken from the BNL review[15] of Oconee Probabilistic Risk Assessment[16] (OPRA). The mission time was taken to be 24 hours.

2.8 Common Cause Failures

Table 2.5 lists the beta factors used in the NSAC-84 and the BNL analyses. The reference sources of the beta factors used in the BNL analysis are also listed.

The use of a beta factor for the diesel generators increased the conditional probability of station blackout given a loss of offsite power by approximately 10%. The use of beta factors for the various pumps had a much smaller effect on the unavailability of multiple systems because they only affect individual systems whereas multiple system failures are dominated by maintenance unavailability of essential buses or failure of the support systems.

2.9 Unavailability of Components Due to Maintenance

NSAC-84 used the Zion control room operating log book records to estimate the maintenance unavailability of components during shutdown. The result is summarized in Table 2.6 (Table 4-3 of NSAC-84). Table 2.6 shows that each charging pump is individually unavailable 15.4% of the time. Table 2.6 also shows that both charging pumps are simultaneously unavailable due to maintenance, with probability $6.5x10^{-2}$ (for Phase 3 of a refueling only). Therefore, maintenance unavailability is the dominant cause of the unavailability of the charging pumps. In the quantitative analysis of the systems unavailabilities and the core damage frequency, NSAC-84 did not consider the maintenance unavailability of the charging pumps nor the safety injection pumps. Maintenance unavailabilities were included in the BNL analysis.

2.10 Use of the Safety Injection System

Technical specifications typically require that the safety injection system be made inoperable during plant shutdown and cooldown. This would be done by switching the pumps to the "PULL-TO-LOCK" position, racking out the supply breakers of the pumps, and closing the pump discharge MOVs. Therefore, the system can not be made operable from the control room. However, in most cases a lot of time would be available for the operators to make the system operable. Therefore, the safety injection system should be included in the analysis. In NSAC-84, no credit was given to the safety injection system in the loss-of-cooling event tree. In the BNL analysis, the safety injection system was modelled as the last top event in the loss-of-cooling event tree. If the safety injection system can not be made available, the core damage frequency due to loss-of-cooling events has been estimated to increase by at least an order of magnitude.

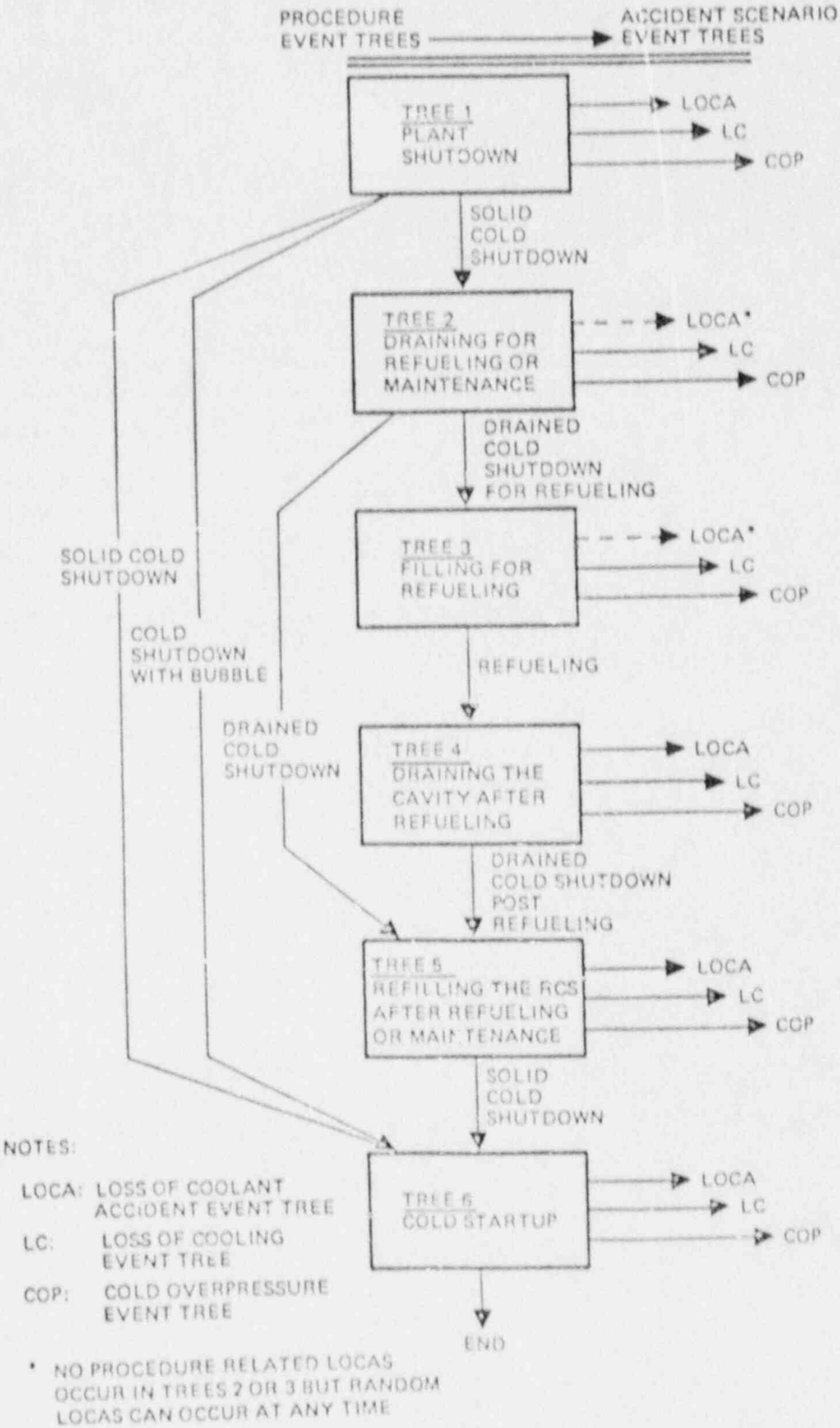


Figure 2.1. Event tree map (Figure 2-1, NSAC-84).

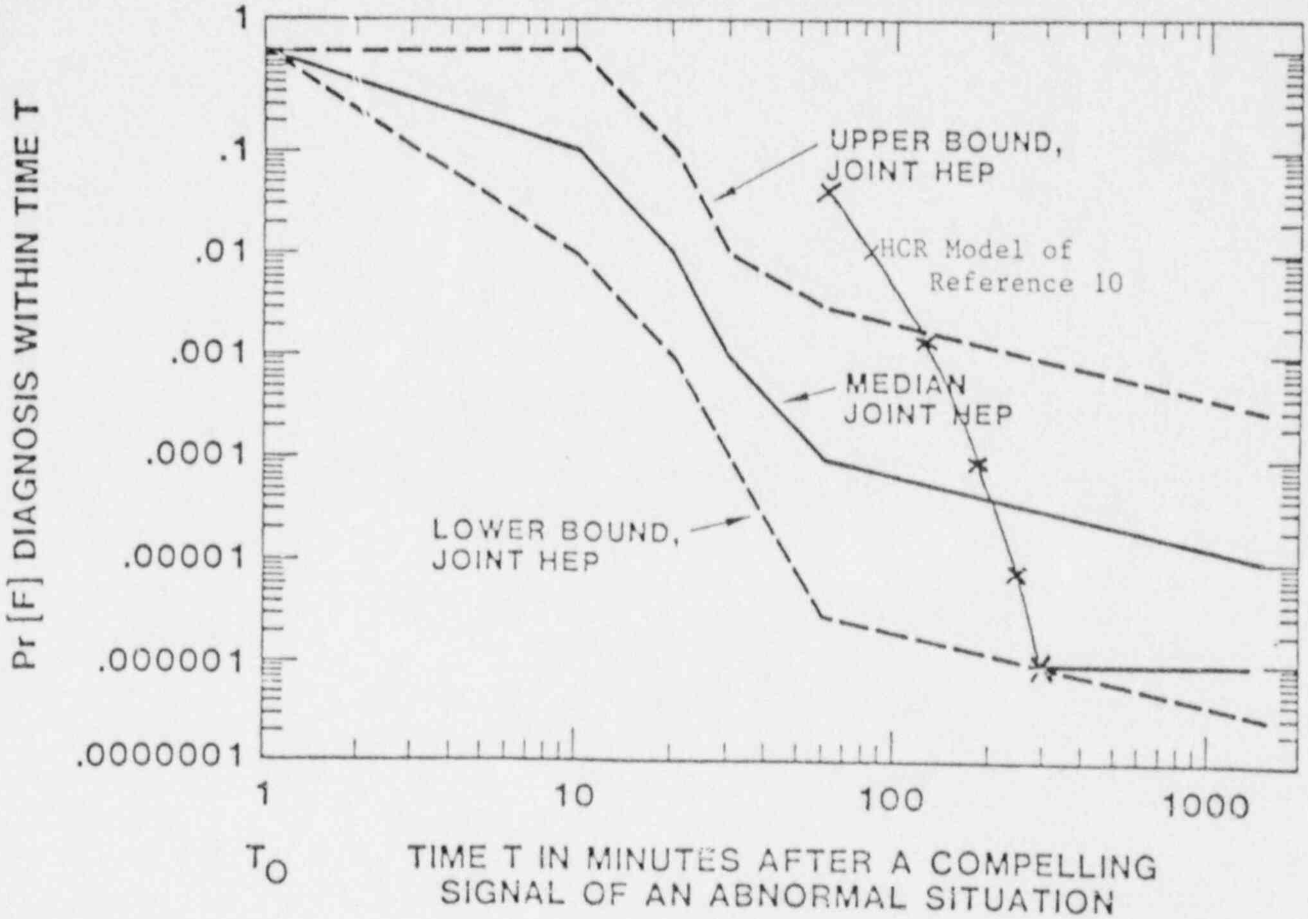


Figure 2.2. A comparison of the HCR model used with the model of the Handbook for Human Reliability Analysis.[11]

Table 2.1
Probabilities of Power States Given a Loss of Offsite Power (NSAC-84 Results)

State	Failed	Success	Mean
1	-	147, 148, 149	3.05×10^{-1}
2	$\overline{147}$	148, 149	4.69×10^{-1}
3	$\overline{148}$	147, 149	6.11×10^{-2}
4	$\overline{149}$	147, 148	4.51×10^{-2}
5	$\overline{147}$, $\overline{148}$	149	5.49×10^{-2}
6	$\overline{147}$, $\overline{149}$	148	5.57×10^{-2}
7	$\overline{148}$, $\overline{149}$	147	4.05×10^{-3}
8	$\overline{147}$, $\overline{148}$, $\overline{149}$	No Power	4.96×10^{-3}

Table 2.2
Frequency That a Loss of Offsite Power Occurs and The Electric Power is in a Given Power State (NSAC-84 Results)

Power State (Buses Available)	Refueling	Drained Maintenance	Nondrained Maintenance
147, 148, 149	5.06×10^{-3}	1.93×10^{-3}	2.67×10^{-3}
148, 149	7.78×10^{-3}	2.29×10^{-3}	3.17×10^{-3}
147, 149	1.01×10^{-3}	1.83×10^{-4}	2.54×10^{-4}
147, 148	7.48×10^{-4}	1.83×10^{-4}	2.54×10^{-4}
149	9.11×10^{-4}	2.30×10^{-4}	3.18×10^{-4}
148	9.24×10^{-4}	2.30×10^{-4}	3.18×10^{-4}
147	6.72×10^{-5}	1.63×10^{-5}	2.26×10^{-5}
No Power	8.23×10^{-5}	1.96×10^{-5}	2.71×10^{-5}

Table 2.3
Durations and Characterization of Phases
of Three Types of Outages

Outages	Phase	Start* (hr)	End* (hr)	Duration (hr)	Plant Conditions
Refueling	1	54	167	113	RCS Cooling Down, RCS Draining
Refueling	2	167	587	420	RCS Drained to Hot Leg Midplane, SG Eddy Current Test
Refueling	3	587	1087	500	Refueling Cavity Filled, Fuel Shuffling, RCS Filled, Vessel Head Off
Refueling	4	1087	1996	909	RCS Filled, Maintenance
Drained Maintenance	1	21	83	62	RCS Cooling Down, RCS Draining
Drained Maintenance	2	83	179	96	RCS Drained, Maintenance
Drained Maintenance	3	179	982	803	RCS Filled, Maintenance
Nondrained Maintenance	1	21	146	125	RCS Filled, Maintenance

*Time after shutdown.

Table 2.4
Core Uncovery Time and Human Error Probability of Failure to Diagnose when the RCS is Drained

Time at Which Loss of Cooling Occurs (Hours After Shutdown)	Core Uncovery Time* (Hour)	Probability** (Failure to Diagnose)
83	2.7	$2.3*10^{-4}$
103	3.0	$1.0*10^{-4}$
123	3.2	$6.2*10^{-5}$
143	3.4	$3.7*10^{-5}$
167	3.6	$2.3*10^{-5}$
179	3.7	$1.8*10^{-5}$
187	3.8	$1.4*10^{-5}$
207	3.9	$1.1*10^{-5}$
227	4.1	$6.6*10^{-6}$
247	4.3	$4.1*10^{-6}$
267	4.4	$3.2*10^{-6}$
287	4.5	$2.5*10^{-6}$
307	4.7	$1.6*10^{-6}$
327	4.8	$1.2*10^{-6}$
347	5.0	10^{-6}
367	5.1	10^{-6}
387	5.2	10^{-6}
407	5.4	10^{-6}
427	5.5	10^{-6}
447	5.6	10^{-6}
467	5.8	10^{-6}
487	5.9	10^{-6}
507	6.0	10^{-6}
527	6.2	10^{-6}
547	6.3	10^{-6}
567	6.4	10^{-6}
587	6.5	10^{-6}

*Determined using the model of Appendix B.
**Calculated using Equation (1).

Table 2.5
Beta Factors Used in NSAC-84 and BNL Analyses

	NSAC-84	BNL	Reference (BNL)
RHR Pumps:			
Failure to start	5.0×10^{-2}	9.4×10^{-2}	NUREG/CR-2098
Failure to run	---	4.1×10^{-2}	NUREG/CR-2098
AFW Pumps:			
Failure to start	---	1.48×10^{-1}	NUREG/CR-2098
Failure to run	---	1.48×10^{-1}	NUREG/CR-2098
Charging Pumps:			
Failure to start	---	1.96×10^{-1}	NUREG/CR-2098
Failure to run	---	2.8×10^{-2}	NUREG/CR-2098
Diesel Generators:			
Failure to start	---	0.05	EPRI-NP-3967
PORV: Failure to open	---	7.0×10^{-2}	EPRI NP-3967

Table 2.6
Component Unavailability Data During an Outage
(Table 4-3 of NSAC-84)

Component	Unavailability		
	Mean	Median	Variance
Essential[a,b] 4,160V Buses			
47	1.12-2	1.19-2	5.41-5
48	6.56-3	6.32-3	1.80-6
49	8.16-3	9.10-3	1.65-5
Nonessential[b] 4,160V Buses			
42	7.51-3	5.91-3	2.30-5
43	1.08-2	1.36-2	3.30-5
44	8.56-3	7.88-3	1.87-5
45	2.33-2	1.54-2	6.30-5
Nonessential[c] 480V Buses	1.61-2	1.26-2	1.57-4
Instrument Buses	4.81-3	4.40-3	1.23-5
Diesel Generators	4.56-2	1.72-2	2.26-3
Batteries	4.81-3	4.40-3	1.23-5
Inverters	4.81-3	4.40-3	1.23-5
RHR Pumps	6.06-2	3.81-2	6.28-3
Centrifugal Charging Pumps	1.54-1	1.54-1	1.63-3
Both Centrifugal Charging Pumps	6.52-2	7.17-2	1.53-3
Service Water and Component Cooling Pumps	1.48-2	1.08-2	9.85-4

Note: Exponential notation is indicated in abbreviated form; i.e., 1.12-2 = 1.12×10^{-2}.

[a]Values applicable to essential 480V buses.

[b]Where no maintenance duration time has been given, a generic out of service time of 12 hours has been assigned.

[c]Values applicable to all nonessential 480V buses.

3. QUANTIFICATION OF THE BNL SHUTDOWN MODEL

3.1 Generic Data Collection and Analysis

This section provides discussions on the collection and analysis of the generic data that are needed for the generic assessment. Section 3.1.1 discusses the operational experience of loss of DHR. Section 3.1.2 uses this operational experience to estimate the frequencies of initiating events that will lead to loss of DHR. Section 3.1.3 discusses the estimation of the mean diagnosis time of loss-of-cooling events. Section 3.1.4 discusses the generic data for component f. ilures.

3.1.1 Operational Experience of Loss of DHR

Three sources of operational experience related to loss of RHR were used in the generic data analysis. The three sources cover different periods of time. Table 3.1 summarizes a total of 177 of loss of RHR events. Of these, 86 events occurred in the period from 1976 to 1981. These events were identified in NSAC-52[17] and described in Appendix A of that report. Some events in NSAC-52 were excluded from Table 3.1 on the basis that they occurred prior to initial criticality. The major data source of NSAC-52 was Licensee Event Reports (LERs). The data for the period 1982-1983 were obtained by the Office for Analysis and Evaluation of Operational Data (AEOD)[18] from LERs and NRC reports. The AEOD report[18] stated that 56 events were covered by that study and descriptions of 45 events that occurred in 1982 or 1983 were provided in Appendix A of the AEOD report. Table 3.1 includes all 45 events that were listed in Appendix A of the AEOD report. The data for the period from 1984 to 1986 were collected by BNL using the Sequence Coding Search System[19] (SCSS). The LER search using SCSS was performed on March 4, 1987. From this, 163 events were identified by specifying keywords of RHR systems, PWRs, and failures. By reviewing the abstracts of these events, 46 events were judged to be loss of DHR events. The abstracts of these events are listed in Appendix B to this report. For the convenience of quantitative analysis, these events have been classified into five types of failures: 1) spurious isolation of RHR suction valves, 2) overdraining the RCS, 3) failure to maintain RCS level, 4) loss of RCS coolant, and 5) other failures. Table 3.2 summarizes the operational experience according to this classification. Descriptions of the events covered by NSAC-52[17] and the AEOD report[16] were reviewed and the events were classified accordingly.

The following items concerning the operational experience of loss of RHR are also noted:

1. The operational experience summarized in Tables 3.1 and 3.2 is mainly based on LERs. There may be more events that were not reported. For example, NSAC-84 states that 16 events of spurious isolation by the RHR suction valves occurred at Zion in the period from 1975 to 1982, while only one event is found in the LER data base.

2. Table 3.1 lists the annual frequency of loss of RHR events as a function of time. It can be seen that there is no obvious indication that the frequency is reducing.

3.1.2 Estimation of the Frequencies of Initiating Events Leading to Loss of DHR

The operational experience summarized in Table 3.2 has been used to estimate the frequency of loss of DHR due to different causes. It can be estimated, using the Gray Book,[20] that all PWRs have accumulated approximately 504 years of operating experience from 1976 to 1986. The number of hours that a plant stays in a shutdown condition is the sum of the numbers of hours that the plant stays in 3 types of outages, i.e.:

0.747 refueling/year * 1996 hour/refueling
+ 1.932 drained maintenance/year * 982 hour/drained maintenance
+ 1.121 nondrained maintenance/year * 146 hour/nondrained maintenance
= 3550 hours/year.

Therefore, the total experienced operating time of the RHR system is estimated to be:

504 year * 3550 hours/year = $1.79 * 10^6$ hours.

The frequencies of different initiating events that cause RHR systems to become unavailable are estimated below and are summarized in Table 3.3.

a. Spurious Isolation of RHR Suction Valves. The frequency of spurious isolation signals is simply the number of spurious isolation events divided by the number of RHR system operation hours, namely:

   $64/1.79\times10^6$ hour = 3.58×10^{-5}/hour.

b. Overdraining. This event was modelled as a failure on demand. The probability of overdraining was estimated by dividing the number of overdraining events by the estimated number of drain down operations. It was assumed that the frequencies of the three types of outages estimated for Zion in NSAC-84 were representative of those for the PWR population. The RCS is drained once per drained maintenance and twice per refueling outage. Therefore, the number of drain down operations is:

   504 year * (frequency of drained maintenance + 2 * frequency of refueling)
   = 504 * (1.932 + 2 * 0.747)
   = 1.73×10^3
   The probability of overdraining is equal to
   number of overdraining events/1.73×10^3
   = $21/1.73\times10^3$
   = 1.21×10^{-2}

c. Inadequate Inventory. This represents failures to maintain vessel level. For the purposes of this study, it has been assumed that such an event may occur only when the RCS is in the drained condition. It was modelled as an event that occurs with a constant frequency. The frequency that such an event occurs was estimated by dividing the number of events experienced by the estimated total time that the plants have spent in the drained condition. It was assumed that the duration of time that a plant stays in the drained condition is 420 hours for a refueling outage and 96 hours for a drained maintenance outage. Therefore, the total time that PWRs have spent in the drained condition is

504 years * (frequency of refueling * 420 hours + frequency of drained maintenance * 96 hours)
= 504 * (0.747 * 420 + 1.93 * 96)
= 2.52×10^5 hours,
and the frequency of failure to maintain vessel level is
16 Events/2.52×10^5 hour = 6.35×10^{-5}/hour.

d. LOCA. The experienced LOCA-type events were due to leaking valve packing glands, stuck open RHR relief valves, and improper valve alignment. The frequency of LOCA was estimated in the same way the frequency of spurious isolation signal was estimated, namely:

9 Events/1.79×10^6 hours = 5.03×10^{-6}/hour.

e. Spurious Containment Spray. Such events have the potential to cause rapid loss of RCS inventory and were modelled separately from the other LOCA-type events. The frequency of such events is

2 Events/1.79×10^6 hours = 1.12×10^{-6}/hour.

3.1.3 Estimates of Mean RHR Recovery Time and Mean Diagnosis Time for Loss of Cooling Events

Descriptions of the 177 loss-of-RHR events that are categorized in Table 3.1 were reviewed in an attempt to identify the diagnostic time. The descriptions of more than 50% of the events provided information that was used to determine the duration of time that the RHR system is not available. This duration was assumed to be the diagnosis time plus the time it takes to restore decay heat removal, and therefore would be a conservative estimate of the diagnosis time. It has also observed that loss-of-cooling events that are caused by overdraining, inadequate inventory and LOCAs tend to have longer recovery times. The reason is that it takes longer to restore the RHR system to an operable condition following an overdraining event as such efforts as venting air out of the RHR pumps may be required. It would be too conservative to use these RHR recovery times to estimate the diagnosis time for those particular events. It was, therefore, decided to use the experienced recovery times of those loss-of-cooling events that were caused by spurious closure of RHR suction valves and certain other failures to estimate the diagnosis time (as discussed in Section 2.6). The 49 events of spurious suction valve closure have a total of 675 minutes of RHR diagnosis and recovery time. Thirty-two events due to other failures have a total diagnosis plus recovery time of 523 minutes. Therefore, the average diagnosis and recovery time is approximately 15 minutes. This 15 minute time period has been conservatively applied as the input value of $T_{1/2}$ in Equation 2.1 for all of the diagnosis human error probability (HEP) calculations. Twenty-eight events caused by overdraining or inadequate inventory have a total of 1372 minutes of RHR recovery time. Therefore, the average RHR recovery time for such events is 49 minutes. This time is used in Section 3.2.2 to determine the time available for diagnosis for these types of events. The following is a summary of the estimated mean RHR recovery time and mean diagnosis time. They are used in the HCR model described in Section 2.6.1.

	Mean RHR Recovery Time	Mean Diagnosis Time
Overdraining Events	49 minutes	15 minutes
Other Loss of Cooling Events	15 minutes	15 minutes

3.1.4 Generic Component Failure Data

The component failure data used in NSAC-84 are Zion specific. In the generic analysis of this section, generic data was used. The generic data in the Oconee PRA[13] have been used in the quantitative analysis of Sections 3.2 to 3.5. Table 3.4 provides a comparison of the generic data and the Zion-specific data. Due to a lack of other data bases, it has been assumed that the maintenance unavailabilities used for Zion (NSAC-84) are representative of PWRs in general.

3.2 Loss of Cooling

NSAC-84 defined six phases during a refueling outage which are illustrated in Figure 2.1. Each phase was modelled in NSAC-84 using a procedural event tree. Sequences in the procedural event tree may lead to three types of initiating events: 1) LOCA, 2) loss of cooling, and 3) cold overpressurization. Instead of using the procedural event trees approach, BNL used a fault tree model for the operating train of the RHR system to model loss of cooling during each of the phases defined in Section 2.5. The BNL fault tree was derived from the RHR system fault tree in NSAC-84.

Section 3.2.1 discusses the quantification of the initiating event of loss of cooling. Section 3.2.2 discusses the quantification of the loss of cooling event trees. Given that the operating train has failed, a loss of cooling event tree was used to analyze the mitigation of the initiating event in the BNL analysis. This event tree has similar structure as the loss of cooling event tree in NSAC-84.

3.2.1 Frequency of Loss of Decay Heat Removal Capability

NSAC-84 classified plant outages into three types: 1) refueling, 2) drained maintenance, and 3) nondrained maintenance. The frequencies of these various outages were estimated in Section 3.3 of NSAC-84. During the eight year period, 1975-1982, there were 49 maintenance outages with durations greater than 50 hours and 12 refueling outages for the two Zion units. The overall frequency of an outage at Zion is then:

$$\frac{(49 + 12)}{2 * 8 \text{ years}} = 3.8 \text{ per year} .$$

Note that the eight year period is eight calendar years not reactor years. Therefore, the frequency of an outage is 3.8 per calendar year.

NSAC-84 also estimated the following:

f_1 = 2.95×10^{-1} (fraction of all outages where the primary system was kept water-solid).

f_2 = 0.0 (fraction of all outages where a steam or nitrogen bubble was maintained in the pressurizer)

f_3 = 7.05×10^{-1} (fraction of all outages where the primary system was drained)

f_4 = 7.21×10^{-1} (fraction of all "drained" outages due to maintenance or repair only)

f_5 = 2.79×10^{-1} (fraction of all "drained" outages due to refueling)

Therefore, frequency of refueling = $3.8 * f_3 * f_5$ = 0.747 per year,
frequency of drained maintenance = $3.8 * f_3 * f_4$ = 1.93 per year,
frequency of nondrained maintenance = $3.8 * f_1$ = 1.12 per year.

Section 2.5 defines the phases of the three types of outages. They are summarized in Table 2.3. The durations of the phases were used as the mission times for the operating train of the RHR system. Some of the basic events in the fault tree for the operating train of RHR system were characterized by failure rates. The probability of failure for a component with failure rate λ per hour is λ * MISSION TIME.

Figure 3.1 is a diagram of the RHR system taken from NSAC-84 (Figure A-2a). Table 3.5 lists the fault tree structure for the operating train of the RHR system in the form of input to the IBM PC version of the FTAP program.[21] It was derived from the RHR fault tree model in NSAC-84. Support systems such as electric power, component cooling water, and service water were included in the fault tree as basic events. Table 3.6 lists the basic events, the descriptions of the basic events, the failure rates, the sources of the failure rates, and the basic event probabilities. The basic event probabilities in Table 3.6 were calculated for phase 2 of the refueling outage with a mission time of 420 hours. For the other event trees, different mission times were used as discussed in Section 2.5.

The fault trees for each of the phases of shutdown were used to calculate the conditional probability that the operating train of the RHRS becomes unavailable in the given phase. The frequency that a loss of DHR event occurs in a given phase of shutdown is the frequency of the phase multiplied by the conditional probability of loss of DHR. The total frequency of loss of DHR during shutdown or refueling is the sum of the frequencies for all phases, i.e., 3.21×10^{-1} per year.

3.2.2 Loss-of-Cooling Event Tree

The BNL loss-of-cooling event trees are similar to those of NSAC-84 except that the safety injection system has been added. Figures 3.2 to 3.9 show the BNL loss-of-cooling event trees for the various phases of shutdown. These event trees have similar structure as those in NSAC-84, but the frequencies of the sequences are different. The generic component failure data in Table 3.4 were used in calculating the basic event probabilities used in the fault trees for the core damage sequences. The use of the data in Table 3.3 is discussed below:

a. Spurious Isolation of the RHR Suction Valves. The frequency of spurious isolation signals, 3.58×10^{-5}/hour, was used in the analysis. It was assumed that the spurious signal may occur any time during an outage, i.e., in all phases of shutdown, the spurious signal is a failure mode of the RHR system.

b. Overdraining. During an outage, the RCS may be drained in two situations, draining after cooldown for maintenance or refueling and draining after refueling. The overdraining event, with probability of 1.21×10^{-2} is modelled as a basic event in the fault trees of the RHR system in these phases. The needed operator response to this event is similar to that for a spurious isolation of the RHR suction valves. The same sequences (the lower branch of CV) in the loss of cooling event tree are used to model the mitigation of the initiating events.

c. Inadequate Inventory. This event occurs when the operator fails to maintain the vessel level with the RCS in the drained condition. The needed operator responses to such events are tripping the RHR pump, restoring vessel level, and restoring decay heat removal. They are similar to the needed responses to a spurious isolation of the RHR suction valves. In the loss-of-cooling event tree, such events are modelled in the same way the spurious isolation is modelled, i.e., the frequency of inadequate inventory events is added to the frequency of spurious isolation signals.

The quantification of the top events of the loss-of-cooling event tree is discussed in the following:

CV - This top event deals with the reliability of the operating train of the RHR system (RHRS). The branches under this top event differ from those for other top events in that both branches under this top event represent failure of the operating train of the RHRS. The success branch for the top event is omitted from the event tree. The probabilities associated with these branches in Figures 3.2 to 3.9 are the probabilities of the failure modes of the RHRS. Figures 3.2, 3.4 to 3.6, and 3.8 to 3.9 have two branches under this top event. The lower branch represents loss of RHR suction due to spurious isolation of the suction valves. The probability of this branch is the product of the frequency of spurious signal and the duration of the phase of the outage. The required operator responses to such a failure mode are 1) tripping the operating RHR pump, 2) reopen the suction valves, and 3) restart the RHR pump. The upper branch under this top event represents all other failure modes, e.g., operating pump fails to run. The probability of this branch is the sum of the probabilities of the cutsets representing these failure modes. Figures 3.3 and 3.7 apply to the phases in which the RCS is drained to the hot leg midplane. An additional branch of overdraining is included in the event trees.

HE - In response to the initiating event, the operator needs to 1) diagnose the situation including deciding on the appropriate actions to take and then 2) carry out the actions. If the operator fails to respond successfully to the initiating event, core damage is assumed to result. The human error probability (HEP) for the HE event depends on the initiating event and the time available for the operator to respond. The human error model described in Section 2.6 is used to model this event. The HEPs used in each phase of an outage are estimated as follows:

Refueling Phases

Refueling, Phase 1 - In this phase, the RCS is filled and the plant just shut down. NSAC-84 estimated that 3.8 hours will be available if loss of cooling occurs 6 hours after shutdown. As discussed in Section 2.6, the time available for diagnosis is the time to core uncovery minus the time needed to carryout the action. Using this 3.8 hours - 15 minutes in Equation 2.1 to calculate the HEP of failure to diagnose, 2.6×10^{-5} is obtained. The probability of HE is the sum of the probability of failure to diagnose, 2.6×10^{-5} and the probability of failure to implement the action, 10^{-5}, i.e., $2.6\times10^{-5} + 10^{-5} = 3.6\times10^{-5}$.

Refueling, Phase 2 - In this phase, the RCS is drained to the hotleg midplane. Therefore, events such as overdraining and failure to maintain inventory represent an additional failure mode of the RHR system. Figure 3.3 is the loss-of-cooling event tree for this phase. It differs from Figure 3.2 in that a third branch under the top event "CV" is added to analyze such failure modes. This failure mode is characterized by long RHR recovery time, i.e., a mean recovery time of 49 minutes was estimated in Section 3.1.3 using the generic experience. Two types of human errors were considered, i.e., failure to diagnose and failure to restore cooling. The time to core uncovery can be determined using the thermal model in Appendix A. This is the time available for the operator to diagnose the initiating event, decide on the needed actions and carry out the actions. Section 2.6.2 provides an estimate of 1×10^{-4} for the HEP of the failure to restore RHR cooling event, based on the HEP used in existing PRAs for similar types of events. Since the mean RHR recovery time is 49 minutes, the time available for diagnosis is the total available time minus 49 minutes. Equation 2.1 is then used to determine the HEP of failure to diagnose. The average HEP over the duration of this phase is 3×10^{-5}. The probability of HE for Sequence 18 is therefore $10^{-4} + 3\times10^{-5} = 1.3\times10^{-4}$. Similar calculation was done for Sequences 6 and 12. There, 15 minutes instead of 49 minutes was subtracted from the time available and a smaller probability of failure to carryout the action, 10^{-5}, was used.

Refueling, Phase 3 - In this phase, the refueling cavity is filled and significant time is available for operator diagnosis. A limiting HEP of 10^{-6} is used for failure to diagnose and 10^{-5} is used for failure to carryout the action. The probability of HE is $10^{-6} + 10^{-5} = 1.1\times10^{-5}$.

Refueling, Phase 4 - In this phase, the RCS is filled and the decay heat is low. The same HEP as that used for Phase 3 was used.

Drained Maintenance Phases

Drained Maintenance, Phase 1 - This phase is similar to Phase 1 of a refueling outage. The same HEPs are used. The event tree is shown in Figure 3.6.

Drained Maintenance, Phase 2 - Similar to Phase 2 of a refueling outage, a third branch under the top event "CV" is used to model the failure mode of overdraining the RCS. The average diagnostic HEP over the duration of the phase can be calculated using the thermal model of Appendix A and Equation 2.1. This results in an HEP of $7x10^{-4}$ for failure to diagnose. This is higher than that of phase 2 of a refueling outage, because mid-loop conditions are reached sooner and the decay heat is higher for a drained maintenance outage. The HEP of 10^{-4} is used for failure to restore RHR given a successful diagnosis. The HEPs for the two types of human errors are used in Figure 3.7 to quantify the HE event of Sequence 18. Similar calculation was done for Sequences 6 and 12. The probability of failure to diagnose was found to be $1.47x10^{-4}$.

Drained Maintenance, Phase 3 - In this phase, the RCS is filled and significant time is available for operator diagnosis. Similar to Phase 3 of a refueling outage, the HEP for HE is taken to be $1.1x10^{-5}$.

Nondrained Maintenance Phases

Nondrained Maintenance - This phase is similar to Phase 1 of a refueling outage. The same human error probabilities are used. The event tree is shown in Fig. 3.9.

RH - Normal RHR restored. This top event appears in sequences 5, 11, and 17 of the event trees. The quantification of these sequences requires linking the fault trees for four systems (RHRS, auxiliary feedwater system, safety injection system, and CVCS). In sequence 5, the RHR system fault tree includes both the operating train and the standby train. The mission time for the operating train is the mission time for the particular phase of the shutdown. The mission time for the standby train is assumed to be 24 hours. Included in the fault tree are maintenance of the RHR pumps, maintenance of the 4 kV buses, and support systems. In sequences 11 and 17, the same RHRS fault tree is used except that the mission time for the operating train is assumed to be 24 hours. Because, in these sequences, the initial loss of decay heat removal was caused by loss of suction and the operator is successful, the rest of the RHR system should, in principle, be available.

SG - Steam generator cooling. The steam generators are not available after the primary system is drained. Therefore, in phases other than phase 1 the steam generators are not considered available. For phase 1, the fault tree for the auxiliary feedwater system found in the Zion PSS was used. Included in this fault tree are maintenance of the auxiliary feedwater pumps, maintenance of buses and support systems.

BF - Boil and feed. The fault tree for this function was derived from NSAC-84. Included in the fault tree are maintenance of the charging pumps, maintenance of the buses and support systems. In particular, the simultaneous maintenance of both charging pumps is included in phase 3 of a refueling outage as per NSAC-84.

SI - Safety Injection System. As is discussed in Section 2.10, the safety injection system has been included in the loss-of-cooling event tree as an alternate method of providing makeup to the RCS. The fault tree for this system has been derived from the Zion PSS. Included in the fault tree are maintenance of the safety injection pumps, maintenance of the buses and support systems. The maintenance unavailability of the safety injection pumps has been assumed to be the same as that for the RHR pumps.

The results of the quantification of the loss-of-cooling event tree are shown in Figures 3.2 to 3.9 and are summarized in Table 3.7. It can be seen, from Table 3.7, that the dominant contributor to core damage frequency came from phases in which the RCS is partially drained. A sensitivity calculation was performed to determine the effect of the duration that a plant stays in the drained condition on the core damage frequency. If the duration is doubled, the core damage frequency due to loss of cooling increases approximately 50%. The probability shown under the top event SI in sequences 5, 11, and 17 of Figures 3.2 to 3.9 is actually the probability that all four system functions (RH, SG, BF, and SI from above) are not available.

3.3 Loss of Offsite Power

The frequency of loss of offsite power has been taken from NUREG-1032,[22] i.e., 0.088 per year or 10^{-5} per hour. The duration of the three types of shutdowns (refueling, drained maintenance, and nondrained maintenance) as well as the durations of the specific phases are listed in Table 2.3 and are used in the following calculation. The frequencies of the three types of outage have been estimated in Section 3.2.1 to be 0.747, 1.93, and 1.12 per year, respectively. The frequency that a loss of offsite power occurs in a particular type of outage is calculated as:

  frequency of the type of outage,
* frequency of loss of offsite power, and
* the duration of the outage.

As an example, for a refueling outage the frequenc of loss of offsite power would be:

$$0.747 \text{ per year} * 1\times10^{-5} \text{ per hour} * 1996 \text{ hours} = 1.49\times10^{-2} \text{ per year.}$$

Figures 3.10 to 3.12 are the BNL event trees for loss of offsite power for the three types of outages. The top events in these event trees have the same meaning as the same top events in the loss-of-cooling event trees. The last top event in these event trees does not appear in the loss-of-cooling event trees. It considers recovery of ac power. There are two core damage sequences in each loss of offsite power event tree. The frequency of sequence 6 is simply the frequency of the initiating event times the human error probability of failure to diagnose. This probability was assumed to be negligibly small because various alarms would be available and relatively long time is available for diagnosis.

The quantification of sequence 5 involves a time dependent analysis. First, the conditional probability that the safety systems, i.e., RHRS, charging system, and safety injection are not available (given a loss of offsite power) is evaluated by linking the fault trees for the systems. For refueling outages and drained maintenance outages, it is assumed that the steam generators are not available for decay heat removal. The unavailability of the systems is found to be 7.86×10^{-3} for these types of outages. It is dominated by the unavailability of the diesel generators. Table 3.8 lists the basic event probabilities used in the quantification of the onsite ac power systems. It can be seen that the maintenance unavailability of DGs during an outage is higher than that for power operations. It has been assumed that simultaneous maintenance on the DGs is not allowed. Also, essential 4 kV buses may be under maintenance during an outage.

The recovery of ac power is modelled using the time to core uncovery determined in Appendix B and the nonrecovery probabilities assessed in the ASEP[23] for Zion. Figure 3.13 shows the time to core uncovery curve determined in Appendix B and the probabilities of nonrecovery assessed in ASEP. The lower axis in Figure 3.13 is the time (T_{LDHR}) at which the decay heat removal capability is assumed to have been lost as a result of the loss of offsite power. The y axis is the time (T_{CU}) at which core uncovery occurs given that the decay heat removal capability is lost at T_{LDHR}. The time to core uncovery curve is determined assuming the RCS is drained to the mid-plane of hot leg and the coolant temperature is 100°F. The upper X axis in Figure 3.13 is the probability that ac power is not restored by either restoring offsite power or restoring a diesel generator. The four dots in Figure 3.13 are probabilities taken from ASEP for Zion, i.e.

Time (hr)	P(Nonrecovery of AC)
2	0.3
4	0.07
5	0.04
8	0.02

The circles plotted in Figure 3.13 are the results of NUREG-1032[22] for the probability that offsite power is not restored by the specified times. They are calculated using Figure A.1 of NUREG-1032. For example, the frequency that a loss of offsite power occurs and its duration exceeds two hours can be calculated using Figure A.1 of NUREG-1032. It is approximately 0.016 per year. The frequency of loss of offsite power of duration greater than 0.5 hour is estimated in NUREG-1032 to be 0.088 per year. The ratio of the two frequencies is the probability that a loss of offsite power event exceeds two hours, i.e., 0.016/0.088 = 0.18. It can be seen that the ac nonrecovery probability used in ASEP for Zion is not very different from the offsite power nonrecovery probability given in the generic data base. For calculational convenience, a staircase function is fitted through the dots as shown in Figure 3.13 and represents

Time Interval (hr)	P(Nonrecovery of AC Power)
2-3	0.3
3-4.5	0.07
4.5-6.5	0.04
6.5-11.5	0.02
Beyond 11.5	0.01

Using the time to core uncovery curve and the above staircase function for nonrecovery probability, the probability that ac power is not recovered can be expressed as a function of the time at which decay heat removal capability is lost. For example, using the time to core uncovery curve in Figure 3.13, the time to core uncovery is 3 hours if decay heat removal capability is lost at 105 hours after shutdown. It becomes 4.5 hours if decay heat removal capability is lost at 280 hours.

Using the staircase function for nonrecovery probability, the probability that ac power is not recovered is 0.07 if 3 to 4.5 hours is available. Therefore, the probability that ac power is not recovered is 0.07, if decay heat removal capability is lost in the time interval 105 to 280 hours after shutdown. Similarly, we get

Time at Which Decay Heat Removal Capability is Lost (hours)	P(Nonrecovery of AC Power)
72-105	0.3
105-280	0.07
280-580	0.04
580-1435	0.02
Beyond 1435	0.01

This is also listed in Table 3.9 and Figure 3.13.

The probability that a loss-of-offsite power occurs in the time intervals listed in Table 3.9 can be calculated as the product of the frequency of loss of offsite power, $1*10^{-5}$/hour, and the lengths of the interval, e.g., the first phase of a refueling outage is the time interval from 54 hours to 167 hours after shutdown

Prob (loss of offsite power in 54 to 167 hours)
= $1*10^{-5}$/hour * (167-54)
= $1.13*10^{-3}$.

