UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

APR 5 1991

MEMORANDUM FOR: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

FROM: Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

SUBJECT: PROPOSED RESOLUTION OF GI-130, "ESSENTIAL SERVICE WATER SYSTEM FAILURES AT MULTI-UNIT SITES"

Enclosed for CRGR review is the proposed resolution package for Generic Issue 130 (GI-130). The RES staff has completed the work necessary to resolve this generic issue. The background, technical findings, bases for resolution, cost/benefit analyses, recommendations, and conclusions are highlighted in the executive summary of Draft NUREG-1421, the regulatory analysis prepared by the staff (Enclosure 1).

GI-130 evaluates the potential safety consequences of Essential Service Water System (ESWS) failures at the multi-unit sites which have a two-pump-per-unit ESW configuration, with a crosstie between units. The ESWS is required in pressurized water reactors (PWR) to provide cooling capability for safe shutdown of the reactor during normal operation and accident conditions. Typical safety equipment supported by the ESWS under accident conditions are component cooling water heat exchangers, containment spray heat exchangers, high pressure injection pump oil coolers, emergency diesel generators, and auxiliary building ventilation coolers. Also, the reactor cooling pump (RCP) seals are cooled indirectly by the ESWS via the Component Cooling Water System (CCWS). Loss of RCP seal cooling could lead to seal failure, resulting in a small LOCA. Thus, failure of the ESWS function could lead to serious complications to a safe plant shutdown.

Initially, the proposed resolution involved (1) changes in the technical specifications and emergency procedures to improve the availability of ESW in both units via the crosstie between them, and the accident management of a loss of ESW using existing design features for recovery; and (2) the installation of a backup RCP seal cooling system independent of ESW to provide seal cooling for at least 8-10 hours following loss of ESW.

The Office of AEOD concurred with the recommended resolution and the Office of the General Counsel had no legal objections. The Office of NRR has advocated that the decision on the portion of the backfit providing a backup RCP seal cooling capability independent of ESW be deferred until GI-23, "Reactor Coolant Pump Seal Failures" reaches the decision point for resolution. The matter of RCP seal failures will then be addressed for the GI-130 plants as part of GI-23 resolution.

We accept this approach advocated by NRR. However, we both agree that some interim resolution is appropriate for these multi-unit sites. Therefore, we have revised our resolution package accordingly to recommend only changes in the technical specifications and emergency procedures to improve service water system availability, as currently discussed in our regulatory package.

The revised draft generic letter (Enclosure 2) provides for the deferral of the part of the recommended resolution involving a backup RCP seal cooling system to the resolution of GI-23, "Reactor Coolant Pump Seal Failures" pending completion of public review and comment on that generic issue.

The enclosed Regulatory Analysis, NUREG-1421, has also been appropriately revised. Please note that, in lieu of deleting significant portions from our regulatory analysis, we have retained all information used by the staff in the Decision Rationale, Chapter 6, of the Regulatory Analysis. Instead, we have confined our revision to Chapter 7, Implementation, reflecting our decision to proceed at this time only with the recommendation to improve technical specifications and emergency procedures. We believe it important that licensees/applicants have the benefit of the complete NRC staff evaluation, as each individual plant decides on the applicability of the analyses to their respective plant-specific configurations.

The revised pages of both documents are appropriately marked in the right margin to indicate revisions/additions.

The major supporting document, NUREG/CR-5526 (Enclosure 3), is also provided.

If you have any questions regarding this issue please contact Demetrios Basdekas of my staff on (X23943).

Eric S. Beckjord

Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

Enclosures:
1. Draft NUREG-1421 "Regulatory Analysis for GI-130"
2. Draft Generic Letter
3. Draft NUREG/CR-5526, "Analysis of Risk Reduction Measures Applied to Shared ESW Systems at Multi-Unit Sites"

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DRAFT REVISION 3

ENCLOSURE 1

NUREG-1421

REGULATORY ANALYSIS FOR THE RESOLUTION OF GENERIC ISSUE 130: ESSENTIAL SERVICE WATER SYSTEM FAILURES AT MULTI-UNIT SITES

Draft Report for Comment

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ABSTRACT

The Essential Service Water System (ESWS) is required to provide cooling in nuclear power plants during normal operation and accident conditions. Typical equipment supported by the ESWS are component cooling water heat exchangers, containment spray heat exchangers, high pressure injection pump oil coolers, emergency diesel generators, and auxiliary building ventilation coolers. Failure of the ESWS function could lead to severe consequences. This report presents the regulatory analysis for GI-130 "Essential Service Water System Failures at Multi-Unit Sites." The risk reduction estimates, cost benefit analyses, and other insights gained during this effort have shown that implementation of the recommendations will significantly reduce risk and that these improvements are warranted in accordance with the Backfit Rule, 10 CFR Part 50.109(a)(3).

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EXECUTIVE SUMMARY

This report provides supporting information, including a value-impact analysis, for the Nuclear Regulatory Commission's (NRC's) resolution of Generic Issue 130, "Essential Service Water System Failures at Multi-Unit Sites." This issue addresses the concerns regarding the Essential Service Water (ESW) system at multi-unit PWR sites having two ESW trains per unit with a crosstie capability (fourteen reactor units at seven sites). Typical components cooled by the ESW system under normal and accident conditions are the component cooling heat exchangers, containment spray heat exchangers, high pressure injection pump oil coolers, emergency diesel generators, and auxiliary building ventilation coolers. The ESW system is also used for cooling the reactor coolant pump (RCP) seals, typically indirectly via the component cooling water system (CCWS) or the charging pumps.

This issue was initially identified as a result of the safety evaluation related to the limiting condition for operation (LCO) relaxation program for Byron Unit 1. ESW system support from Byron Unit 2 via the crosstie between the two units was not available while Unit 2 was under construction. To support the LCO relaxation program, Byron Unit 1 performed a probabilistic risk assessment (PRA) of the ESW system. The insights derived from that study indicated that the core damage frequency due to the unavailability of a two train (one pump per train) ESW system could present a significant risk to the public health and safety, particularly if one ESW pump from the adjacent unit via an ESW system crosstie is not available.

At multi-unit sites, crossties are usually provided between the ESW systems of the adjacent unit to enhance operational flexibility; however, the Technical Specifications (TS) for these plants have typically not placed any operability requirements in the adjacent unit's ESW system, particularly during shutdown modes 5 and 6.

This regulatory analysis is partly based on a modified reliability analysis performed by Brookhaven National Laboratory (BNL) for the Byron plant. The PRA model was modified to reflect the multi-unit configuration and the assumption of having an ESW system failure as an initiating event for the accident sequence. Also, it was determined that a more recent value for RCP seal LOCA probability based on the data developed in NUREG-1150 should be established for the present analysis. A model was developed to incorporate the probability of an RCP seal LOCA as a function of time and leakage rate of the reactor coolant pump seal. In addition, both short and long term recovery actions which might affect the final outcome were examined.

The results of the analysis indicate that the core damage frequency (CDF) due to ESW system failure is estimated to be 1.52E-04 per reactor year. The staff examined seven possible alternatives to lower the CDF, and estimated that the potential reductions in CDF range from 1.37E-05 to 9.13E-05 per reactor year. A detailed description of modeling and assumptions used in the analysis are presented in NUREG/CR-5526.

A cost-benefit evaluation of the possible alternatives indicate that cost-effective options are available. One or more of these alternatives have the potential for significantly reducing the risk due to loss of ESW. Table ES.1 provides a summary of the best estimate cost-benefit ratios for each of the alternatives examined. Comparison of the best estimate cost-benefit ratios for all the alternatives against a guideline cost-benefit ratio of $1000/person-rem shows that all the alternatives are cost-beneficial except Alternative 4 entailing a separate intake structure. The regulatory analysis used these cost-benefit calculations as partial basis for considering a proposed resolution to GI-130. The proposed resolution is a combination of Alternative 6 (or 6a) plus Alternative 5 to provide a backup means of RCP seal cooling plus additional ESW technical specifications and emergency procedures.