For the initial phase of a refueling, the RCS is filled. Therefore, the SGs can be used to remove decay heat. NSAC-84 estimated that 3.8 hours will be available before core damage occurs, if a loss of cooling occurs 6 hours after shutdown with the RCS at 425 psig, 350°F and a bubble in the pressurizer. This time is a conservative estimate for the time to core uncovery due to loss of decay heat removal capability during a nondrained maintenance, because the RHR system can only be started when RCS temperature reaches 350°F. Assuming 3.8 hours is available, Table 3.9 is used to determine the probability that ac power is not recovered, i.e., 0.07. Therefore, the core damage frequency due to a loss of offsite power during the first phase of a refueling outage

Frequency (core damage due to loss of decay heat removal capability as a result of loss of offsite power in 54 to 167 hours after a refueling shutdown)
= frequency (refueling outages)
  * P(loss of offsite power in 54 to 167 hours)
  * Unavailability of safety systems due to loss of offsite power
  * P(ac power not recovered before core damage occurs, given loss of offsite power in 54 to 167 hours and safety systems unavailable)
= 0.747/year
  * 10^{-5}/hour*(167-54) hours
  * $2.54*10^{-4}$
  * 0.07
= $1.5*10^{-8}$/year.

Similar calculations can be done for other time intervals. The frequency that core damage occurs due to loss of offsite power in a refueling outage is

$$0.747/\text{year} * 1.0*10^{-5}/\text{hour} * \{(167-54)\text{ hours} * 0.07 * 2.54 * 10^{-4} + [(280-167)\text{ hours} * 0.07 + (587-280)\text{ hours} * 0.04 + (1996-587)\text{ hours} * 0.01] * 7.86*10^{-3}\}$$
$$= 2.03*10^{-6}/\text{year},$$

where $2.54x10^{-4}$ and $7.86*10^{-3}$ are the conditional probabilities that the safety systems are not available, given a loss of offsite power with the RCS in the filled or drained condition.

Similarly, for a drained maintenance outages:

$$1.932/\text{year} * 1.0*10^{-5}/\text{hour} * \{62\text{ hour} * 0.07 * 2.54 * 10^{-4} + [(105-83)\text{ hour} * 0.3 + (179-105)\text{ hour} * 0.07 + 803 * 0.01] * 7.86*10^{-3}\}$$
$$= 3.03*10^{-6}/\text{year}.$$

For nondrained maintenance outages:

$$1.121/\text{year} * 10^{-5}/\text{hour} * 2.54*10^{-4} * 125\text{ hour} * 0.07 = 2.49*10^{-8}/\text{year}.$$

Figures 3.10 to 3.12 are the loss of offsite power event trees for the generic analysis. Table 3.10 summarizes the results of the loss of offsite power analysis.

3.4 LOCA Event Trees

Table 3.3 provides the frequencies of two types of LOCAs. The first type can be characterized by a stuck open RHR relief valve, the leakage rate for this case is relatively low. Therefore, more time may be available for the operator to isolate the LOCA. However, if the LOCA occurs in the RHR system, isolating the LOCA may require complete isolation of the RHR system. The second type of LOCA, for example, could be due to inadvertent opening of the containment spray header valves. The leakage rate in this case is much higher and much less time would be available before the RHR pump would lose its NPSH. LOCA event trees are developed for the two types of LOCAs by modifying the loss-of-cooling event trees. The quantification of the LOCA event trees are also similar to that of the loss-of-cooling event trees. Figures 3.14 to 3.29 represent the LOCA event trees for the eight different phases (spanning the three types of outages) for each of the two types of LOCAs. The numbers at the end of the core damage sequences in the event trees are conditional probabilities of core damage given a LOCA. Tables 3.11 to 3.16 summarize the quantification of the LOCA event trees.

The first top event in the LOCA event trees asks questions about isolation of the LOCA. If a LOCA such as a stuck open RHR relief valve occurs when the RCS is not drained, more than an hour will be available before the RCS will be drained to a level that causes the RHR pump to cavitate. A probability of 0.03 is used (Figures 3.14, 3.18 and 3.21) for failure of the operators to isolate the LOCA before the RHR system is lost. It is obtained by using 1 hour in Equation (2.1). If such a LOCA occurs when the RCS is drained to the hot leg mid-plane then the RHR pump will rapidly lose suction. Two LOCA events in the data base (Trojan on April 25, 1978 and Davis Besse-1 on April

18, 1980) occurred when the RCS was partially drained. In both cases, the RHR pump lost its NPSH. Therefore, it is conservatively assumed that if a LOCA occurs when the RCS is drained, the probability that the operator fails to isolate the LOCA before the pump loses its NPSH is one (Figures 3.15 and 3.19).

If the containment spray header valves are inadvertently opened with the RCS filled, the RCS inventory will be decreasing at approximately 3000 gpm, and the vessel level will reach the mid-plane of the hot leg in less than 20 minutes. It is assumed that the probability that the operator fails to isolate the containment spray header valves before the RHR pump loses its NPSH is 0.1 (Figures 3.22, 3.25, 3.26, 3.28, and 3.29). A probability of one is used (Figures 3.23 and 3.27) if the LOCA occurs when the RCS is drained.

Quantification of top event HE in the LOCA event trees is done by the same approach used to quantify the HE event in the loss of cooling event trees. In general, each phase of an outage is of a different condition, and the time available for diagnosis and response is different. As an example, the quantification of the LOCA (stuck open relief valve) event trees for the four phases of a refueling outage is discussed here.

Phase 1 of a Refueling Outage (Figure 3.14) - In Sequence 6, the HE event is conditional upon successful isolation of the LOCA. Therefore, the operator should have known that the DHR capability is affected by the LOCA. The same human error probability as that used in the loss of cooling event tree is used for this event, i.e., 2.6×10^{-5} for failure to diagnose and 10^{-5} for failure to implement needed action. It is assumed that the LOCA occurs in the RHR system and isolation of the LOCA disables the system. In Sequences 5 and 11 the RHR system is unavailable. In Sequence 12, the HE event is conditional on failure to isolate the LOCA, such that the RHR system loses its suction. This scenario is similar to a loss-of-cooling event caused by overdraining the RCS. The core uncovery time can be determined using the model in Appendix B. It is assumed that the LOCA occurs in the middle of this phase, i.e., 110.5 hours after shutdown. The core uncovery time is 3.1 hours. The time available for diagnosis is 3.1 hours minus 49 minutes which is the estimated average time needed to restore RHR. Equation (2.1) gives a probability of 7×10^{-4} for this time. The probability of failure to implement the recovery actions is taken to be the same as that used in the loss of cooling event tree, i.e., 10^{-4}. The probability of HE in Sequence 12 is the sum of the probabilities of the two types of human errors, i.e., $7\times10^{-4} + 10^{-4} = 8\times10^{-4}$. Similar analyses have been done for Phase 1 of a drained maintenance outage or a nondrained outage. These are shown in Figures 3.18 and 3.21. The human error probabilities are different because the starting points and ending points of the phases are different (i.e., different decay heat rates). RHR initiation occurs sooner in a maintenance outage than a refueling outage.

Phase 2 of a Refueling Outage (Figure 3.15) - In this phase, a LOCA occurs when the RCS is drained. It is assumed that the probability of failure to isolate the LOCA is one. In Sequence 11, steam generator cooling is not available either. In Sequence 12, the same human error probability as that assumed in the loss-of-cooling analysis is used. Similar analysis is done for Phase 2 of a drained maintenance outage and is shown in Figure 3.19.

Phases 3 and 4 of a Refueling Outage (Figures 3.16 and 3.17) - In these phases, the decay heat is low and plenty of time is available for operator action. A limiting human error probability of 10^{-6} has been used for probability of failure to diagnosis. A human error probability of 10^{-5} is used for failure to take the correct recovery action. The difference between the two phases is that in Phase 3 both charging pumps may be under maintenance.

3.5 Base Case Results

Table 3.17 summarizes the base case results of the BNL shutdown model. The analysis was done for a plant with the same configuration as the Zion plant. It is called a generic analysis, because the initiating event frequencies were estimated using generic experience and generic component failure data were used. Only three initiating events were considered in the analysis, i.e., loss of cooling, loss of offsite power, and LOCA. Other initiating events such as low temperature overpressurization and loss of component cooling water are not considered. The focus of this study is the functional loss of shutdown cooling capability. The loss of offsite power and small LOCA initiators are developed as specific cases within this overall context.

3.6 Proposed Improvements

In this section, several design/procedural improvements for the Zion design configuration are considered based upon the BNL shutdown risk model.

3.6.1 Upgraded Instrumentation for RHRS and Upgraded Emergency Procedures

This improvement requires the availability of an alarming trend recorder for monitoring RHR pump conditions (flow, discharge pressure, motor current) to provide the operator with early warning of a potential loss of RHR suction, and the availability of emergency procedures for restoration of RHR capability. It is assumed in the base case calculation that the instrumentation for the RHR system includes indications for RHR loop flow, RHR pump discharge pressure, and annunciation for RHR pump cooling water flow and high RHR pump discharge pressure. To derive an estimate of the risk-reduction benefits provided by availability of adequate instrumentation and alarms, abnormal operating procedures, and administrative controls, BNL has assumed that such features are available as part of this improvement.

These proposed design and procedural changes will improve the ability of the operator to respond to a loss of RHR and may improve the reliability of the RHR system itself by allowing the operator to respond prior to a loss of RHR. This latter benefit is considered to be small compared to the former and has, therefore, not been included in the model. The reduction in core damage frequency as the result of the improvement is estimated by assuming perfect operator response to loss of RHR, i.e., the operator will trip the RHR pump if its suction is lost, and will restore DHR if it is required. This corresponds to changing the human error probabilities in the event trees to zero.

3.6.2 Upgraded Vessel Level Indication

This improvement requires the availability of highly reliable, redundant instrumentation, with control room readout and alarm, for monitoring the water level and temperature within the reactor vessel during drained RCS operations. However, no low level alarm is assumed available.

With the proposed upgraded vessel level instrumentation, the operator is not likely to over-drain the reactor vessel during a drain-down operation. The decrease in the frequency of loss of DHR is estimated by setting the human error probability of overdraining to zero.

3.6.3 Removal of Auto Closure Interlock (ACI)

The removal of the ACI will reduce the frequency of spurious isolation by the RHR suction valves. It is assumed that this frequency simply reduces to that for spurious closure of MOVs. In NSAC-84, the generic population mean for the frequency of spurious closure of MOVs without an ACI was estimated to be 2.33×10^{-8} per hour. For two RHR suction valves in series, the frequency of spurious isolation becomes 4.66×10^{-8} per hour. Another potential benefit of the removal of ACI is that the RHR relief valve may be available to relieve the pressure in case of a low temperature overpressurization (LTOP). LTOP is considered in generic issue 94 and is not within the scope of this study.

The removal of the auto closure interlock has the potential to increase the interfacing LOCA frequency at a given plant. The concern is that only one RHR suction valve might be closed during plant startup and the other valve may be inadvertently left open. The single failure of the closed suction valve could then lead directly to an overpressurization event. Two plants (Kewaunee and Callaway) proposed[24-25] to replace the auto closure interlock with some features that will reduce the chance that one valve may be left open during plant heatup. Both plants have two drop lines.

Kewaunee's features are:

1. Control room annunciator entitled, "RHR Abnormal Lineup" which alarms whenever the RCS pressure is above 700 psig and any one isolation valve is not fully closed.

2. Interlock that ties together the closing circuits of the two valves in a single drop line so that they both close (but do not open) together on the actuation of either control switch to the "close" position.

Callaway's features are:

1. Removal of ACI to one valve in each drop line.

2. Alarms if either of the above valves is open above the interlock setpoint.

The two proposals were reviewed and accepted by NRC.[24-25] As noted in Reference 22, Westinghouse analyzed the proposal to remove the ACI on the Kewaunee plant's DHR suction valves, and concluded that such a modification would be a safety improvement.

There are two things that must be stressed here concerning this proposed improvement. First, the above two plants that have received NRC approval for changing the ACI did not remove the protection. In both instances, alternative means were also applied and it is on this basis that modification to ACI has been included herein as a proposed improvement. The second thing is that the core damage frequency of an interfacing system LOCA is orders of magnitude lower than that calculated herein for loss-of-cooling incidents at shutdown. Therefore, even if the alternative means provided when removing the ACI were to yield a slightly higher core damage frequency for interfacing system LOCAs, the overall predicted core damage frequency should still be lowered. If this is not the case for a given plant, this proposed improvement should not be implemented.

3.7 CDF-Related Results

In Table 3.18 the core damage frequency for all of the shutdown initiators is provided. Of the three proposed improvements, only the upgraded RHR instrumentation had any effect on the LOCA core damage frequency.

It can be seen in Table 3.19 that removal of ACI is the most effective way to reduce the frequency of loss of cooling but the least effective in reducing the core damage frequency. Upgraded instrumentation for the RHR system is the most effective way to reduce the core damage frequency. The reductions in the frequency of loss of cooling events and core damage as a result of the upgraded level instrumentation and the removal of ACI are additive, because these improvements simply reduce the frequency of the loss-of-cooling initiating event. The upgraded instrumentation for RHRS and upgraded emergency procedures will affect the human error probabilities used in the analysis. The reduction in core damage frequency due to this improvement can not be simply added to those for the other improvements.

It can be seen from Table 3.20 that each of the three proposed changes leads to significant reduction in core damage frequency due specifically to loss-of-cooling events. Removal of ACI is very effective in reducing the frequency of loss of cooling, but its reduction in core damage frequency is smaller than that for upgraded vessel level instrumentation, because the spurious isolation of an RHR suction valve may occur any time during a shutdown while overdraining is postulated only when the RCS is already partially drained. This is an important distinction because interruption of the RHR system during most of the time over a given shutdown yields ample time for operator recovery; whereas, loss of level instrumentation during a partially drained condition represents a much more vulnerable scenario (see Table 3.7).

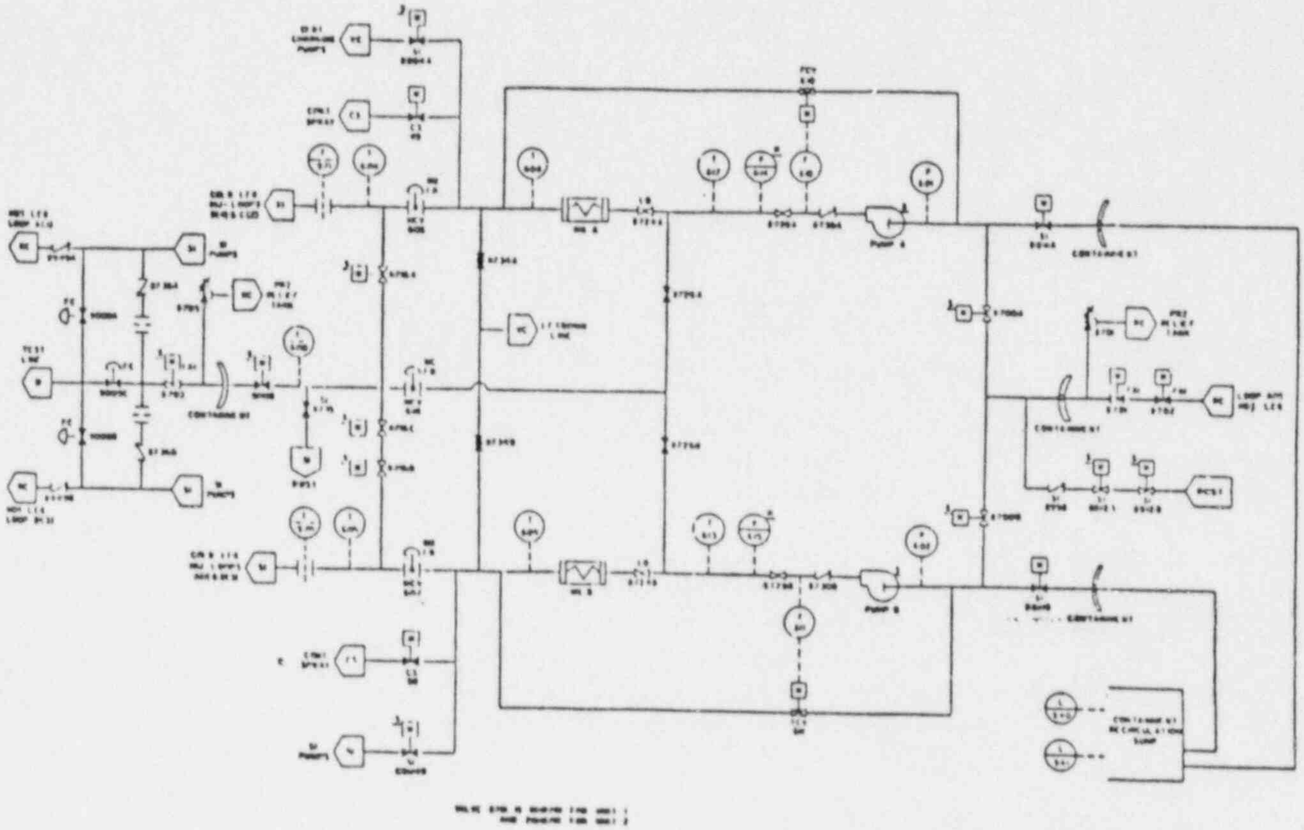
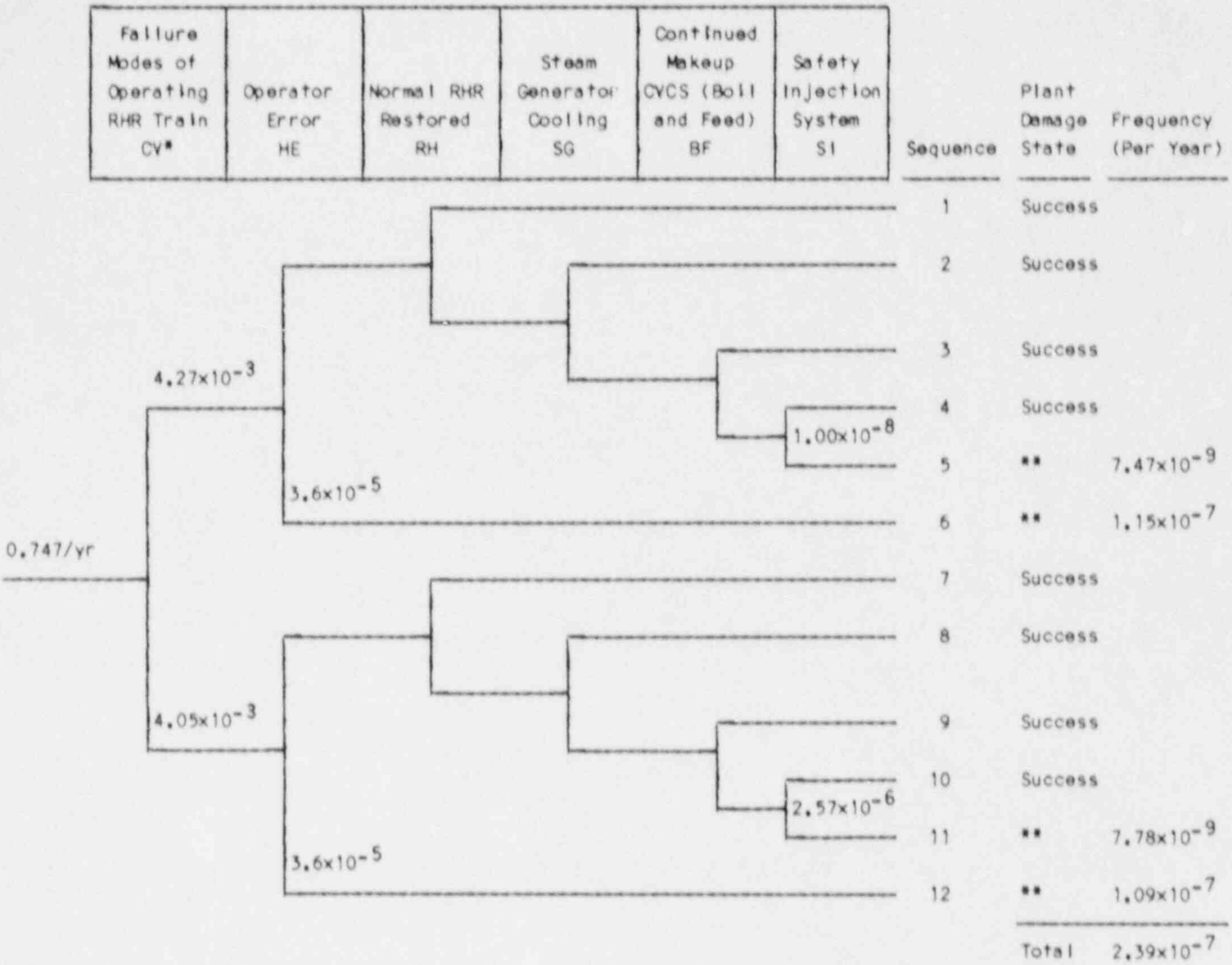


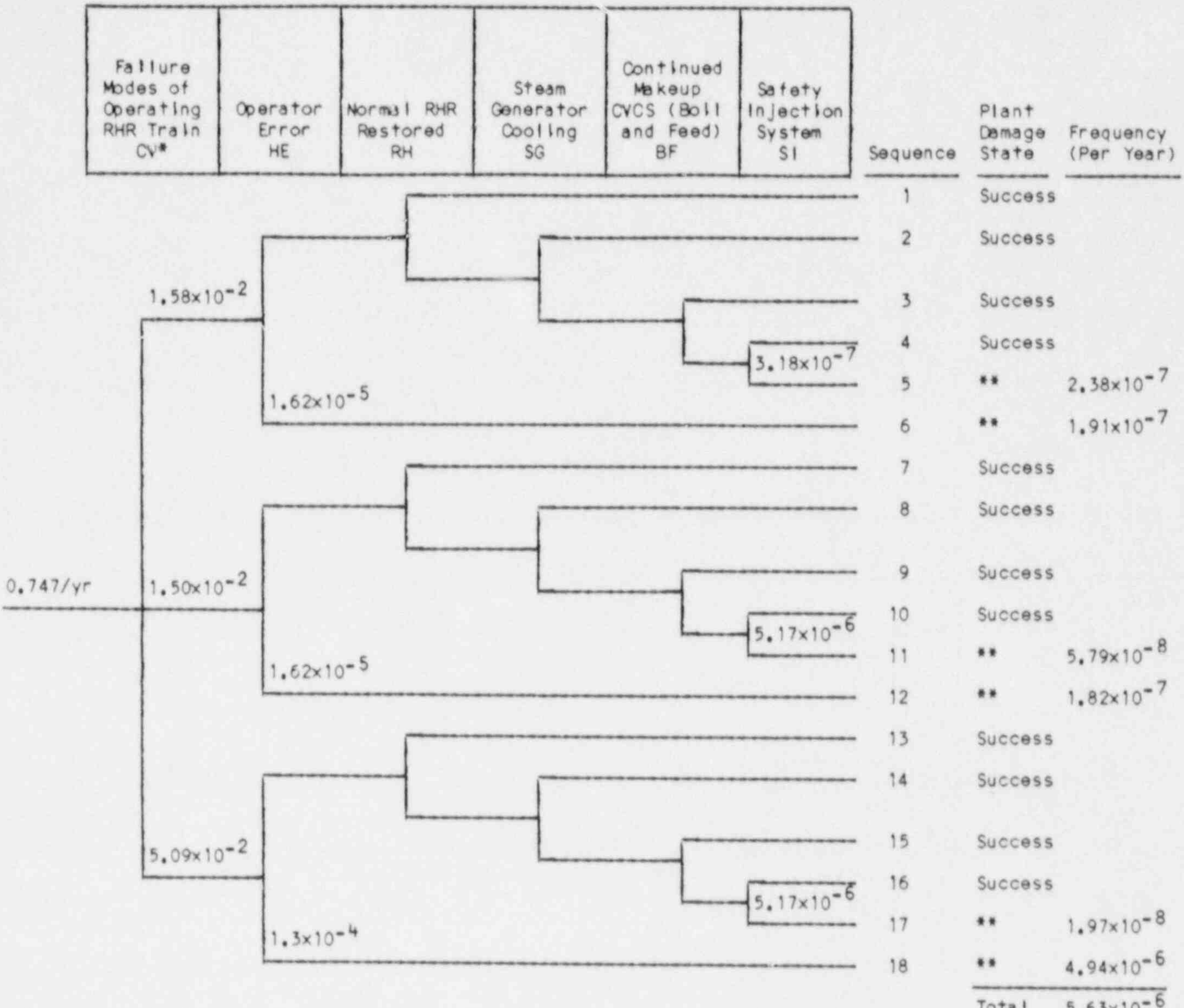
Figure 3.1. Residual heat removal system (Figure A-2a of NSAC-84).



*Lower branch of CV represents failure of RHRS due to spurious closure of suction valves. Upper branch represents other failure modes of operating RHR train.

**Core Damage

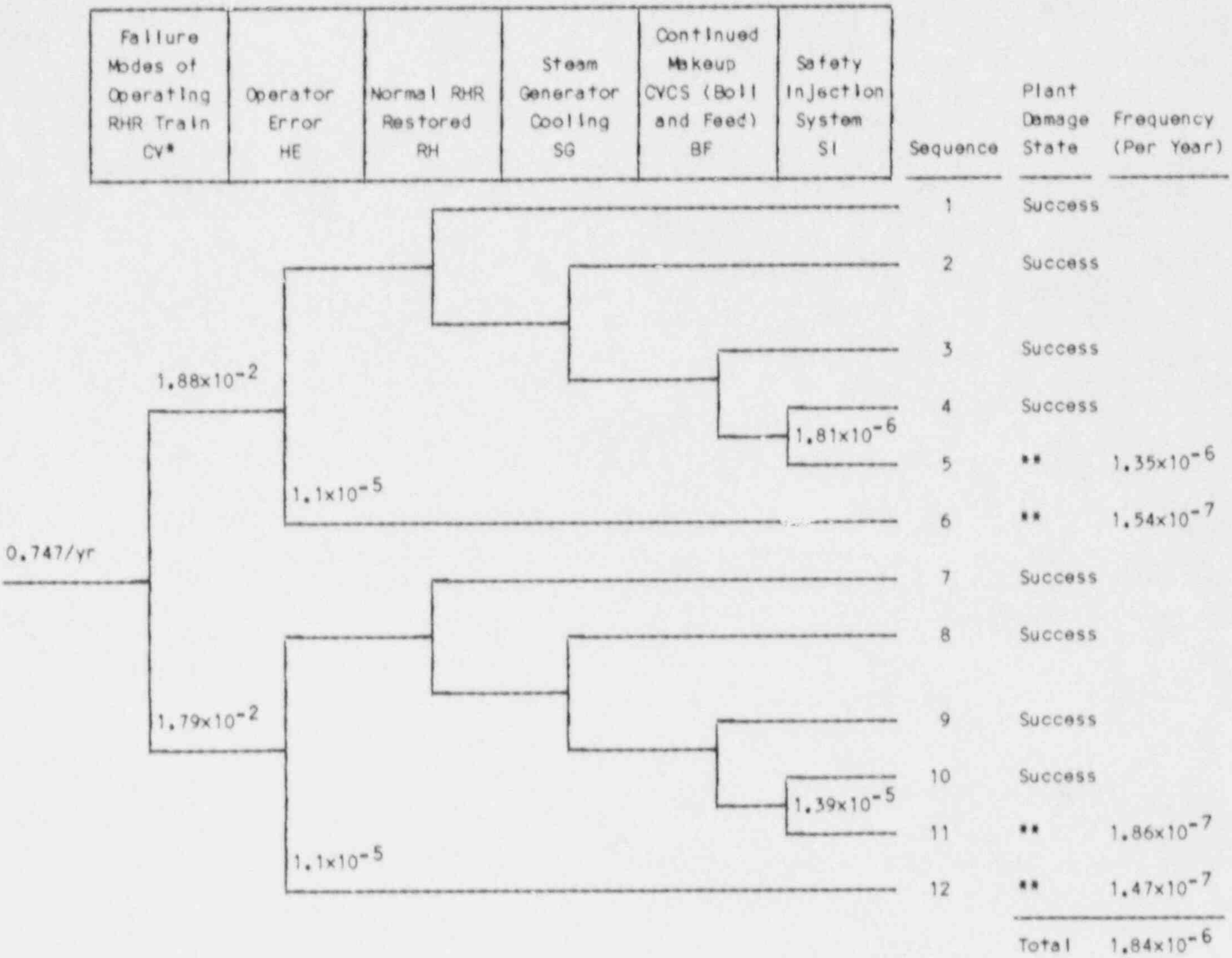
Figure 3.2. Loss of cooling event tree for Phase 1 of refueling outage.



*Lower branch of CV represents loss of suction due to overdraining. Middle branch represents loss of RHRS due to spurious closure of suction valves. Upper branch represents other failure modes of operating RHR train.

**Core Damage

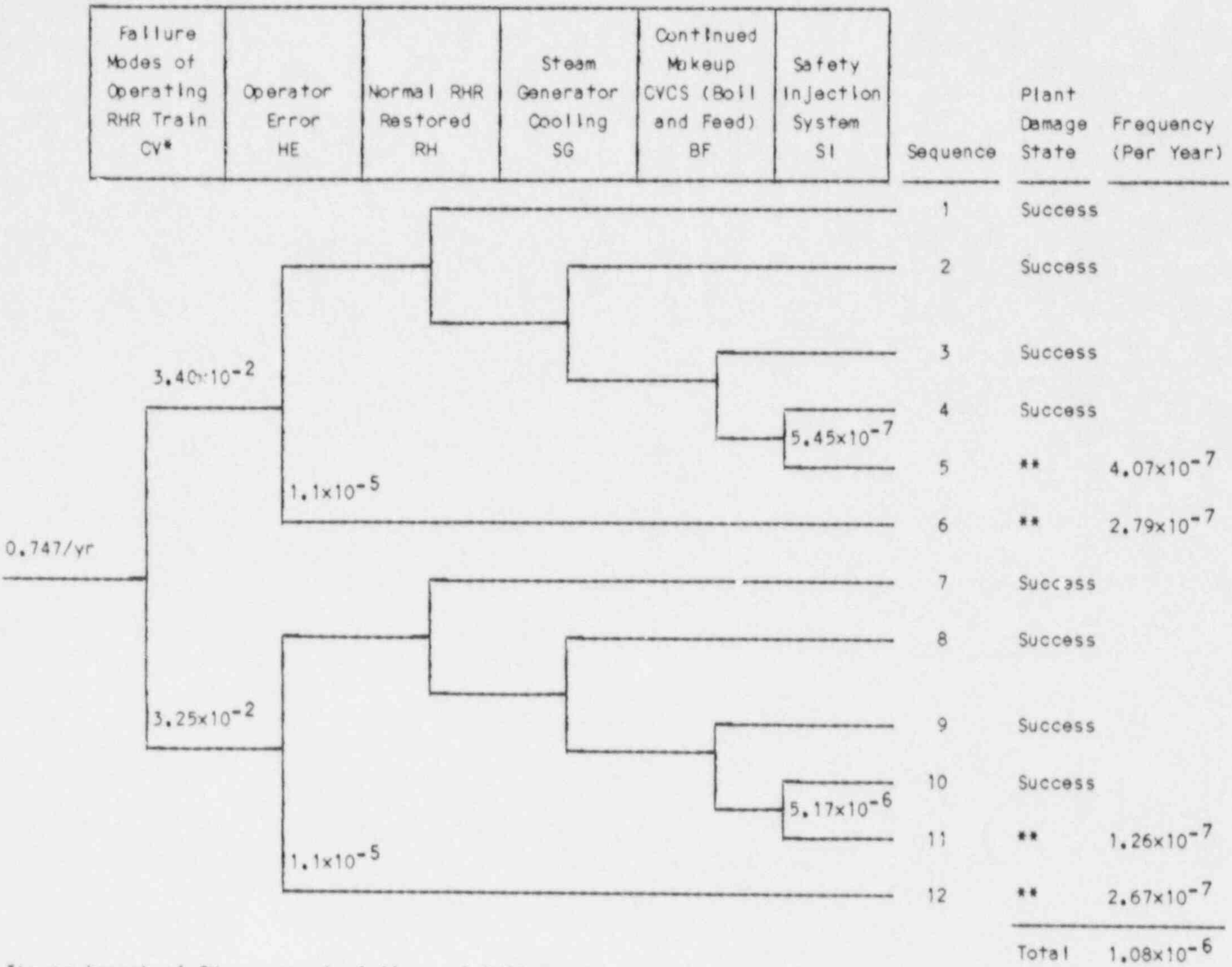
Figure 3.3. Loss of cooling event tree for Phase 2 of refueling outage.



*Lower branch of CV represents failure of RHRS due to spurious closure of suction valves or loss of suction due to overdraining. Upper branch represents other failure modes of operating RHR train.

**Core Damage

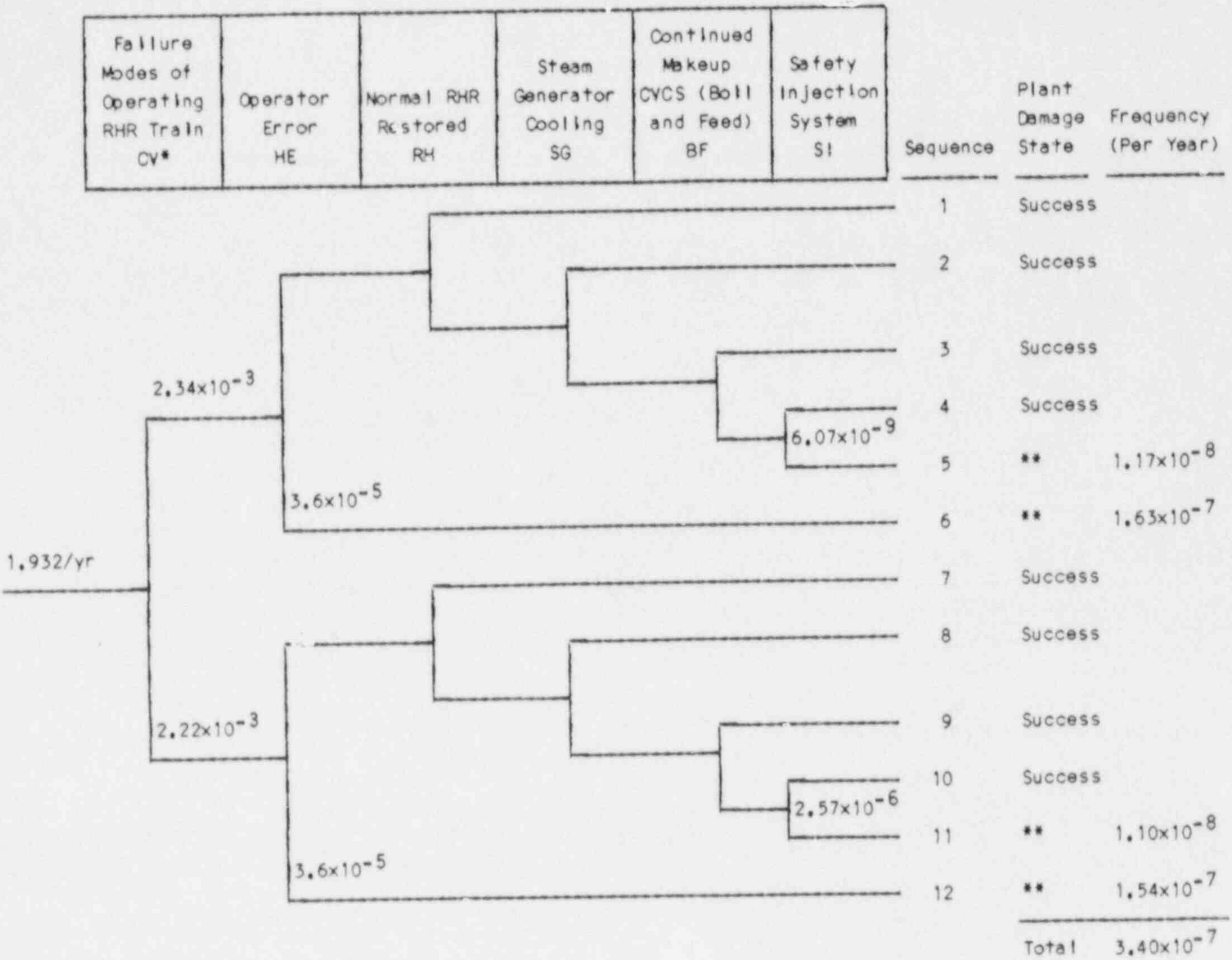
Figure 3.4. Loss of cooling event tree for Phase 3 of refueling outage.



*Lower branch of CV represents failure of RHRS due to spurious closure of suction valves or loss of suction due to overdraining. Upper branch represents other failure modes of operating RHR train.

**Core Damage

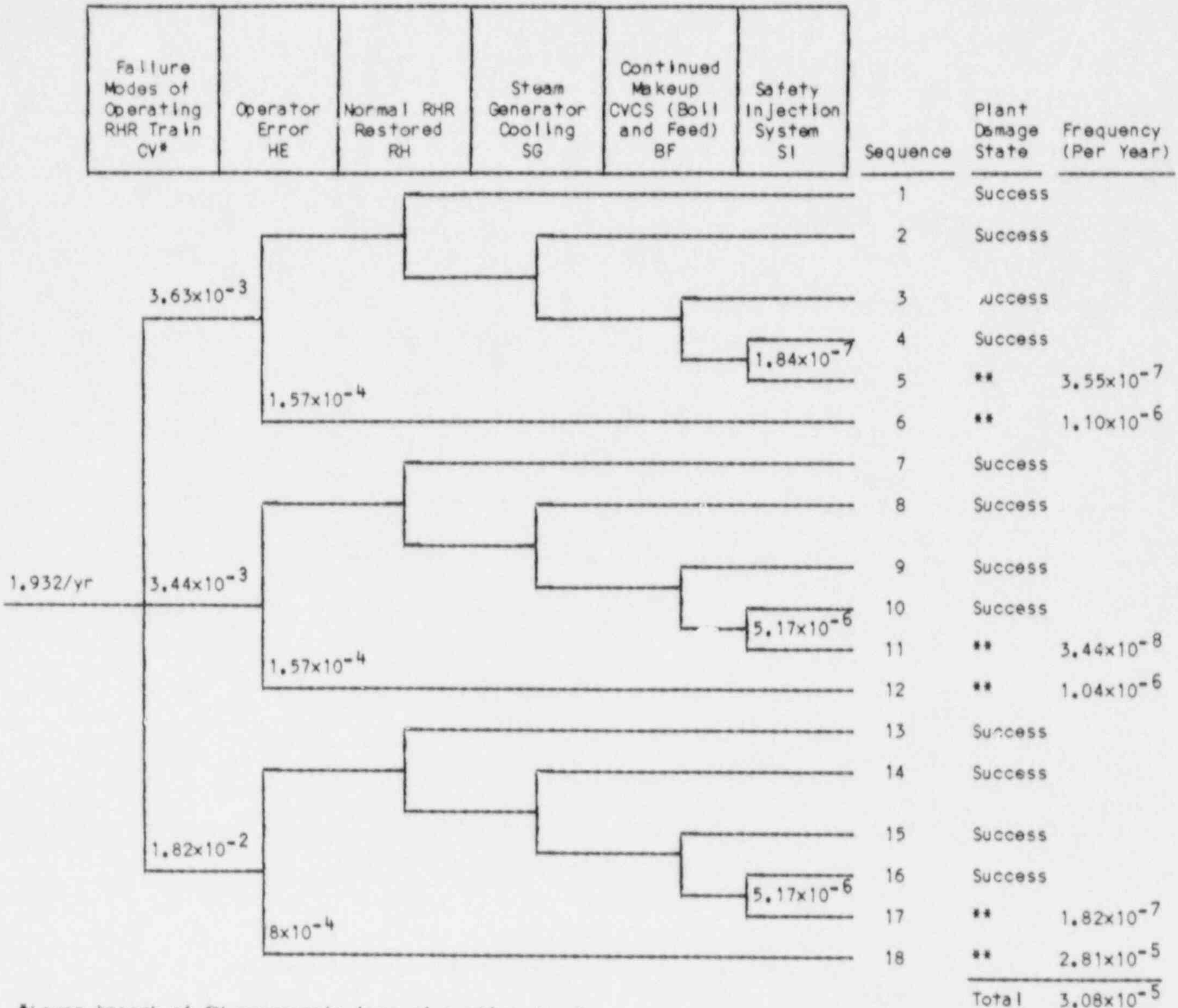
Figure 3.5. Loss of cooling event tree for Phase 4 of refueling outage.



*Lower branch of CV represents failure of RHRS due to spurious closure of suction valves or loss of suction due to overdraining. Upper branch represents other failure modes of operating RHR train.

**Core Damage

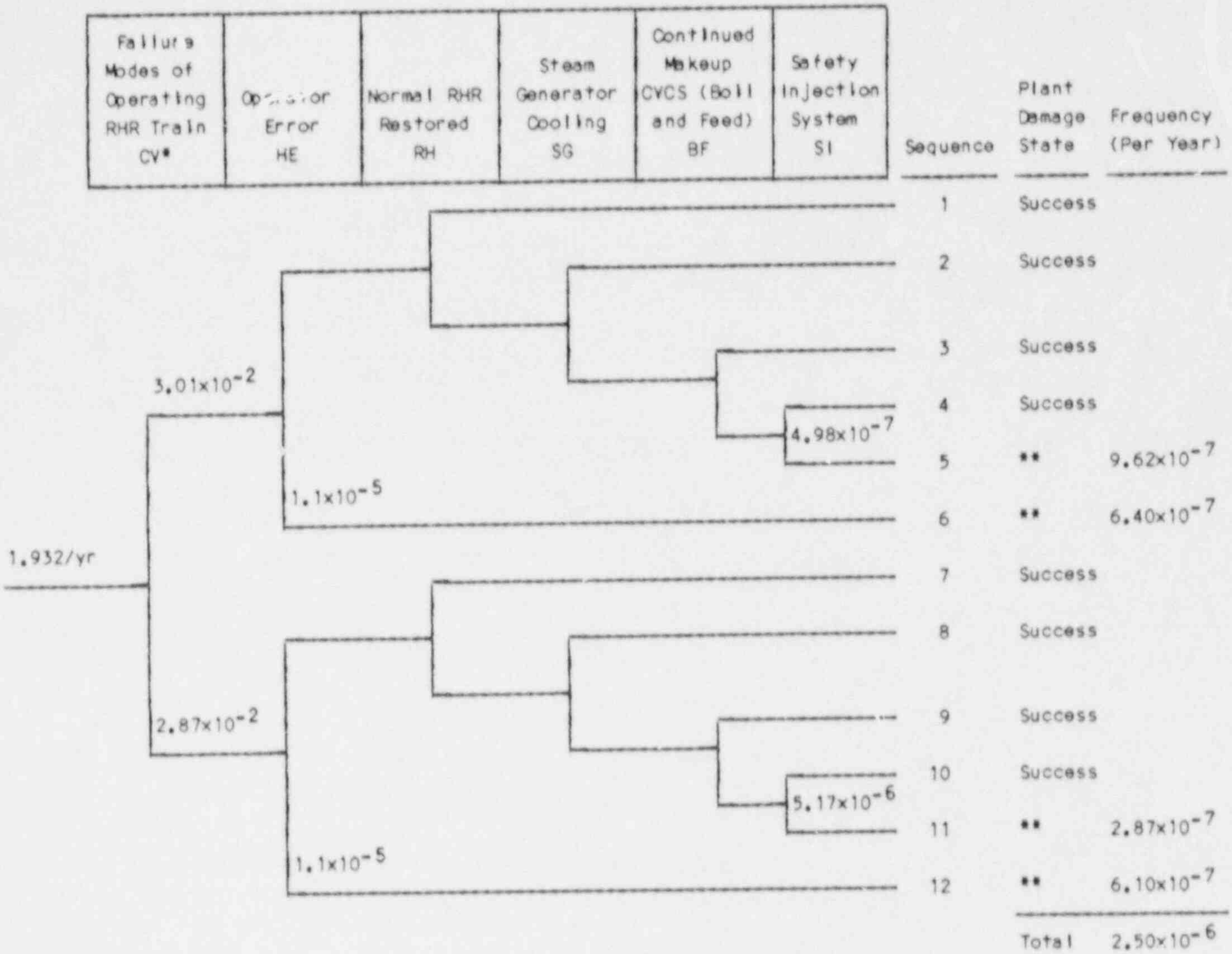
Figure 3.6. Loss of cooling event tree for Phase 1 of drained maintenance outage.