The cost-benefit ratios were also calculated for the case of licence renewal for an additional term of 20 years, or a remaining plant life of 50 years. A comparison of the results shows that the cost-benefit ratios for all analyzed backfit alternatives are considerably lower for extended plant life. Even so, Alternative No. 4, Separate Intake Structure, still remains appreciably higher than the $1,000/person-rem guideline at a cost-benefit ratio of $2,285/person-rem.

Of interest to the decision process on this generic issue are the insights and views available in related PRA documentation in the open literature. Although still not finalized, the preliminary PRA work available in NUREG-1150, "Severe Accidents Risks: An Assessment for Five U.S. Nuclear Power Plants" (plus supporting documentation) is a source of extensive risk analyses information one might turn to for an understanding of ESW vulnerabilities. An examination of the NUREG-1150 documentation of the three PWRs that were studied indicates that the analyst considered that the ESW redundancy for two of the three PWRs was large enough that a complete loss of ESW as an event-initiator was deemed not credible (eight pumps available in Sequoyah, Unit 1). None of the five plants in the NUREG-1150 study is a GI-130 plant; however, it is worthwhile to note that one of the PWRs (Zion) identified the service water contribution to risk to be substantial (approximately 1.5E-4/RY). This contribution for Zion was approximately 42% of the total core damage frequency due to all causes.

Another PRA work available in the open literature is NSAC-148, "Service Water Systems and Nuclear Plant Safety," dated May 1990. Although it is only a compilation of earlier PRA results for six plants performed by the industry, it is useful to note that a greater appreciation of the service water system's contribution to plant risk has moved the industry to initiate a program to improve service water performance. The limited guidance available in NSAC-148 is a step in the right direction. The wide range of core damage frequencies (due to LOSW) over the six plants studied suggests large variability in plant-specific ESW configurations. The average CDF due to LOSW for the six plants was 6.55E-05/RY, with a range of 2.33E-04/RY-to-"negligible" contribution. Many details of these six PRAs are not included in NSAC-148 and, therefore, must be considered to be used only with a great caution. The overall message that the service water system provides an important safety function which could be a substantial contributor to overall plant risk tends to lend added credence to the GI-130 conclusions.

Table ES.1
Best Estimate Cost-Benefit Ratios ($/Person-Rem)

Alternative		Total Cost/ Benefit without Averted onsite Costs	Total Cost/ Benefit/With Averted onsite Costs
1.	No Action	---	---
2.	Additional Crosstie	433	238
3.	Electrical Cross-Connection	80	Note 1
4.	Separate Intake Structure	3847	3651
5.	Technical Specification Modifications + Procedures	25	Note 1
6.	High Pressure Pump for RCP Seal Cooling	862	684
6a.	Firewater for Thermal Barrier Cooling	37	Note 1
7.	Combination 6 + 5	756	574
7a.	Combination 6a + 5	39	Note 1

Note 1: Including averted onsite costs resulted in a net cost savings.

1. STATEMENT OF THE PROBLEM

This issue was identified in 1986 (Refs. 1, 2) as a result of the Byron Unit 1 vulnerability to core-damage sequences in the absence of the availability of Byron Unit 2 (not operational at the time). Because of the licensing considerations of Byron Units 1 and 2 and the immediate need to make a third ESW pump available to Byron Unit 1 via a crosstie with one of the two Byron Unit 2 ESW pumps, the Byron Unit 1 concern was treated as a plant-specific issue. However, the Byron plant-specific issue raised questions relative to multi-unit sites that have only two ESW pumps/unit with a crosstie capability between them.

Fourteen units at seven sites having the basic Byron ESW configuration were evaluated as part of this issue. These multi-unit sites have two ESW pumps per unit (one per train) with a sharing of one of the two pumps with the other unit via a crosstie between the two units. Evaluation of other design configurations of ESW systems in LWRs, including those of single unit sites, will be performed under GI-153, "Loss of Essential Service Water in LWRs."

It should be noted that the success criteria for the ESW systems in providing adequate cooling capability during normal, accident, and post-accident conditions are design-specific, depending on the plant configuration, the capacities of the ESW pumps, and equipment dependencies on ESW cooling. Although the success criteria may be as varied as the ESW systems, this evaluation assumed a generic set of success criteria as a representative model for purposes of quantifying the events leading to possible core-damage accidents. These generic criteria are discussed below and apply only to multi-unit sites having two ESW pumps/plant with a crosstie capability between them.

During normal operation, one ESW pump per unit provides adequate cooling to systems such as CCW, RCP seals and air conditioning and ventilation systems. The second ESW pump per unit is assumed to be normally in a standby mode. Because of load shedding (isolation of non-essential equipment), one ESW pump per unit is assumed capable of handling accident and cooldown heat loads. Typical equipment cooled by the ESW under these conditions are the CCW heat exchangers, containment spray heat exchangers, diesel generators, and auxiliary building ventilation coolers. With one plant in power operation, and the second plant in the shutdown or refueling modes of operation, the criteria assume one ESW pump can provide adequate cooling to shut down the operating plant through the crosstie connection, should the need arise.

A survey of operational experience (Refs. 3 and 4) shows that a number of different components in the ESW system may fail to perform their intended function in a variety of ways. However, review of operating experience has indicated that there are specific dominant failure modes for the ESW system associated with failures of certain components. Such failures have involved the traveling screens and/or other common cause problems at the intake structure leading to the partial or complete loss of the water supply. The ESW pumps and their electrical power supply are other important contributors to the partial or total loss of the ESW system. All ESW systems at the GI-130 multi-unit sites are safety systems, and their designs are plant-specific with plant-specific equipment, crosstie capability, and ESW operability needs for successful accident mitigation operations.

A comprehensive review and evaluation of the operating experience with ESWS has been performed and is reported in NUREG/CR-5526 (Ref. 3). Excluding system fouling(sediment, biofouling, corrosion, erosion), the total number of plant events involving a possible complete loss of the ESW function was 12 (Ref. 3, Appendix B). System fouling data were noted, but excluded from the current analysis due to the earlier resolution of Generic Issue 51, "Improving the Reliability of Open Cycle Service Water Systems" (see also the discussion in Chapter 6). The total number of PWR years during this period of data retrieval was calculated to be 667 reactor-years.

In 1980, one event involved a complete loss of ESW at San Onofre, Unit 1. At 100% power, a shaft on the operating salt water cooling (SWC) pump sheared due to vibration. This event then involved the additional failure of the normal standby pump (discharge valve failed to open) as well as the failure of a second auxiliary standby pump (lost prime). This led to a complete loss of ESW flow for about 15 minutes, at which time a fourth pump was manually crossconnected from the traveling screen wash system to establish cooling water flow.

A detailed examination of the loss of ESW events indicates that a number of events occurred in Modes 5 and 6 (shutdown) and some of them may not have been a complete loss of ESW in terms of total stoppage of ESW flow, even though the ESW system might have been declared inoperable.

The difference of the ESW system between power and shutdown operation is primarily the actual heat load and equipment affected by the loss of ESW. In addition, the actual administrative requirements imposed by the technical specifications also differ, and make these two operational modes more distinct.

To calculate the initiating event frequency for loss of ESW, the total operating ESW-system-years for all PWRs of 667 reactor-years was divided into two parts as follows:

487 reactor-years-at-power
180 reactor-years-at-shutdown

Finally, the respective loss of all ESW frequencies were calculated to be 1.1E-03 per reactor-year-at-power, and 3.2E-02 per reactor-year-at-shutdown (with one pump running and one at standby), and 2.9E-01 per reactor-year-at-shutdown (with one pump running and the other in maintenance). These numbers then were weighted for the various operational states of each unit and their respective time fractions, before calculating the CDF values, as discussed in Section 4.1.1.