*Lower branch of CV represents loss of suction due to overdraining. Middle branch represents loss of RHRS due to spurious closure of suction valves. Upper branch represents other failure modes of operating RHR train.

**Core Damage

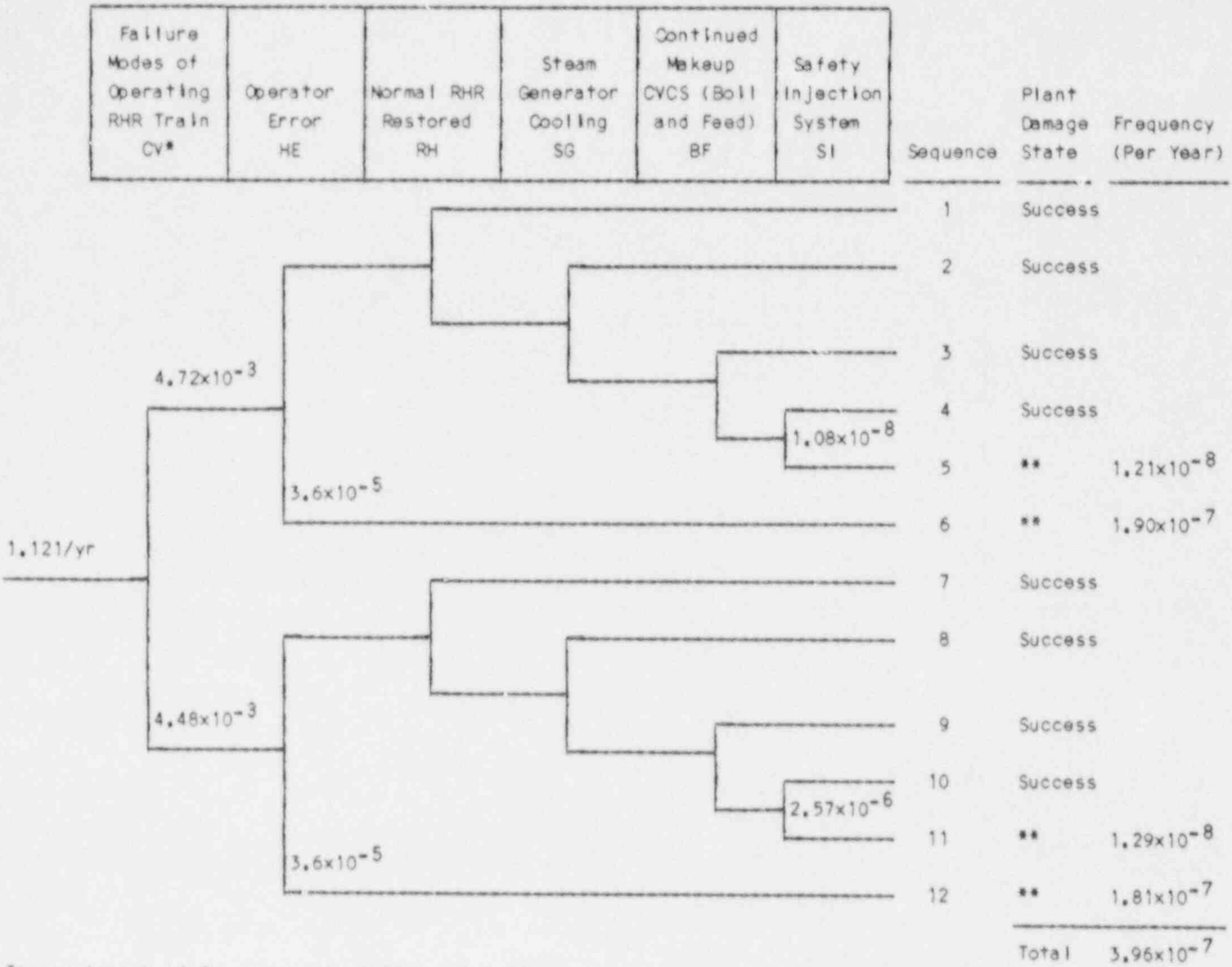
Figure 3.7. Loss of cooling event tree for Phase 2 of drained maintenance.



*Lower branch of CV represents failure of RHRS due to spurious closure of suction valves or loss of suction due to overdraining. Upper branch represents other failure modes of operating RHR train.

**Core Damage

Figure 3.8. Loss of cooling event tree for Phase 3 of drained maintenance outage.



*Lower branch of CV represents failure of RHRS due to spurious closure of suction valves or loss of suction due to overdraining. Upper branch represents other failure modes of operating RHR train.

**Core Damage

Figure 3.9. Loss of cooling event tree for nondrained maintenance.

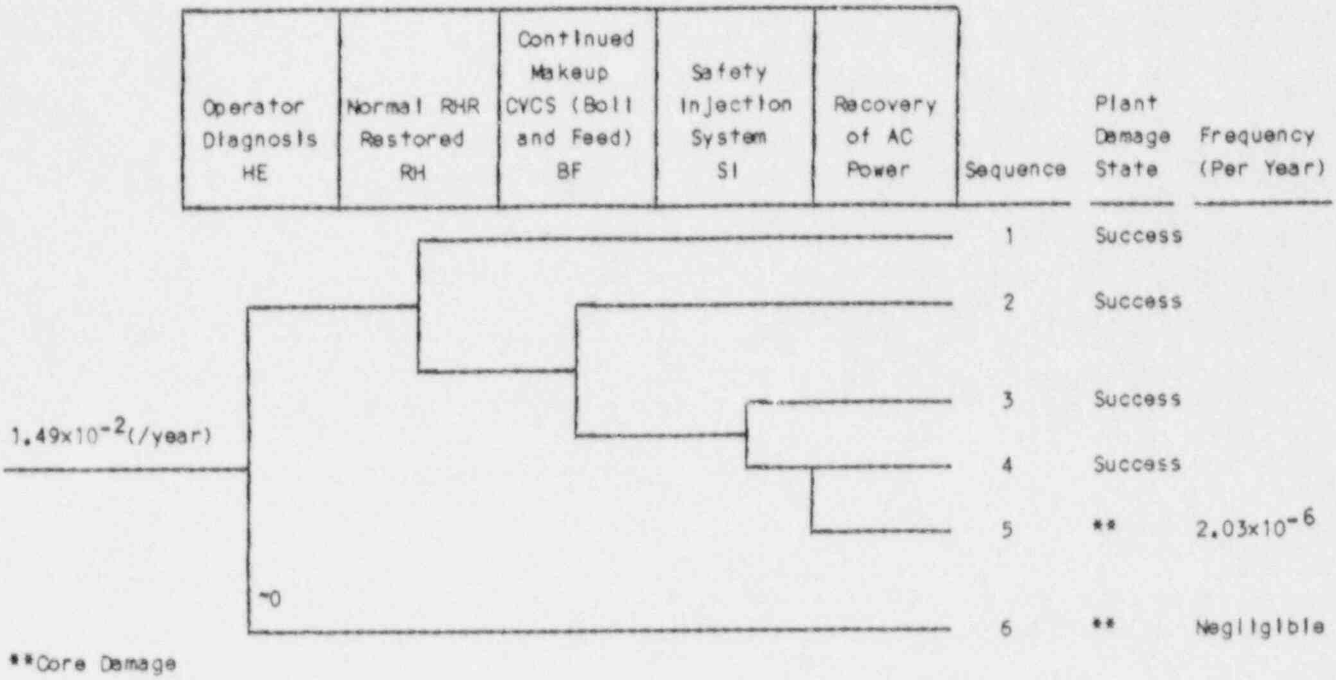


Figure 3.10. Event tree for loss of offsite power during a refueling outage - generic.

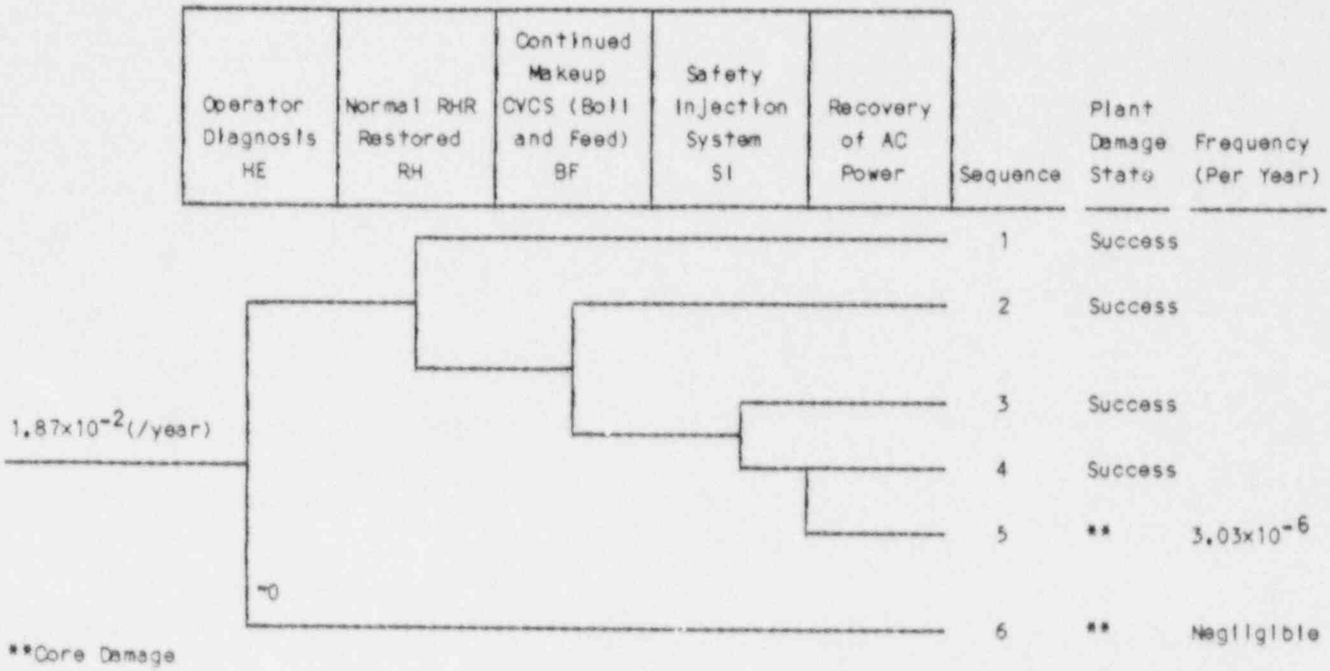


Figure 3.11. Event tree for loss of offsite power during a drained maintenance outage - generic.

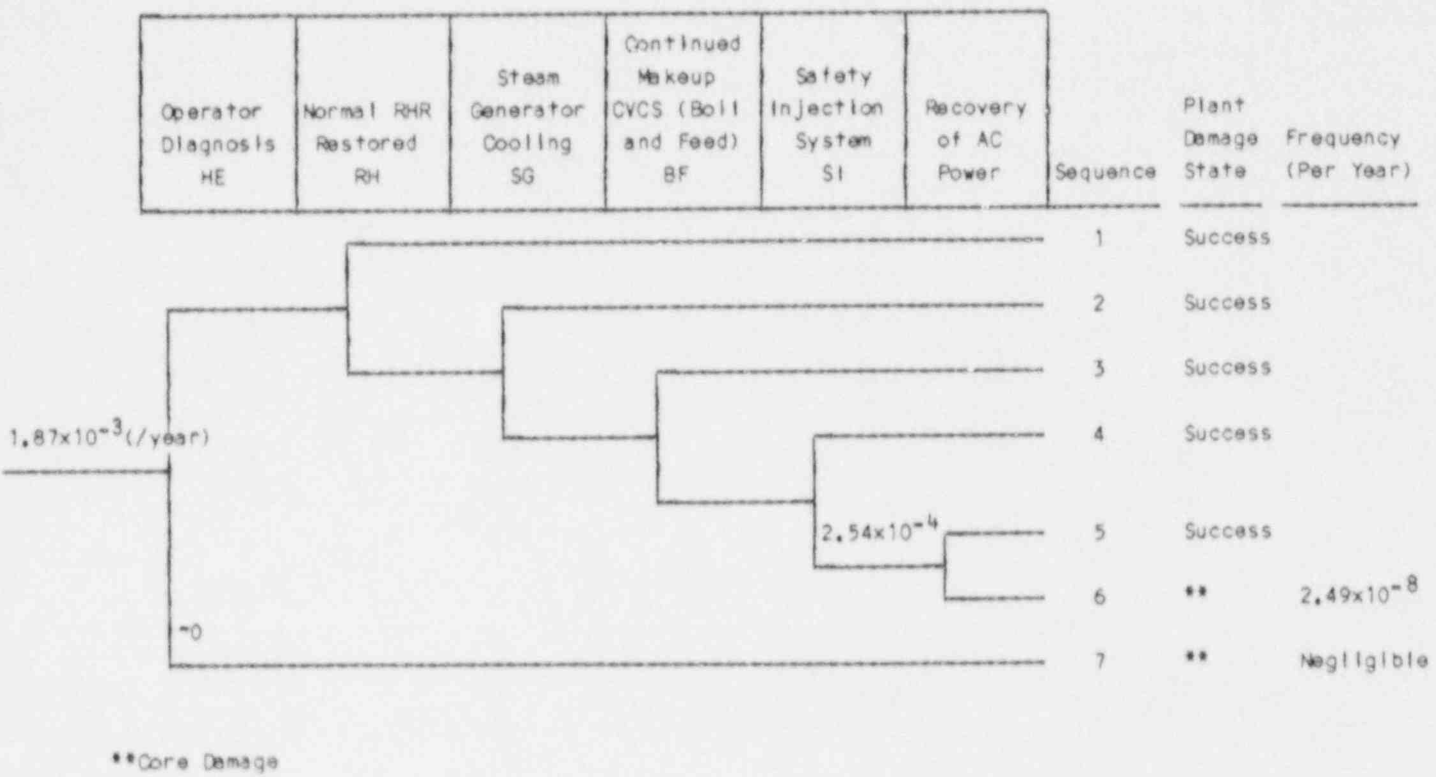


Figure 3.12. Event tree for loss of offsite power during a nondrained maintenance outage - generic.

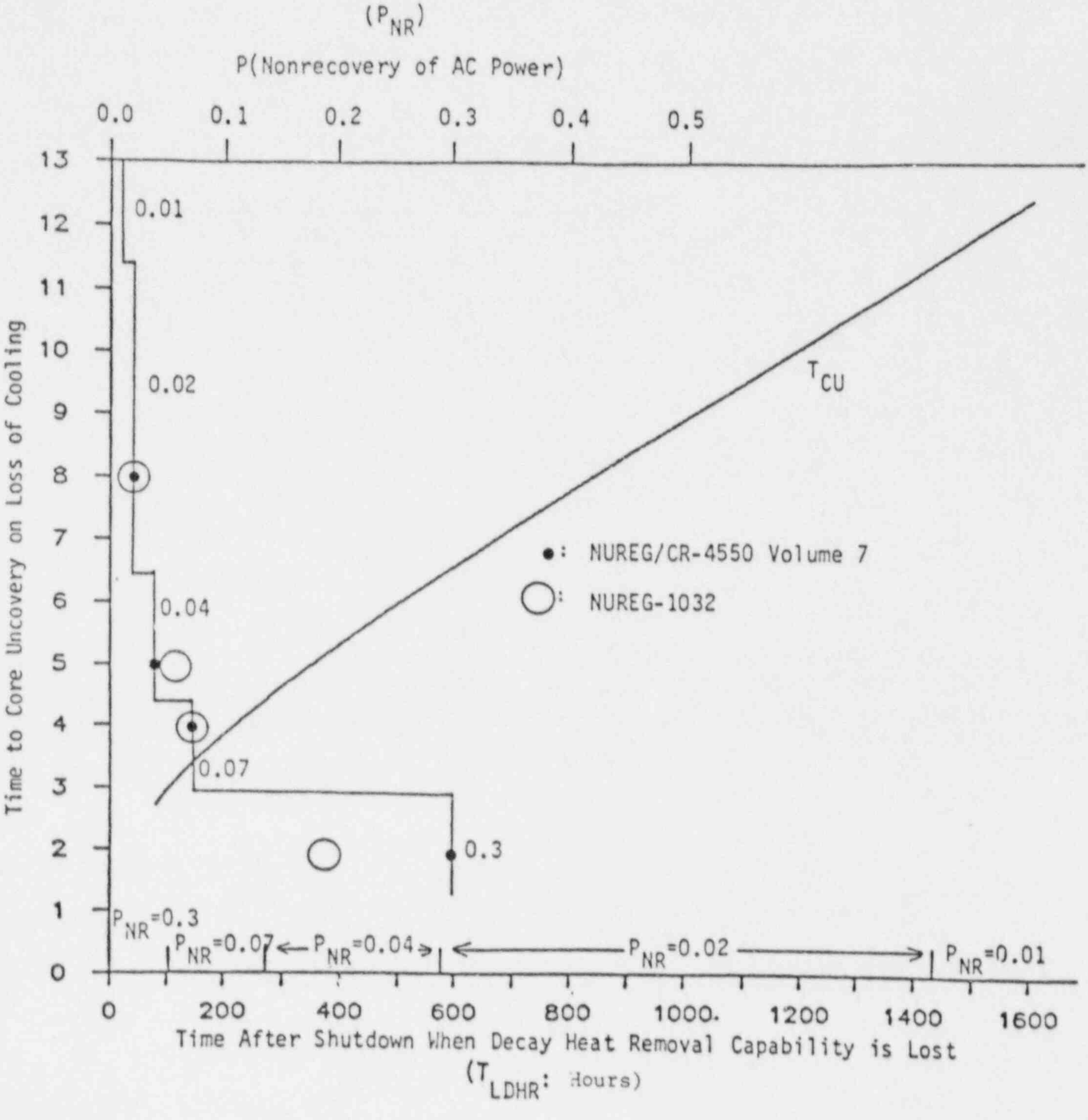


Figure 3.13. Model of ac power recovery.

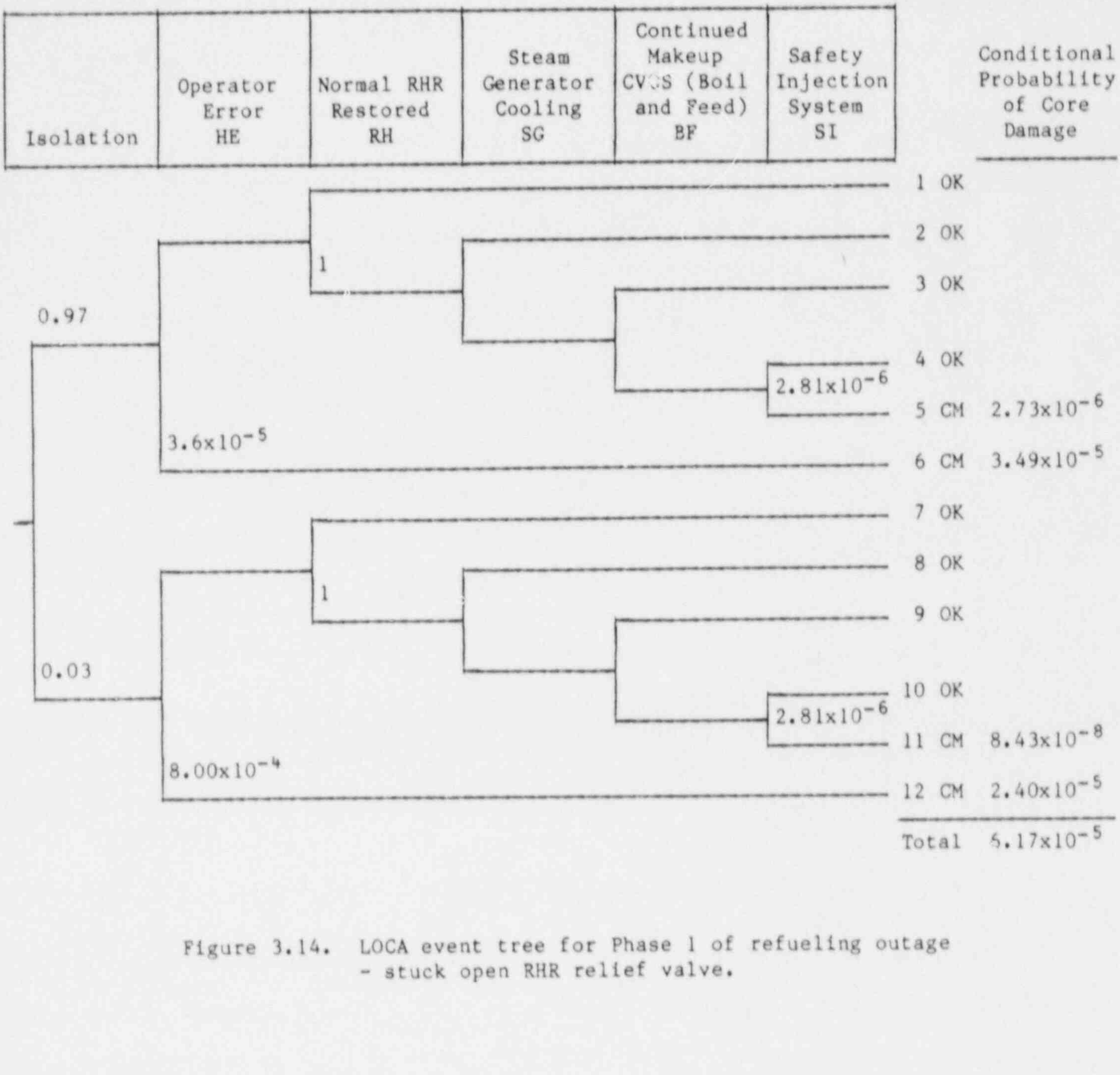


Figure 3.14. LOCA event tree for Phase 1 of refueling outage - stuck open RHR relief valve.

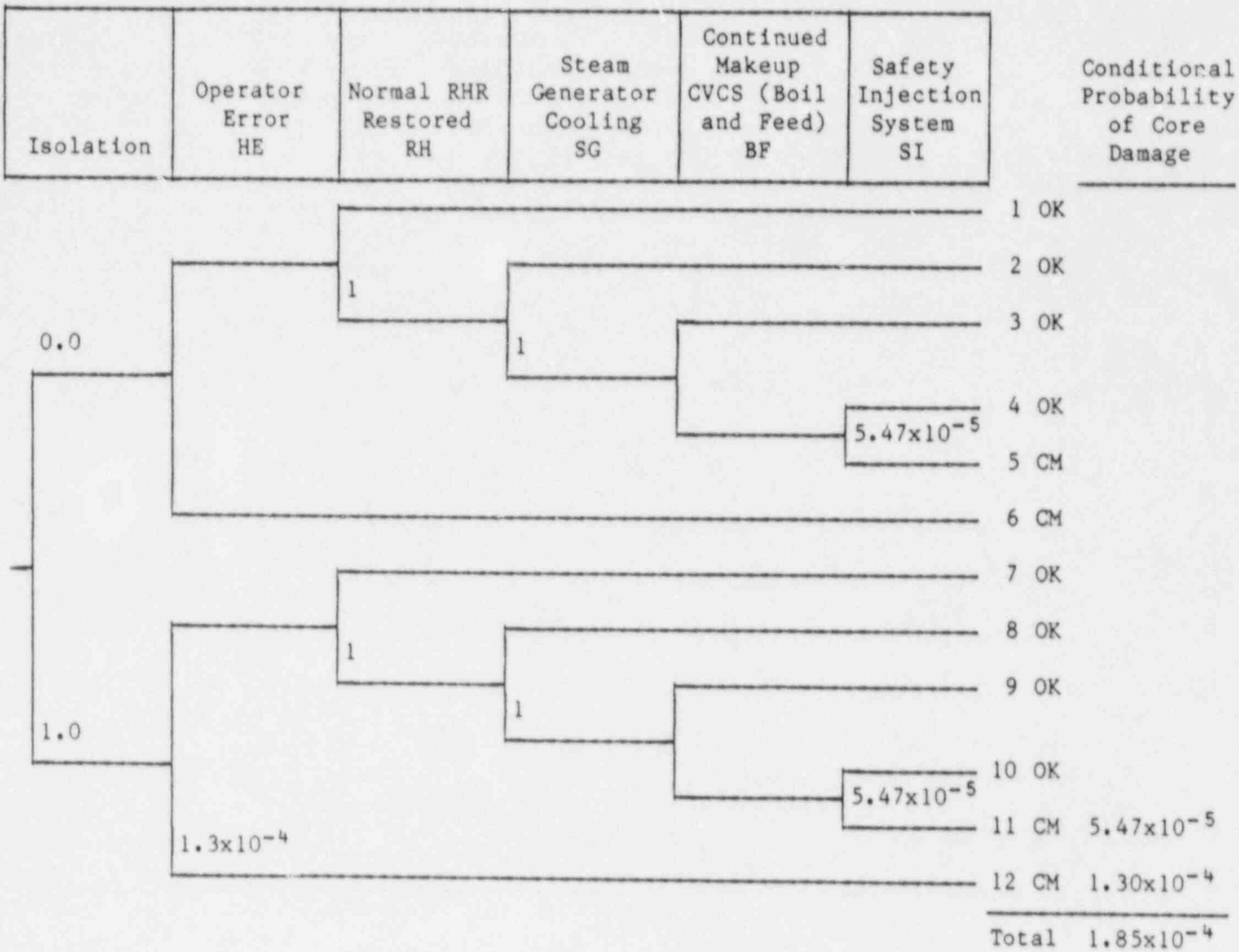


Figure 3.15. LOCA event tree for Phase 2 of refueling outage - stuck open RHR relief valve.

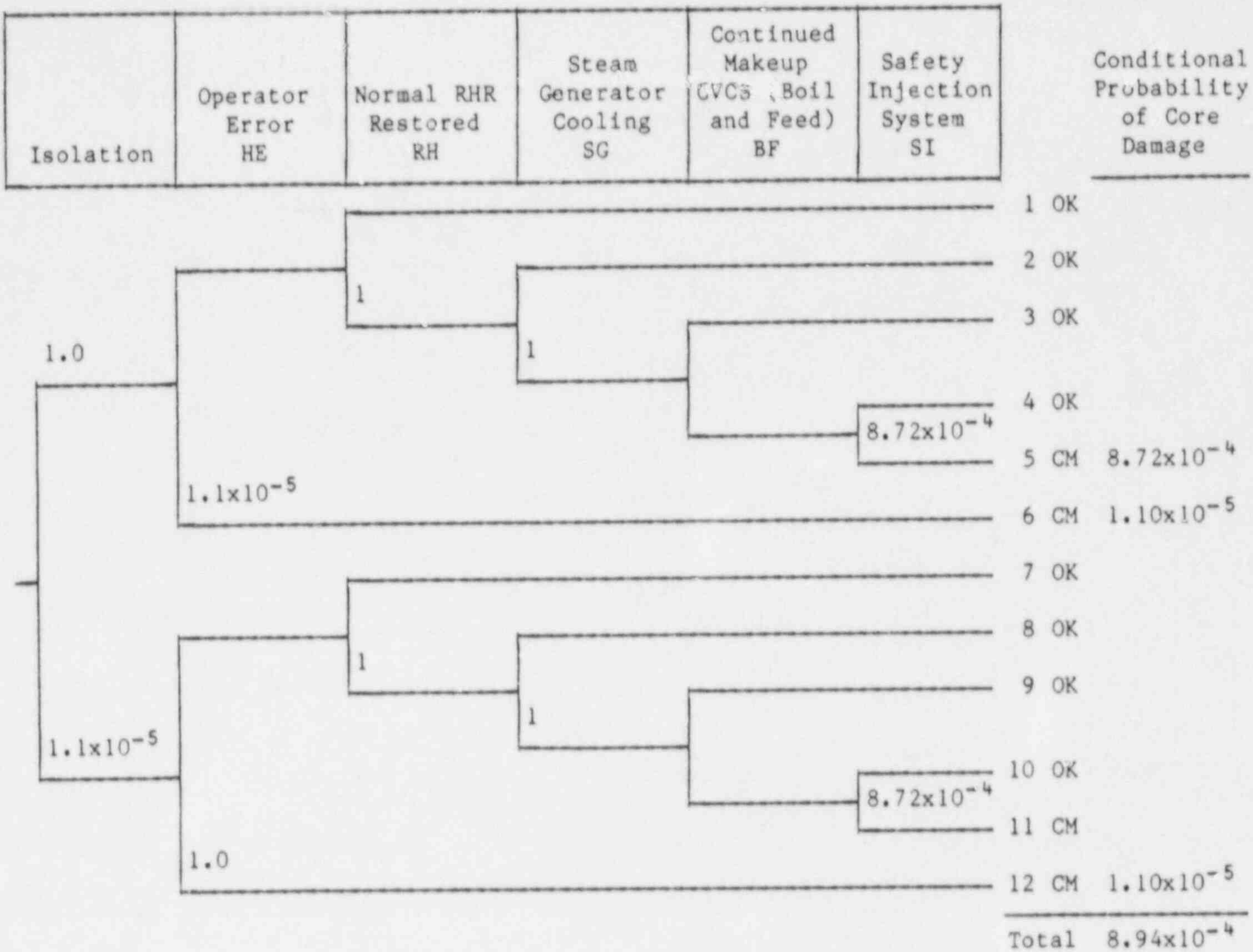


Figure 3.16. LOCA event tree for Phase 3 of refueling outage - stuck open RHR relief valve.

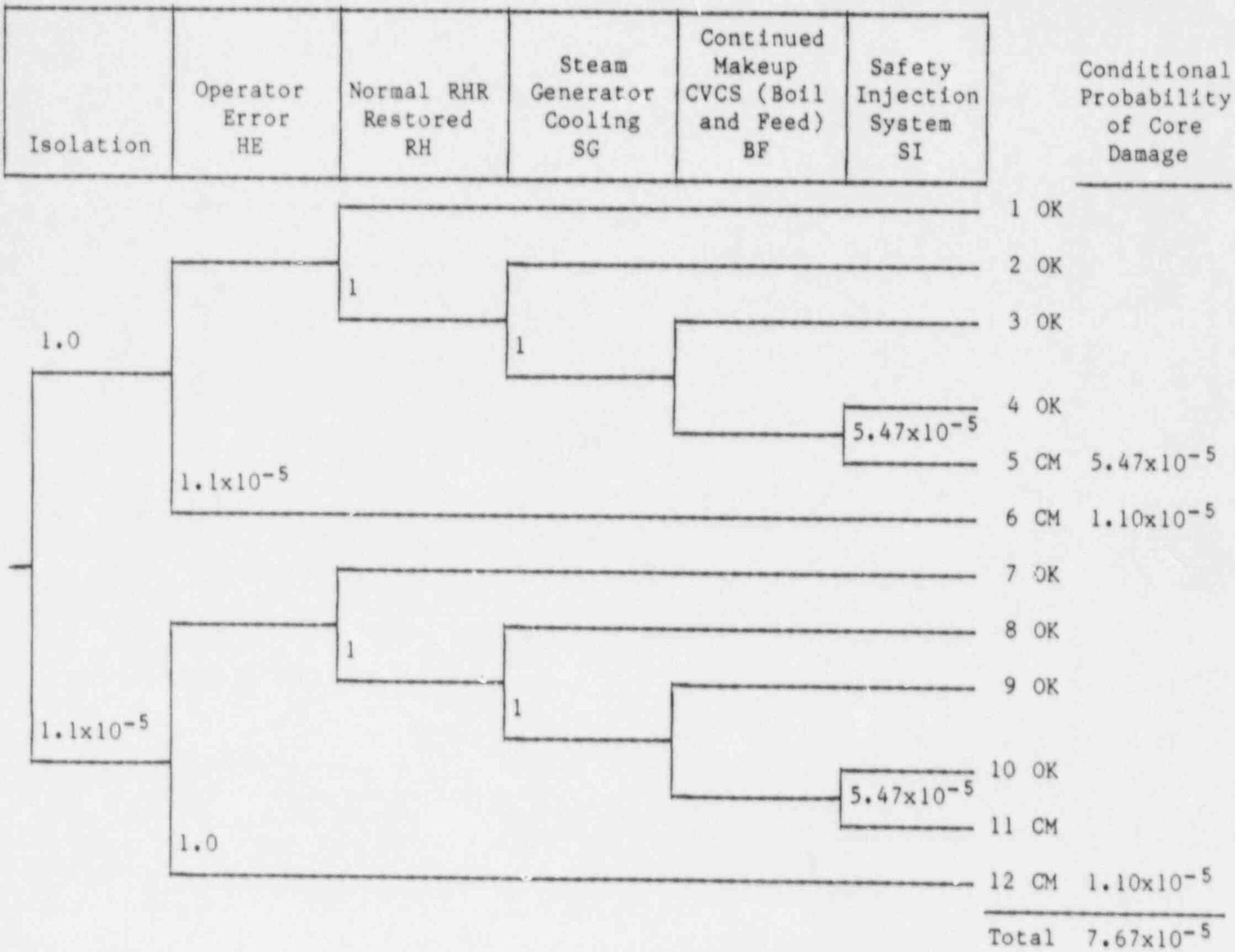


Figure 3.17. LOCA event tree for Phase 4 of refueling outage - stuck open RHR relief valve.

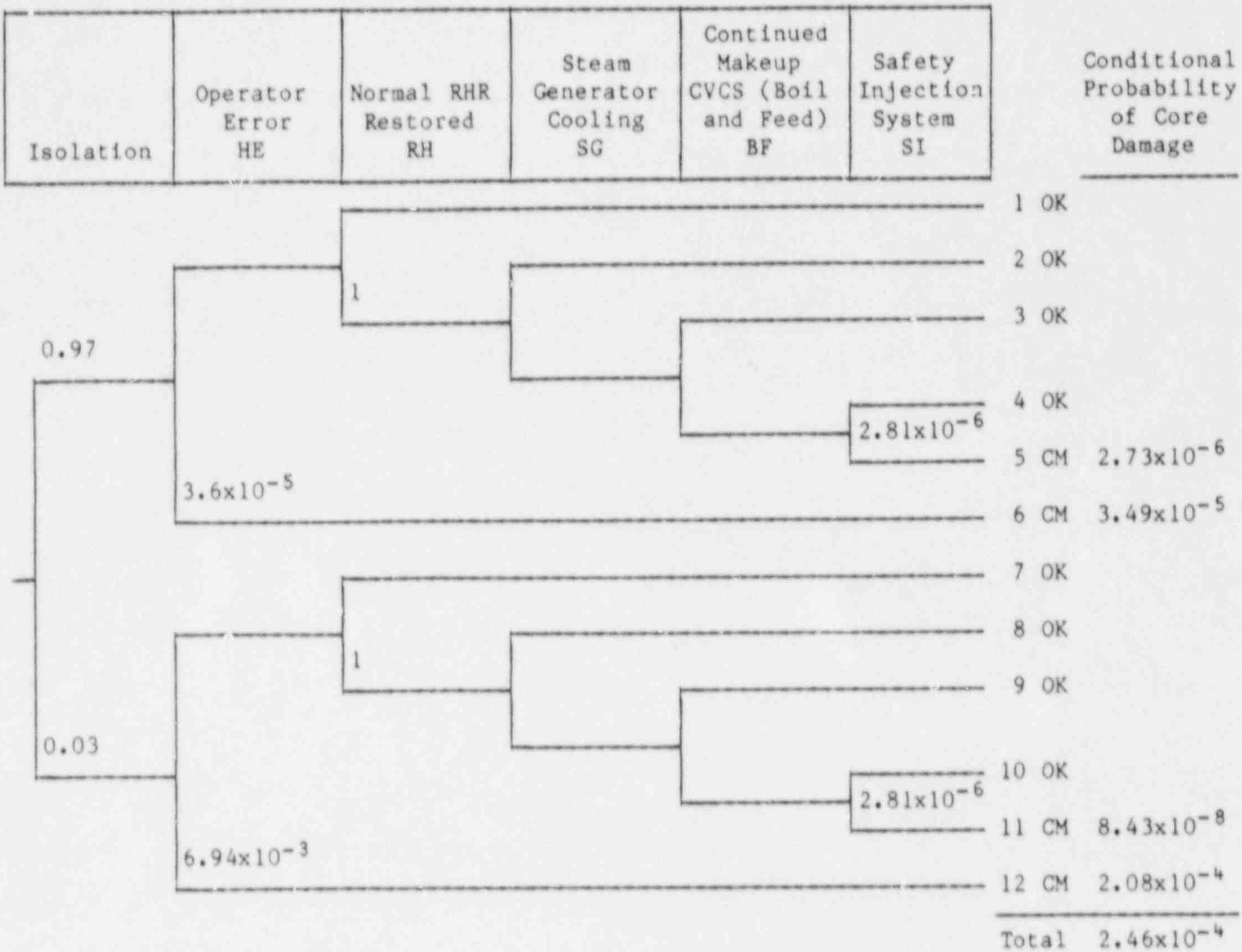


Figure 3.18. LOCA event tree for Phase 1 of drained maintenance - stuck open RHR relief valve.

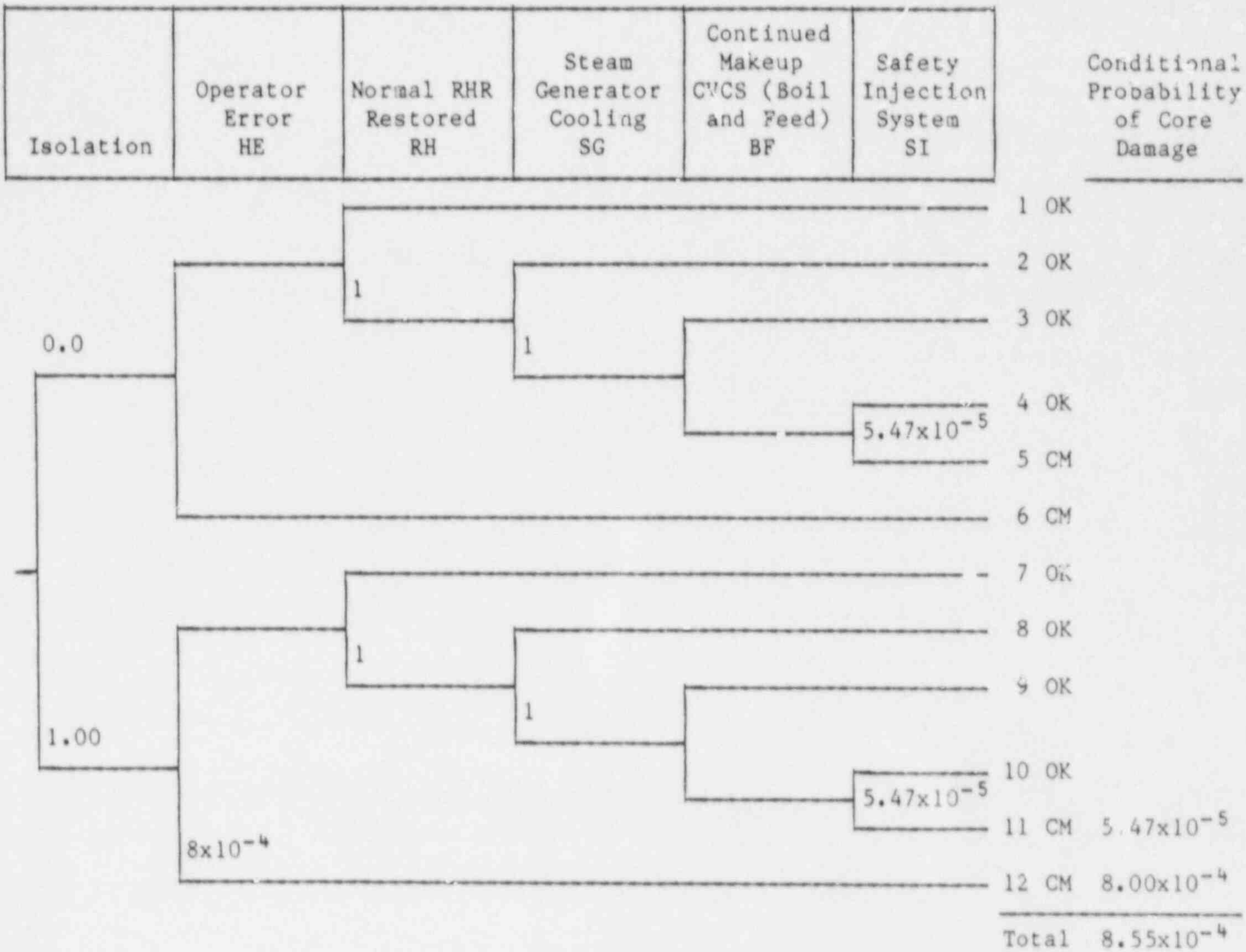


Figure 3.19. LOCA event tree for Phase 2 of drained maintenance - stuck open RHR relief valve.