Should a loss of the ESW system function fail to be recovered, the resulting core-damage accident could lead to significant risk to the public. The most dominant sequence is the reactor coolant pump seal loss of coolant accident (RCP-LOCA). This specific sequence is the subject of GI-23, "Reactor Coolant Pump Seal Failures" (Ref. 7). This study estimated the total core damage frequency (CDF) attributable to the loss of ESW for seven two-unit sites (Chapter 4) and the cost-effectiveness of several alternative modifications (Chapter 5) which could lower this CDF.

2. OBJECTIVE

The purpose of the Generic Issue 130 program is to evaluate the safety adequacy of a two-pump ESW system in existing multi-unit PWR power plant sites, and to examine the cost-effectiveness of alternative measures for reducing the overall vulnerability to ESW system failures.

Probabilistic methods were used to assess the CDF, the potential reduction in risk of the modifications, and their cost-effectiveness. The overall objective for resolution of GI-130 is that contribution from loss of the ESW system should be a small percentage of the total CDF due to all causes.

For USI A-45, the staff recommended in NUREG-1289 that the frequency of events related to DHR failure leading to core damage should be reduced to a level (around 1.0E-5/RY) so that the probability of such an accident in the next 30 years would be about 0.03 based on a population of around 110 plants. A similar core damage objective (1.0E-5/RY) was noted in USI A-44 covering station blackout. These objectives are also consistent with the recently issued guidance to the staff (Ref. 6) setting a goal for CDF of less than 1.0E-04/RY from all contributors. To meet such a goal the staff has aimed for the benchmark that a single contributor to the CDF contributes no more than 10% of the above suggested value, or no more than 1.0E-05/RY. The application of the safety goal guidance and the objectives of previously resolved USIs, as discussed above, to GI-130 was limited to using them as general guidelines to the decision process described in Chapter 6. Rigid application of such a quantitative objective to define an absolute requirement was not made. Since the ESW vulnerability issue is only a fraction of the total contribution to risk due to all causes, the current safety goal guidance that the overall mean frequency of a large release should be less than 1 in 1,000,000 per year is not directly usable to this case. This is partly because an overall PRA due to all causes was not in the scope of GI-130. However, consistent with current policy guidance in References 5 and 6, a judgement was made that, in light of the safety goals and available knowledge, the recommendation to backfit selected design and operational improvements to reduce risk due to ESW failures is warranted (Chapter 6).

3. ALTERNATIVE RESOLUTIONS

There were several alternatives considered for the resolution of Generic Issue 130. These alternatives are described below.

3.1 Alternative 1 - No Action

Under this alternative there would be no new regulatory requirements. Consistent with existing regulations, this alternative does not preclude a licensee, or an applicant for an operating license, from proposing to the NRC staff design changes intended to enhance the reliability/operability of the Essential Service Water System and its components on a plant-specific basis.

3.2 Alternative 2 - Install Additional Crosstie

The ESW systems of the seven multi-unit sites analyzed under GI-130 are cross-connected through pipe connections and isolation valves. This arrangement allows the operator of one unit to utilize the ESW cooling capacity of the other unit under most circumstances. In most cases, the crosstie isolation valves can be remotely operated. A hardware failure to open the isolation valves, should the need arise, could result in adverse conditions. A parallel cross-connection could reduce the possibility of this kind of failure, and in addition would allow for more flexible maintenance options.

3.3 Alternative 3 - Provide Electrical Power Cross- Connection

In general, the electrical power supplies to the ESW trains are separated and have no cross-connection capability, i.e., the Train A ESW pump cannot be powered from electrical Train B (or Diesel B). This alternative investigated the implementation of crossties between the electrical trains of the unit with respect to the operation of the two ESW pumps (Trains A and B). The cross-connection of electrical power supply of other electrical components, such as MOVs was not considered as part of this alternative because of their less significant potential contribution to risk as observed in the operational experience failure data.