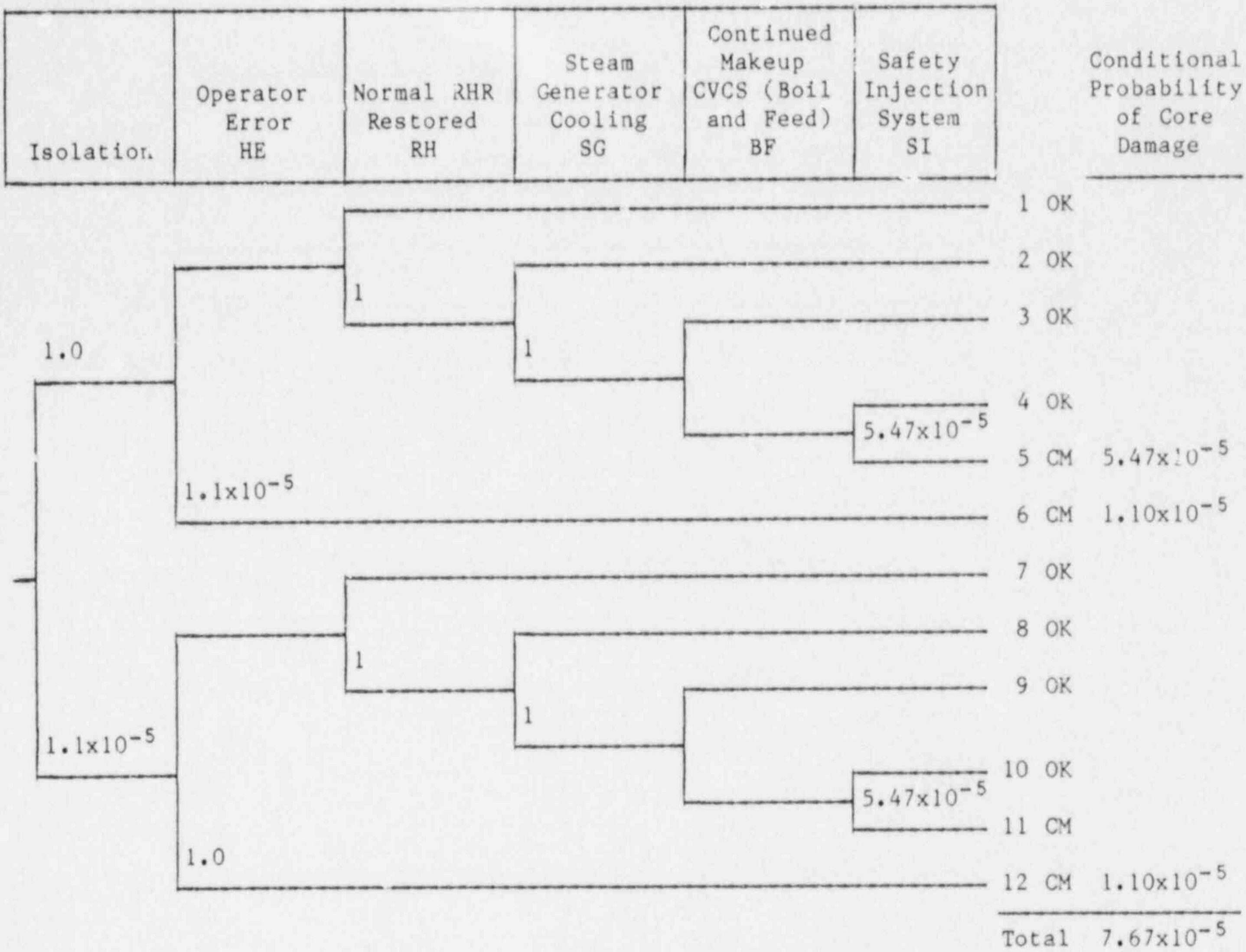


Figure 3.20. LOCA event tree for Phase 3 of drained maintenance - stuck open RHR relief valve.

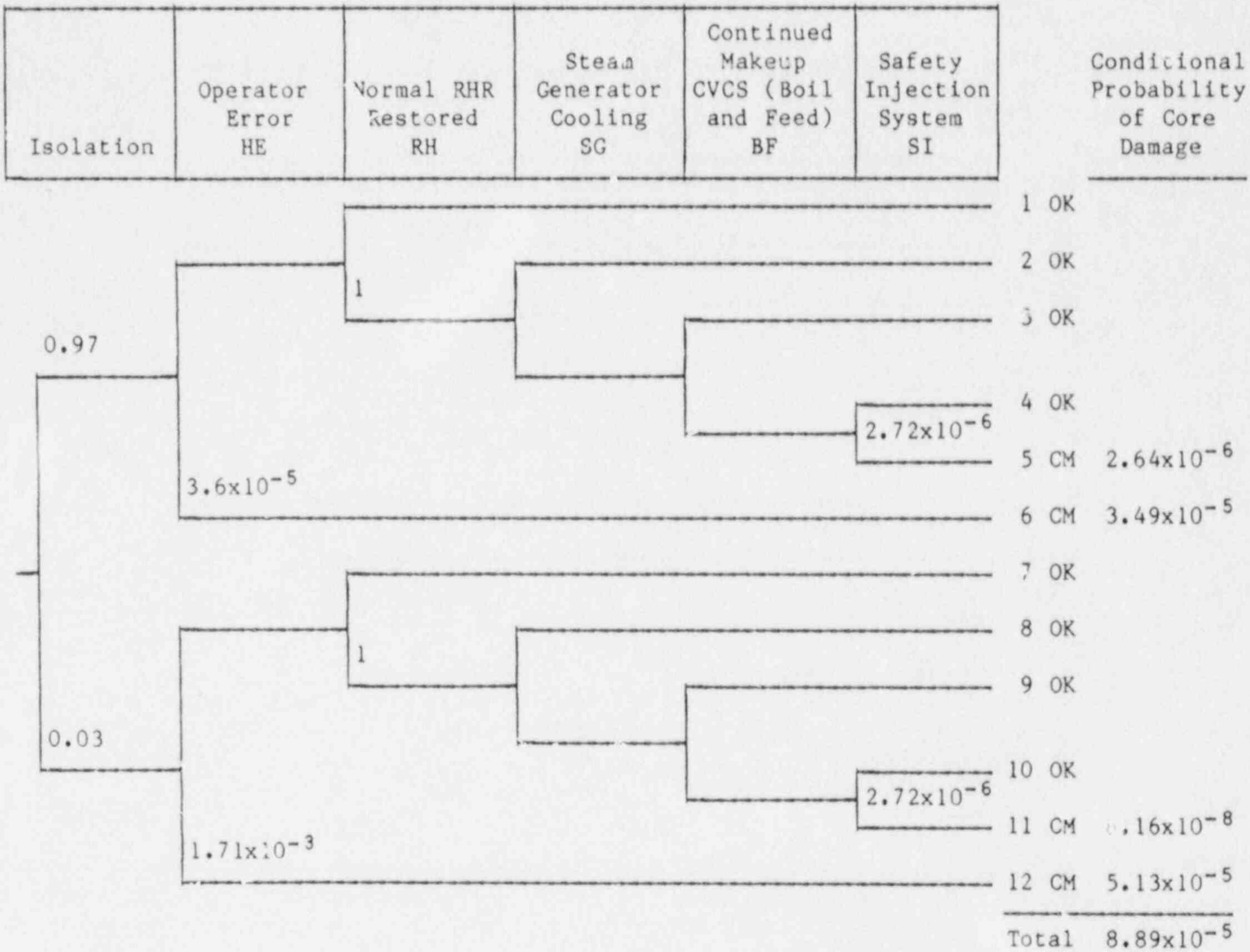


Figure 3.21. LOCA event tree for nondrained maintenance - stuck open RHR relief valve.

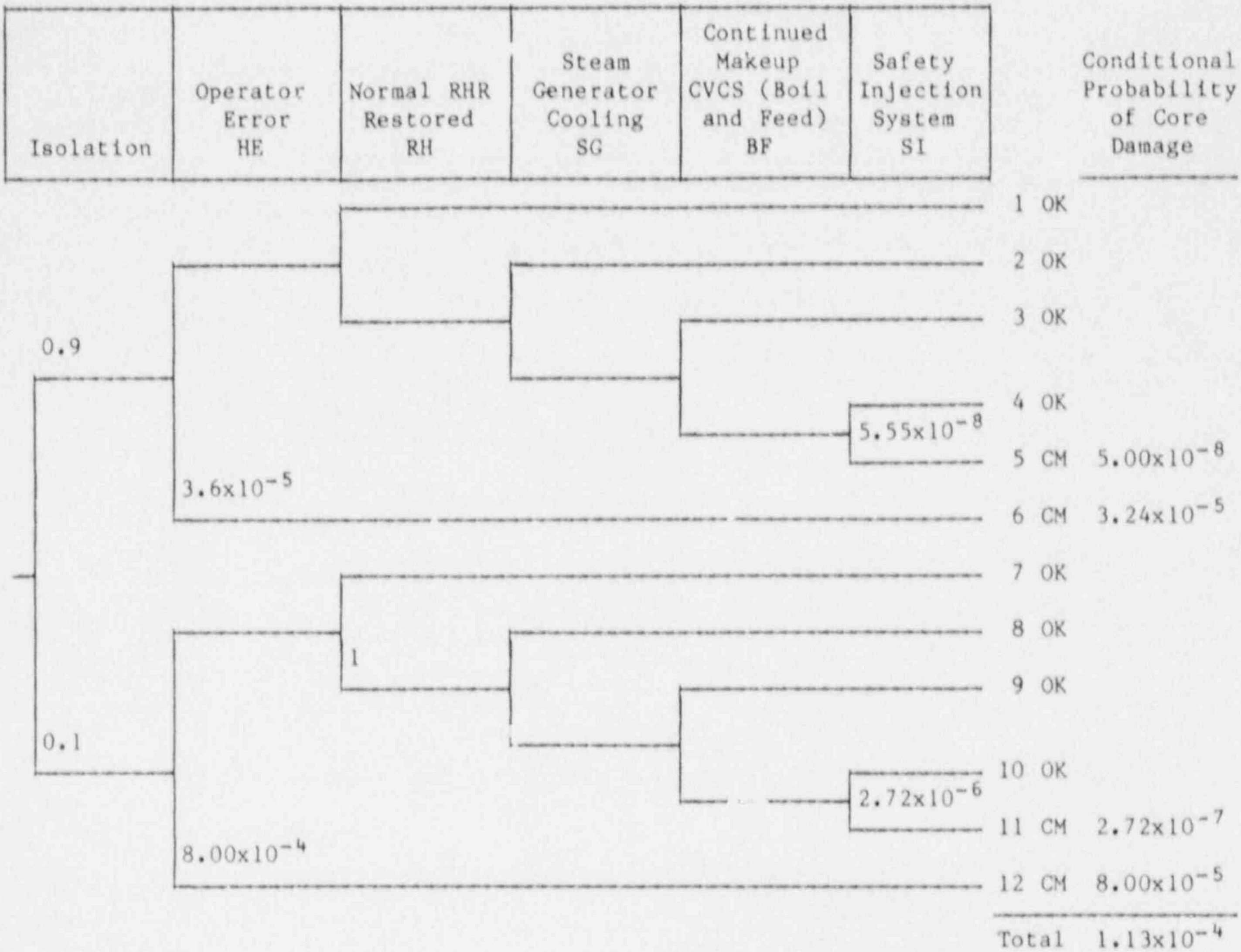


Figure 3.22. LOCA event tree for Phase 1 of refueling outage - inadvertent opening of containment spray header valves.

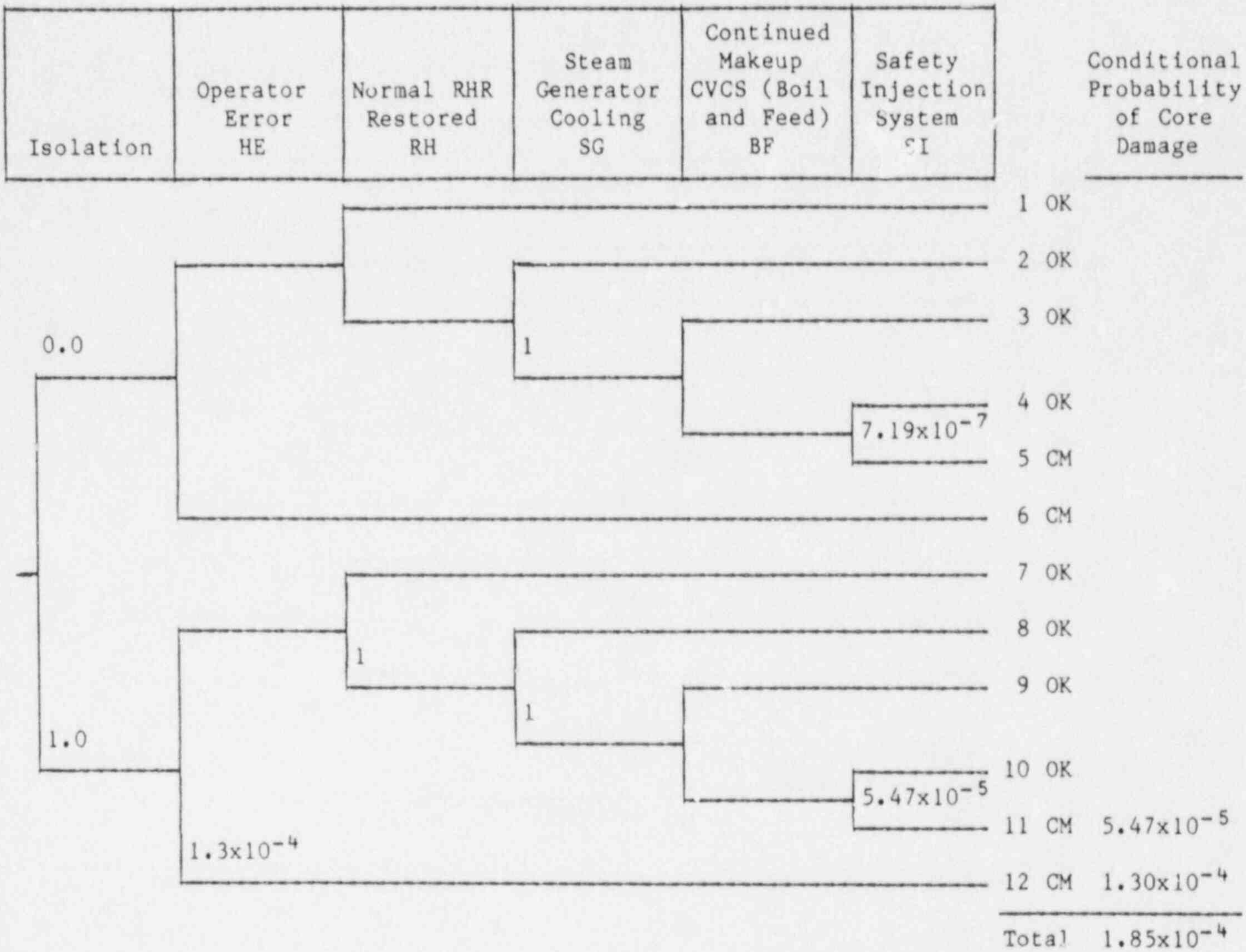


Figure 3.23. LOCA event tree for Phase 2 of refueling outage - inadvertent opening of containment spray header valves.

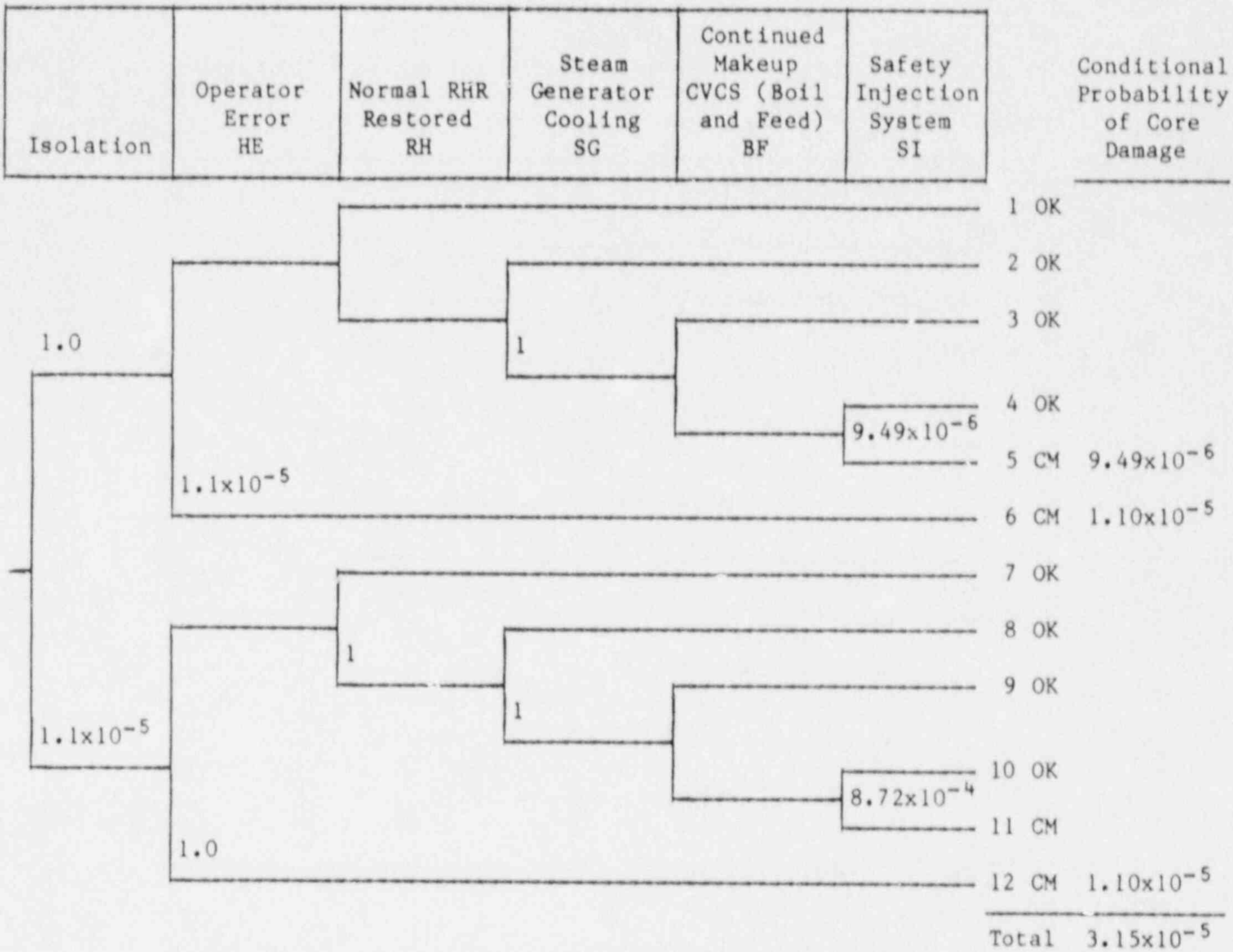


Figure 3.24. LOCA event tree for Phase 3 of refueling outage - inadvertent opening of containment spray header valves.

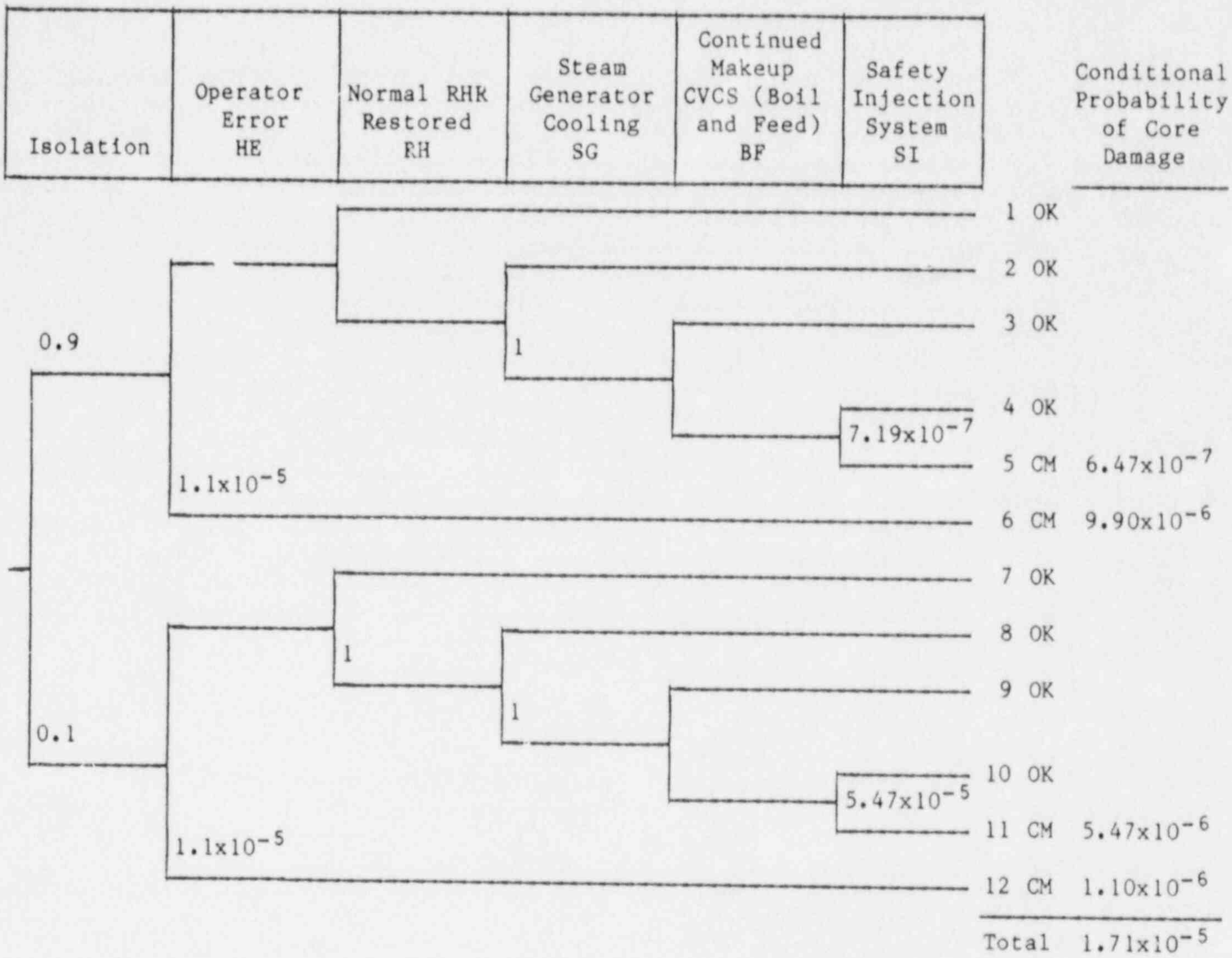


Figure 3.25. LOCA event tree for Phase 4 of refueling outage - inadvertent opening of containment spray header valves.

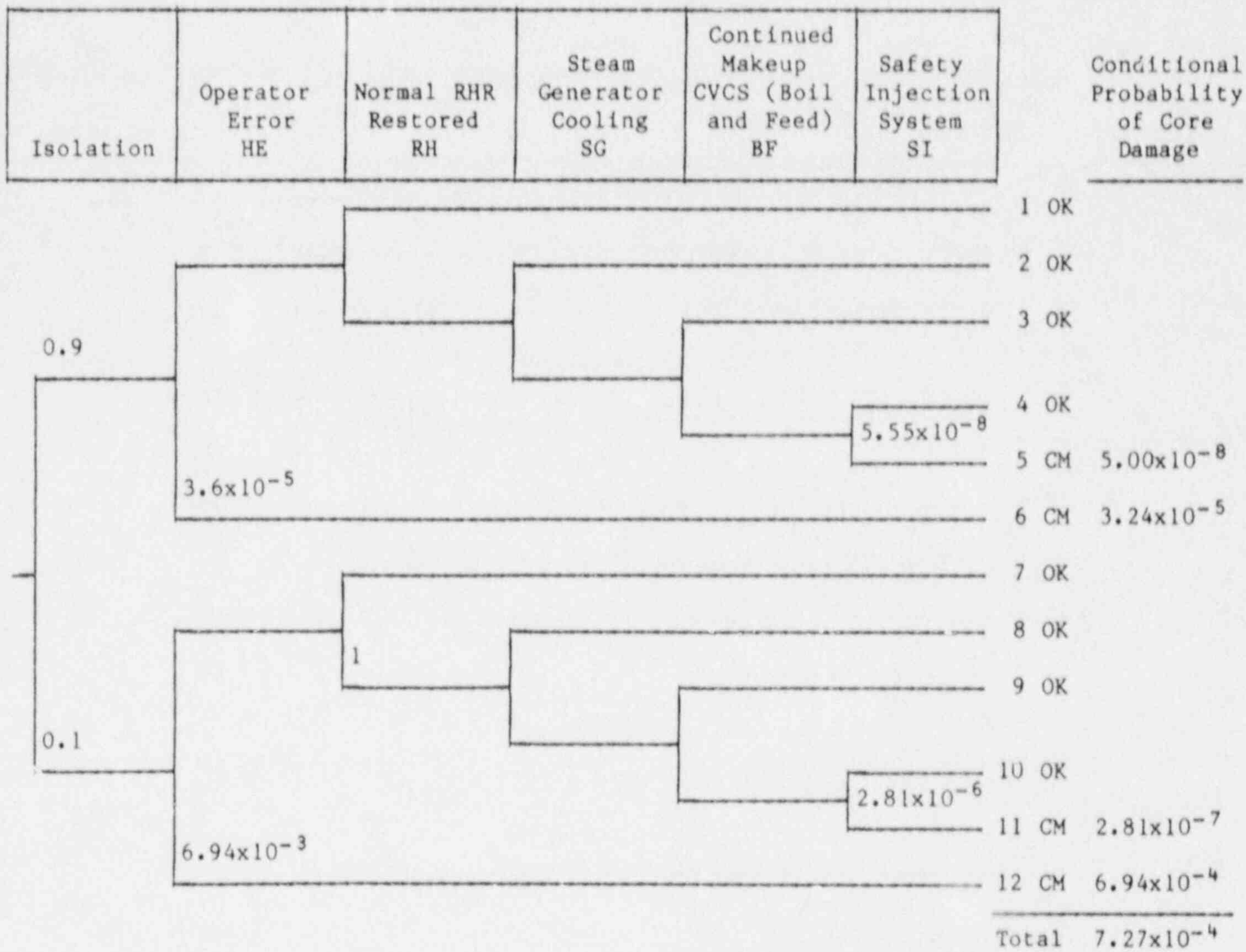


Figure 3.26. LOCA event tree for Phase 1 of drained maintenance - inadvertent opening of containment spray header valves.

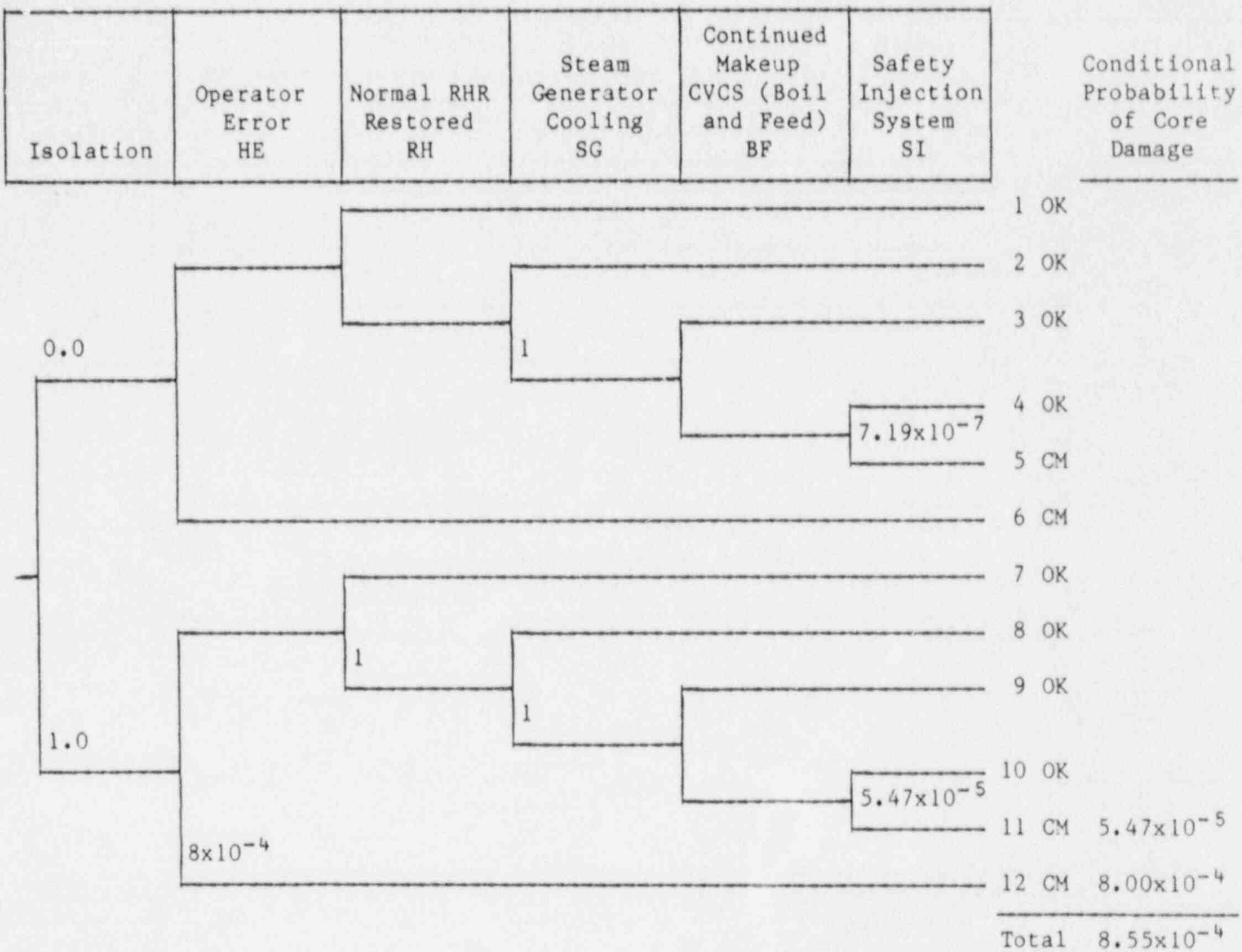


Figure 3.27. LOCA event tree for Phase 2 of drained maintenance - inadvertent opening of containment spray header valves.

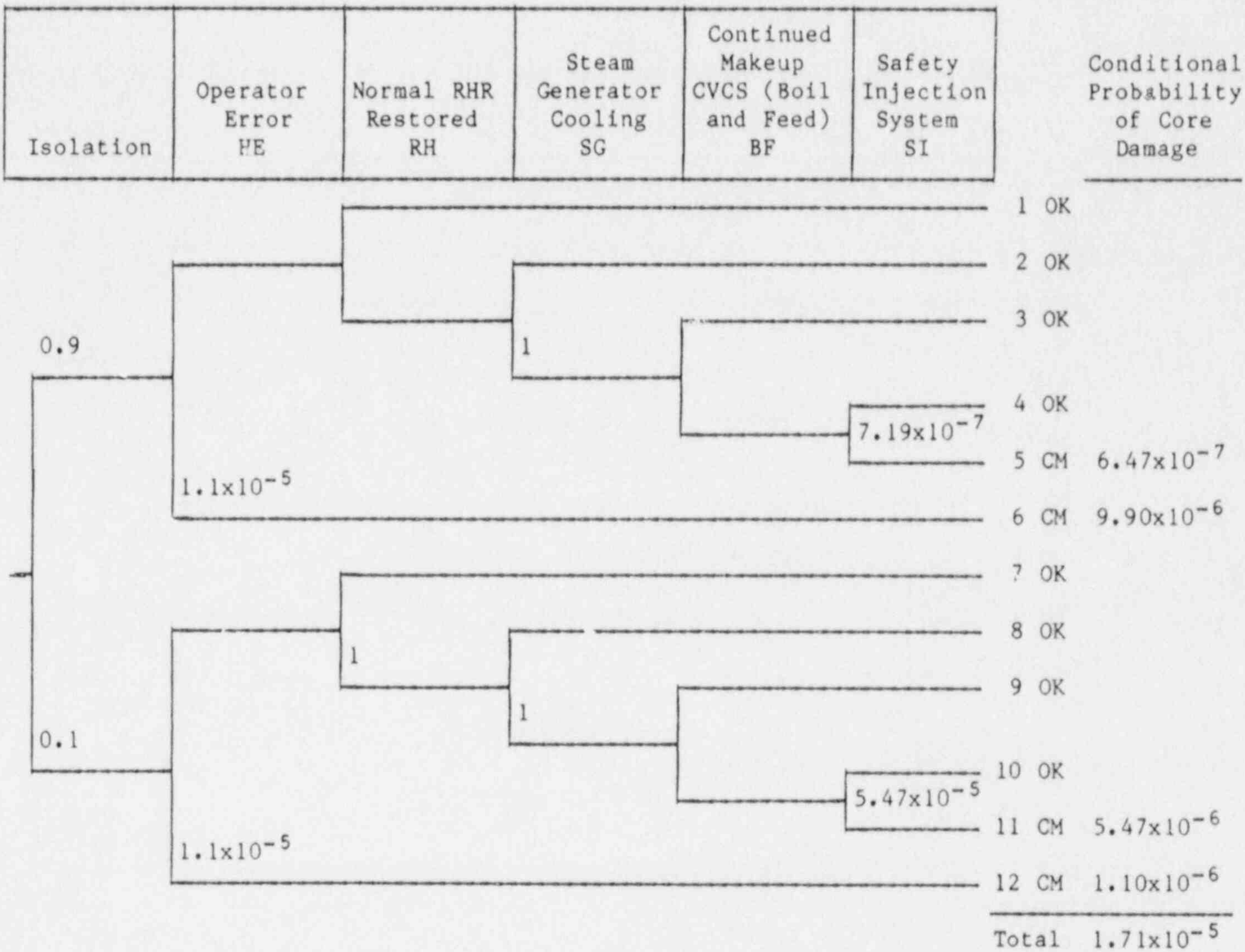


Figure 3.28. LOCA event tree for Phase 3 of drained maintenance - inadvertent opening of containment spray header valves.

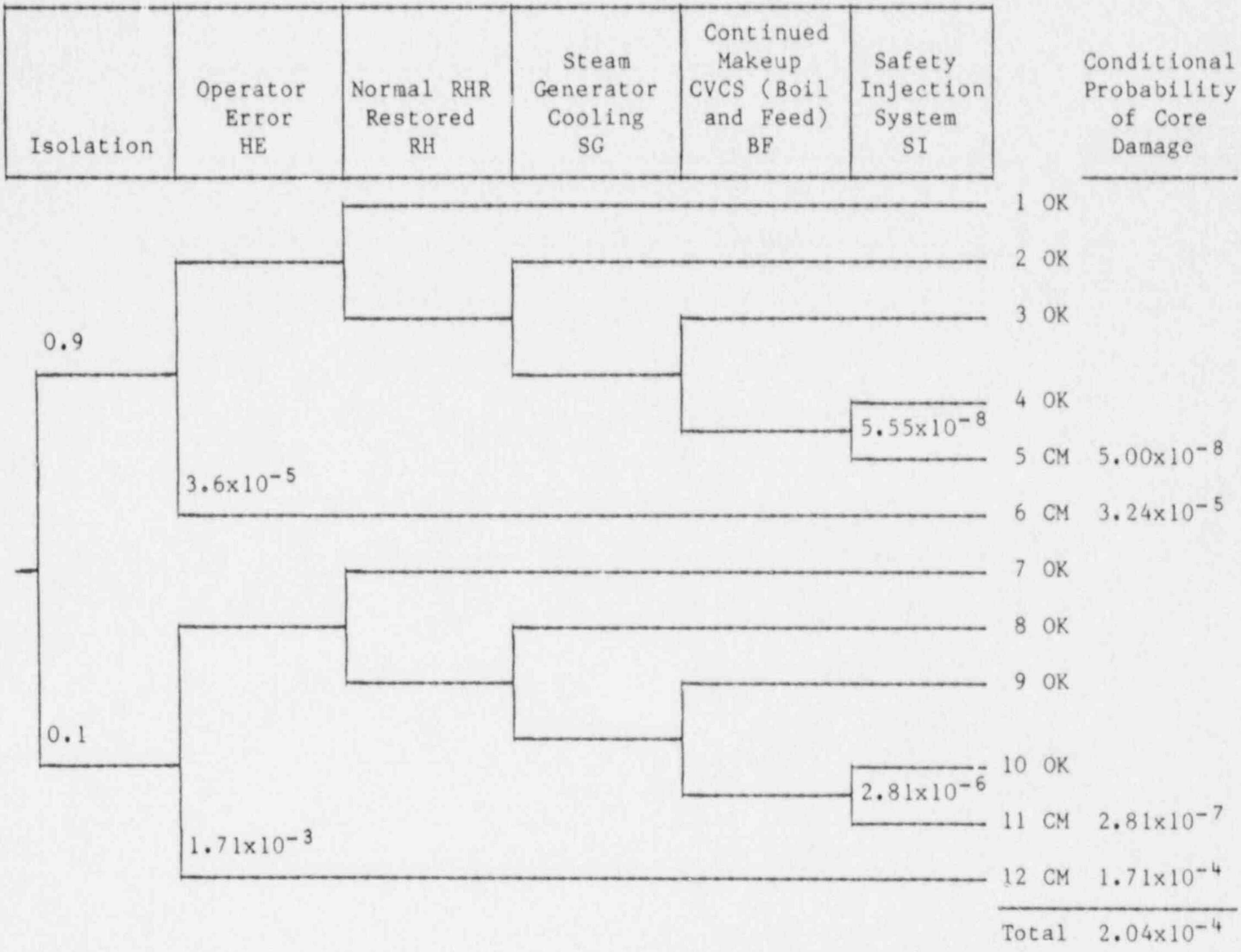


Figure 3.29. LOCA event tree for nondrained maintenance - inadvertent opening of containment spray header valves.

Table 3.1
Operational Experience of Loss of DHR

Plant	1976	1977	1978	1979	1980	1981	1982	1983	1984	1985	1986	Total
ANO-2				2					1			3
Beaver Valley-1			1	1	4	2	1	1				10
Callaway-1									1			1
Calvert Cliffs-1			2	1	5		1	1			1	11
Calvert Cliffs-2			2	1		2	3	2				10
Catawba-1										1	1	2
Cook-1										1		1
Cook-2									1			1
Crystal River		1	2	2	2	1					1	9
Davis-Besse			5	1	10	2						18
Diablo Canyon-1										2	1	3
Diablo Canyon-2											1	1
Farley-1			2		2	1				1		6
Farley-2								1				1
Fort Calhoun			1									1
Ginna								2	1			3
Haddam Neck									1			1
Indian Point-3	1								1			2
Main Yankee						2						2
McGuire-1							2	1				3
McGuire-2								1	2			3
Millstone-2				1		1	1					3
North Anna-1				1	1		2	2				6
North Anna-2							4	3	1			8
Oconee-1						1						1
Oconee-2						1						1
Oconee-3										1		1
Palisades			1			1						2
Palo Verde-2											1	1
Rancho Seco						1	1			1	3	6
Salem-1	2			3			1		1			7
Salem-2						2		7	1			10
San Onofre-1					1					1		2
San Onofre-2											1	1
Sequoyah-1						1	1			2		4
Sequoyah-2								1				1
St. Lucie-1			1					1				2
Summer-1								1	2	1		4
Surry-1								1			1	2
Surry-2											1	1
Trojan		1	5			1						7
Turkey Point-3								2		1		3
Turkey Point-4						2			1		1	4
Waterford-3											2	2
Yankee Rowe											1	1
Zion-1							1		1			2
Zion-2										1	2	3
Total Number of Events	3	3	21	13	25	21	18	27	15	13	18	177
Number of Operating Plants	36	38	41	41	43	47	49	52	54	60	62	
Number of Events / Number of Operating Plants	0.08	0.08	0.51	0.32	0.58	0.45	0.37	0.52	0.28	0.22	0.29	

Table 3.2
Classification of Loss of DHR Events

	NSAC-52 1976-1981	AEOD 1982-1983	Sub-Total 1976-1983	BNL 1984-1986	Total 1976-1986
Spurious Isolation	23	20	43	21	64
Overdraining	4	9	13	8	21
Inadequate Inventory	8	5	13	3	16
LOCA	7	0	7	2	9
Spurious Containment Spray	2	0	2	0	2
Others	42	11	53	12	65
Total	86	45	131	46	177

Table 3.3
Frequencies of Initiating Events That Lead to Loss of DHR

Initiating Event	Frequency/Probability
A. Spurious Isolation of Suction Valves	3.58×10^{-5}/hour
B. Overdraining	1.21×10^{-2}
C. Inadequate Inventory	6.35×10^{-5}/hour
D. LOCA	5.03×10^{-6}/hour
E. Spurious Containment Spray	1.12×10^{-6}/hour

Table 3.4
Comparison of Generic Data With Zion Specific Data

	NSAC-84/ZPSS	Generic Data
RHR Pump		
Failure to Start	1.68×10^{-3}(/demand)	5×10^{-4}(/demand)
Failure to Run	7.26×10^{-5}(/hour)	2×10^{-5}(/hour)
Centrifugal Charging Pump		
Failure to Start	7.21×10^{-4}(/demand)	5×10^{-4}(/demand)
Failure to Run	1.76×10^{-6}(/hour)	2×10^{-5}(/hour)
Motor-Driven AFW Pump		
Failure to Start	5.02×10^{-3}(/demand)	5×10^{-4}(/demand)
Failure to Run	9.87×10^{-5}(/hour)	2.0×10^{-5}(/hour)
Turbine-Driven AFW Pump		
Failure to Start	1.15×10^{-2}(/demand)	4×10^{-3}(/hour)
Failure to Run	7.63×10^{-6}(/hour)	2×10^{-5}(/hour)
SI Pump		
Failure to Start	7.21×10^{-4}(/demand)	5×10^{-4}(/demand)
Failure to Run	1.55×10^{-5}(/hour)	2×10^{-5}(/hour)
DG		
Failure to Start	1.82×10^{-2}(/demand)	3×10^{-2}(/demand)
Failure to Run	5.97×10^{-3}(/hour)	3×10^{-3}(/hour)*
Bus - Open Circuit	1.91×10^{-8}(/hour)	3.6×10^{-6}(/hour)
Circuit Breaker - Transfer Open	2.32×10^{-7}(/hour)	1.6×10^{-7}(/hour)
MOV		
Transfer Closed	5.28×10^{-8}(/hour)	2.3×10^{-7}(/hour)
Failure to Open	1.55×10^{-3}(/demand)	4.0×10^{-3}(/demand)
AOV		
Failure to Operate	1.44×10^{-3}(/demand)	9×10^{-4}(/demand)
Transfer Closed	5.28×10^{-8}(/hour)	2.3×10^{-7}(/hour)
Manual Valve - Transfer Closed	5.28×10^{-8}(/hour)	3.4×10^{-8}(/hour)
Check Valve - Failure to Open	4.32×10^{-5}(/demand)	1.0×10^{-4}(/demand)
RHR HX - Rupture	7.13×10^{-7}(/hour)	4.56×10^{-6}(/hour)

*This failure rate is taken from NUREG/CR-2815.

Table 3.5
Fault Tree of the Operating Train of RHR System

Top Event or Intermediate Gate Output	Gate Type	Input to the Gate					
RM	+	A-RHR	B1-RHR	C1-RHR	HDR-RHR	HE-RHR	LCCW
		LSW	LOSP	RV-RHR	DRAIN		
HDR-RHR	*	MO8809A	XTI-RHR				
XTI-RHR	+	D-RHR	MO8809B				
R1-RHR	+	A-BUS	B-BKR	DA-RHR	MO8700A	MV8708	
C1-RHR	+	HXA-RHR	MV8724A	MO9412A	MV9504A	MV9507A	AOV606
D-RHR	+	MO8716A	MO8716B	MO87160			

Note: See Table 3.6 for basic event description and basic event probabilities.

Table 3.6
Basic Event Probabilities Used in Quantification of Frequency of Loss of DHR
(Phase 2 of a Refueling Outage)

Basic Event	Description	Failure Rate/Prob. (Per Hr./ Demand)	Source	** Mission Time (Hr.)	Basic Event Prob.
A-RHR	Spurious Isolation Signal	3.58×10^{-5}	*	420	1.50E-02
A-BUS	4 kV Bus Failure	3.6E-06	OPRA	420	1.51E-03
B-BKR	4 kV Bus Feed Breaker Transfers Open	1.6E-07	OPRA	420	6.72E-05
PA-RHR	RHR Pump Fails to Run	2.0E-05	OPRA	420	8.4E-03
HXA-RHR	RHR Heat Exchanger Failure	4.56E-06	OPRA	420	1.92E-03
MO8700A	MOV8700A Transfers Closed	2.30E-07	OPRA	420	9.66E-05
MV8728A	MV8728A Transfers Closed	2.30E-07	OPRA	420	9.66E-05
MV8724A	MV8724A Transfers Closed	2.30E-07	OPRA	420	9.66E-05
MO9412A	MOV9412A Transfers Closed	2.30E-07	OPRA	420	9.66E-05
MV9504A	MV9504A Transfers Closed	2.30E-07	OPRA	420	9.66E-05
MV9807A	MV9507A Transfers Closed	2.30E-07	OPRA	420	9.66E-05
AOV606	AOV606 Transfers Closed	2.30E-07	OPRA	420	9.66E-05
MO8809A	MOV8809A Transfers Closed	2.30E-07	OPRA	420	9.66E-05
MO8716A	MOV8716A Transfers Closed	2.30E-07	OPRA	420	9.66E-05
MO8716B	MOV8716B Transfers Closed	2.30E-07	OPRA	420	9.66E-05
MO8716C	MOV8716C Transfers Closed	2.30E-07	OPRA	420	9.66E-05
MO8809B	MOV8809B Transfers Closed	2.30E-07	OPRA	420	9.66E-05
LCCW	Loss of Component Cooling Water System	8.26E-08	ZPSS	420	3.47E-05
LSW	Loss of Service Water System	1.07E-07	ZPSS	420	4.49E-05
LOSP	Loss of Offsite Power	1.00E-05	NUREG-1032	420	4.20E-03
RV-RHR	Human Error Induced Overdraining	1.21E-02	*	---	2.42×10^{-2}
DRAIN	Failure to Maintain Level	6.35E-05	*	420	2.67×10^{-2}

*See Section 3.1.1.
**The mission time for phase 2 of a refueling outage is used in this table as an example.

Table 3.7
Summary Results for Loss of Cooling Event Trees - Generic
(CDF Per Year)

Outage Type	Phases 1	2	3	4	Total
Refueling	2.39×10^{-7}	5.63×10^{-6}	1.84×10^{-6}	1.08×10^{-6}	8.79×10^{-6}
Drained Maintenance	3.40×10^{-7}	3.08×10^{-5}	2.50×10^{-6}		3.36×10^{-5}
Nondrained Maintenance	3.96×10^{-7}				3.96×10^{-7}
Total					4.28×10^{-5}

Table 3.8
Basic Event Probabilities Used for Onsite Power System

		Shutdown (BNL/NSAC-84)	Power Operation (ZPSS)
Unavailability of DG due to maintenance		$4.56*10^{-2}$	$3.44*10^{-2}$
Unavailability of Essential 4 kV Bus Due to maintenance	47	$1.12*10^{-2}$	0
	48	$6.53*10^{-3}$	0
	49	$8.16*10^{-3}$	0
DG failure to start		$1.82*10^{-2}$	$1.82*10^{-2}$
DG failure to run (/hour)		$5.97*10^{-3}$	$5.97*10^{-3}$
β factor for DG Failure to start:		0.05	0
Bus-fail open (/hour)		$1.91*10^{-8}$	$1.91*10^{-8}$
Breaker-transfer open (/hour)		$2.32*10^{-7}$	$2.32*10^{-7}$

Table 3.9
Probability of Nonrecovery of AC Power

Time at Which Decay Heat Removal Capability is Lost (Hours)	P(Nonrecovery of AC Power
72-105	0.3
105-280	0.07
280-580	0.04
580-1435	0.02
Beyond 1435	0.01

Table 3.10
Summary Results for Loss of Offsite Power

Sequence	Nondrained Maintenance (146 Hrs.) 1.121/Year	Drained Maintenance (982 Hrs.) 1.93/Year	Refueling (1996 Hrs.) 0.747/Year	Total
5	2.49×10^{-8}	$3.03*10^{-6}$	2.03×10^{-6}	5.08×10^{-6}
6	2.62×10^{-8}	1.72×10^{-7}	3.43×10^{-8}	2.33×10^{-7}
Total	5.11×10^{-8}	3.84×10^{-6}	2.50×10^{-6}	5.31×10^{-6}

Table 3.11
Core Damage Frequency Due to LOCAs in a Refueling Outage
- Stuck Open RHR Relief Valve

	Phase 1	Phase 2	Phase 3	Phase 4	Total
Duration (hour)	113	420	500	909	
Frequency (per year)	0.747	0.747	0.747	0.747	
f(LOCA) (per year)	4.25×10^{-4}	1.58×10^{-3}	1.88×10^{-3}	3.42×10^{-3}	7.31×10^{-3}
P(CD/LOCA)	6.17×10^{-5}	1.85×10^{-4}	8.94×10^{-4}	7.67×10^{-5}	
f(CD) (per year)	2.68×10^{-8}	2.92×10^{-7}	1.68×10^{-6}	2.62×10^{-7}	2.26×10^{-6}

Table 3.12
Core Damage Frequency Due to LOCAs in a Drained Maintenance
- Stuck Open RHR Relief Valve

	Phase 1	Phase 2	Phase 3	Total
Duration (hour)	62	96	803	
Frequency (per year)	1.932	1.932	1.932	
f(LOCA) (per year)	6.03×10^{-4}	9.33×10^{-4}	7.8×10^{-3}	9.34×10^{-2}
P(CD/LOCA)	2.46×10^{-4}	8.55×10^{-4}	7.67×10^{-5}	
f(CD) (per year)	1.48×10^{-7}	7.98×10^{-7}	5.98×10^{-7}	1.54×10^{-6}

Notes for Tables 3.11 and 3.12:
Frequency = Frequency of the phase of outage.
f(LOCA) = Frequency that a LOCA occurs in the phase.
P(CD/LOCA) = Conditional probability of core damage given a LOCA.
f(CD) = Frequency of core damage.

Table 3.13

Core Damage Frequency Due to LOCAs in a Nondrained Maintenance - Stuck Open RHR Relief Valve

Duration (hour)	125
Frequency (per year)	1.121
f(LOCA) (per year)	7.05×10^{-4}
P(CD/LOCA)	8.89×10^{-5}
f(CD) (per year)	6.27×10^{-8}

Table 3.14

Core Damage Frequency Due to Spurious Containment Spray in a Refueling Outage

	Phase 1	Phase 2	Phase 3	Phase 4	Total
Duration (hour)	113	420	500	909	
Frequency (per year)	0.747	0.747	0.747	0.747	
f(Spray) (per year)	9.45×10^{-5}	3.51×10^{-4}	4.18×10^{-4}	7.61×10^{-4}	1.62×10^{-3}
P(CD/Spray)	1.13×10^{-4}	1.85×10^{-4}	3.15×10^{-5}	1.71×10^{-5}	
f(CD) (per year)	1.07×10^{-8}	6.49×10^{-8}	1.32×10^{-8}	1.30×10^{-8}	1.02×10^{-7}

Notes for Tables 3.13 and 3.14:
Frequency = Frequency of the phase of shutdown.
f(Spray) = Frequency that a spurious spray ocurs in the phase.
f(LOCA) = Frequency that a LOCA occurs in the phase.
P(CD/LOCA) = Conditional probability of core damage given a LOCA.
P(CD/Spray) = Conditional probability of core damage given a spurious containment spray.
f(CD) = Frequency of core damage.

Table 3.15
Core Damage Frequency Due to Spurious Containment Spray in a Drained Maintenance Outage

	Phase 1	Phase 2	Phase 3	Total
Duration (hour)	62	96	803	
Frequency (per year)	1.932	1.932	1.932	
f(Spray) (per year)	1.34×10^{-4}	2.08×10^{-4}	1.74×10^{-3}	2.08×10^{-3}
P(CD/Spray)	7.27×10^{-4}	8.55×10^{-4}	1.71×10^{-5}	
f(CD) (per year)	9.74×10^{-8}	1.78×10^{-7}	2.98×10^{-8}	3.05×10^{-7}

Table 3.16
Core Damage Frequency Due to Spurious Containment Spray in a Nondrained Maintenance

	Phase 1
Duration (hour)	125
Frequency (per year)	1.121
f(Spray) (per year)	1.57×10^{-4}
P(CD/Spray)	2.04×10^{-4}
f(CD) (per year)	3.20×10^{-8}

Notes for Tables 3.15 and 3.16:
Frequency = Frequency of the phase of shutdown.
f(Spray) = Frequency that a spurious spray occurs in the phase.
P(CD/Spray) = Conditional probability of core damage given a spurious containment spray.
f(CD) = Frequency of core damage.

Table 3.17
Summary of Results for Generic Shutdown Risk

Initiating Event	Frequency (Per Year)
Loss of Cooling	$4.28x10^{-5}$
LOCA	$4.30x10^{-6}$
Loss of Offsite Power	$5.08*10^{-6}$
Total	$5.22*10^{-5}$

Table 3.18
Summary of Core Damage Frequency Results for a Generic Plant at Shutdown
(Loss-of-Cooling + LOCA + LOOP)

Initiating Events	Core Damage Frequency (Per Year)				
	Base Case	I1	I2	I3	I1+I2
Loss of Cooling	$4.28x10^{-5}$	$4.27x10^{-6}$	$9.59x10^{-6}$	$3.94x10^{-5}$	$4.07x10^{-6}$
LOCA	$4.30x10^{-6}$	$2.63x10^{-6}$	$4.30x10^{-6}$	$4.30x10^{-6}$	$2.63x10^{-6}$
Loss of Offsite Power	$5.08x10^{-6}$	$5.08x10^{-6}$	$5.08x10^{-6}$	$5.08x10^{-6}$	$5.08x10^{-6}$
Total	$5.22x10^{-5}$	$1.20x10^{-5}$	$1.90x10^{-5}$	$4.88x10^{-5}$	$1.18x10^{-5}$

I1 = Upgraded instrumentation for RHR pumps.
I2 = Upgraded vessel level indication.
I3 = Removal of auto closure interlock.

Table 3.19
Summary of Benefits of the Proposed Improvements for a Generic Plant
(Loss-of-Cooling + LOCA + LOOP)

	Base Case	I1	I2	I3	I1+I2
f(LC) (per year)	3.21×10^{-1}	3.21×10^{-1}	2.49×10^{-1}	1.71×10^{-1}	2.49×10^{-1}
Δf(LC) (per year)	N.A.	~0	7.20×10^{-2}	1.19×10^{-1}	7.20×10^{-2}
% Reduction		0%	22%	60%	22%
CDF (per year)	5.22×10^{-5}	1.20×10^{-5}	1.90×10^{-5}	4.88×10^{-5}	1.18×10^{-5}
ΔCDF (per year)	N.A.	4.02×10^{-5}	3.32×10^{-5}	3.40×10^{-6}	4.04×10^{-5}
% Reduction		77%	64%	7%	77%

Table 3.20
Summary of Benefits of the Proposed Improvements for Loss-of-Cooling Events

	Base Case	I1	I2	I3	I1+I2
f(LC) (per year)	3.21×10^{-1}	3.21×10^{-1}	2.49×10^{-1}	1.7×10^{-1}	2.49×10^{-1}
Δf(LC) (per year)	N.A.	~0	7.20×10^{-2}	1.19×10^{-1}	7.20×10^{-2}
% Reduction		0%	22%	60%	22%
Loss-of-cooling CDF (per year)	4.28×10^{-5}	4.27×10^{-6}	9.59×10^{-6}	3.94×10^{-5}	4.07×10^{-6}
Loss-of-cooling ΔCDF (per year)	N.A.	3.85×10^{-5}	3.32×10^{-5}	3.40×10^{-6}	3.87×10^{-5}
% Reduction		90%	78%	8%	90%

Notes for Tables 3.19 and 3.20:
I1 = Upgraded instrumentation for RHR pumps.
I2 = Upgraded vessel level indication.
I3 = Removal of auto closure interlock.
f(LC) = Frequency of loss of cooling.
CDF = Core damage frequency.

4. CONSEQUENCE AND RISK CALCULATIONS

The shutdown risk is calculated as the product of the frequency of core damage and the consequence of a core damage, i.e., Risk = f(core damage) * consequence. Section 4.1 presents a containment event tree that is used to investigate the integrity of the containment and the operability of the containment spray system. The end states of the containment event tree are PWR release categories. The frequency of a release category is the product of the core damage frequency and the sum of the probabilities of the sequences in the containment event tree that result in the release category, i.e.,

$$\begin{aligned} f(\text{release category } i) &= f(\text{core damage}) * P(\text{category } i) \\ &= f(\text{core damage}) * \sum P(\text{sequence } j) \qquad (4.1) \end{aligned}$$

where the summation is over all sequences that result in category i.

Sections 4.2 to 4.5 provide the calculations of the consequences of the release categories. The risk can be calculated as follows:

$$\begin{aligned} \text{Risk} = f(\text{core damage}) * \sum_i \big[& P(\text{release category } i) \\ & * \text{consequence (release category } i) \big] \qquad (4.2) \end{aligned}$$

Section 4.6 presents the results of the risk calculations using the above equation.

4.1 Containment Event Tree

Figure 4.1 is the containment event tree that was used to estimate the potential releases of radioactivity resulting from a core damage event during shutdown. Available information is not sufficient to allow satisfactory quantification of the event tree and it was therefore decided to perform sensitivity calculations using the probabilities shown in Figure 4.1. As a result, the risk calculations are also conditional on these assumed probability values. The ranges of probability values were chosen to simply represent a high, medium, and low probability associated with each of the top events.

EH - Equipment Hatch Closed. During an outage, the equipment hatch may be open. It is not known what fraction of time in an outage the hatch is open. Typically, Technical Specifications only require containment isolation while fuel is being shuffled. Information from the plants indicates that one or two hours will be needed to close the steel hatch door and bolt it down. This means that there may be sufficient time for operators to close the hatch. However, they may not start closing the hatch early enough. In the loss of cooling events that have occurred, no attempted closing of the equipment hatch was reported.

CP - Containment Penetrations Closed. For test or maintenance purposes, some containment penetrations may be opened in an outage, e.g., spare penetrations may be opened to pull cable through. The penetrations can be closed in time to prevent release of radioactivity, however, the operators may not start doing that until it is too late.

CS - Containment Spray System. The availability of the containment spray system depends on the core damage scenario. If core damage occurs when the refueling cavity is filled, there may not be sufficient water in the refueling water storage tank. If the core damage occurs due to a station blackout, there will be no power for the sprays. If the core damage is due to operator failure to respond to the initiating event, the system may not be started by the operators.

4.2 Source Terms

The characteristics of fission product release are called the source term. The factors of interest are the timing and duration of the release, the release fractions of the various isotope groups and the plume energy. The timing of the release is used for radioactive decay. The duration of release is used to account for continuous releases. The release fractions are for groupings of isotopes that have similar chemical characteristics. The energy of the release is used to determine the height to which the plume might rise. There are many possible source terms depending on the characteristics of the accident sequence and the status and performance of the containment systems. Source terms of similar characteristics, which are expected to result in similar offsite consequences, are grouped together into release categories. Thus, a large number of source terms can be reduced to a smaller and more manageable number of release categories. An example of this process is given in WASH-1400.[26] Some of the release categories generated in WASH-1400 were used for the offsite consequence calculations in the present study.

Four WASH-1400[26] release categories, namely PWR 2, PWR 4, PWR 5, and PWR 7 were used to bound potential fission product releases for accidents at shutdown for the range of containment conditions of interest. The time of the releases was assumed to be five days. The duration of the release was assumed to be ten hours (the maximum duration allowed by the consequence code) since the material was assumed to leak out of the open containment. The release fractions are given in Table 4.2.

These accident categories were chosen to provide bounds on the release fractions that might occur during a shutdown event. PWR 2 is associated with failure of core cooling systems and core melting concurrent with the failure of containment spray and heat removal systems and is representative of a large, unmitigated fission product release. It was used to provide an upper bound on the results. PWR 4 involves failure of the core cooling systems and the containment spray injection system after a loss-of-coolant accident, together with a failure of the containment system to properly isolate. PWR 5 is like PWR 4 except that the containment spray injection would operate. PWR 7 is a core meltdown due to a failure in the core cooling systems with the containment sprays operating. PWR 2B is a PWR 2 with sprays operating, the release fractions are divided by five except for the noble gases to account for decontamination of the aerosols by the sprays.

WASH-1400 source terms were used for convenience and to provide a point of reference with previous studies that were based on WASH-1400 methodology. The only changes made were to allows for the much later release times and lower plume energy expected in the present study compared with the WASH-1400 values. However, the WASH-1400 source terms were calculated for accidents from power operation and care must be taken when applying them to accidents at shutdown. In addition, there has been considerable source term research since the publication of WASH-1400 which can influence our prediction of fission product release during a severe accident.

PWR 2 represents a very severe release category and was calculated with a higher decay heat level than would be expected for accidents at shutdown. Thus, as the core is melting in the reactor pressure vessel, the driving force for the fission product release in PWR 2 is higher than would be expected for accidents at shutdown. Therefore, a source term calculation at the lower decay heat level might predict less fission product release during this stage of core meltdown. In addition, after the core debris melts and penetrates the reactor pressure vessel, WASH-1400 predicts that the core attacks the concrete. During this concrete attack a significant quantity of the more refractory fission products was predicted to be released in WASH-1400. However, for accidents at shutdown with the lower decay heat level the concrete attack may not be as extensive and therefore lower fission product release would be expected.

Based on the above arguments it would appear that PWR 2 is a rather conservative upper bound for accidents at shutdown and that a more mechanistic analysis would predict lower fission product release fractions. However, phenomena that were not modelled in WASH-1400 such as revaporization of previously retained fission products from the primary system have been shown to result in higher releases than predicted in WASH-1400 for some fission product groups under certain accident conditions. Therefore, unless a mechanistic calculation is performed with the new computer codes available (NUREG-0956) for the accident conditions of interest it is difficult to conclude that PWR 2 is overly conservative. In addition, a sensitivity study is provided in Section 4.5 which indicates that the person-rem calculations are not linearly related to the source term so that uncertainty in PWR 2 does not result in a corresponding uncertainty in person-rem. The sensitivity study also shows that a five day decay instead of a two day decay has actually no influence on the person-rem calculations.

4.3 Consequence Codes

Consequence calculations were performed using the newer MACCS[4] codes instead of the older CRAC2.[27-28] These consequence codes consider five processes that account for most of the ways in which people can accumulate a radiation dose after radioactivity has been released to the atmosphere from an accident:

a. Inhalation;
b. Cloudshine (external exposure from passing cloud);
c. Groundshine (external exposure from deposited material);
d. Ingestion; and
e. Inhalation of resuspended material.

The first three mechanisms are by far the most important in contributing to potential high-dose early effects. Lower doses leading to latent effects can come from any of the pathways, especially if interdiction does not preclude ingestion and cleanup does not reduce contamination.

The techniques of consequence analysis are discussed in the Reactor Safety Study (WASH-1400)[26] and the PRA Procedures Guide,[29] and, therefore, the details are omitted here.

4.4 Site Specific Data

Two specific sites were considered: Zion and a "generic" site using the average U.S. population density of about 100 people per square mile. The site specific data included the plant power level (Zion = 3250 MWT), the population distribution, land use data, and the site meteorology. The population data was provided by the NRC. The land use data was taken from the Statistical Abstracts of the United States and is listed in the CRAC2 User's Guide.[25] The generic site used the Zion power level and state averaged land use data for Illinois.

The weather data consists of hourly weather observations of wind speed, wind direction, stability class, and precipitation. The data is not taken from a single year, but is averaged in a manner that represents the long-term average weather behavior. This data is sorted into 29 weather categories (called bins), as discussed in the CRAC2 Model Manual,[28] so that low probability weather conditions can be adequately sampled.

The weather summaries for Zion are given in Table 4.1. The stability is ranked in six categories (A, B, C, D, E, F) ranging from the most dispersive to the least dispersive. Category A, with rapid dispersion, represents a sunny afternoon with low wind speeds. Category F, with little spread of the plume with distance, would occur late at night or just before dawn if wind speeds were very low. In addition, there are weather bins for rain conditions, both at time of release and at later times, and for changing wind condition which produces a slowing down of the plume. Both of these conditions could produce higher doses at greater distances than would otherwise occur.

4.5 Results of Consequence Calculations

The results for the Zion site and the generic site (population density of 100 people per square mile) are given in Table 4.3. The results are tabulated for person-rem within 50 miles and within 500 miles using the Zion site and MACCS. The total economic cost is also given. This includes the costs of emergency action, decontamination, interdiction and food disposal from both farmland and residential areas. It does not include medical care. The generic site results were calculated using the Zion site meteorology and for PWR 2 only.

The Zion site analysis using the MACCS code represents a complete set of results and these are the results that have been used in the risk calculations discussed in Section 4.6. The results calculated for the Zion site are roughly a factor of three higher than the rural generic site results although the relationship is not linear. This, however, does not indicate that the Zion site represents an average U.S. site. The Zion site population density is one of the highest in the nation. The factor of three between Zion and the generic site would be expected to envelope a vast majority of the U.S. sites. What is most important is the relative risk reduction attainable from the various potential improvements rather than any absolute magnitudes calculated for any given site.

The results of the sensitivity calculations are given in Table 4.4. Note that a reduction in the source term is not linearly related to a reduction in person-rem at either 50 or 500 miles. This is because of a land decontamination and interdiction criterion that people not be exposed to more than 25 rem in 30 years. Furthermore, people can live on land that is interdicted for crops. Of course during the emergency phase of an accident high doses are possible. This all means that a smaller source term can lead to higher doses over 30 years since land may not be interdicted and that total costs can vary widely because of the protective measures taken.

It can also be seen in Table 4.4 that there is not much difference in dose for a PWR2 type accident that occurs two days after shutdown instead of five days. This is because most of the short lived isotopes have decayed and because it is the long lived isotopes such as CS-137 that contribute most to the 30 year dose.

4.6 Risk Calculations

The consequences have been calculated using MACCS within 50 miles of the Zion site (as discussed above) and have been applied in Equation 4.2 to calculate the risk for the base case and the reductions in risk as a result of the proposed improvements. Table 4.5 summarizes the results for the BNL shutdown risk model representing a generic plant. This table includes the contributions from all three initiating event categories (i.e., loss of cooling, small LOCA, and LOOP). Table 4.6 summarizes the results for just the loss-ofcooling initiating events.

4.7 Risk-Related Results

The person-rem entries in Table 4.6 represent the overall quantification of the containment event tree (Figure 4.1) for all 27 possible combinations of the sensitivity variables. Line 28 of the table assumes that release category PWR7 would result given a core damage and the equipment hatch and containment penetrations are closed. The ranking of the effectiveness of each proposed improvement is the same as that shown for core damage. That is, the most effective improvement is associated with upgraded RHR instrumentation followed by upgraded vessel level indication and removal of ACI, respectively.

In order to present an integrated approach to the effective resolution of this generic issue, the following insights are provided from the sensitivity study to supplement the proposed hardware/procedural improvements discussed in Section 3.

The following comments are based upon the column of Table 4.6 labeled "Base Case." The remaining columns follow closely to the Base Case and therefore will not be discussed. From the first nine lines of the table, it can be seen that the containment penetrations have no effect on the dose if the equipment hatch is open. Also, comparing line one with line 19 shows a factor of 3.5 reduction in dose by having a fairly large probability that the equipment hatch will be closed. Coupling this with the insight from Table 3.7 that the most vulnerable period during shutdown is while the RCS is partially drained, yields the following conclusion. Consideration should be given to developing a Technical Specification requirement that would require the equipment hatch and any other large penetration to be closed during the drain-down conditions.

In terms of containment spray, comparing line one to line three yields a factor of two in dose reduction with the containment open. Comparing line 25 with line 27 yields essentially the same results with the containment closed. This suggests the possibility of a Technical Specification requirement requiring some level of containment spray availability be maintained throughout the entire shutdown period.

Looking at the effect of containment penetrations and comparing line 19 with line 25 yields almost a factor of 22 reduction in dose with the containment spray unavailable, and a factor of 14 reduction in dose (line 21/line 27) with the sprays operating. Table 3.7 shows that the phases in which the RCS is partially drained contribute 85% to the total core damage frequency due to loss of cooling events. This suggests that, in a prioritization scheme, a Technical Specification addressing closing the containment penetrations while in a partially drained condition is more effective than a requirement for containment spray availability. To further illustrate this point, comparing line one to line 25 yields a factor of 77 reduction in postulated dose by simply maintaining containment integrity during the entire outage and a factor of 65 (i.e., 77*.85) for maintaining containment integrity during partially drained conditions. Containment spray availability then contributes another factor of 2 as discussed above.

It is most important to note that the actual doses listed in Table 4.6 are dependent upon the site-specific modelling assumptions and would be changed if different assumptions (sites) were investigated, however, the relative relationship of the various factors calculated above would not be expected to significantly change.

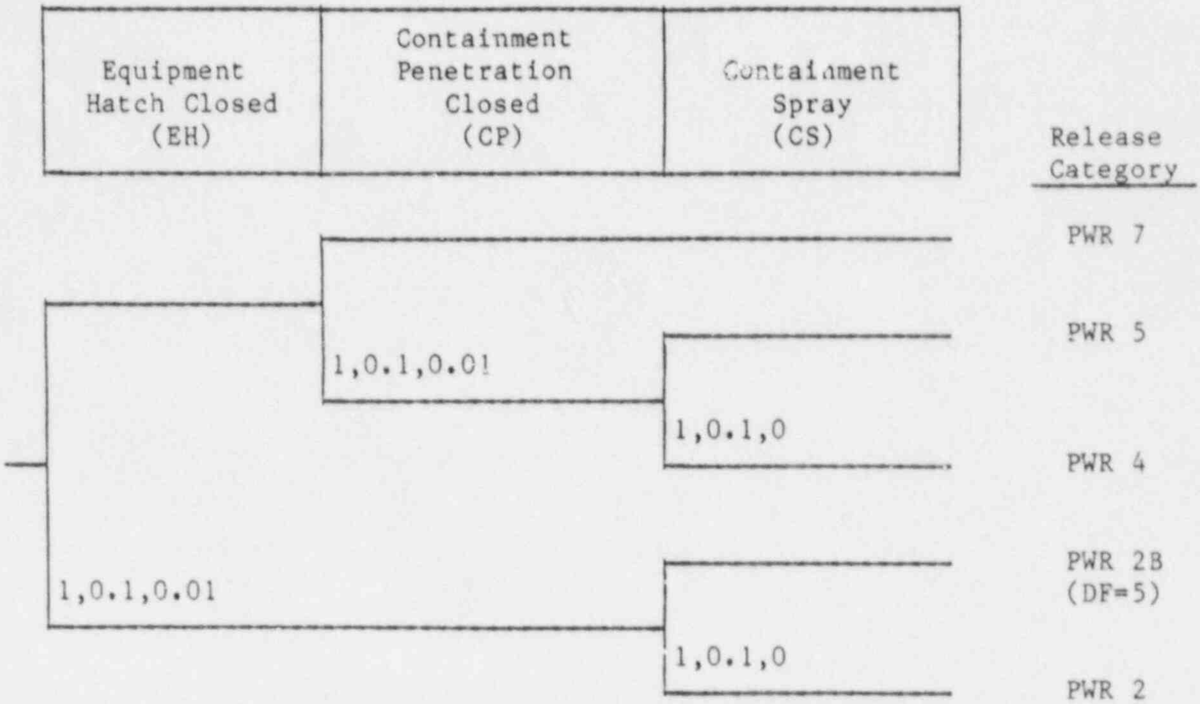


Figure 4.1 Containment event tree for a shutdown accident.

Table 4.1
One Year of Zion Meteorological Data
Summarized Using Weather Bin Categories

Weather Bin	Number of Sequences	Percent
1. R (0)	566	6.46
2. R (0-5)	58	0.66
3. R (5-10)	152	1.74
4. R (10-15)	134	1.53
5. R (15-20)	114	1.30
6. R (20-25)	103	1.18
7. R (25-30)	92	1.05
8. S (0-10)	27	0.31
9. S (10-15)	23	0.26
10. S (15-20)	12	0.14
11. S (20-25)	24	0.27
12. S (25-30)	25	0.29
13. A-C 1,2,3	1222	13.95
14. A-C 4,5	1990	22.72
15. D 1	25	0.29
16. D 2	290	3.31
17. D 3	739	8.44
18. D 4	582	6.64
19. D 5	207	2.36
20. E 1	50	0.57
21. E 2	462	5.27
22. E 3	559	6.38
23. E 4	266	3.04
24. E 5	24	0.27
25. F 1	67	0.76
26. F 2	493	5.63
27. F 3	364	4.16
28. F 4	90	1.03
29. F 5	0	0.00
	8760	100.0

R = Rain starting within indicated interval (miles).
S = Windspeed slowdown occurring within indicated interval (miles).
A-C, D,E,F = Stability categories.
1(0-1), 2(1-2), 3(2-3), 4(3-5), 5(GT 5) = Wind speed intervals (meters/second).

Table 4.2
Release Fractions

Release	Xe	I	Cs	Te	Sx	Ru	La
PWR 2	0.9	0.7	0.5	0.3	0.06	0.02	0.004
PWR 4	0.6	0.09	0.04	0.03	0.005	0.003	0.0004
PWR 5	0.3	0.03	0.009	0.005	0.001	0.0006	0.00007
PWR 7	6×10^{-3}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}

Table 4.3
Results of Consequence Calculations

Release Category	MACCS Person-Rem (50 Miles)	MACCS Person-Rem (500 Miles)	Total Economic Cost ($)
Zion			
PWR 2	2.37E7	1.03E8	8.12E9
PWR 4	6.71E6	1.54E7	6.62E8
PWR 5	1.99E6	3.95E6	2.05E8
PWR 7	4.53E3	6.99E3	3.00E4
PWR 2B	1.25E7	3.25E7	1.93E9
Generic Site			
PWR 2	4.38E6	5.6E7	3.15E9

Note: The above results represent the estimated dose that would occur given an event in that particular release category. The above results do not reflect the probability associated with each release category. Tables 4.5 and 4.6 take into account the release category probabilities.

Table 4.4
Sensitivity Calculations

Release	Person-Rem 50 Miles	Person-Rem 500 Miles
PWR2	2.37E7	1.03E8
PWR2/10	7.91E6	1.85E7
PWR2/100	1.23E6	2.43E6
PWR2-2 day decay	2.46E7	1.04E8

Table 4.5
Summary of Risk Calculations - Loss of Cooling + LOCA + LOOP

	Containment Event Tree Probabilities			Risk (Person-rem/year)					Risk Reduction (Person-rem/year)			
	EH	CP	CS	Base Case	I1	I2	I3	I1+I2	I1	I2	I3	I1+I2
1	1.00	1.00	1.00	1237.14	284.40	450.30	1156.56	279.66	952.74	786.84	80.58	957.48
2	1.00	1.00	0.10	710.96	163.44	258.78	664.66	160.72	547.52	452.18	46.31	550.25
3	1.00	1.00	0.00	652.50	150.00	237.50	610.00	147.50	502.50	415.00	42.50	505.00
4	1.00	0.10	1.00	1237.14	284.40	450.30	1156.56	279.66	952.74	786.84	80.58	957.48
5	1.00	0.10	0.10	710.96	163.44	258.78	664.66	160.72	547.52	452.18	46.31	550.25
6	1.00	0.10	0.00	652.50	150.00	237.50	610.00	147.50	502.50	415.00	42.50	505.00
7	1.00	0.01	1.00	1237.[illegible]	284.40	450.30	1156.56	279.66	952.74	786.84	80.58	957.48
8	1.00	0.01	0.10	710.[illegible]6	163.44	258.78	664.66	160.72	547.52	452.18	46.31	550.25
9	1.00	0.01	0.00	652.50	150.00	237.50	610.00	147.50	502.50	415.00	42.50	505.00
10	0.10	1.00	1.00	438.95	100.91	159.77	410.36	99.23	338.04	279.18	28.59	339.72
11	0.10	1.00	0.10	186.76	42.93	67.98	174.60	42.22	143.83	118.78	12.16	144.54
12	0.10	1.00	0.00	158.74	36.49	57.78	148.40	35.88	122.25	100.96	10.34	122.86
13	0.10	0.10	1.00	155.43	35.73	56.57	145.31	35.14	119.70	98.86	10.12	120.29
14	0.10	0.10	0.10	82.85	19.05	30.16	77.46	18.73	63.81	52.70	5.40	64.12
15	0.10	0.10	0.00	74.79	17.19	27.22	69.92	16.91	57.60	47.57	4.87	57.88
16	0.10	0.01	1.00	127.08	29.21	46.25	118.80	28.73	97.86	80.82	8.28	98.35
17	0.10	0.01	0.10	72.46	16.66	26.38	67.74	16.38	55.8[illegible]	46.09	4.72	56.08
18	0.10	0.01	0.00	66.40	15.26	24.17	62.07	15.01	51.13	42.23	4.32	51.39
19	0.01	1.00	1.00	359.13	82.56	130.72	335.74	81.18	276.57	228.41	23.39	277.95
20	0.01	1.00	0.10	134.34	30.88	48.90	125.59	30.37	103.46	85.44	8.75	103.97
21	0.01	1.00	0.00	109.36	25.14	39.81	102.24	24.72	84.22	69.56	7.12	84.64
22	0.01	0.10	1.00	47.26	10.86	17.20	44.18	10.68	36.39	30.06	3.08	36.58
23	0.01	0.10	0.10	20.04	4.61	7.30	18.74	4.53	15.44	12.75	1.31	15.51
24	0.01	0.10	0.00	17.02	3.91	6.19	15.91	3.85	13.11	10.82	1.11	13.17
25	0.01	0.01	1.00	16.07	3.69	5.85	15.02	3.63	12.38	10.22	1.05	12.44
26	0.01	0.01	0.10	8.61	1.98	3.14	8.05	1.95	6.63	5.48	0.56	6.67
27	0.01	0.01	0.00	7.79	1.79	2.83	7.28	1.76	6.00	4.95	0.51	6.03
28	0.00	0.00	0.00	0.24	0.05	0.09	0.22	0.05	0.18	0.15	0.02	0.18

I1 = Upgraded instrumentation for RHR pumps.
I2 = Upgraded vessel level indication.
I3 = Removal of autoclosure interlock.

Table 4.6
Summary of Risk Calculations - Loss of Cooling Events

	Containment Event Tree Probabilities			Risk (Person-rem/year)					Risk Reduction (Person-rem/year)			
	EH	CP	CS	Base Case	I1	I2	I3	I1+I2	I1	I2	I3	I1+I2
1	1.00	1.00	1.00	1014.36	101.20	227.28	933.78	96.46	913.16	787.08	80.58	917.90
2	1.00	1.00	0.10	582.94	58.16	130.62	536.63	55.43	524.78	452.32	46.31	527.50
3	1.00	1.00	0.00	535.00	53.38	119.88	492.50	50.88	481.63	415.13	42.50	484.13
4	1.00	0.10	1.00	1014.36	101.20	227.28	933.78	96.46	913.16	787.08	80.58	917.90
5	1.00	0.10	0.10	582.94	58.16	130.62	536.63	55.43	524.78	452.32	46.31	527.50
6	1.00	0.10	0.00	535.00	53.38	119.88	492.50	50.88	481.63	415.13	42.50	484.13
7	1.00	0.01	1.00	1014.36	101.20	227.28	933.78	96.46	913.16	787.08	80.58	917.90
8	1.00	0.01	0.10	582.94	58.16	130.62	536.63	55.43	524.78	452.32	46.31	527.50
9	1.00	0.01	0.00	535.00	53.38	119.88	492.50	50.88	481.63	415.13	42.50	484.13
10	0.10	1.00	1.00	359.91	35.91	80.64	331.31	34.22	324.00	279.26	28.59	325.68
11	0.10	1.00	0.10	153.13	15.28	34.31	140.97	14.56	137.85	118.82	12.16	138.57
12	0.10	1.00	0.00	130.15	12.99	29.16	119.82	12.38	117.17	100.99	10.34	117.78
13	0.10	0.10	1.00	127.44	12.71	28.55	117.32	12.12	114.73	98.89	10.12	115.32
14	0.10	0.10	0.10	67.93	6.78	15.22	62.54	6.46	61.16	52.71	5.40	61.47
15	0.10	0.10	0.00	61.32	6.12	13.74	56.45	5.83	55.20	47.58	4.87	55.49
16	0.10	0.01	1.00	104.19	10.40	23.35	95.92	9.91	93.80	80.85	8.28	94.29
17	0.10	0.01	0.10	59.41	5.93	13.31	54.69	5.65	53.49	46.10	4.72	53.76
18	0.10	0.01	0.00	54.44	5.43	12.20	50.11	5.18	49.01	42.24	4.32	49.26
19	0.01	1.00	1.00	294.46	29.38	65.98	271.07	28.00	265.08	228.48	23.39	266.46
20	0.01	1.00	0.10	110.15	10.99	24.68	101.40	10.47	99.16	85.47	8.75	99.67
21	0.01	1.00	0.00	89.67	8.95	20.09	82.55	8.53	80.72	69.58	7.12	81.14
22	0.01	0.10	1.00	38.75	3.87	8.68	35.67	3.68	34.88	30.07	3.08	35.06
23	0.01	0.10	0.10	16.43	1.64	3.68	15.13	1.56	14.79	12.75	1.31	14.87
24	0.01	0.10	0.00	13.95	1.39	3.13	12.85	1.33	12.56	10.83	1.11	12.63
25	0.01	0.01	1.00	13.18	1.31	2.95	12.13	1.25	11.86	10.22	1.05	11.92
26	0.01	0.01	0.10	7.06	0.70	1.58	6.50	0.67	6.36	5.48	0.56	6.39
27	0.01	0.01	0.00	6.38	0.64	1.43	5.88	0.61	5.75	4.95	0.51	5.78
28	0.00	0.00	0.00	0.19	0.02	0.04	0.18	0.02	0.17	0.15	0.02	0.18

I1 = Upgraded instrumentation for RHR pumps.
I2 = Upgraded vessel level indication.
I3 = Removal of autoclosure interlock.

5. SUMMARY

In this study, BNL has developed a generic PWR shutdown risk model derived from a Zion-specific study reported in NSAC-84. Information and data taken from NSAC-84 for the BNL model included: frequency and overall duration of 3 types of outages as well as component shutdown maintenance unavailabilities. Modifications applied to the NSAC-84 model included:redefinition of the phases of an outage, new estimates of the durations of phases (in particular, the duration that a plant stays in the partially drained condition), and the modelling of human cognitive errors. In the BNL analysis, generic shutdown data were collected and used to estimate the frequencies of initiating events. The overall results of the BNL analysis in terms of core damage frequency and loss-of-cooling initiating event frequency are given in Tables 3.18 and 3.19. Table 3.20 provides similar results for just the loss-of-cooling initiating events.

A containment event tree was developed to assess the status of containment integrity given that a core damage event occurs while in shutdown. Due to insufficient data on the top events in the event tree, sensitivity calculations were done for the quantification of the containment event tree. Four release categories from WASH-1400 were used to represent the source terms. The release categories were PWR 2, PWR 4, PWR 5, and PWR 7 and were chosen to provide bounds on the release fractions that might occur during a shutdown event. The consequences of the release categories in the containment event tree were assessed using the MACCS code. The risk and the reductions in risk achieved by the proposed improvements are shown in Table 4.5 for the overall results of this study and in Table 4.6 for just the loss-of-cooling initiating events.

5.1 Conclusions

The BNL estimate of overall core damage frequency resulting during shutdown from this study is 5.22×10^{-5} per year. This includes three types of initiating events: 1) loss-of-cooling, 2) LOCA, and 3) LOOP. Loss-of-cooling events were estimated to represent a core damage frequency of 4.28×10^{-5} per year. Core damage frequency during shutdown is typically not included in PRAs. Adding approximately 5×10^{-5} per year to a plant's overall core damage frequency would in most cases represent a non-trivial contribution. Therefore, BNL identified three potential design/procedural improvements that would serve to reduce the risk during reactor shutdown conditions. The benefits of the proposed design/procedural improvements can be measured in terms of the frequency of occurrence of loss of cooling events, the frequency of core damage, and the risk. Since the person-rem reduction is simply the reduction in core damage frequency times the consequence of a core damage, core damage frequency and loss-of-cooling frequency have been chosen for discussion.

1 - Upgraded instrumentation for RHR pumps with emergency procedures for shutdown conditions

The upgraded instrumentation proposed for the RHR pumps would provide an alarm to alert the operators that RHR capability has been affected and information that would allow the operators to easily identify the cause of the problem. The attendant proposed emergency procedures would help the operators to determine the corrective actions needed to restore the RHR capability. The benefit of this design improvement is that the operators would be able to respond to loss of RHR events in a timely fashion. The 90% reduction in core damage frequency calculated for this improvement was derived by assuming that given the improvement the operators will always be able to diagnose the situation and initiate proper corrective actions.