3.4 Alternative 4 - Provide Separate Intake Structure

The intake structure is usually a single structure divided into separate bays by concrete walls. There are a number of screens installed to prevent the intake from passing large foreign objects. The common mode failure of these screens may occur as a loss of the common inlet and/or common water source. The whole intake structure or screens could be affected by events such as flooding or freezing.

The alternative considered here is a completely separate intake structure and swing pump serving as a redundant intake source of ESW water. It may be located on the same water source, but in a physically separate location. An alternate design, which would provide additional independence/diversity, would be to install the additional intake structure on a physically separate water source (e.g., pond or lake). The separate intake structure alternative includes the structure, screens, associated motors, valves and piping. A swing ESW pump would also be made available to either unit with redundant electrical power supplies. Common mode failure considerations are assumed to play a primary role in the design and installation of the new structure (such as heated spaces in areas of the country subject to freezing conditions).

3.5 Alternative 5-Modify Technical Specifications (TS) Requirements

In operating modes 5 and 6 (shutdown and refueling, respectively), the status of ESW pumps is uncertain because TS typically do not require that the ESW pumps be operational in these shutdown modes. This alternative partially involves imposing an explicit operability requirement on at least one of the ESW pumps of a unit while in modes 5 and 6 to provide backup for the other unit ESW system. An additional improvement is the testing of the unit crosstie valves to provide greater assurance of operability, thereby reducing the hardware failure assumptions on the crosstie valves. Also, this alternative includes additional credit for improvements in emergency procedures for recovering from a LOSW accident.

3.6 Alternative 6 -Provide Independent RCP Seal Cooling System

This alternative provides an independent water supply and distribution system for backup cooling of the RCP seals in case of ESW loss. Preventing an RCP seal failure and, hence, a small break LOCA would remove a substantial risk contributor associated with this issue. This alternative is also a consideration in Generic Issue 23, "Reactor Coolant Pump Seal Failures." A proposed resolution for GI-23 has recently been reported (Ref. 7). An objective of the proposed resolution of GI-23 is to reduce the probability of seal failure, thus making it a relatively small contributor to total core-damage frequency.

3.7 Alternative 7 - Combine Alternatives 5 and 6 (TS Changes and Independent RCP Seal Cooling)

Under this alternative, a combination of two or more alternatives discussed above could result in greater risk reduction. The combination of Alternatives 5 and 6, namely technical specifications (TS) changes regarding limits on taking equipment out of service during shutdown operations, cross-tie testing requirements, and procedures improvement combined with an independent RCP seal cooling system, could be expected to result in a more substantial CDF reduction and still be cost-effective.

4. TECHNICAL FINDINGS

The BNL evaluation of failures of ESW system at multi-unit sites included a determination of the initiating frequency of loss of ESW system, core damage frequency due to ESW failure, dose consequence analysis and cost benefit analysis. The detailed evaluation is found in NUREG/CR-5526 (Ref. 3).

4.1 Core Damage Frequency Analysis

The core damage vulnerability caused by the failure of the ESW system may be estimated by developing a full scale PRA model including initiating event frequency categories, event tree and fault tree analysis and incorporation of support system dependencies. The PRA model was then appropriately modified to reflect various plant operating configurations to analyze the consequences of the loss of ESW function in each operating state as shown in Table 4.1.1.

To facilitate the present analysis, BNL selected an existing Byron Unit 1 PRA model (Ref 2.) which was previously developed and which examined the ESW system of a single unit (Byron Unit 2 was not operational at the time). The Byron model was modified by BNL to include the effects of multi-unit configuration, and short term/long-term recovery actions. Additionally, the probability of RC pump seal LOCA was established based on a more recent pump seal failure model as described in NUREG/CR-4550 (Ref. 8), and incorporated in the present analysis.