2 - Upgraded vessel level indication

Given reliable vessel level indication, the operators will be able to avoid overdraining in a draindown operation and will be able to maintain proper vessel level when the RCS is drained to the hotleg midplane. The benefit of this upgrade is a reduction of the initiating frequency of loss-of-cooling events and an attendant reduction in core damage frequency. A potential 22% reduction in the frequency of loss-of-cooling events was calculated by setting the probability of overdraining to zero. This upgrade results in an 78% reduction in calculated core damage frequency per year.

3 - Removal of auto-closure interlocks on the RHR suction valves

With this design change, the frequency of spurious closure of an RHR suction valve would be significantly reduced. This design change reduces the frequency of loss-of-cooling events and reduces the calculated core damage frequency. Due to the large number of already experienced spurious isolation events, this event is an important contributor to the estimated frequency of loss-of-cooling events. Therefore, the proposed design change results in a 60% reduction in the initiator frequency of loss-of-cooling. The reduction in calculated core damage frequency based upon implementation of this possible upgrade is 8%.

Consequence calculations were also carried out for the base case and the three proposed improvements. The results of these calculations are presented in the framework of sensitivity study found in Section 4. The sensitivity study addresses the possibilities of 1) having the equipment hatch open and not being able to close it, 2) having a containment penetration open and not being able to seal it, and 3) the potential for reducing the source terms given containment spray availability. Insights from the sensitivity study are discussed in Section 4.7 and are summarized in Section 5.3.

5.2 Discussions on Applicability of the Generic Analysis Results to a Specific Plant

The following items concerning the applicability of the results to a specific plant are noted:

1. Some newer plants have two RHR suction lines while Zion has only one. This difference is judged to have a small effect on the frequency of loss of RHR events. Because most of the time during an outage only one RHR train is operating, a spurious isolation signal to the suction valves of the operating train will lead to loss of cooling. The type of situation in which the second suction line helps most is when one suction valve has a mechanical failure and can

not be opened by any means. However, the probability of such failure may be small compared with that for other failure modes, e.g., loss of suction due to overdraining. This should be evaluated on a case-by-case basis.

2. A dominant cause of core damage during a shutdown is due to the failure of the operator to respond. Operator performance depends on the information available to him. The analysis in this report assumes that the instrumentation and annunciators available for the operator are those given in Table 2.2 of NSAC-84. Therefore, there is no alarm assumed to be in the generic plant control room for low RHR pump suction pressure. In the latter stages of this study BNL was informed that such an alarm is being installed at Zion and this should help the operators to respond to events such as loss of pump suction.

3. The BNL analysis used the Zion plant configuration but did not consider the loss-of-component-cooling-water event as an initiating event, because the dependence of safety systems on component cooling water and service water systems may vary from plant to plant. For Zion, loss of component cooling water may lead directly to core damage.[23] For Byron,[30] loss of service water may lead to core damage. For other plants, some safety systems may not depend on component cooling water, or service water. An NRC survey of PWRs found that approximately 16 plants are potentially vulnerable to such types of dependence on support systems. Such issues can only be addressed on a plant-specific basis.

4. The BNL analysis (as well as the NSAC-84 analysis) has assumed that the progression to cold shutdown conditions will proceed in an orderly and unhurried fashion. This is a major assumption of the study as it factors into the determination of how much time is avail- able for operator actions as a function of the decay heat rate of the core. The most significant benchmark of this time frame would be the modelled assumption that vessel level would not be drained to the hotleg midplane until 83 hours after reactor trip. Technical Specification allowed cooldown rates, if actually followed, could yield a drained condition in as short a period as one day. Reference 2 identified a scenario that may cause core uncovery to occur sooner, i.e., a loss of cooling event occurs when the reactor coolant system is drained to the midplane of the hot leg with the cold leg opened due to maintenance of a reactor coolant pump, and the system becomes pressurized and forces coolant to flow out of the cold leg opening. The risk associated with any such practice is not bounded by this analysis. Such actions could represent a significant increase in shutdown risk based upon a much more limited time available for operator recovery actions.

5.3 Insights With Respect to Technical Specifications

The following is a listing of the insights derived throughout this study concerning Technical Specifications. The purpose of this listing is to simply highlight the observations that have been made.

1. When the availability of the safety injection (SI) system during shutdown was included in the model, the calculated core damage frequency was reduced by about an order of magnitude. Current Technical Specifications (TS) require the disabling of SI as reactor pressure is reduced on the way to shutdown. Insights offered on this subject are the following: 1) disabling of the SI system should be done in a manner that would allow minimum effort for restoration and 2) simultaneous maintenance on all trains should be avoided.

2. NSAC-84 gives the maintenance unavailability of both charging pump trains as 6% (in Procedure Event Tree 3 of refueling outage). This represents a relatively high unavailability and consideration could be given to avoiding simultaneous maintenance in this system.

3. Based on items 1 and 2 above, both of which address systems with injection capability, alternative consideration might be given to a Technical Specification requiring at least one train from either system be kept in an operable condition during shutdown. It is recognized that the SI system would be in a bypassed condition and in that case, "operable" would mean free from maintenance activities.

4. In reviewing the Zion Technical Specifications, it was found that, during shutdown conditions, if offsite power is available, both diesels could be in simultaneous maintenance. BNL has been informed that diesel power from the second Zion unit could be transferred to the first if both diesel generators were in maintenance and a loss of offsite power occurred. Although this may not therefore be a concern for Zion, if there are other plants with similar latitude in their Technical Specifications with no similar local power source, these plants would exhibit a higher vulnerability to core damage during shutdown.

5. The following insights were derived from the sensitivity study on the containment event tree and are listed by priority:

   a. During partially drained conditions within the reactor coolant system (RCS), consideration could be given to a requirement that the equipment hatch either be in place or be in a position such that it could be closed quickly.

   b. Also during partially drained conditions, consideration could be given to assuring that the containment penetrations are either closed or in a state in which they could be resealed quickly.

   c. During shutdown conditions, consideration could be given to a requirement that a train of containment spray be kept available.

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APPENDIX A: Determination of Time to Core Uncovery Given a Station Blackout During an Outage

When a postulated station blackout occurs during an outage, the decay heat is not removed, and the RCS will heat up. Steam will begin to discharge through the openings of the RCS i.e. the relief valves and the PORVs. This loss of RCS inventory continues until decay heat removal capability is restored or the core becomes uncovered. A simple thermal model that determines the time to core uncovery given a station blackout event, is described in this appendix. The model estimates the energy that is needed to heatup the RCS inventory and boil-off the coolant above the top of the active core, and determines the time that is needed for the integrated decay heat to be equal to that amount. Figure A.1 shows the result of the model. It can be seen that as much as ten hours may be available before core uncovery occurs if station blackout occurs late in an outage.

A.1 Energy Needed to Result in Core Uncovery

The most vulnerable condition of a plant during an outage is when the RCS is drained to the mid-plane of the hot leg nozzles, and the RCS is open. This could occur when the steam generator manway is removed for steam generator maintenance. This is assumed to be the initial condition of the RCS for this analysis. Specifically, the following initial conditions are assumed:

- The RCS is drained to hot leg mid-plane.
- The average coolant temperature is 100°F.
- One train of the RHR system is operating to remove decay heat.
- The RCS hotleg is vented to the containment and is at atmospheric pressure.

In the very last stages of finalizing this report, it was brought to BNL's attention that a different set of initial conditions has been shown to provide less time to core uncovery than those above. The difference is that the cold leg (instead of the hot leg) is assumed to be vented. The claim is that with the cold leg vented the reactor heatup/pressurization will force inventory out the cold leg and this will expend the available inventory faster than the simple boil-off model described herein. This information came too late for consideration in this study, however, the simple solution is to also vent the hot leg thereby negating the pressurization scenario.

When a station blackout occurs, the RCS coolant will heat up to 212°F and boiling will start in the active core region. Steam will leave the RCS through openings in the RCS. The following water volumes in different regions of the RCS are estimated using the Zion FSAR and NSAC-84:

Volume (hot leg center line to core mid-plane)
 = 1790 ft^3
Volume (top of core to bottom of core)
 = Volume (Active Core) + Volume (Annulus)
 = 665 ft^3 + 449 ft^3
 = 1114 ft^3
Volume (below the core) = 1050 ft^3.

At 100°F, the specific volume of water is 0.0161 ft^3/lbm. The water mass in different regions is:

Mass (hot leg centerline to core mid-plane)
= 1790/0.016
= 1.11×10^5 lbm
Mass (top of core to bottom of core)
= 1114/0.0161
= 6.92×10^4 lbm
Mass (below the core)
= 1050/0.0161
= 6.52×10^4 lbm

It is assumed that core uncovery occurs when the water level drops to core mid-plane. Therefore, the water above the core mid-plane needs to be heated to 212°F and then converted to steam. It is also assumed that the rest of the water in the system, including water below the core, is at 212°F when core uncovery occurs.

The energy needed to heat up th water from 100°F to 212°F is

$$[h_f(212°F,\ 1\ atm) - h_f(100°F,\ 1\ atm)] * (1.11*10^5\ lbm + 6.92*10^4\ lbm/2 + 6.52*10^4\ lbm)$$
$$= (180.17-68.04) * 2.11*10^5$$
$$= 2.37*10^7\ BTU$$

The energy needed to boil the water from the hot leg mid-plane to the core mid-plane is

$$h_{fg} * 1.11*10^5\ lbm$$
$$= 970.3\ BTU/lbm * 1.11*10^5\ lbm$$
$$= 1.08*10^8\ BTU$$

The total energy that is needed to result in core uncovery

$$= 2.37*10^7 + 1.08*10^8$$
$$= 1.32*10^8\ BTU$$

The energy needed to heat up the reactor vessel internals is estimated to be only a few per cent of this energy and is conservatively ignored in the calculation.

A.2 Determination of Time to Core Uncovery Using the Decay Heat Curve

The following equation expresses the decay power as a function of the time, τ (sec.), after shutdown and the time, T_0 (sec.), that the plant had been operating before shutdown.

$$P(\tau) = P_0 * 0.1[(\tau-T_0 + 10)^{-0.2} - (\tau + 10)^{-0.2} + 0.87\ (\tau + 2*10^7)^{-0.2} - 0.87(\tau-T_0 + 2*10^7)^{-0.2}]$$

where P_0 is the power of the reactor, i.e., 3250 MWt for Zion. It is taken from Reference 31. The energy generated from time T_1 to T_2 is simply the integral of the equation from T_1 to T_2. If a station blackout occurs at T_1, the time, at which the energy generated from decay heat is equal to what is needed for core uncovery to occur, can thus be determined. The time to core uncovery curve in Figure A.1 is calculated assuming T_0 is one year.

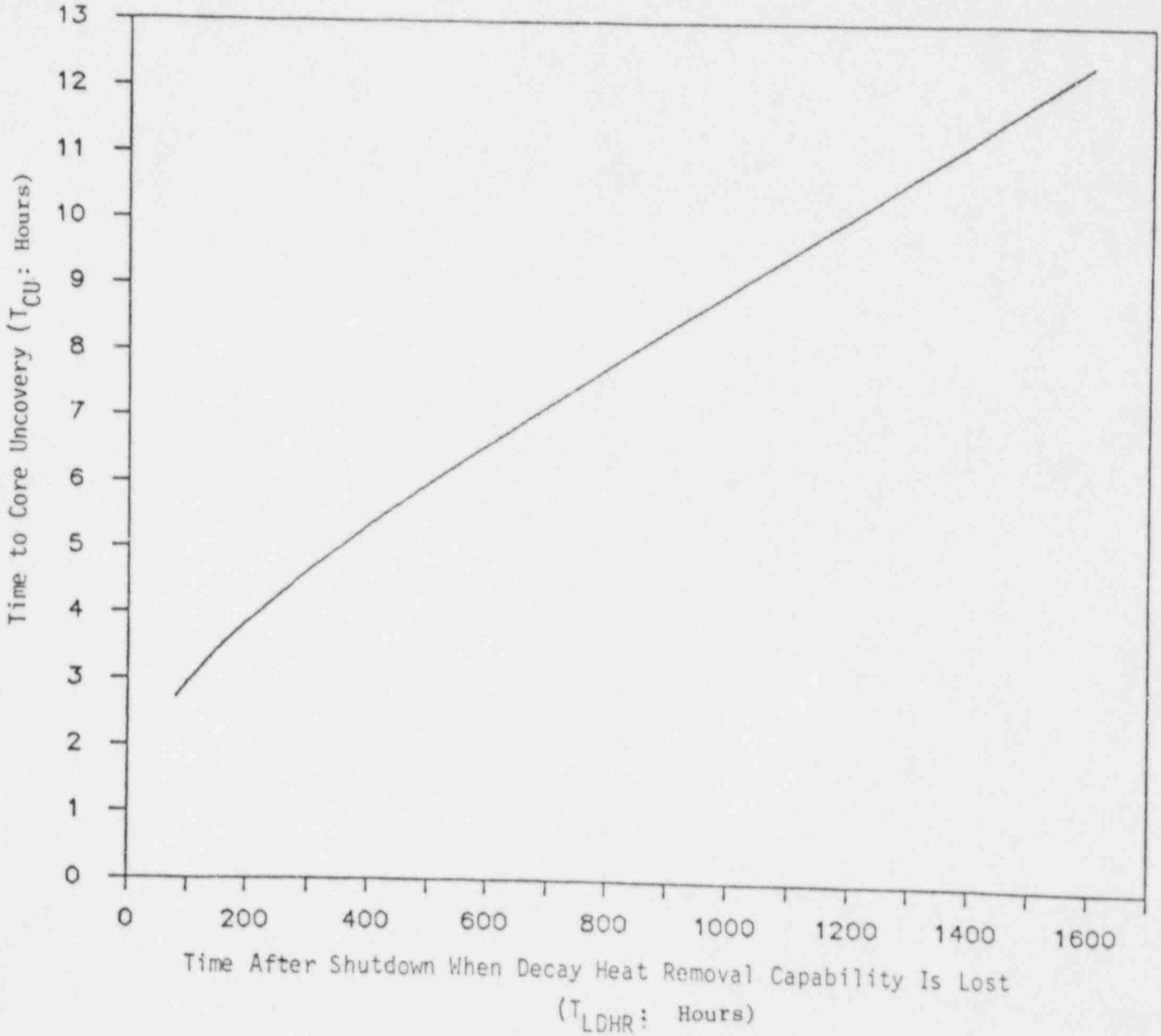

Figure A.1 Time to core uncovery due to loss of cooling.

APPENDIX B: Survey of Operational Experience of Loss of RHR at PWRs in 1984 to 1986

A. Spurious Isolation of RHR Suction Valves..........21 Events
B. Overdraining of RCS................................ 8 Events
C. Failure to Maintain RCS Level...................... 3 Events
D. Loss of RCS Coolant................................ 2 Events
E. Others...12 Events

A. Spurious Isolation of RHR Suction Valves

Table A.1 LER SCSS DATA 03-04-87

****

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
250	1985	036	0	8512100414	196717	10/25/85

****

DOCKET:250 TURKEY POINT 3 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: FLORIDA POWER & LIGHT CO.
SYMBOL: FPL

COMMENTS
STEP 1: MODEL BF22F.

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

REFERENCE LERS:
1 250/83-019 2 251/84-027

ABSTRACT
POWER LEVEL - 000%. ON OCTOBER 25, 1985, WHILE UNIT 3 WAS IN SHUTDOWN, THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM FLOW WAS INTERRUPTED FOR APPROXIMATELY 27 MINUTES DUE TO THE AUTOMATIC CLOSURE OF MOV-3-750. THIS VALVE IS LOCATED IN THE SINGLE RHR PUMP SUCTION LINE ORIGINATING FROM THE HOT LEG OF THE REACTOR COOLANT SYSTEM (RCS), AND IT IS DESIGNED TO CLOSE TO PROTECT THE RHR SYSTEM FROM OVER-PRESSURIZATION WHEN THE RCS PRESSURE EXCEEDS 465 PSIG. RHR WAS RE-ESTABLISHED APPROXIMATELY 27 MINUTES LATER BY OPENING THE VALVE AND REMOVING POWER TO THE VALVE'S MOTOR OPERATOR. DURING THE PERIOD IN WHICH THE VALVE REMAINED CLOSED, THE RCS TEMPERATURE ROSE 20F, I.E., FROM 110F TO 130F. MOV-3-750 WAS RETURNED TO SERVICE AND PERFORMED SATISFACTORILY AFTER REPLACING A MALFUNCTIONING RELAY. A FAILED RELAY, PC-403-A-2, IN THE PRESSURE COMPARATOR FOR THE PRESSURE CONTROLLER PC-403 CAUSED TWO BLOWN FUSES IN THE COMPARATOR, WHICH RESULTED IN AN ERRONEOUS HIGH PRESSURE SIGNAL CLOSING RHR VALVE MOV-3-750. IMMEDIATE CORRECTIVE ACTIONS TAKEN WERE AS FOLLOWS: 1) THE 3B RHR PUMP WAS STOPPED WHEN MOV-3-750 CLOSED. 2) MOV-3-750 WAS MANUALLY OPENED AND ITS POWER REMOVED BY RACKING OPEN ITS BREAKER. 3) RHR PUMP 3B WAS THEN RESTARTED. 4) FAILED RELAY PC-403-A-2 WAS REPLACED ALONG WITH TWO BLOWN FUSES AND MOV-3-750 WAS RESTORED TO SERVICE AFTER VERIFICATION OF OPERABILITY. SIMILAR OCCURRENCES: LERS 250 83-19 AND 251 84-27.

Table A.2 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
251	1984	027	0	8501100117	192575	11/30/84

****

DOCKET:251 TURKEY POINT 4 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: FLORIDA POWER & LIGHT CO.
SYMBOL: FPL

REPORTABILITY CODES FOR THIS LER ARE:
15 10 CFR 50.73(a)(2)(vii): Single failure criteria.

REFERENCE LERS:
1 250/83-019

ABSTRACT
POWER LEVEL - 000%. ON 11-30-84, WHILE UNIT 4 WAS AT REFUELING SHUTDOWN (RFSD) CONDITIONS, THE RHR SYSTEM FLOW WAS INTERRUPTED FOR APPROX 4 MINS. THE ROOT CAUSE STEMMED FROM THE CLOSURE OF MOV-4-751, ISOLATION VALVE IN THE RHR PUMP SUCTION LINE, CAUSED BY A MALFUNCTION IN PRESSURE CONTROLLER PC-405B, FAILING HIGH PRODUCING A FALSE INDICATION OF HIGH RCS PRESSURE THUS ACTIVATING THE PROTECTIVE INTERLOCK. THIS INTERLOCK PREVENTS RHR SYSTEM OVERPRESSURIZATION BY CLOSING MOV-4-751 UPON HIGH RCS PRESSURE. THE OPERATORS WERE ALERTED TO THE CONDITION BY THE OVERPRESSURE MITIGATING SYSTEM (OMS) HIGH PRESSURE ALERT ANNUNCIATOR. UPON ACTUATION OF THE OMS, CONTROLLING IN THE LOW PRESSURE SETTING (415 PSIG), POWER OPERATED RELIEF VALVES (PORVS), PCV-456 AND PCV-455C CYCLED OPEN TO RELIEVE RCS PRESSURE THUS PERFORMING THEIR INTENDED FUNCTION. CORRECTIVE ACTIONS INCLUDED: 1) THE B RHR PUMP WAS STOPPED, 2) THE OPERATING CHARGING PUMP WAS STOPPED AND PRESSURE WAS CONTROLLED BY PRESSURE CONTROL VALVE, PCV-145, 3) MOV-4-751 WAS SUCCESSFULLY OPENED BY BYPASSING THE PRESENT CLOSING SIGNAL AND RACKING OPEN ITS BREAKER, 4) I&C REPLACED PC-405B AND RELEASED IT TO OPERATIONS. THE RESPECTIVE BREAKER FOR MOV-4-751 WAS RACKED IN, THUS RETURNING THE RHR SYSTEM TO NORMAL OPERATION. DURING TRANSIENT RCS PRESSURE INCREASED FROM 350 PSIG TO 415 PSIG AND NO RCS HEATUP WAS OBSERVED. SIMILAR OCCURRENCES: LER 250-83-19.

Table A.3 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
251	1986	006	0	8604210370	198907	03/15/86

****

DOCKET:251 TURKEY POINT 4 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: FLORIDA POWER & LIGHT CO.
SYMBOL: FPL

REPORTABILITY CODES FOR THIS LER ARE:
13 10 CFR 50.73(a)(2)(iv): ESF actuations.
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.
15 10 CFR 50.73(a)(2)(vii): Single failure criteria.

REFERENCE LERS:
1 250/85-029 2 250/85-036 3 250/86-003 4 251/86-004

ABSTRACT

POWER LEVEL - 000%. ON 3-15-86, WHILE UNIT 4 WAS IN A SCHEDULED REFUELING OUTAGE MODE 6, WORK WAS PROGRESSING TO DE-ENERGIZE AND REPLACE A VITAL BUS FEEDER BREAKER 4P08. WHEN BREAKER 4P08-3 WAS OPENED, THE RHR PUMP SUCTION VALVE WENT CLOSED. UPON RECEIPT OF THE LETDOWN ISOLATION ALARM, THE RHR PUMP WAS STOPPED, THE BREAKER RE-ENERGIZED, THE VALVE REOPENED, THE PUMP RESTARTED AND FLOW RESTORED IN APPROXIMATELY 5 MINS. TECH SPEC ACTION STATEMENT 3.10.7.2 WAS ENTERED DURING THE APPROXIMATE 5 MINS OF FLOW LOSS. THERE WAS NO NOTICEABLE INCREASE IN THE 93 F SYSTEM TEMPERATURE. WHEN BREAKER 4P08-20 WAS OPENED, A PROCESS RADIATION MONITOR RACK WAS LOST, CAUSING THE CONTAINMENT VENTILATION SYSTEM TO ISOLATE AND THE CONTROL ROOM VENTILATION SYSTEM TO ISOLATE AND SWITCH OVER TO THE RECIRCULATION MODE, AS DESIGNED. THE PURGE VALVES WERE SECURED PER TECH SPEC ACTION STATEMENT 3.10.2.A, BY REMOVING POWER FUSES. NO SIGNIFICANT INCREASE IN ACTIVITY WAS RECORDED ON THE PLANT VENT EFFLUENT MONITORING SYSTEM DURING THIS EVENT; OTHER MEANS OF MONITORING CONTAINMENT ACTIVITY WERE AVAILABLE. NO RELEASE PATH TO OUTSIDE CONTAINMENT WAS AVAILABLE. WHEN DE-ENERGIZED, FEEDER BREAKER FP08 REPLACEMENT WAS COMPLETED, POWER WAS RESTORED AND THE SYSTEMS WERE THEN RETURNED TO THEIR NORMAL LINE-UP.

Table A.4 LER SCSS DATA 03-04-87

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
312	1986	030	0	8701130230	202380	12/08/86

DOCKET:312 RANCHO SECO TYPE:PWR
REGION: 5 NSSS:BW
ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: SACRAMENTO MUNICIPAL UTIL. DISTRICT
SYMBOL: SMU

COMMENTS
STEP 1: CAUSE XX - SWITCHING OF POWER SOURCES TO ALLOW MAINTENANCE ON STARTUP TRANSFORMER.

REPORTABILITY CODES FOR THIS LER ARE:
13 10 CFR 50.73(a)(2)(iv): ESF actuations.
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.
15 10 CFR 50.73(a)(2)(vii): Single failure criteria.

REFERENCE LERS:
1 312/82-015 2 312/85-016 3 312/86-016 4 312/86-024

ABSTRACT
POWER LEVEL - 000%. THE PLANT WAS IN COLD SHUTDOWN, REMOVING DECAY HEAT VIA THE DECAY HEAT REMOVAL SYSTEM (DHS) TRAIN "B" ON DECEMBER 8, 1986. STARTUP TRANSFORMER NO. 1 WAS SCHEDULED FOR ROUTINE PREVENTIVE MAINTENANCE. A LOSS OF THE 4A BUS POWER, ATTENDANT DIESEL GENERATOR START, AND DECAY HEAT SYSTEM (DHS) ISOLATION OCCURRED DURING THE TRANSFER OF THE SOURCE TRANSFORMER AT 2:18 PM ON DECEMBER 8, 1986. AN AUTOMATIC FEATURE OF THE NUCLEAR SERVICE BUS IS A FIVE-SECOND LIMIT ON HAVING TWO SOURCES FEEDING THE BUS. THE CONTROL ROOM OPERATOR CLOSED STARTUP TRANSFORMER NO. 2 SUPPLY BREAKER 4A10 ONTO THE 4A BUS. WHEN THE OPERATOR OPENED THE SUPPLY BREAKER (4A01) FROM STARTUP TRANSFORMER NO. 1, BREAKER 4A10 FROM STARTUP TRANSFORMER NO. 2 HAD JUST COMPLETED THE AUTOMATIC FIVE-SECOND RUN-OUT AND HAD TRIPPED OPEN. THESE EVENTS LEFT THE 4A BUS WITHOUT EITHER THE NORMAL OR ALTERNATE SUPPLY. AN ATTENDANT RESULT WAS THAT WHEN POWER WAS RESTORED, DHS SUCTION VALVE HV-20001 CLOSED AS WOULD BE EXPECTED IN THIS SITUATION CAUSING THE DHS ISOLATION. THE POWER SUPPLIES TO BOTH HV-20001 AND HV-20002 ARE CURRENTLY RACKED OUT. THE PURPOSE FOR THE DHS SYSTEM VALVE INTERLOCKS IS TO PREVENT OVER=PRESSURING THE DHS PIPING WITH RCS PRESSURE. SINCE THE RCS IS "OPEN TO ATMOSPHERE," THERE IS NO NEED FOR THE INTERLOCKS TO PROTECT THE DHS PIPING FROM OVER-PRESSURE.

Table A.5 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
275	1985	005	0	3502250761	193345	01/20/85

****

DOCKET:275 DIABLO CANYON 1 TYPE:PWR
REGION: 5 NSSS:WE
ARCHITECTURAL ENGINEER: PGEC
FACILITY OPERATOR: PACIFIC GAS & ELECTRIC CO.
SYMBOL: PGE

COMMENTS
STEP 3: CAUSE XX-REMOVED FROM SERVICE FOR SURVEILLANCE TEST.

REPORTABILITY CODES FOR THIS LER ARE:
15 10 CFR 50.73(a)(2)(vii): Single failure criteria.

REFERENCE LERS:
1 275/84-004 2 275/85-006

ABSTRACT
POWER LEVEL - 000%. AT 2303 PST, 1-20-85, WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN) BOTH RHR TRAINS BECAME INOPERABLE FOR APPROX 6 MINS. THIS EVENT WAS CAUSED BY A PLANT TECHNICIAN CHECKING THE WRONG BREAKER AND VERIFYING IT AS BEING OPEN. WHEN OVERPRESSURE PROTECTION CHANNEL PT-403 WAS REMOVED FROM SERVICE, AN INTERLOCK BETWEEN THE PROTECTION CHANNEL AND RHR PUMP INLET VALVE, MOV 8702, RESULTED IN VALVE CLOSURE AND BOTH RHR LOW FLOW ALARM. AT 2309, MOV 8702 WAS REOPENED. AT 2312 RHR PUMP 1-1 WAS RESTARTED AND RHR FLOW ESTABLISHED. ALL TECH SPEC ACTION STATEMENTS WERE MET. AN INCIDENT REVIEW BOARD MET AND MADE RECOMMENDATIONS TO REVISE SURVEILLANCE TEST PROCEDURES (STP'S) I-68A AND I-69A. THE PROCEDURES WILL INFORM THE TECHNICIAN THAT THE BREAKER MAY BE FOUND OPEN OR CLOSED AND, IF FOUND CLOSED, OPERATIONS DEPARTMENT SHOULD BE NOTIFIED TO OPEN IT. ALSO, THE EVENT WAS REVIEWED WITH ALL AFFECTED PERSONNEL STRESSING THE IMPORTANCE OF VERIFYING THE CORRECT BREAKER. SIMILAR EVENT 275/84-004.

Table A.6 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
275	1985	006	0	8503010423	193346	01/25/85

****

DOCKET:275 DIABLO CANYON 1 TYPE:PWR
REGION: 5 NSSS:WE
ARCHITECTURAL ENGINEER: PGEC
FACILITY OPERATOR: PACIFIC GAS & ELECTRIC CO.
SYMBOL: PGE

REPORTABILITY CODES FOR THIS LER ARE:
13 10 CFR 50.73(a)(2)(iv): ESF actuations.
15 10 CFR 50.73(a)(2)(vii): Single failure criteria.

REFERENCE LERS:
1 275/85-005 2 275/84-004

ABSTRACT

POWER LEVEL - 000%. AT 1750 PST, 1-25-85, WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN), A LOSS OF VITAL 4KV BUS VOLTAGE RESULTED IN THE AUTOSTARTS OF DG 1-2, CONTAINMENT FAN COOLER SYSTEM 1-5, AND AUX SALTWATER PUMP 1-2, AND THE TRANSFER OF THE CONTROL ROOM VENTILATION SYSTEM TO MODE 4. IN ADDITION, FOR APPROX 2 MINS, THE DECAY HEAT REMOVAL CAPABILITY WAS LOST WHEN THE CLOSURE OF THE LOOP 4 RHR SUCTION VALVE (MOV-8702) RESULTED IN BOTH RHR TRAINS BEING ISOLATED FROM THE RCS. THE RHR SUCTION VALVE WAS SUBSEQUENTLY OPENED AND RHR FLOW ESTABLISHED WITHIN 2 MINS. ALL OTHER AFFECTED EQUIPMENT AND SYSTEMS WERE RETURNED TO THEIR NORMAL STANDBY CONDITIONS. INVESTIGATION HAS SHOWN THAT THE CAUSE OF THIS EVENT WAS MISADJUSTMENT OF THE AUX SWITCHES ON THE BUS G FEEDER BREAKERS (HG 13 AND 14). THE AUX SWITCHES WERE ADJUSTED TO A NEW TOLERANCE AND THE BREAKERS WERE TESTED WITH SATISFACTORY RESULTS. TO PREVENT RECURRENCE, PROCEDURE E-51.2, "4.16KV CIRCUIT BREAKER PM (PREVENTATIVE MAINTENANCE)," IS BEING REVISED TO IDENTIFY THE SPECIFIC AUX SWITCH ADJUSTMENT REQUIRED FOR THE BUS FEEDER BREAKERS. SIMILAR EVENTS 275/85-004 AND 85-005.

Table A.7 LER SCSS DATA 03-04-87

****

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
275	1986	012	0	8610100683	201273	09/08/86

****

DOCKET:275 DIABLO CANYON 1 TYPE:PWR
REGION: 5 NSSS:WE
ARCHITECTURAL ENGINEER: PGEC
FACILITY OPERATOR: PACIFIC GAS & ELECTRIC CO.
SYMBOL: PGE

COMMENTS
OTHER REPORTABILITY - 10 CFR 50.72(B)(2)(III)(B). STEP 9: FAILURE TO REPORT SIGNIFICANT EVENT WITHIN 10 CFR 50.72 4-HOUR TIME LIMIT.

WATCH-LIST CODES FOR THIS LER ARE:
941 REPORT ASSOCIATED WITH 10 CFR 50.72

REPORTABILITY CODES FOR THIS LER ARE:
10 10 CFR 50.73(a)(2)(i): Shutdowns or technical specification violations.
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.
21 OTHER: Voluntary report, special report, Part 21 report, etc.

REFERENCE LERS:
1 323/86-002

ABSTRACT
POWER LEVEL - 000%. AT 2314 PCT ON SEPTEMBER 8, 1986, WITH THE UNIT IN MODE 5 (COLD SHUTDOWN), AN INSTRUMENTATION AND CONTROLS (I&C) TECHNICIAN INADVERTENTLY GROUNDED A POWER SUPPLY WHILE INSTALLING A MODIFICATION IN A SOLID STATE PROTECTION SYSTEMS (SSPS) CABINET. THE MOMENTARY GROUNDING OF THE POWER SUPPLY CAUSED RELAY ACTUATION WHICH RESULTED IN THE CLOSURE OF RESIDUAL HEAT REMOVAL (RHR) VALVE 8702 AND AN RHR LOW FLOW ALARM. IN RESPONSE TO THE RHR LOW FLOW ALARM, THE OPERATING RHR PUMP WAS SECURED BY A LICENSED OPERATOR. RHR VALVE 8702 WAS REOPENED FROM THE CONTROL ROOM. THE RHR PUMP WAS RESTARTED AT 2316 PDT, AND NO SEAL DAMAGE WAS OBSERVED. A SIGNIFICANT EVENT REPORT WAS NOT FILED WITHIN THE 4-HOUR TIME REQUIREMENT OF 10 CFR 50.72. THE SIGNIFICANT EVENT REPORT WAS MADE AT 1744 PDT, SEPTEMBER 9, 1986. THE EVENT WAS REVIEWED AT AN I&C TAILBOARD MEETING EMPHASIZING ATTENTION TO ENERGIZED AND POTENTIALLY ENERGIZED CIRCUITS WHEN WORKING ON ELECTRICAL COMPONENTS. THE CIRCUMSTANCES AND LESSONS LEARNED FROM THE EVENT WILL BE EVALUATED FOR POSSIBLE INCLUSION IN THE GENERIC NEW EMPLOYEE TRAINING PROGRAM FOR I&C PERSONNEL. ADDITIONAL TRAINING ON 10 CFR 50.72 REPORTING REQUIREMENTS WILL BE PROVIDED FOR ALL APPLICABLE PERSONNEL.

Table A.8 LER SCSS DATA 03-04-87

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
304	1986	001	0	8602060332	198307	01/03/86

DOCKET:304 ZION 2 TYPE:PWR
REGION: 3 NSSS:WE
ARCHITECTURAL ENGINEER: SLXX
FACILITY OPERATOR: COMMONWEALTH EDISON CO.
SYMBOL: CWE

COMMENTS
OTHER REPORTABILITY- TECH. SPEC. 6.6.3.H. STEP 1: EFF IX- POWER FLUCTUATION.

REPORTABILITY CODES FOR THIS LER ARE:
21 OTHER: Voluntary report, special report, Part 21 report, etc.

ABSTRACT
POWER LEVEL - 000%. AT 1547 ON 1-3-86 UNIT 2 WAS SHUT DOWN FOR A REFUELING OUTAGE AND THE RCS WAS FILLED SOLID WITH NO BUBBLE IN THE PRESSURIZER. A MOMENTARY FLUCTUATION OF OUTPUT OF INVERTER POWER SUPPLY BUS 213 (CAUSE UNKNOWN) CAUSED THE CHARGING FLOW CONTROL VALVE, 2VC-FCV121, TO FAIL TO THE 20% DEMAND POSITION, AND ALSO CAUSED 2MOV-RH8701 THE RHR PUMP SUCTION ISOLATION VALVE TO FAIL CLOSED. THIS INCREASED CHARGING FLOW FROM 39 TO 190 GPM, AND ISOLATED LETDOWN FLOW RESULTING IN LIFTING OF THE PRESSURIZER POWER OPERATED RELIEF VALVES (PORV'S). WHILE INVESTIGATING THE CAUSE, BUS 213 WAS AGAIN DEENERGIZED AND THE PORV'S AGAIN LIFTED. THE CAUSE OF THE BUS OUTPUT FLUCTUATION IS CURRENTLY UNKNOWN. THIS EVENT IS REPORTABLE SINCE TECH SPEC 6.6.3.H REQUIRES A 30 DAY WRITTEN REPORT ON ACTUATION OF THE OVERPRESSURE PROTECTION SYSTEM.

Table A.9 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
311	1984	002	0	8403130384	189358	02/09/84

****

DOCKET:311 SALEM 2 TYPE:PWR
REGION: 1 NSSS:WE
ARCHITECTURAL ENGINEER: PSEG
FACILITY OPERATOR: PUBLIC SERVICE ELECTRIC & GAS CO.
SYMBOL: PEG

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

ABSTRACT
POWER LEVEL - 000%. ON FEBRUARY 9, 1984, DURING A MAINTENANCE SHUTDOWN, RESIDUAL HEAT REMOVAL COMMON SUCTION VALVE (2RH1) INADVERTENTLY SHUT WHILE TESTING WAS BEING PERFORMED ON THE PRESSURIZER OVERPRESSURE PROTECTION SYSTEM. THIS RESULTED IN A LOSS OF RHR FLOW THROUGH THE REACTOR COOLANT SYSTEM. THE BREAKERS FOR THE RHR COMMON SUCTION VALVES WERE NOT TAGGED, AS REQUIRED, PRIOR TO POPS TESTING. THE CONTROLS FOR THESE VALVES, LOCATED ON THE CONTROL ROOM CONSOLE, CONTAINED RED BEZEL COVERS WHICH INDICATED THAT THE VALVES ALREADY CONTAINED SHIFT SUPERVISOR TAGS FOR A PREVIOUS JOB. SINCE THE TAGS WERE NOT REQUIRED FOR PERSONNEL SAFETY, POPS TESTING WAS AUTHORIZED WITH THE USE OF THE EXISTING TAGS. UNKNOWN TO THE SHIFT SUPERVISOR, THESE TAGS HAD BEEN TEMPORARILY RELEASED, AND THE RED BEZEL COVERS HAD NOT BEEN REMOVED. TECH SPEC ALLOW RHR TO BE REMOVED FROM SERVICE FOR UP TO TWO HOURS, PROVIDED THERE ARE NO OPERATIONS WHICH WOULD RESULT IN A REDUCTION OF REACTOR COOLANT SYSTEM BORON CONCENTRATION. RHR FLOW WAS REESTABLISHED WITHIN SEVENTEEN MINUTES. A SYSTEM WILL BE ESTABLISHED FOR UPDATING THE STATUS OF THE CONTROL ROOM CONSOLE BEZEL COVERS, WHENEVER TAGGING RELEASES OR REQUESTS ARE INITIATED. THIS EVENT IS REPORTABLE IN ACCORDANCE WITH 10CFR50.73(A) (2) (V). AS A RESULT OF THIS OCCURRENCE, A SYSTEM WILL BE ESTABLISHED FOR UPDATING STATUS OF CONTROL ROOM CONSOLE BEZEL COVERS, WHENEVER TAGGING RELEASES OR REQUESTS ARE INITIATED.

Table A.10 LER SCSS DATA 03-04-87

****

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
312	1985	016	1	8611170020	201865	08/08/85

****

DOCKET:312 RANCHO SECO TYPE:PWR
REGION: 5 NSSS:BW
ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: SACRAMENTO MUNICIPAL UTIL. DISTRICT
SYMBOL: SMU

COMMENTS
EVENTS OCCURRED ON 8/8/85, 8/14/85, 12/29/85, 12/30/85, AND 12/31/85.

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

ABSTRACT
POWER LEVEL - 000%. ON AUGUST 8 AND AUGUST 14, 1985, WHILE IN COLD SHUTDOWN, THE DECAY HEAT REMOVAL SYSTEM (DHS) SUCTION BLOCK VALVE (HV-20002) AUTOMATICALLY CLOSED ON A HIGH REACTOR COOLANT SYSTEM (RCS) PRESSURE SIGNAL, THUS RESULTING IN A TEMPORARY LOSS OF THE DHS SYSTEM CAPABILITY. IN BOTH CASES, DHS FLOW WAS RE-ESTABLISHED IN ELEVEN MINUTES OR LESS, AND NO NOTICEABLE INCREASES IN THE INCORE TEMPERATURES WERE DETECTED. HV-20002 IS DESIGNED TO CLOSE AUTOMATICALLY WHEN THE RCS PRESSURE EXCEEDS 255 PSIG. THE RCS PRESSURE RECORDED BY OPERATIONS PERSONNEL AT THE TIME OF THE EVENTS WAS APPROXIMATELY 230 PSIG. ALTHOUGH NO DEFINITE REASON FOR THE VALVE CLOSURES WAS DETERMINED, AN INVESTIGATION OF THE EVENTS INDICATED THAT VOLTAGE SPIKES ON PRESSURE TRANSMITTER PT-21099 CIRCUITRY CAUSED THE BLOCK VALVE TO CLOSE. PT-21099 WAS REPLACED AND CALIBRATED DURING THE CYCLE 7 REFUELING OUTAGE AND A SUCCESSFUL MAINTENANCE TEST WAS PERFORMED FOLLOWING THE EVENTS TO ENSURE THE PROPER OPERABILITY OF THE DECAY HEAT VALVE INTERLOCK AND ASSOCIATED INSTRUMENTATION. THE SPURIOUS DECAY HEAT ISOLATION SIGNAL WAS TRACED TO IMPROPERLY ROUTING SHIELDED INSTRUMENT CABLE (1R1S04B6A) THROUGH CHANNEL B POWER TRAYS AND CONDUIT TO A PENETRATION, AS DOCUMENTED IN NCR S-5263, REVISION 3.

Table A.11 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
312	1986	016	0	8611130138	201861	10/03/86

****

DOCKET:312 RANCHO SECO TYPE:PWR
REGION: 5 NSSS:BW
ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: SACRAMENTO MUNICIPAL UTIL. DISTRICT
SYMBOL: SMU

COMMENTS
STEPS 5,7: ISYS SF-DHS PUMP ROOM SUMP. STEP 4: CAUSE AX-SNUBBER SEAL REPLACEMENT. STEP 8: EFF IX-VOLTAGE TRANSIENT.

WATCH-LIST CODES FOR THIS LER ARE:
913 UPDATE NEEDED

REPORTABILITY CODES FOR THIS LER ARE:
13 10 CFR 50.73(a)(2)(iv): ESF actuations.
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

REFERENCE LERS:
1 312/78-001

ABSTRACT
POWER LEVEL - 000%. WHILE IN COLD SHUTDOWN ON OCTOBER 3, 1986, DURING INSTRUMENT & CONTROL INVESTIGATION OF ABNORMAL INDICATION ON PANEL H2SFB FOR DECAY HEAT SYSTEM (DHS) "B" ROOM SUMP STACK LIGHTS, SFAS "B" BISTABLES TRIPPED CAUSING HV -20002 TO CLOSE, WHICH TRIPPED DHS "B" PUMP. THE PLANT WAS WITHOUT THE USE OF THE NORMAL DHS FOR APPROXIMATELY 13 MINUTES. DUE TO THE EXTENDED PERIOD THAT THE PLANT HAS BEEN SHUT DOWN, THERE WAS A SMALL, BUT DETECTABLE INCREASE OF REACTOR COOLANT TEMPERATURE. STEPS WERE TAKEN IMMEDIATELY TO RESTORE A DHS TRAIN TO SERVICE IN ACCORDANCE WITH THE INTENT OF TECH SPEC 3.1.1.5. THIS EVENT IS REPORTABLE ACCORDING TO 10 CFR PART 50.73(A)(2)(IV & V). THE CAUSE OF THE INCIDENTS WAS I&C TECHNICIANS TROUBLESHOOTING ABNORMAL INDICATION ON PANEL H2SFB FOR DHS "B" PUMP ROOM (EAST) SUMP STACK LIGHTS (18 INCH LEVEL INDICATION) ON PANEL H2SFB. THE IMMEDIATE CAUSE OF THE SPURIOUS ACTUATION WAS AN ELECTRIC ARC FROM THE SUMP LEVEL STACK LIGHT WHEN "ROLLING-OVER" THE RESPECTIVE BULB. THE ARC INITIATED THE TRIP OF INVERTER "B". AS A LONG TERM CORRECTIVE ACTION, THE DC VITAL POWER SUPPLIES WILL BE MODIFIED TO BE EQUIPPED WITH STATIC TRANSFER SWITCHES.

Table A.12 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
312	1986	024	0	8612240033	202204	11/15/86

****

DOCKET:312 RANCHO SECO TYPE:PWR
REGION: 5 NSSS:BW
ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: SACRAMENTO MUNICIPAL UTIL. DISTRICT
SYMBOL: SMU

COMMENTS
STEP 5: MODEL SPDC 611-250-60.

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.
15 10 CFR 50.73(a)(2)(vii): Single failure criteria.

REFERENCE LERS:
1 312/78-001 2 312/82-015 3 312/86-016

ABSTRACT
POWER LEVEL - 000%. THE PLANT WAS IN COLD SHUTDOWN, REMOVING DECAY HEAT VIA THE DECAY HEAT REMOVAL SYSTEM (DHS) TRAIN "A" ON 11-15-86. AT 1:00 PM, IN PREPARATION FOR A FUSE REPLACEMENT ACTIVITY IN THE S1A BUS INVERTER, S1A BUS POWER WAS MOMENTARILY INTERRUPTED, DHS OVER-PRESSURE BISTABLES TRIPPED, HV-20001 CLOSED WHICH TRIPPED DHS "A" PUMP AS DESIGNED. STEPS WERE TAKEN IMMEDIATELY TO RESTORE A DHS TRAIN TO SERVICE IN ACCORDANCE WITH TECH SPEC 3.1.1.5. THE BASIC CAUSE OF THE INVERTER FAILURE IS THAT THE ORIGINAL DESIGN DID NOT ALLOW FOR TESTABILITY OF THE DEVICE THROUGH THE USE OF A SUBSTITUTE POWER SOURCE. THAT DESIGN DEFICIENCY, IDENTIFIED AS EARLY AS 1979 (NCR'S-1258, REVISION 2), WAS RECOGNIZED BY THE CURRENT ACTION PLAN FOR PERFORMANCE IMPROVEMENT. THERE IS A PREVENTIVE MAINTENANCE PROCEDURE, EM 171A, "STATION INVERTER ROUTINE - STATIC PRODUCTS INVERTERS," THAT IS SCHEDULED TO BE PERFORMED ONCE PER YEAR ON THE INVERTERS, A 120VAC VITAL BUS SYSTEM IMPROVEMENT WAS APPROVED, BUT HAS NOT BEEN IMPLEMENTED YET.

Table A.13 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
317	1986	002	1	8606170212	199689	03/22/86

****

DOCKET:317 CA[illegible]IFFS 1 TYPE:PWR
[illegible] 1 NSSS:CE
ARCHITECTURAL ENG[illegible]: BECH
FACILITY OP[illegible]R: BALTIMORE CAS & ELECTRIC CO.
SYMBOL: BGE

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

REFERENCE LERS:
1 317/83-061

ABSTRACT
POWER LEVEL - 000%. ON 3-22-86 AT 1944, WHILE UNIT 1 WAS IN MODE 5 WITH REACTOR COOLANT SYSTEM PRESSURE AT 210 PSIA AND TEMPERATURE AT 149 DEGREES FAHRENHEIT, SHUTDOWN COOLING SYSTEM FLOW WAS MOMENTARILY INTERRUPTED. TECHNICIANS WERE PERFORMING A TEST UNDER MAINTENANCE ORDER (MO) #206-084-679A TO LIFT A RECENTLY INSTALLED ELECTROMATIC RELIEF VALVE (ERV 404). AS REQUIRED BY THE MO, THE TECHNICIANS INSERTED A GREATER THAN 300 PSIA SIGNAL TO PRESSURE INDICATOR/CONTROLLER (PIC 103-1). THE SHUTDOWN COOLING RETURN MOTOR OPERATED VALVE (MOV 652), WHICH IS IN THE SUCTION LINE OF THE LOW PRESSURE SAFETY INJECTION (LPSI) PUMP #12, WENT FROM OPEN TO CLOSED. LPSI PUMP #12 WAS SECURED AS SOON AS MOV 652 CLOSED. THE TECHNICIANS IMMEDIATELY REDUCED THE INPUT SIGNAL TO LESS THAN 300 PSIA, AND MOV 652 WAS REOPENED. LPSI PUMP #12 WAS RE-STARTED AT 1945 ON 3-22-86, 1 MINUTE AFTER INITIATION OF THE EVENT, AND SHUTDOWN COOLING FLOW WAS REESTABLISHED. IT WAS DETERMINED THAT THE PROCEDURE USED TO TEST ERV 404 DID NOT PROVIDE INSTRUCTIONS FOR ENSURING THAT MOV 652 WOULD NOT BE CLOSED BY THE TEST INPUT SIGNAL. TO PREVENT A RECURRENCE OF THIS EVENT, A REVIEW WILL BE MADE OF APPLICABLE PROCEDURES, AND CHANGES INITIATED AS APPROPRIATE TO ENSURE THAT WHEN AN ERV IS LIFT TESTED, AND SHUTDOWN COOLING IS OPERATIONAL, MOV 652 WILL REMAIN OPEN.

Table A.14 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
323	1986	002	0	8602250154	199498	01/17/86

****

DOCKET:323 DIABLO CANYON 2 TYPE:PWR
REGION: 5 NSSS:WE
ARCHITECTURAL ENGINEER: PGEC
FACILITY OPERATOR: PACIFIC GAS & ELECTRIC CO.
SYMBOL: PGE

COMMENTS
STEP 1: CAUSE XX - NORMAL LINE-UP FOR PLANT CONDITIONS. STEP 3: EFF IH - MOMENTARY POWER LOSS. STEP 4: COMP RLY - UNKNOWN TYPE.

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

REFERENCE LERS:
1 275/85-005 2 275/85-020

ABSTRACT
POWER LEVEL - 000%. AT 0455 PST ON 1-17-86, WHILE ATTEMPTING TO TRANSFER INSTRUMENT AC PANEL PY 2-1A FROM NORMAL TO BACKUP POWER SUPPLY, AN UNLICENSED OPERATOR WENT TO THE WRONG PANEL AND INADVERTENTLY TRANSFERRED INSTRUMENT AC PANEL PY 2-1 TO ITS BACKUP POWER SOURCE. THIS MOMENTARY LOSS OF POWER CAUSED RELAY ACTUATION WHICH RESULTED IN THE CLOSURE OF RESIDUAL HEAT REMOVAL (RHR) VALVE 8702. IN RESPONSE TO THE ENSUING LOSS OF FLOW ALARM, RHR PUMP 2-1 WAS SECURED BY A LICENSED OPERATOR. RHR VALVE 8702 WAS REOPENED FROM THE CONTROL ROOM. RHR PUMP 2-1 WAS RESTARTED, OBSERVED FOR SEAL DAMAGE, AND DECLARED OPERABLE AT 0508 PST, 1-17-86. NO OPERATIONS WERE IN PROGRESS THAT INVOLVED A REDUCTION IN REACTOR COOLANT SYSTEM BORON CONCENTRATION. THUS, THE REQUIREMENTS OF TECH SPEC 3.4.1.4.1 ACTION B WERE MET. TO PREVENT RECURRENCE, THE OPERATOR INVOLVED HAS BEEN COUNSELED, OPERATING PROCEDURES ON TRANSFERRING INSTRUMENT AC PANEL POWER SUPPLIES WILL BE REVISED, AND PANEL IDENTIFICATION LABELS IN THE INSTRUMENT AC PANELS WILL BE UPGRADED.

Table A.15 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
327	1985	020	0	8506240254	194616	05/14/85

****

DOCKET:327 SEQUOYAH 1 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: TVAX
FACILITY OPERATOR: TENNESSEE VALLEY AUTHORITY
SYMBOL: TVA

REPORTABILITY CODES FOR THIS LER ARE:
10 10 CFR 50.73(a)(2)(i): Shutdowns or technical specification violations.
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.
15 10 CFR 50.73(a)(2)(vii): Single failure criteria.

ABSTRACT
POWER LEVEL - 000%. ON 5-14-85 WITH UNIT 1 IN MODE 5 AT 144 F BOTH TRAINS OF RHR WERE INADVERTENTLY ISOLATED BY CLOSURE OF THE TRAIN B SUCTION VALVE. THE SUCTION WAS REESTABLISHED WITHIN 16 MINS AND THERE WAS NO INDICATED CHANGE IN RCS TEMPERATURE. THE ISOLATION OCCURRED WHILE WORK WAS BEING PERFORMED ON THE REACTOR VESSEL LEVEL INSTRUMENTATION SYSTEM (RVLIS) TO REFILL SENSE LINES. RCS WIDE RANGE PRESSURE TRANSMITTER 1-PT-68-66, WHICH IS USED FOR RHR OVERPRESSURE PROTECTION, RECEIVES ITS PROCESS SIGNAL FROM THE RVLIS SENSE LINES AND WAS INCREASED TO APPROX 2000 PSI DURING TESTING (RHR ISOLATION IS AT 700 PSI INCREASING).

Table A.16 LER SCSS DATA 03-04-87

****

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
348	1985	008	0	8506170539	194771	05/06/85

****

DOCKET:348 FARLEY 1 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: BESS
FACILITY OPERATOR: ALABAMA POWER CO.
SYMBOL: APC

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.
15 10 CFR 50.73(a)(2)(vii): Single failure criteria.

ABSTRACT
POWER LEVEL - 000%. AT 0925 ON 5/6/85, BOTH TRAINS OF THE RESIDUAL HEAT REMOVAL SYSTEM (RHR) AND THE OVERPRESSURE MITIGATION SYSTEM (OMS) WERE MADE INOPERABLE BY A COMMON CAUSE. AT 0920 ON 5/6/85, THE SUCTION VALVE FOR THE "A" TRAIN RHR SYSTEM CLOSED. ATTEMPTS TO OPEN THE VALVE FROM THE MAIN CONTROL BOARD WERE UNSUCCESSFUL AND THE OPERATORS STOPPED THE "A" TRAIN RHR PUMP. SIMILARLY, THE SUCTION VALVE FOR THE "B" TRAIN RHR SYSTEM CLOSED. ATTEMPTS TO OPEN THIS VALVE FROM THE MAIN CONTROL BOARD WERE UNSUCCESSFUL AND THE OPERATORS STOPPED THE "B" TRAIN RHR PUMP AT 0925. CLOSING OF THESE VALVES ALSO ISOLATED THE OMS RELIEF VALVES. POWER WAS REMOVED FROM THE TWO VALVES AND THEY WERE MANUALLY OPENED ALLOWING THE "A" TRAIN RHR PUMP TO BE RE-STARTED AT 1012 ON 5/6/85 AND THE "B" TRAIN RHR PUMP TO BE RESTARTED AT 1020. THIS RESTORED BOTH TRAINS OF RHR AND OMS TO OPERABILITY. THIS EVENT WAS CAUSED BY PROCEDURAL INADEQUACY AND PERSONNEL ERROR. POWER WHICH HAD BEEN PROCEDURALLY REMOVED FROM THE VALVES WAS INCORRECTLY RESTORED WHILE AN AUTO CLOSE SIGNAL FROM THE RCS PRESSURE TRANSMITTERS WAS PRESENT.

Table A.17 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
370	1984	002	0	8403120173	189406	01/15/84

****

DOCKET:370 MCGUIRE 2 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: DUKE
FACILITY OPERATOR: DUKE POWER CO.
SYMBOL: DPC

COMMENTS
STEP 3: COMPONENT MEI - TAG. STEP 4: CAUSE AX - TO PERFORM TEST. STEP 6: CAUSE XX - REQUIRED FOR OPERATIONS.

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

REFERENCE LERS:
1 369/81-072 2 370/84-004

ABSTRACT

POWER LEVEL - 000%. DURING FILLING AND VENTING OPERATIONS FOR UNIT 2 STARTUP, OPERATORS CLOSED THE BREAKERS FOR VALVES 2ND-1B AND 2ND-2A ('C' REACTOR COOLANT (NC) LOOP TO RESIDUAL HEAT REMOVAL (ND) PUMPS' ISOLATION VALVES) ON JANUARY 15, 1984. FUSES FOR THE A AND B TRAIN OUTPUT RELAY CABINETS OF THE SOLID STATE PROTECTION SYSTEM (SSPS) HAD BEEN REMOVED ON JANUARY 9 TO PERMIT TRANSMITTER TIME RESPONSE TESTING. NORMALLY CLOSED CONTACTS IN THE CLOSE CIRCUITS OF THE VALVES ARE CONTROLLED BY SSPS OUTPUT RELAYS. WITH SSPS OUTPUTS DE-ENERGIZED, THE CONTACTS COMPLETED THE CIRCUITS, PROVIDING CLOSE SIGNALS FOR 2ND-1B AND 2A. THUS, WHEN THE BREAKERS FOR 2ND-1B AND 2ND-2A WERE CLOSED, THE VALVES IMMEDIATELY CLOSED, ISOLATING ND SUCTION. BOTH ND TRAINS WERE DECLARED INOPERABLE AT 2207, PURSUANT TO TECH SPEC 3.4.1.4.2. UNIT 2 WAS IN MODE 5 WITH THE REACTOR COOLANT LOOPS NOT FILLED AT THE TIME OF THE INCIDENT. OPERATORS RESPONDED BY TRIPPING ND PUMP A AND CHEMICAL AND VOLUME CONTROL (NV) PUMP A AND REOPENING THE BREAKERS FOR 2ND-1B AND 2A. THE VALVES WERE THEN MANUALLY OPENED AND ND PUMP A WAS RESTARTED. THIS INCIDENT IS ATTRIBUTED TO PERSONNEL ERROR. APPROPRIATE MEASURES TO ENSURE CONTROL OVER 2ND-1B AND 2A WERE NOT TAKEN ON JANUARY 9, 1984, WHEN THE SSPS OUTPUT RELAY CABINETS WERE DE-ENERGIZED. PROCEDURES WERE REVISED, AND APPROPRIATE PERSONNEL WILL BE COUNSELED.

Table A.18 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
395	1984	044	0	8411130369	192069	10/02/84

****

DOCKET:395 SUMMER 1 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: GLBT
FACILITY OPERATOR: SOUTH CAROLINA ELECTRIC & GAS CO.
SYMBOL: SCC

COMMENTS
STEP 9: COMPONENT MEI - FUSE HOLDER.

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

ABSTRACT
POWER LEVEL - 000%. ON OCTOBER 2, 1984, THE PLANT WAS IN MODE 5 WITH TRAIN "B" OF THE RESIDUAL CONTROL (I&C) TECHNICIAN REMOVED TWO (2) FUSES IN SOLID STATE PROTECTION OF A MODIFICATION. THE FUSES WERE IMMEDIATELY REPLACED WHEN THE TECHNICIAN HEARD A RELAY ACTIVATE. THE DE-ENERGIZED CIRCUIT CAUSED THE TRAIN "A" RHR SUCTION ISOLATION VALVES XVG-8702 A & B (ONE VALVE IN EACH RHR TRAIN) TO CLOSE. OPERATIONS PERSONNEL IMMEDIATELY RESTORED TRAIN "B" RHR TO SERVICE AFTER THE VALVE CLOSURE. THE CAUSE WAS DETERMINED TO BE DRAWING ERRORS. AT 1700 HOURS DURING PERFORMANCE OF THE SAME MODIFICATION ON SSPS CABINET XPN-7010, A SIMILAR RHR ISOLATION OCCURRED VIA THE TRAIN "B" RHR SUCTION ISOLATION VALVES XVG-8701 A & B (ONE VALVE IN EACH RHR TRAIN). THE I & C TECHNICIAN WAS LIFTING LEADS AFFECTED BY THE MODIFICATION TO PREVENT A REPEAT OF THE PREVIOUSLY MENTIONED ISOLATION WHEN A DEFECTIVE FUSE HOLDER INTERRUPTED POWER TO THE TRAIN "B" CIRCUITRY. OPERATIONS PERSONNEL IMMEDIATELY RESTORED TRAIN "B" RHR TO SERVICE AFTER THE VALVE CLOSURE TO PREVENT A POTENTIAL RECURRENCE, THE LICENSEE INITIATED A DRAWING REVISION AND REPLACED THE DEFECTIVE FUSE HOLDER ON OCTOBER 9 AND OCTOBER 10, 1984, RESPECTIVELY.

Table A.19 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
395	1984	045	0	8411260552	192266	10/18/84

****

DOCKET:395 SUMMER 1 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: GLBT
FACILITY OPERATOR: SOUTH CAROLINA ELECTRIC & GAS CO.
SYMBOL: SCC

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

ABSTRACT
POWER LEVEL - 000%. ON 10-18-84, THE PLANT WAS IN MODE 5 FOR THE FIRST REFUELING OUTAGE WITH TRAIN 'A' OF THE RHR SYSTEM IN SERVICE, RHR TRAIN 'B' OUT-OF-SERVICE FOR ROUTINE MAINTENANCE, AND THE RCS VENTED AT A TEMPERATURE OF APPROX 110 F. AT 1605 HRS A POWER LOSS TO 120V AC DISTRIBUTION PANEL APN-5901 DE-ENERGIZED SOLID STATE PROTECTION SYSTEM (SSPS) CHANNEL I AND CAUSED THE INSTRUMENT CHANNEL FOR RCS WIDE RANGE PRESSURE (PT-403) TO INITIATE AN AUTO-CLOSURE OF THE OPERABLE RHR TRAIN'S SUCTION ISOLATION VALVE XVG-8701A. FOLLOWING DETERMINATION THAT THE POWER LOSS HAD BEEN CAUSED BY PERSONNEL ERROR DURING THE PERFORMANCE OF A PLANT MODIFICATION, OPERATIONS PERSONNEL RESTORED POWER TO APN-5901. XVG-8701A WAS OPENED AND TRAIN 'A' OF THE RHR SYSTEM RETURNED TO OPERABLE STATUS AT 1630 HRS (TOTAL TIME OF RHR ISOLATION WAS APPROX 25 MINS). RCS TEMPERATURE INCREASED FROM 110 F TO 130 F DURING THE EVENT. THE LOSS OF RHR MET THE CONDITIONS OF AN ALERT, AND THE PROPER NOTIFICATIONS WERE MADE IN ACCORDANCE WITH THE EMERGENCY PLAN.

Table A.20 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
395	1985	014	0	8506170551	194851	05/06/85

****

DOCKET:395 SUMMER 1 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: GLBT
FACILITY OPERATOR: SOUTH CAROLINA ELECTRIC & GAS CO.
SYMBOL: SCC

COMMENTS
OTHER REPORTABILITY - SPECIAL REPORT

REPORTABILITY CODES FOR THIS LER ARE:
13 10 CFR 50.73(a)(2)(iv): ESF actuations.
21 OTHER: Voluntary report, special report, Part 21 report, etc.

ABSTRACT

POWER LEVEL - 000%. ON MAY 5, 1985 AT APPROXIMATELY 2200 HOURS, A REACTOR COOLANT SYSTEM (RCS) PRESSURE TRANSIENT RESULTED IN A CHALLENGE OF A RESIDUAL HEAT REMOVAL (RHR) SUCTION RELIEF VALVE. THE PLANT WAS IN COLD SHUTDOWN (MODE 5) WITH RHR SYSTEM (TRAIN "A") IN OPERATION. DIESEL GENERATOR (D/G) SURVEILLANCE TESTING WAS IN PROGRESS AND HAD RESULTED IN A NON-VALID TEST FAILURE DURING AN ATTEMPT TO PARALLEL THE D/G TO THE ESF BUS (XSW-1DB). THE FAILURE TO PARALLEL THE D/G WAS A RESULT OF FAILURE OF THE SPEED CONTROL SWITCH ON THE MAIN CONTROL BOARD. DURING TROUBLESHOOTING ACTIVITIES ON THE D/G, A PERSONNEL ERROR RESULTED IN A LOSS OF ESF BUS (XSW-1DB). MAJOR EQUIPMENT AFFECTED INCLUDED THE LOSS OF THE "B" COMPONENT COOLING WATER (CCW) PUMP, "B" SERVICE WATER (SW) PUMP, AND "B" HVAC CHILLER AND CHILL WATER PUMP. THE LOSS OF CCW FLOW TO THE REACTOR COOLANT PUMP (RCP) REQUIRED THE SHUTDOWN OF THE OPERATING RCP. THE BREAKER WAS RECLOSED TO ESF BUS (XSW-1DB) AND THE BUS WAS RELOADED. UPON RESTART OF THE RCP WITH SOLID PLANT OPERATION, PRESSURE SPIKES OCCURRED WHICH RESULTED IN THE CHALLENGE TO THE TRAIN "A" RHR SUCTION RELIEF VALVE. FOLLOWING THE RELIEF VALVE ACTUATION, AN OPERATOR NOTED THAT PRESSURIZER RELIEF TANK (PRT) LEVEL CONTINUED TO INCREASE APPARENTLY DUE TO A FAILURE OF THE RELIEF VALVE TO RESEAT. APPROXIMATELY SIXTEEN HUNDRED (1600) GALLONS OF RCS INVENTORY WERE RELEASED TO THE PRT.

Table A.21 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
413	1986	044	1	8610090057	201245	08/15/86

****

DOCKET:413 CATAWBA 1 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: DUKE
FACILITY OPERATOR: DUKE POWER CO.
SYMBOL: DPC

COMMENTS
OTHER REPORTABILITY - 10 CFR 50.72(B)(2)(III). STEP 2: COMPONENT RLY - RELAY, UNKNOWN TYPE.

WATCH-LIST CODES FOR THIS LER ARE:
941 REPORT ASSOCIATED WITH 10 CFR 50.72

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.
21 OTHER: Voluntary report, special report, Part 21 report, etc.

REFERENCE LERS:
1 413/84-012 2 413/85-028

ABSTRACT
POWER LEVEL - 000%. ON AUGUST 15, 1986, TECHNICIANS WERE REPLACING A RELAY IN THE TRAIN A SOLID STATE PROTECTION SYSTEM (SSPS) CABINET WHEN A LUG ON THE RELAY SHORTED TO CABINET GROUND AND CAUSED THE OUTPUT RELAY FUSE IN THE SSPS CABINET TO BLOW. WHEN THE FUSE BLEW, POWER WAS LOST TO THE RELAYS THAT CONTROL THE POSITION OF THE A AND B TRAIN SUCTION VALVES FOR THE RESIDUAL HEAT REMOVAL (ND) PUMPS. SUBSEQUENTLY, THESE RELAYS CHANGED STATE AND THE VALVES CLOSED. THE ND SYSTEM WAS INOPERABLE FOR APPROXIMATELY 15 MINUTES BEFORE A NEW FUSE WAS INSTALLED IN THE SSPS CABINET. THE UNIT WAS IN MODE 5, COLD SHUTDOWN, AT THE TIME OF THIS INCIDENT. THIS INCIDENT IS ASSIGNED CAUSE CODE A, PERSONNEL ERROR. WHILE INSERTING A RELAY MOUNTING SCREW, THE TECHNICIAN'S HAND SLIPPED, CAUSING A SHORT AND BLOWING A FUSE IN THE 120 VAC POWER SUPPLY OF THE SSPS OUTPUT BAY. THIS INCIDENT IS REPORTABLE PURSUANT TO 10 CFR 50.73, SECTION (A)(2)(V)(B) AND 10 CFR 50.72, SECTION (B)(2)(III).

B. <u>Overdraining of RCS</u>

Table B.1 LER SCSS DATA 03-04-87

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
244	1984	003	0	8404240283	189208	03/07/84

DOCKET:244 GINNA TYPE:PWR
REGION: 1 NSSS:WE
ARCHITECTURAL ENGINEER: GLBT
FACILITY OPERATOR: ROCHESTER GAS & ELECTRIC CORP.
SYMBOL: RGE

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

ABSTRACT
POWER LEVEL - 000%. ON MARCH 7, 1984, WHILE THE REACTOR WAS IN THE COLD SHUTDOWN CONDITION, THE DRAINDOWN OF THE REACTOR COOLANT SYSTEM (RCS) WAS IN PROGRESS IN PREPARATION FOR THE STEAM GENERATORS' (S/G) ANNUAL INSPECTION. IN THE PROCESS OF DRAINING THE RCS TO THE CVCS HOLDUP TANKS, WHILE PREPARING TO SHIFT FROM DRAINING VIA THE REACTOR COOLANT DRAIN TANK (RCDT) PUMP TO THE LOW PRESSURE PURIFICATION PUMP, VALVES MOV-851A AND B (CONTAINMENT SUMP B SUCTION TO RHR) WERE MISTAKENLY OPENED PRIOR TO SHUTTING VALVE MOV-850A (DOWNSTREAM OF MOV-851A AND UPSTREAM OF RCDT PUMP SUCTION). THIS RESULTED IN WATER BEING DRAINED FROM THE RCS LOOP TO THE SUMP B, WITH POTENTIAL LOSS OF RHR CAPABILITY. A REVIEW WAS MADE OF OPERATING PROCEDURE O-2.3.1 "DRAINING THE REACTOR COOLANT SYSTEM" AND OF O-2.2 "PLANT SHUTDOWN FROM HOT SHUTDOWN TO COLD SHUTDOWN CONDITION" TO SEE IF CLARIFICATIONS WERE REQUIRED. A MINOR CHANGE WAS MADE TO PROCEDURE O-2.2 TO CLARIFY ONE STEP ASSOCIATED WITH MOV-851A AND B. OPERATIONS PERSONNEL HAVE BEEN CAUTIONED ON STRICT ADHERENCE TO OPERATING PROCEDURES.

Table B.2 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
295	1984	031	1	8411130498	197354	09/14/84

****

DOCKET:295 ZION 1 TYPE:PWR
REGION: 3 NSSS:WE
ARCHITECTURAL ENGINEER: SLXX
FACILITY OPERATOR: COMMONWEALTH EDISON CO.
SYMBOL: CWE

COMMENTS
STEP 5: CAUSE XX - PURGE OPERATION.

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

ABSTRACT
POWER LEVEL - 000%. WHILE IN COLD SHUTDOWN, DRAINING THE RCS IN PREPARATION FOR SG PRIMARY-SECONDARY LEAK TESTING THE RCS LEVEL DROPPED BELOW THE SUCTION LINE FOR THE RHR PUMP AS A RESULT OF AN IMPROPER VALVE LINEUP WHICH GAVE FALSE INDICATION OF THE RCS LEVEL. THE RHR PUMP WAS STOPPED WHEN IT WAS NOTICED THAT THE MOTOR AMPERAGE WAS FLUCTUATING. THE VALVE LINEUP WAS CHECKED AND THE LINEUP ERROR CORRECTED. RCS LEVEL WAS INCREASED TO NORMAL AND THE RHR PUMP WAS RESTARTED. RCS TEMPERATURE INCREASED FROM 110 DEGREES F TO 147 DEGREES F DURING THE 45 MINS THE PUMP WAS OFF. NO ABNORMAL CONDITIONS DEVELOPED AS A RESULT OF THIS EVENT. STATION PROCEDURES WILL BE REVISED TO PROHIBIT SIMULTANEOUS DRAINING AND PURGING OPERATIONS, A PROCEDURE FOR LOSS OF RHR WILL BE PREPARED. RETRAINING WILL BE CONDUCTED IN PROPER VALVE LINEUP PROCEDURES.

Table B.3 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
339	1984	008	1	8501210143	196425	10/16/84

****

DOCKET:339 NORTH ANNA 2 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: SWXX
FACILITY OPERATOR: VIRGINIA ELECTRIC & POWER CO.
SYMBOL: VEP

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

ABSTRACT
POWER LEVEL - 000%. ON 10-16-84, WITH NORTH ANNA UNIT 2 IN MODE 5 A COMPLETE LOSS OF RHR CAPABILITY OCCURRED WHEN BOTH RHR PUMPS WERE UNABLE TO OPERATE DUE TO THE INTRODUCTION OF AIR INTO THE RHR SYSTEM. THE INCIDENT OCCURRED DURING THE DRAIN DOWN OF THE RCS, WHEN THE LEVEL OF THE RCS WAS BEING MONITORED VIA A STANDPIPE OFF THE CENTERLINE OF ONE OF THE RCS LOOPS. THE ISOLATION VALVE TO WHICH THE STANDPIPE WAS ATTACHED BECAME CLOGGED SOMETIME DURING THE DRAIN DOWN AND FALSELY INDICATED 64 INCHES ABOVE CENTERLINE WHEN IN FACT THE LEVEL WAS BELOW THE RHR SUCTION LINE (BELOW CENTERLINE). SUBSEQUENTLY, LETDOWN FROM THE RCS WAS ISOLATED AND MAKEUP INITIATED. RHR CAPABILITY WAS REGAINED 2 HRS AFTER INITIATION OF THE EVENT. RCS LEVEL INDICATION WAS MOVED TO AN ALTERNATE TAP OFF LOOP CENTERLINE AND INDICATED SATISFACTORILY.

Table B.4 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
361	1986	007	0	8605050244	198985	03/26/86

****

DOCKET:361 SAN ONOFRE 2 TYPE:PWR
REGION: 5 NSSS:CE
ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: SOUTHERN CALIFORNIA EDISON CO.
SYMBOL: SCE

REPORTABILITY CODES FOR THIS LER ARE:
9 10 CFR 50.36(c)(2): Limiting conditions for operation.
10 10 CFR 50.73(a)(2)(i): Shutdowns or technical specification violations.
11 10 CFR 50.73(a)(2)(ii): Unanalyzed conditions.
13 10 CFR 50.73(a)(2)(iv): ESF actuations.
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.
15 10 CFR 50.73(a)(2)(vii): Single failure criteria.

REFERENCE LERS:
1 361/82-002

ABSTRACT
POWER LEVEL - 000%. MARCH 26, 1986 AT 2208 WITH UNIT 2 IN COLD SHUTDOWN, THE SHUTDOWN COOLING SYSTEM (SDCS) EXPERIENCED A TOTAL LOSS OF FLOW FOR A PERIOD OF FORTY-NINE MINUTES. THIS OCCURRED WHILE REACTOR COOLANT SYSTEM (RCS) LEVEL WAS BEING REDUCED TO REPAIR A LEAKING COLD LEG STEAM GENERATOR NOZZLE DAM WHICH HAD BEEN INSTALLED TO ALLOW WORK IN STEAM GENERATOR CHANNEL HEADS. USING THE ESTABLISHED LEVEL INDICATION, WHICH WAS LATER FOUND TO BE IN ERROR, THE RCS WAS DRAINED TO A LEVEL WHERE VORTEXING OCCURRED AT THE RCS/SDCS SUCTION CONNECTION CAUSING THE SDCS/LPSI PUMPS TO EVENTUALLY BECOME AIRBOUND. THE PUMPS WERE STOPPED AND THE SYSTEM VENTED, REESTABLISHING SDCS FLOW AT 2257. CONCURRENT WITH THE RESTORATION OF SDCS FLOW, BOTH GAS CHANNELS OF THE FUEL HANDLING ISOLATION SYSTEM ACTUATED ON HIGH NOBLE GAS AS A RESULT OF THE RCS DEGASING. THE HIGH PRESSURE SAFETY INJECTION SYSTEM WAS USED TO MAKE-UP TO THE RCS UNTIL SDCS FLOW RETURNED TO A STABLE STATE. THE CAUSE OF THE EVENT WAS ERRONEOUS LEVEL INDICATION RESULTING IN THE OPERATORS NOT RECOGNIZING THE RCS LOW LEVEL CONDITION PRIOR TO COMPLETE LOSS OF SDCS FLOW. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO PREVENT SDCS/LPSI PUMP DAMAGE, RESTORE SDCS FLOW TO A STABLE STATE AND RECALIBRATE THE LEVEL INDICATORS. CHANGES IN PLANT DESIGN, PROCEDURAL REVISIONS, FORMAL CONTROL OF LEVEL INDICATOR INSTALLATION, AND OPERATOR TRAINING WILL BE UNDERTAKEN.

Table B.5 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
368	1984	023	0	8410170302	191777	08/29/84

****

DOCKET:368 ARKANSAS NUCLEAR 2 TYPE:PWR
REGION: 4 NSSS:CE
ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: ARKANSAS POWER AND LIGHT CO.
SYMBOL: APL

COMMENTS
STEP 3: LEVEL INDICATOR IS A TEMPORARY INSTRUMENT USED DURING RCS DRAINING.

REPORTABILITY CODES FOR THIS LER ARE:
10 10 CFR 50.73(a)(2)(i): Shutdowns or technical specification violations.

REFERENCE LERS:
1 368/84-024

ABSTRACT

POWER LEVEL - 000%. ON 8-29-84, THE PLANT WAS IN MODE 5 AND THE RCS LEVEL WAS BEING MONITORED BY A TEMPORARY LEVEL INDICATOR CONNECTED TO THE BOTTOM OF THE RCS HOT LEG AND VENTED TO ATMOSPHERE. A NITROGEN PURGE OF THE RCS WAS IN PROGRESS TO "SWEEP" HYDROGEN FROM THE SYSTEM PRIOR TO MAINTENANCE. THE RCS WAS BEING VENTED VIA THE UPPER VESSEL HEAD VENT AND DUE TO NITROGEN FLOW EXCEEDING VENT FLOW CAPACITY THE RCS BECAME SLIGHTLY PRESSURIZED. THIS RESULTED IN A MANOMETER EFFECT AND INACCURATE INDICATION OF RCS LEVEL. THE LEVEL INDICATION INACCURACY LED TO DRAINING OF THE WATER IN THE RCS HOT LEG BELOW THE MINIMUM LEVEL FOR ADEQUATE SHUTDOWN COOLING PUMP SUCTION. SDC LOOP FLOW INDICATION BEGAN OSCILLATING BETWEEN 2000 AND 4000 GPM INDICATING CAVITATION OF THE SDC PUMP. CONSEQUENTLY THE "B" SDC PUMP AND NITROGEN PURGE WERE SECURED. DECAY HEAT REMOVAL ALIGNMENT WAS SHIFTED TO THE "A" SDC LOOP AND NORMAL FLOW OF APPROX 3000 GPM WAS ESTABLISHED. DURING THE PERIOD SDC FLOW WAS OFF, RCS BULK AVERAGE TEMPERATURE INCREASED FROM APPROX 140 TO 205 DEGREES F RESULTING IN A CHANGE FROM MODE 5 TO MODE 4. TO PREVENT RECURRENCE THE TEMPORARY LEVEL SYSTEM REFERENCE LEG HAS BEEN CHANGED FROM VENTING TO ATMOSPHERE TO VENTING TO THE PRESSURIZER STEAM SPACE. CHANGES HAVE BEEN MADE TO NORMAL AND ABNORMAL OPERATING PROCEDURES TO IMPROVE SYSTEM AND OPERATOR RESPONSE TO SIMILAR EVENTS.

Table B.6 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
370	1984	001	0	8402270304	189371	01/09/84

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DOCKET:370 MCGUIRE 2 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: DUKE
FACILITY OPERATOR: DUKE POWER CO.
SYMBOL: DPC

COMMENTS
STEPS 1 AND 8: CAUSE XX - DRAINING OPERATIONS

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

REFERENCE LERS:
1 370/83-092

ABSTRACT
POWER LEVEL - 000%. ON DEC 31, 1983 AT 1640, DURING DRAINING OPERATIONS OF THE REACTOR COOLANT (NC) SYSTEM, RESIDUAL HEAT REMOVAL (ND) PUMP B WAS OBSERVED TO HAVE ZERO DISCHARGE FLOW. PUMP B MOTOR AMPERAGE WAS LOW, AND THE ND SYSTEM PRESSURE AND PUMP B DISCHARGE PRESSURE WERE EQUAL. BASED ON THESE FACTORS, ND PUMP B WAS TRIPPED AND ND TRAIN B WAS DECLARED INOPERABLE AT 1650. THE FWST TO ND PUMP ISOLATION VALVE WAS TWICE CYCLED TO PROVIDE CORE COOLING AND RAISE NC SYSTEM LEVEL WITH WATER FROM THE FUELING WATER STORAGE TANK, WHILE VENTING THE ND SUCTION LINE AND PUMP B. THE CORE TEMPERATURE RATE OF RISE DECREASED AFTER THE 1ST WATER ADDITION, AND THE 2ND ADDITION RESULTED IN DECREASED CORE TEMPERATURES. ND PUMP B WAS RESTARTED AT 1720, AND FLOW WAS RESTORED. ON JAN 9, 1984 OPERATORS WERE DECREASING LEVEL IN THE REACTOR COOLANT LOOPS WHEN A COMPUTER ALARM FOR LOW ND PUMP A DISCHARGE PRESSURE WAS RECEIVED. FLUCTUATIONS IN ND PUMP A MOTOR AMPERAGE WERE NOTED AND SIMULTANEOUS FLUCTUATIONS IN DISCHARGE PRESSURE AND FLOW ALSO OCCURRED. AFTER THE "LOW ND FLOW" ANNUNCIATOR ALARMED, ND PUMP A WAS TRIPPED AT 1246, AND ND TRAIN A WAS INOPERABLE. OPERATORS MANUALLY OPENED THE ND SYSTEM TO FWST ISOLATION VALVE, RAISING THE REACTOR COOLANT LOOP LEVEL WITH WATER FROM THE FWST. THE SUCTION LINE AND PUMP WERE VENTED, AND THE PUMP WAS RESTARTED AT 1348. THESE INCIDENTS ARE DUE TO INADEQUATE GUIDELINES RECORDING THE WATER LEVEL TO BE MAINTAINED IN THE REACTOR COOLANT LOOPS DURING ND OPERATION.

Table B.7 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
382	1986	015	0	8608190325	200720	07/14/86

****

DOCKET:382 WATERFORD 3 TYPE:PWR
REGION: 4 NSSS:CE
ARCHITECTURAL ENGINEER: EBAS
FACILITY OPERATOR: LOUISIANA POWER & LIGHT CO.
SYMBOL: LPL

REPORTABILITY CODES FOR THIS LER ARE:
15 10 CFR 50.73(a)(2)(vii): Single failure criteria.

ABSTRACT
POWER LEVEL - 000%. ON JULY 14, 1986 WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS IN MODE 5 (COLD SHUTDOWN) WHEN OPERATIONS PERSONNEL WERE DRAINING THE REACTOR COOLANT SYSTEM (RCS) (AB) TO FACILITATE THE REPLACEMENT OF THE SEAL PACKAGE FOR REACTOR COOLANT PUMP 2A. THE RCS WAS BEING MAINTAINED BY DRAINING INTO THE REFUELING WATER STORAGE POOL (RWSP) (VIA THE LOW PRESSURE SAFETY INJECTION PUMP B MINI-RECIRCULATION VALVES, SI-120B, -121B) AND HOLDUP TANKS (VIA THE CHEMICAL VOLUME CONTROL SYSTEM (CB) PURIFICATION ION SI-423). AT 0113 HOURS OPERATIONS PERSONNEL SECURED DRAINING THE RCS BY CLOSING SI-423. HOWEVER, OPERATIONS PERSONNEL NEGLECTED TO CLOSE SI-120B AND -121B RESULTING IN RCS INVENTORY BEING PUMPED INTO THE RWSP. IN ADDITION, BECAUSE OF INSUFFICIENT NITROGEN PRESSURE, LOCAL REACTOR VESSEL LEVEL INDICATION WAS SUSPECT. AT 0317 HOURS LPSI PUMP B BEGAN CAVITATING. OPERATIONS IMMEDIATELY SECURED THE PUMP, TERMINATING SHUTDOWN COOLING (SDC) (BP). AT 0658 HOURS SDC WAS RESTORED BY A PROCESS OF REFILLING THE RCS AND CYCLING THE LPSI PUMPS TO RESTORE FLOW. (SINCE THE RCS TEMPERATURE INCREASED TO THE POINT OF LOCALIZED BOILING, THE LPSI PUMPS WERE SUBJECTED TO STEAM BINDING). THIS EVENT WAS DUE TO SIMULTANEOUSLY USING MORE THAN ONE METHOD OF DRAINING THE RCS, AND INACCURATE LEVEL INDICATION. THESE PROBLEMS WILL BE CORRECTED BY PLANT MODIFICATION AND PROCEDURAL CHANGES.

Table B.8 LER SCSS DATA 03-04-87

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
413	1985	028	0	8506070691	197983	04/22/85

DOCKET:413 CATAWBA 1 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: DUKE
FACILITY OPERATOR: DUKE POWER CO.
SYMBOL: DPC

COMMENTS

OTHER REPORTABILITY - 10CFR50.72(B)(2)(III); STEP 1: CAUSE AX - REQUIRED MAINTENANCE; STEP 7: PRIMARY COOLANT SYSTEM DRAINING; STEP 11: EFFECT KX - MINIFLOW VALVE CYCLING OPEN AND CLOSED; STEP 19: EFFECT HF - UNEXPLAINED LEVEL DECREASE AFTER PUMP STARTED.