4.1.1 Initiating Event Frequency

The initiating event frequency representing the loss of ESW for multi-unit site operations was derived initially from operational experience for single unit PWR operations. This LOSW initiating event frequency was then modified, to reflect multi-unit PWR sites. As the system configuration for various operating states may be different, the respective LOSW initiating event frequency for each of these operating states was determined separately. An approximation method involving the combination of the experience data with an analytical technique was used. A multi-unit ESW system fault tree was developed similar to the existing model of Byron Unit 1. This modified model represents the unavailability of the second unit to supply ESW to the first Unit, given the complete LOSW in the first unit. The fault tree is provided in Appendix D of Reference 3. Table 4.1.2 lists the initiating event frequency for each operating state. This frequency was calculated on the basis of the operational experience reflected by the base initiator, and then multiplied by a modifier corresponding to the respective operating states of the two units derived from a fault three analysis (Ref. 3).

4.1.2 ESW System and RCP Seal LOCA Recovery

The event tree established in Reference 2 indicated that the small LOCA due to RC pump seal failure and AFW system failure are the dominant accident sequences. It was decided to use a more recent model for seal LOCA probability. The RC pump seal failure probabilities are based on the model developed in Reference 8 which provides the probability of a seal leakage as a function of the leak rate and elapsed time after the loss of seal cooling.

A simplified recovery model was also developed by BNL in Reference 3 for the sequences relative to the failure of the ESW system. The recovery model consists of a number of recovery factors which are established based on the particular failure mode and the time available.

Operating experience data bases regarding ESW systems consisting mostly of LER submittals were searched by BNL and, as also confirmed by NUREG-1275 (Ref. 4), the ESW system failure duration has varied from less than 1 hour to a few days before ESW system recovery. The data suggest that there are approximately three characteristic time periods of system recovery. The first time period involves ESW failures which may be recovered within one hour and consists of a large fraction of the ESW events (approximately 70% of the total). The second time period involving more problematical hardware or other failures, extends up to 5 hours. About 90-96% of all events may be recovered in this time. The last group of events are such that recovery may take a relatively long time and generally involve the most serious hardware problems. It is estimated that by the end of 24 hours only about 1% of the events were not recovered.

4.1.3 Relative Time Fractions

Since the average time of operation varies with different operating configurations, it is necessary to estimate the relative time fractions for each operating mode. The relative time fractions essentially represent the average length of time period of the specific multi-unit operating state coupled with the arrangement of the ESW Systems. Maintenance or test-related outage time of ESW equipment must also be accounted for in the system's average time fraction. The ESW flow requirement may be satisfied through the unit crossties utilizing the ESW pumps of the other unit. Based on discussions with utilities, it was assumed that the crossties are used about 10% of the time during the shutdown period.

The most dominant time fraction is that of the power operating arrangement, i.e., both units at power and one ESW train of each unit running with the other in standby.

4.1.4 Core Damage Frequency

The core damage frequency (CDF) due to LOSW was calculated using the following expression:

$$CDF = \Sigma_i \lambda_i \text{ (state)* } P_i \text{ (Sequence)* } RT_i$$

Where λ_i is the state-dependent initiating event frequency given that the unit is in this state for the full year, and RT_i is the relative time fraction of the ith state while P_i is the ith sequence probability (conditional core damage probability).

The dominant sequence conditional core damage probabilities are summarized in Table 4.1.3. The sum of all the sequences during power operation results in P (power operation) = 1.03E-01 which reflects the conditional probability of core damage given a complete loss of ESW during power operations. The corresponding value for shutdown is P (shutdown) = 2.82E-02. The most dominant contributor for all sequences, including shutdown, is the RCP seal LOCA; P (Seal LOCA) = 6.8E-02, which is approximately 65% of P (power operation).

The core damage frequencies due to various accident sequences are summarized in Tables 4.1.4, and 4.1.5. The most dominant sequence is the RCP seal LOCA: CDF (Seal LOCA) = 8.8E-05 per reactor-year, which is about 60% of the total CDF due to ESW loss of 1.5E-04 per reactor-year.