WATCH-LIST CODES FOR THIS LER ARE:

941 REPORT ASSOCIATED WITH 10 CFR 50.72

REPORTABILITY CODES FOR THIS LER ARE:

14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

21 OTHER: Voluntary report, special report, Part 21 report, etc.

ABSTRACT

POWER LEVEL - 000%. ON APRIL 22, 1985, FROM 2039:21 TO 2051:17 HOURS, BOTH TRAINS OF RESIDUAL HEAT REMOVAL (RHR) WERE INOPERABLE. THIS WAS A RESULT OF RHR TRAIN A BEING DECLARED INOPERABLE ON APRIL 20, 1985, AT 1600 HOURS, FOR THE PERFORMANCE OF VARIOUS TRAIN A RELATED WORK REQUESTS, AND RHR PUMP B BEING SECURED ON APRIL 22, 1985, AT 2039:21 HOURS DUE TO LOSS OF PUMP SUCTION. ALSO, TECH SPEC 3.4.1.4.2 WAS VIOLATED ON APRIL 22, 1985, AT 0522 HOURS WHEN REACTOR COOLANT (RC) SYSTEM DRAINING BEGAN WITH RHR TRAIN A INOPERABLE. CATAWBA UNIT 1 WAS IN MODE 5 (COLD SHUTDOWN) WHEN THESE INCIDENTS OCCURRED. FALSE RC SYSTEM LEVEL INDICATION APPARENTLY CONTRIBUTED TO THE LOSS OF RHR PUMP B SUCTION. HOWEVER, THE CAUSE OF THE FALSE LEVEL INDICATION IS NOT KNOWN AT THIS TIME. WITH RHR TRAIN A INOPERABLE, THE LIMITING CONDITIONS FOR OPERATION OF TECH SPEC 3.4.1.4.2 WERE NOT MET. HOWEVER, PRIOR TO BEGINNING RC SYSTEM DRAINING, A DECISION HAD BEEN MADE TO ALLOW DRAINING TO BEGIN WITH RHR TRAIN A INOPERABLE. THEREFORE, THIS INCIDENT IS ALSO CLASSIFIED AS A PERSONNEL ERROR. AFTER RHR PUMP B WAS SECURED, CENTRIFUGAL CHARGING PUMP (CCP) A WAS ALIGNED TO THE REFUELING WATER STORAGE TANK (RWST) AND STARTED TO RESTORE RC SYSTEM LEVEL. RHR PUMP B WAS THEN VENTED AND RE-STARTED AT 2051:17 HOURS. ON APRIL 24, 1985, AT 1843 HOURS, AN OPERABLE RHR TRAIN A FLOWPATH WAS ESTABLISHED.

C. Failure to Maintain Vessel Level

Table C.1 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
304	1985	028	0	8601240058	198222	12/14/85

****

DOCKET:304 ZION 2 TYPE:PWR
REGION: 3 NSSS:WE
ARCHITECTURAL ENGINEER: SLXX
FACILITY OPERATOR: COMMONWEALTH EDISON CO.
SYMBOL: CWE

COMMENTS
STEP 4: OPERATOR LOWERED RPV LEVEL FOR MAINTENANCE WORK.

WATCH-LIST CODES FOR THIS LER ARE:
943 ALERT

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

REFERENCE LERS:
1 304/83-036

ABSTRACT

POWER LEVEL - 000%. ON 12-14 AT 3:25, 2B RHR PUMP BECAME AIRBOUND AS A RESULT OF VORTEXING. UNIT 2 WAS IN COLD SHUTDOWN WITH THE REACTOR HEAD INSTALLED BUT NOT TENSIONED AND THE RCS VENTED TO ATMOSPHERE. 2B RHR PUMP HAD BEEN IN OPERATION PROVIDING DECAY HEAT REMOVAL WITH RHR LETDOWN IN PROGRESS AND 2B CHARGING PUMP PROVIDING MAKE-UP FLOW TO THE RCS. DECAY HEAT REMOVAL WAS LOST FOR 75 MINS WITH A RCS CHANGE IN TEMPERATURE OF 15 DEGREES F. THE UNIT HAD BEEN SHUTDOWN FOR APPROX 100 DAYS THEREFORE THE SAFETY SIGNIFICANCE WAS MINIMAL. THE CAUSE OF THE EVENT WAS IDENTIFIED TO BE INADEQUATE PROCEDURES COUPLED WITH THE LACK OF KNOWLEDGE OF THE LEVEL AT WHICH THE RHR PUMPS BEGIN TO CAVITATE. AS A CONTRIBUTING FACTOR, THERE WERE PROBLEMS FOUND WITH THE LEVEL INDICATION. TO PREVENT RECURRENCE, PROCEDURES WILL BE REVIEWED AND CHANGED REFLECTING THE LESSONS LEARNED. TRAINING WILL BE CONDUCTED ON RCS LEVEL MEASUREMENT AND LOSS OF RHR SUCTION. THE RCS LEVEL SYSTEM WILL BE MODIFIED IN ORDER TO PROVIDE RELIABLE REMOTE LEVEL INDICATION DURING ALL REFUELING CONFIGURATIONS.

Table C.2 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
316	1984	014	0	8406280310	190615	05/21/84

****

DOCKET:316 COOK 2 TYPE:PWR
REGION: 3 NSSS:WE
ARCHITECTURAL ENGINEER: AEPS
FACILITY OPERATOR: INDIANA & MICHIGAN ELECTRIC CO.
SYMBOL: IME

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

ABSTRACT
POWER LEVEL - 000%. WITH THE UNIT IN COLD SHUTDOWN (MODE 5) AND THE REACTOR COOLANT SYSTEM AT HALF-LOOP, THE CONTROL ROOM OPERATORS STARTED A SECOND RESIDUAL HEAT REMOVAL (RHR) PUMP IN PREPARATION FOR REMOVING THE OPERATING RHR PUMP FROM SERVICE. WITH BOTH PUMPS RUNNING, FLOW BECAME EXCESSIVE FOR THE HALF-LOOP CONDITION CAUSING CAVITATION AND AIR BINDING OF BOTH PUMPS. BOTH PUMPS WERE OUT OF SERVICE FOR APPROX 25 MINS WHILE THEY WERE BEING VENTED WHICH IS WITHIN THE 1 HR ACTION STATEMENT TIME LIMIT OF TECH SPEC 3.4.1.3. TO PREVENT RECURRENCE THE PROCEDURE WHICH CONTROLS THE OPERATION OF THE RHR PUMPS HAS BEEN CHANGED TO INCLUDE SPECIFIC INSTRUCTIONS TO STOP THE OPERATING PUMP PRIOR TO STARTING THE SECOND PUMP WHILE AT HALF-LOOP.

Table C.3 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
327	1985	040	0	8511180159	197485	10/09/85

****

DOCKET:327 SEQUOYAH 1 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: TVAX
FACILITY OPERATOR: TENNESSEE VALLEY AUTHORITY
SYMBOL: TVA

COMMENTS
STEP 2: CAUSE XX-NORMAL SYSTEM OPERATION.

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

ABSTRACT
POWER LEVEL - 000%. ON 10-9-85 AT 1807 CST DURING COLD SHUTDOWN, SWAP OVER FROM 'B' TRAIN TO 'A' TRAIN RHR RESULTED IN BOTH TRAINS BECOMING INOPERABLE DUE TO AIR INJECTION INTO THE SUCTION OF THE PUMPS. THIS REQUIRED BOTH PUMPS TO BE VENTED AND REQUIRED RCS LEVEL TO BE RAISED FROM 695'1" TO 695'5" TO PREVENT A POSSIBLE RECURRENCE OF THE VORTEX PROBLEM. SUCTION FOR RHR COMES FROM THE LOOP 4 HOT LEG WHICH HAS A CENTER LINE OF 695'5". THE CAUSE FOR THE LOSS OF FLOW CAN BE ATTRIBUTED TO THE ADDITIONAL SUCTION CAUSED BY PLACING THE STANDBY RHR PUMP INSERVICE COUPLED WITH THE LOW RCS LEVEL OF 695'1". SYSTEM OPERATING INSTRUCTION (SOI)-74," "RHR SYSTEM," IS BEING REVISED TO CHANGE THE LOWER RCS OPERATING LIMIT FROM 695'0" TO 695'6" AND WILL REQUIRE THE OPERATING PUMP TO BE REMOVED FROM SERVICE PRIOR TO STARTING THE STANDBY PUMP. THE UNIT WAS IN COLD SHUTDOWN WITH ONLY A 0.2 DEGREES F RISE IN RCS TEMPERATURE RESULTING FROM THE EVENT. TECH SPEC 3.4.1.4 ACTION (B) SAYS THAT "...WITH NO RHR LOOPS IN OPERATION, SUSPEND ALL OPERATIONS INVOLVING A REDUCTION IN BORON CONCENTRATION OF THE RCS." AT THE TIME OF THIS EVENT, THE CHEMICAL VOLUME CONTROL SYSTEM (CVCS) (MAKEUP SYSTEM) WAS TAGGED OUT OF SERVICE; THEREFORE, NO VIOLATIONS OF TECH SPECS OCCURRED.

D. Loss of Reactor Coolant Through RHRS

Table D.1 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
029	1986	010	0	8608040099	200402	06/27/86

****

DOCKET:029 YANKEE ROWE TYPE:PWR
REGION: 1 NSSS:WE
ARCHITECTURAL ENGINEER: SWXX
FACILITY OPERATOR: YANKEE ATOMIC ELECTRIC CO.
SYMBOL: YAE

COMMENTS
STEP 1: EFFECT DX - FAILED. STEP 2: CAUSE AX - PUMP SHUT DOWN TO REPAIR SEAL.

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

ABSTRACT
POWER LEVEL - 000%. ON JUNE 27, 1986, AT 0137 HOURS' DURING A MAINTENANCE OUTAGE WITH THE PLANT IN MODE 5, MAIN COOLANT WAS INADVERTENTLY DRAINED TO THE LOW PRESSURE SURGE TANK (LPST). THIS COULD HAVE RESULTED IN A LOSS OF SHUTDOWN COOLING. THIS EVENT OCCURRED WHILE TRANSFERRING TO THE ALTERNATE METHOD OF SHUTDOWN COOLING PER PROCEDURE OP-2162, ATTACHMENT C. PERFORMANCE OF THIS PROCEDURE WAS NECESSARY BECAUSE OF THE FAILURE OF THE SHUTDOWN COOLING PUMP'S SHAFT SEAL. DURING THE EVOLUTION, APPROXIMATELY 2000 GALLONS OF WATER WAS DRAINED FROM THE PRESSURIZER AND MAIN COOLANT PRESSURE DROPPED FROM 100 PSIG TO 10 PSIG. THE PRESSURIZER DID NOT EMPTY. THE CONTROL ROOM OPERATOR (CRO) IMMEDIATELY SECURED THE LPST COOLING PUMP AND THE PRIMARY AUXILIARY OPERATOR (PAO) ISOLATED THE FLOW PATH. THE CRO STARTED ALL THREE CHARGING PUMPS AND RESTORED PRESSURIZER LEVEL AND PRESSURE. THE ROOT CAUSE OF THIS OCCURRENCE HAS BEEN ATTRIBUTED TO PERSONNEL ERROR. WHILE CONDUCTING THE ALTERNATE SHUTDOWN COOLING VALVE LINEUP, CH-V-654 WAS NOT FULLY SHUT, WHICH RESULTED IN A MAIN COOLANT SYSTEM TO LPST FLOW PATH. THE PAO THOUGHT THAT THE VALVE HAD COMPLETED ITS FULL TRAVEL WHEN HE OPERATED THE MANUAL VALVE. THIS OCCURRENCE WAS REVIEWED WITH THE APPROPRIATE PLANT PERSONNEL AND THE NEED FOR STRICT PROCEDURAL COMPLIANCE WAS EMPHASIZED. THIS IS THE FIRST OCCURRENCE OF THIS NATURE. THERE WERE NO ADVERSE EFFECTS TO THE PUBLIC HEALTH OR SAFETY AS THE RESULT OF THIS OCCURRENCE.

Table D.2 LER SCSS DATA 03-04-87

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
483	1984	016	0	8408240193	191322	07/17/84

DOCKET:483 CALLAWAY 1 TYPE:PWR
REGION: 3 NSSS:WE
ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: UNION ELECTRIC CO.
SYMBOL: UEC

REPORTABILITY CODES FOR THIS LER ARE:
11 10 CFR 50.73(a)(2)(ii): Unanalyzed conditions.

ABSTRACT
POWER LEVEL - 000%. ON 7-17-84 THE RCS DEPRESSURIZED TO 0 PSIG AND THE PRIMARY SEAL ON REACTOR COOLANT PUMP 'C' (RCP 'C') WAS DAMAGED. THE PLANT WAS IN MODE 5, WATER SOLID WITH THE RCS AT 380 PSIG AND 180 F PRIOR TO THIS EVENT. THE CAUSE OF THE RCS PRESSURE TRANSIENT WAS DETERMINED TO BE IMPROPER SEQUENCE OF VALVE OPERATION IN THE 'A' RESIDUAL HEAT REMOVAL PUMP SURVEILLANCE PROCEDURE RESTORATION. RHR TRAIN 'B' WAS ALIGNED TO TAKE A SUCTION AND DISCHARGE TO THE RCS, AND RHR TRAIN 'A' WAS BEING RESTORED FROM THE SURVEILLANCE DURING WHICH THE SUCTION AND DISCHARGE WERE ALIGNED TO THE REFUELING WATER STORAGE TANK (RWST). THE PROCEDURE REQUIRED OPENING THE TRAIN 'B' RHR INJECTION BALANCE LINE ISOLATION VALVE (EJ-HV-8716B) PRIOR TO ISOLATING THE RHR INJECTION BALANCE LINE FROM THE RWST BY CLOSING BN-8717. THUS, THE RHR PUMP WAS TAKING SUCTION FROM THE RCS AND DISCHARGING TO THE RWST, WHICH IMMEDIATELY DEPRESSURIZED THE RCS. RCP SEAL DAMAGE OCCURRED WHEN THE RCS DEPRESSURIZED TO 0 PSIG. THE SEAL WAS REPLACED AND RCP 'C' RETURNED TO SERVICE ON 8-6-84. A TEMPORARY CHANGE NOTICE WAS ISSUED TO CORRECT THE RHR SURVEILLANCE PROCEDURE. SIMILAR PROCEDURES WERE ALSO REVIEWED FOR IMPACT ON PLANT CONDITIONS.

E. Others

Table E.1 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
213	1984	014	0	8409280261	191741	08/24/84

****

DOCKET:213 CONNECTICUT YANKEE TYPE:PWR
REGION: 1 NSSS:WE
ARCHITECTURAL ENGINEER: SWXX
FACILITY OPERATOR: CONN. YANKEE ATOMIC POWER CO.
SYMBOL: COY

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

ABSTRACT
POWER LEVEL - 000%. WHILE IN THE REFUELING MODE A TOTAL LOSS OF NORMAL OFFSITE POWER WAS INITIATED BY STARTING A LARGE PUMP. POWER WAS BEING SUPPLIED BY ONE OFFSITE LINE AND STATION SERVICE TRANSFORMER. AUTOMATICALLY, BOTH DIESEL GENERATORS STARTED AND UNNECESSARY LOADS WERE SHED. THE AUTOMATIC CLOSURE OF ONE DG OUTPUT CIRCUIT BREAKER WAS DELAYED APPROX 20 MINS. CAUSES OF BOTH ANOMALIES: (1) A DIFFERENTIAL RELAY CURRENT TRANSFORMER WIRE WAS FOUND PULLED FROM ITS TERMINAL LUG. INRUSH CURRENT OF STARTING THE PUMP APPEARED AS AN INTERNAL TRANSFORMER FAULT CAUSING ISOLATION OF THE STATION SERVICE TRANSFORMER. THE WIRE PULL OCCURRED EARLIER THE SAME DAY WHEN MAINTENANCE ACTIVITIES WERE PERFORMED IN CLOSE PROXIMITY, (2) DIESEL VOLTAGE REGULATOR WAS LEFT SLIGHTLY BELOW THE BREAKER VOLTAGE PERMISSIVE RELAY SETPOINT WHEN IT HAD BEEN PREVIOUSLY SHUTDOWN. THE RELAY EVENTUALLY CLOSED DUE TO VIBRATION OF RESETTING NEARBY RELAYS AND/OR VOLTAGE AND FREQUENCY OPERATING VARIATIONS. CORRECTIVE ACTIONS: (1) A STATION DIRECTIVE TO LIMIT ACCESS NEAR ELECTRICAL EQUIPMENT PANELS, (2) REVISION OF OPERATING PROCEDURES TO ADJUST DIESEL VOLTAGE REGULATOR WELL ABOVE THE PERMISSIVE SETPOINT PRIOR TO SHUTDOWN, (3) INSPECTIONS FOR OTHER OPEN TERMINATIONS, (4) INITIATION OF PROCEDURE AND TRAINING ENHANCEMENTS, (5) INITIATION OF PERMISSIVE SETPOINT EVALUATIONS.

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Table E.2                    LER SCSS DATA                    03-04-87
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DOCKET  YEAR  LER NUMBER  REVISION  DCS NUMBER   NSIC      EVENT DATE
  272   1984     013         0      8407090030  190656      06/02/84
**********************************************************************
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DOCKET:272 SALEM 1 TYPE:PWR
REGION: 1 NSSS:WE
ARCHITECTURAL ENGINEER: PSEG
FACILITY OPERATOR: PUBLIC SERVICE ELECTRIC & GAS CO.
SYMBOL: PEG

COMMENTS
WATCH 975 - LOSS OF ONSITE POWER AT UNIT 1 AND UNIT 2.

WATCH-LIST CODES FOR THIS LER ARE:
975 POSSIBLE SIGNIFICANT EVENT

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

ABSTRACT

POWER LEVEL - 000%. ON JUN 2, 1984, POWER WAS INTERRUPTED BETWEEN THE 500 KV YARD AND THE 13 KV BUS, RESULTING IN A LOSS OF ONSITE POWER TO THE UNIT 1 AND UNIT 2 4 KV GROUP AND VITAL BUSSES. UNIT 1 WAS IN A REFUELING OUTAGE AT THE TIME WITH THE REACTOR DEFUELED, AND UNIT 2 WAS IN COLD SHUTDOWN. UNIT 2 EMERGENCY DIESELS STARTED AND LOADED IN THE BLACKOUT MODE; UNIT 1 EMERGENCY DIESELS AND 1B VITAL BUS WERE CLEARED AND TAGGED FOR MAINTENANCE. UNIT 2 RHR PUMPS WERE REMOVED FROM SERVICE BY THE SEC SEQUENCER, RESULTING IN A LOSS OF RESIDUAL HEAT REMOVAL FLOW. POWER WAS RESTORED TO ALL GROUP BUSSES WITHIN THIRTY SECONDS. CONTROL OF VITAL BUS LOADS WAS REGAINED, AND RHR WAS IMMEDIATELY RESTORED. UNIT 2 VITAL BUSSES WERE THEN TRANSFERRED TO STATION POWER AND THE DIESELS WERE SECURED. THE EVENT WAS THE RESULT OF A NUCLEAR CONTROL OPERATOR OPENING THE WRONG 500 KV CIRCUIT SWITCHGEAR. THIS WAS DUE TO NOT FULLY UNDERSTANDING THE SWITCHGEAR CONTROLS THAT WERE AVAILABLE TO HIM, AND NOT READING THE LABEL ON THE CONSOLE CONTROL PRIOR TO ITS OPERATION. THIS EVENT WAS AGGRAVATED BY RELAYING THE ORDER TO UNIT 2 CONTROL ROOM VIA THE UNIT 1 CONTROL ROOM NCO. THE INDIVIDUAL INVOLVED WAS COUNSELED AND REPRIMANDED FOR HIS ACTIONS ASSOCIATED WITH THE EVENT. TWO NEWSLETTER ITEMS DISCUSSED THE INCIDENT AND CAUSES. DUE TO THE LOSS OF RHR, THIS EVENT IS REPORTABLE IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(V)(B).

Table E.3 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
280	1986	017	0	8606190479	199656	05/24/86

****

DOCKET:280 SURRY 1 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: SWXX
FACILITY OPERATOR: VIRGINIA ELECTRIC & POWER CO.
SYMBOL: VEP

COMMENTS
STEP 2: COMP RLY-HFA RELAY.

REPORTABILITY CODES FOR THIS LER ARE:
13 10 CFR 50.73(a)(2)(iv): ESF actuations.
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

ABSTRACT

POWER LEVEL - 000%. ON MAY 24, 1986 UNIT 1 WAS AT REFUELING SHUTDOWN WITH REACTOR CAVITY FLOODED AND FORCED CIRCULATION IN SERVICE; UNIT 2 WAS AT 100% POWER. DUE TO MAINTENANCE AND DESIGN CHANGE WORK IN PROGRESS ON UNIT 1, NUMEROUS ELECTRICAL BUSSES WERE CROSS TIED. AMONG THESE WERE 1H AND 1J 4160V EMERGENCY BUSSES AND VITAL BUSSES 1-II AND 1-IV. #1 EMERGENCY DIESEL GENERATOR WAS OUT OF SERVICE. AT APPROXIMATELY 1520 HOURS, RESERVE STATION SERVICE FEEDER BREAKER 15D1 OPENED. THIS RESULTED IN AN UNDERVOLTAGE TRANSIENT SENSED AT 1J EMERGENCY BUS. #3 EMERGENCY DIESEL GENERATOR AUTO STARTED AND ASSUMED LOAD. BY DESIGN, THE 1J STUB BUS BREAKER OPENED DURING THE TRANSIENT WHICH RESULTED IN THE LOSS OF THE OPERATING 1B RESIDUAL HEAT REMOVAL AND 1B COMPONENT COOLING PUMPS. THE STUB BUS BREAKER WAS RESET AND THE COMPONENTS WERE RETURNED TO SERVICE. NUMEROUS SPURIOUS TRIP SIGNALS, ALARMS AND HI CONSEQUENCE LIMITING SAFEGUARDS SIGNAL WERE GENERATED DURING THE TRANSIENT.

Table E.4 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
281	1986	004	0	8603210132	198535	02/19/86

****

DOCKET:281 SURRY 2 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: SWXX
FACILITY OPERATOR: VIRGINIA ELECTRIC & POWER CO.
SYMBOL: VEP

REPORTABILITY CODES FOR THIS LER ARE:
10 10 CFR 50.73(a)(2)(i): Shutdowns or technical specification violations.

ABSTRACT
POWER LEVEL - 000%. ON 2-19-86 WITH UNIT 2 A COLD SHUTDOWN, OPERATORS WERE PERFORMING A TEST OF THE COMPONENT COOLING (CC) CHECK VALVES IN THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM. DURING THIS TEST, AN OPERATOR MADE AN INCORRECT VALVE LINEUP WHICH RESULTED IN THE ISOLATION OF CC FLOW TO THE 'A' RHR HEAT EXCHANGER AND RHR FLOW TO THE 'B' RHR HEAT EXCHANGER FOR APPROX. 10 MINUTES. DURING THIS PERIOD, RCS TEMPERATURE AND PRESSURE WERE CLOSELY MONITORED AND NO ABNORMAL INCREASES WERE NOTED. THE ROOT CAUSE OF THIS EVENT WAS HUMAN ERROR IN THAT THE OPERATOR FAILED TO FOLLOW THE STEPS IN THE WRITTEN PROCEDURE WHICH WOULD HAVE ENSURED THE PROPER VALVE LINEUP. A CONTRIBUTING FACTOR WAS POOR COMMUNICATION BETWEEN THE CONTROL ROOM OPERATOR AND THE OPERATOR PERFORMING THE VALVE LINEUP. THE OPERATORS INVOLVED IN THIS EVENT PREPARED A REPORT DESCRIBING THE CIRCUMSTANCES WHICH LED TO THIS ERROR AND IT WILL BE PLACED IN THE OPERATOR'S REQUIRED READING MANUAL. THIS EVENT WILL ALSO BE EVALUATED BY THE HUMAN PERFORMANCE EVALUATION COORDINATOR.

Table E.5 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
285	1985	011	0	8601220554	198163	12/14/85

****

DOCKET:285 FT. CALHOUN 1 TYPE:PWR
REGION: 4 NSSS:CE
ARCHITECTURAL ENGINEER: GIBB
FACILITY OPERATOR: OMAHA PUBLIC POWER DISTRICT
SYMBOL: OPP

COMMENTS
STEP 3: COMP RLY-RELAY CONTROLLING HIGH VOLTAGE BREAKER.

REPORTABILITY CODES FOR THIS LER ARE:
13 10 CFR 50.73(a)(2)(iv): ESF actuations.

ABSTRACT

POWER LEVEL - 000%. ON 12-14-85 AT 1010 HRS WHILE IN A REFUELING SHUTDOWN, A DC BUS AND 2 AC INSTRUMENT BUSES WERE LOST AS WELL AS ALL ESSENTIAL 480V BUSES. THE LOSS OF POWER INITIATED SAFEGUARD SIGNALS: PRESSURIZER PRESSURE LOW SIGNAL, SAFETY INJECTION ACTUATION SIGNAL, CONTAINMENT ISOLATION ACTUATION SIGNAL AND VENTILATION ISOLATION ACTUATION SIGNAL. ALSO LOST DUE TO THE POWER FAILURE WERE SHUTDOWN COOLING, COMPRESSED AIR, TURBINE PLANT COOLING WATER AND SOME CONTROL ROOM INDICATIONS. THE POWER FAILURE OCCURRED DUE TO PERSONNEL ERROR AND AN ALTERED ELECTRICAL DISTRIBUTION SYSTEM LINEUP DUE TO TESTING, MAINTENANCE AND MODIFICATION WORK. THE TECHNICIAN INADVERTENTLY TRIPPED THE RELAY CONTROLLING THE BREAKER THAT WAS PROVIDING THE 161 KV POWER TO THE 4160V 1A4 SAFEGUARDS BUS AND WHICH IN TURN POWER ALL 480V BUSES INCLUDING THE BATTERY CHARGERS. WITH THE LOSS OF THE BATTERY CHARGERS, DC BUS #2 BECAME INOPERABLE BECAUSE BATTERY #2 WAS DISCONNECTED FOR MAINTENANCE. ALSO, AC INSTRUMENT BUSES B AND D WERE INOPERABLE AS THEY ARE POWERED FROM DC BUS #2. THIS RESULTED IN A PARTIAL LOSS OF CONTROL ROOM INDICATIONS. CORRECTIVE ACTION INCLUDED RESTORING POWER TO THE 480V BUSES WITHIN 15 MINS. A MEETING WAS HELD WITH THE INDIVIDUALS INVOLVED BEFORE ALLOWING THEM TO RETURN TO THEIR TESTING.

Table E.6 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
286	1984	015	1	8507180323	196194	11/16/84

****

DOCKET:286 INDIAN POINT 3 TYPE:PWR
REGION: 1 NSSS:WE
ARCHITECTURAL ENGINEER: UECX
FACILITY OPERATOR: POWER AUTHORITY OF THE STATE OF NY
SYMBOL: PNY

COMMENTS
STEP 1: PSYS SW - UNKNOWN BLDG; COMPONENT MSF - ROOF, EFFECT DX - METAL ROOF PIECES BLEW OFF; STEP 9: MODEL FUSETRON FRN; STEP 12: MODEL OT40; STEP 14: MODEL DS-532.

REPORTABILITY CODES FOR THIS LER ARE:
13 10 CFR 50.73(a)(2)(iv): ESF actuations.
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

ABSTRACT
POWER LEVEL - 000%. ON 11-16-84, WITH THE REACTOR IN THE COLD SHUTDOWN CONDITION, A PHASE TO PHASE FAULT ACROSS THE STATION AUXILIARY TRANSFORMER (ST) BUSWORK CAUSED A LOSS OF NORMAL OFFSITE POWER TO THE UNIT. BOTH OPERABLE EMERGENCY DG'S STARTED AS REQUIRED. DURING THE TEMPORARY LOSS OF NORMAL OFFSITE POWER, SEVERAL BREAKERS IN THE PLANT'S ELECTRICAL DISTRIBUTION SYSTEM FAILED TO OPERATE. THE PLANT OPERATORS RESTORED STATION POWER THROUGH AN ALTERNATE OFFSITE SOURCE, AND RESTARTED ALL NECESSARY EQUIPMENT. THE FAULT WAS FOUND TO HAVE BEEN CAUSED BY A PIECE OF METAL WHICH WAS BLOWN ONTO THE A AND B PHASE BUSWORK OF THE STATION TRANSFORMER BY HIGH WINDS.

Table E.7 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
287	1985	003	0	8511260208	198261	10/15/85

****

DOCKET:287 OCONEE 3 TYPE:PWR
REGION: 2 NSSS:BW
ARCHITECTURAL ENGINEER: DKBE
FACILITY OPERATOR: DUKE POWER CO.
SYMBOL: DPC

COMMENTS
OTHER REPORTABILITY - VOLUNTARY REPORT; FIFTH OCCURRENCE OF VALVE FAILING TO OPEN

REPORTABILITY CODES FOR THIS LER ARE:
21 OTHER: Voluntary report, special report, Part 21 report, etc.

ABSTRACT
POWER LEVEL - 000%. ON 10-15-85 AT 0955 HRS, AN UNSUCCESSFUL ATTEMPT TO OPEN AN ELECTRIC MOTOR OPERATED VALVE WAS MADE FROM THE UNIT 3 CONTROL ROOM. UNIT 3 WAS IN HOT SHUTDOWN AFTER COMING OFF-LINE FOR MAINTENANCE. THE VALVE IS REQUIRED TO OPEN IN ORDER TO INITIATE THE DECAY HEAT REMOVAL COOLING MODE. THE CAUSE OF THE INCIDENT WAS THE TORQUE SWITCH SETTINGS ON THE VALVE. ROTORK NUCLEAR ACTUATOR SETTINGS WERE NOT SET HIGH ENOUGH TO OPERATE THE VALVE UNDER SYSTEM PRESSURE. THE EMO VALVE TORQUE SWITCH SETTINGS WERE NOT SPECIFIED IN THE DESIGN MODIFICATION PACKAGE USED TO REPLACE THE VALVE ACTUATOR WITH A NEW ROTORK NUCLEAR ACTUATOR. THE CORRECTIVE ACTION WAS TO OPEN THE VALVE FROM THE VALVE ACTUATOR CONTACTORS AT THE MOTOR CONTROL CENTER, BYPASSING THE VALVE ACTUATOR'S TORQUE SWITCH LIMIT CONTROL CIRCUIT. THE ANALYSIS SUPPORTING THE LICENSING BASIS FOR OCONEE DOES NOT REQUIRE THE IMMEDIATE OPENING OF THIS VALVE. THE FAILURE TO IMMEDIATELY OPEN THIS VALVE ONLY RESULTS IN A DELAY IN THE INITIATION OF DECAY HEAT REMOVAL COOLING MODE.

Table E.8	LER SCSS DATA	03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
302	1986	003	2	8609080235	201003	02/02/86

****

DOCKET:302 CRYSTAL RIVER 3 TYPE:PWR
REGION: 2 NSSS:BW
ARCHITECTURAL ENGINEER: GLBT
FACILITY OPERATOR: FLORIDA POWER CORPORATION
SYMBOL: FPC

COMMENTS
THREE PREVIOUS SIMILAR EVENTS; STEP 9: EFFECT DX - UNSPECIFIED DAMAGE.

REPORTABILITY CODES FOR THIS LER ARE:
14 10 CFR 50.73(a)(2)(v): Event that could have prevented fulfillment of a safety function.

ABSTRACT
POWER LEVEL - 000%. ON FEBRUARY 2, 1986, CRYSTAL RIVER UNIT 3 WAS IN MODE 5 WHILE PERFORMING REPAIRS ON A REACTOR COOLANT PUMP. THE REACTOR COOLANT SYSTEM WAS VENTED TO THE REACTOR BUILDING ATMOSPHERE AND DRAINED BELOW THE LEVEL OF THE REACTOR COOLANT PUMPS. AT 2148 HOURS, DECAY HEAT PUMP 1B TRIPPED DUE TO A MOTOR OVERLOAD CAUSED BY A PUMP SHAFT FAILURE. START-UP OF THE REDUNDANT PUMP WAS DELAYED BECAUSE AN ISOLATION VALVE ON THE SUCTION SIDE OF THE PUMP COULD NOT BE OPENED FROM THE CONTROL ROOM. THE VALVE WAS MANUALLY OPENED AND SYSTEM OPERATION WAS RESTORED AT 2212 HOURS. ON FEBRUARY 14, 1986, THE "B" TRAIN OF THE DECAY HEAT REMOVAL SYSTEM WAS BEING REFILLED AND MOVEMENT OF THE PUMP AND PIPING WAS NOTICED. EXAMINATION OF PIPE RESTRAINTS IN THE SYSTEM REVEALED THAT SEVERAL PIPE HANGERS WERE LOOSE OR DAMAGED. ALL DAMAGED EQUIPMENT HAS BEEN REPAIRED. BOTH DECAY HEAT PUMPS HAVE BEEN REBUILT. DECAY HEAT REMOVAL SYSTEM OPERATING PROCEDURES HAVE BEEN REVISED TO ADDRESS MINIMUM REQUIRED REACTOR COOLANT LEVEL AND PROVIDE IMPROVED FILL AND VENT INSTRUCTIONS. NEW BREAKER AND TORQUE SWITCH SETTINGS HAVE BEEN ESTABLISHED FOR THE ISOLATION VALVE. PREVENTATIVE MAINTENANCE PROCEDURES WILL REQUIRE PERIODIC LUBRICATION OF THE VALVE DRIVE SHAFT.

Table E.9 LER SCSS DATA 03-04-87

****

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
304	1986	005	1	8612150078	202103	01/17/86

****

DOCKET:304 ZION 2 TYPE:PWR
REGION: 3 NSSS:WE
ARCHITECTURAL ENGINEER: SLXX
FACILITY OPERATOR: COMMONWEALTH EDISON CO.
SYMBOL: CWE

COMMENTS
STEP 1: COMP MEI-LIGHT SOCKET.

REPORTABILITY CODES FOR THIS LER ARE:
13 10 CFR 50.73(a)(2)(iv): ESF actuations.

ABSTRACT
POWER LEVEL - 000%. ON 1-17-86, AT 8:05 AM THE UNIT 2 RESERVE FEED BREAKER 2432 (SUPPLYING POWER FROM THE SYSTEM AUXILIARY TRANSFORMER TO SERVICE BUS 243) TRIPPED WHILE AN ELECTRICIAN WAS REPAIRING THE "CLOSED" (RED) LIGHT SOCKET ON THE MAIN CONTROL BOARD. LOSS OF POWER TO SERVICE BUS 243 RESULTED IN A LOSS OF POWER TO ENGINEERED SAFETY FEATURE BUS 248 AND THUS RESIDUAL HEAT REMOVAL (RHR) 2B PUMP, WHICH IS POWERED OFF OF BUS 248, TRIPPED. ALL EQUIPMENT AFFECTED BY THIS LOSS OF POWER FUNCTIONED PROPERLY. SPECIFICALLY, DIESEL GENERATOR 2A AUTOSTARTED AND CARRIED OUT LOADS ASSOCIATED WITH ENGINEERED SAFETY FEATURE BUS 248. THE OPERATOR STARTED RHR PUMP 2A TO MAINTAIN REACTOR COOLANT SYSTEM TEMPERATURE. THE UNIT WAS IN COLD SHUTDOWN AT THE TIME. THE RESERVE FEED BREAKER TRIPPED BECAUSE THE ELECTRICIAN ACCIDENTALLY SHORTED OUT THE TRIP COIL. THE OPERATOR ALLOWED THE ELECTRICIAN TO FINISH REPAIRING THE LIGHT SOCKET WHILE THE DIESEL GENERATOR CARRIED THE ESF BUS 248 LOADS. TO PREVENT THIS PROBLEM FROM RECURRING, TRAINING FOR ALL OPERATING AND ELECTRICAL MAINTENANCE PERSONNEL WILL BE IMPLEMENTED SO THAT MAINTENANCE ACTIVITIES WILL NOT BE ALLOWED ON CLOSED BREAKERS

Table E.10	LER SCSS DATA	03-04-87

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
315	1985	046	0	8510170049	196728	09/07/85

DOCKET:315 COOK 1 TYPE:PWR
REGION: 3 NSSS:WE
ARCHITECTURAL ENGINEER: AEPS
FACILITY OPERATOR: INDIANA & MICHIGAN ELECTRIC CO.
SYMBOL: IME

REPORTABILITY CODES FOR THIS LER ARE:
13 10 CFR 50.73(a)(2)(iv): ESF actuations.
15 10 CFR 50.73(a)(2)(vii): Single failure criteria.

ABSTRACT
POWER LEVEL - 000%. ON 9-7-85 AT 0720 HRS WITH UNIT 1 IN MODE 5 POWER WAS LOST TO THE CONTROL ROOM INSTRUMENT BUS DISTRIBUTION CIRCUITS FOR CHANNEL 3 AND 4. THIS RESULTED IN VARIOUS ESF REACTOR TRIP SIGNALS AND LOSS OF THE RHR PUMPS. CHANNEL 3 AND 4 CIRCUITS WERE BEING POWERED BY AN ALTERNATE SOURCE WHILE THE NORMAL POWER SOURCE WAS OUT OF SERVICE. THE CIRCUIT BREAKER FOR CHANNEL 3 TRIPPED AS A RESULT OF AN INADEQUATELY TERMINATED LEAD. A LICENSED OPERATOR INVESTIGATING THE POWER LOSS THOUGHT THE CHANNEL 4 CIRCUIT BREAKER HAD TRIPPED ALSO. THE OPERATOR THEN ATTEMPTED TO RESET THE BREAKERS BY OPENING THEN CLOSING THE BREAKER. THIS RESULTED IN THE CHANNEL 4 BREAKER BEING MOMENTARILY DE-ENERGIZED. THIS CAUSED VARIOUS ESF REACTOR TRIP SIGNALS AND THE LOSS OF RHR PUMPS (DUE TO THE REFUELING WATER STORAGE TANK LEVEL INDICATION READING LOW FROM POWER LOSS). THIS PLACED THE UNIT IN A LCO PER TECH SPEC 3.4.1.3. THE RHR SYSTEM WAS MADE OPERABLE WITHIN 2 MINS AFTER LOSS. TO PREVENT RECURRENCE THE OPERATOR HAS BEEN COUNSELED NOT TO TAKE IMMEDIATE ACTIONS WHERE THE SITUATION DOES NOT REQUIRE IT.

Table E.11 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
382	1986	004	0	8604160022	198762	03/14/86

****

DOCKET:382 WATERFORD 3 TYPE:PWR
REGION: 4 NSSS:CE
ARCHITECTURAL ENGINEER: EBAS
FACILITY OPERATOR: LOUISIANA POWER & LIGHT CO.
SYMBOL: LPL

REPORTABILITY CODES FOR THIS LER ARE:
10 10 CFR 50.73(a)(2)(i): Shutdowns or technical specification violations.

ABSTRACT
POWER LEVEL - 000%. ON 3-14-86 WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS IN MODE 5 (COLD SHUTDOWN) (AS A RESULT OF A SCHEDULED SURVEILLANCE/MAINTENANCE OUTAGE WHICH BEGAN ON 3-7-86) WITH BOTH LOOPS OF THE REACTOR COOLANT SYSTEM (AB) DRAINED. AT 1035 HOURS ON 3-14-86 OPERATIONS PERSONNEL, IN PREPARATION FOR MAINTENANCE ON SI-406A, LOOP 2 SHUTDOWN COOLING RETURN RELIEF VALVE, STARTED LOW PRESSURE SAFETY INJECTION (LPSI) PUMP B (BP). THE PRIMARY NUCLEAR PLANT OPERATOR OBSERVED THE FLOW IN THE B SHUTDOWN COOLING (BP) TRAIN TO BE ZERO (0 GPM) AND LPSI PUMP B MOTOR CURRENT TO BE 33 AMPS. THE CONTROL ROOM SUPERVISOR ORDERED THE PUMP SECURED, AND PROCEEDED TO THE B SAFEGUARDS PUMP ROOM TO INVESTIGATE LOCAL CONDITIONS. THE INVESTIGATION REVEALED THAT SI-124B, LOW PRESSURE SAFETY INJECTION PUMP B DISCHARGE VALVE, WAS CLOSED WITH A DANGER TAG AFFIXED TO THE VALVE. THE VALVE WAS MISTAKENLY CLOSED DURING A TAG-OUT WHICH WAS CONDUCTED ON THE 3/13-14/86 MIDNIGHT SHIFT. THE VALVE WAS OPENED AND AT 1154 HOURS ON 3-14-86 THE B LPSI PUMP WAS PLACED INTO SERVICE. THE VALVE WAS INADVERTENTLY CLOSED BECAUSE PLANT OPERATORS DID NOT USE THE CLEARANCE REQUEST SHEET WHEN THEY CONDUCTED THE TAG-OUT. TO PREVENT THIS FROM RECURRING, A REVISION WILL BE MADE TO PROCEDURE UNT-5-003, "CLEARANCE REQUESTS, APPROVAL AND RELEASE", AND THE OPERATIONS SUPERINTENDENT WILL STRESS THE FUNCTION OF CLEARANCE SHEETS WITH OPERATIONS PERSONNEL.

Table E.12 LER SCSS DATA 03-04-87

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DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
529	1986	005	0	8603120359	198340	01/30/86

****

DOCKET:529 PALO VERDE 2 TYPE:PWR
REGION: 5 NSSS:CE
ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: ARIZONA PUBLIC SERVICE CO.
SYMBOL: APS

REPORTABILITY CODES FOR THIS LER ARE:
13 10 CFR 50.73(a)(2)(iv): ESF actuations.

ABSTRACT

POWER LEVEL - 000%. AT 1924 MST ON 1-30-86, PALO VERDE 2 WAS IN MODE 5 WHEN AN UNAUTHORIZED MODIFICATION ON A VITAL POWER INVERTER CAUSED A FAILURE OF THE TRAIN 'A', CLASS 1E, I&C POWER SYSTEM, WHICH RESULTED IN A CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNAL AND A TEMPORARY LOSS OF TRAIN 'A' SHUTDOWN COOLING. THE CAUSE OF THE FAILURE WAS ATTRIBUTED TO INADEQUATE CONTROL OF A MODIFICATION CONSISTING OF A RESISTOR JUMPERED AROUND A CAPACITOR IN THE CIRCUIT. THE MODIFICATION CAUSED AN EXCESSIVELY HIGH CURRENT ON AN INVERTER CIRCUIT BOARD, AND RESULTED IN 3 BLOWN INVERTER FUSES. THE INVERTER LOSS CAUSED A LOSS OF POWER TO A RADIATION MONITORING UNIT, WHICH IN TURN CAUSED THE CREFAS AND THE TEMPORARY TERMINATION OF TRAIN 'A' SDC. AS CORRECTIVE ACTION, ALL INVERTERS WERE INSPECTED FOR ADDITIONAL UNAUTHORIZED MODIFICATIONS, THE BLOWN FUSES WERE REPLACED, AND INVERTER SPECS WERE CHECKED. ADDITIONALLY, WORK CONTROL PROCEDURES WILL BE REVISED TO EMPHASIZE THE IMPORTANCE OF REMOVING ALL TEMPORARY MODIFICATIONS PRIOR TO PUTTING AN ELECTRICAL SYSTEM BACK IN SERVICE.

NRC FORM 335
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12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

This report summarizes a study performed by Brookhaven National Laboratory for the Reactor and Plant Safety Issues Branch, RES of the U.S. Nuclear Regulatory Commission in pursuit of the resolution of NRC Generic Issue 99. Generic Issue 99 focuses on the risk associated with loss of residual heat removal events at PWRs while shut down. Numerous loss of residual heat removal events have occurred at pressurized water reactors (PWRs) in the USA, which were terminated prior to damaging the reactor core. This study estimates the risk from loss of residual heat removal events and investigates ways of lowering this risk.

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