The total CDF due to loss of ESW (1.5E-04 per reactor-year) is judged to be substantial compared to the total due to all causes (typically in a range of about 1.0E-4 to 2.0E-4 per reactor-year). The next section presents the results of an examination of different alternatives which could lower this core damage frequency.

4.1.5 Effects of Potential Improvements on Core Damage Frequency

The potential alternatives for improvements were initially selected in NUREG/CR-5526 (Ref. 3) by considering (a) the dominant failure modes of the ESW system (listed in Table 4.1.4) and (b) the dominant accident sequences contributing to the relatively high CDF. Since there is no single dominant failure mechanism represented in the initiating event frequency, a number of different options were considered including combinations of particular failure modes to reduce the initiating LOSW frequency. The failure modes indicated in Table 4.1.4 are based on actual operating experience.

The base case initiating event frequency was modified to take into account the effects of the particular alternative under consideration. First, the fraction of the initiating event frequency that could be improved by each alternative under consideration was determined using the data listed in Table 4.1.8. Second, the relative change in the ESW system reliability with and without the improvement provides an indication of the potential reduction in the core damage frequency. Fault tree analyses which included the logic modules and/or additional component failure rates that represent the proposed modification were employed to estimate the total system unavailability. The reliability analyses of the improvements were performed for each state or plant configuration, resulting in a calculation of configuration-dependent initiating event frequencies.

As noted in Section 3, the following potential improvements were analyzed regarding their capability to provide a cost-effective reduction in risk due to a LOSW event:

- Additional Crosstie - Reducing the possibility of the malfunction of the cross-connection between units.

- Electrical Power Cross-Connection-Increasing the redundancy of the electrical power supplies to ESW pumps.

- o Separate Intake Structure or Bay with an Additional Swing ESW Pump - Increasing the redundancy of the ultimate heat sink or source of cooling and increasing the availability of the ESW pumps.

- o Changing Technical Specification requirements and emergency procedures.

- o Installation of an independent RCP seal cooling system.

- o Combination of RCP seal cooling system and Technical Specifications/Procedures changes.

The first three alternatives were selected based on considerations regarding the ESW failure mechanisms observed in the PWR operating history data base. A particular operating mode when both ESW pumps of the shut down plant are inoperable (State IId and h) is a concern since there are no explicit Technical Specifications requirements on the ESW system in this operating mode. Therefore, the alternative of imposing additional TS requirements was also analyzed regarding their effect on CDF reduction potential. This alternative also considers additional credit for unit crosstie testing and emergency procedures.

The most dominant contribution to the CDF arises from the failure of the RCP seal upon loss of seal cooling due to the unavailability of the ESW. Therefore, the installation of an independent RCP seal cooling system which would cool the seals in the event of loss of ESW was also evaluated as a potential improvement. The results are summarized in Table 4.1.5.

Table 4.1.1
Operational Status of Multi-Unit Sites

Site's Status	Unit 1	Unit 1 ESW Pump 1	Unit 1 ESW Pump 2	Unit 2	Unit 2 ESW Pump 1	Unit 2 ESW Pump 2
Ia	OP	R	AOT	OP	R	AOT
Ib	OP	R	AOT	OP	R	SB
Ic	OP	R	SB	OP	R	AOT
Id	OP	R	SB	OP	R	SB
IIa	OP	R	AOT	DN	R	AOT
IIb	OP	R	AOT	DN	R	SB
IIc	OP	R	AOT	DN	SB	M
IId	OP	R	AOT	DN	M	M
IIe	OP	R	SB	DN	R	AOT
IIf	OP	R	SB	DN	R	SB
IIg	OP	R	SB	DN	SB	M
IIh	OP	R	SB	DN	M	M
IIIa	DN	R	AOT	OP	R	AOT
IIIb	DN	R	AOT	OP	R	SB
IIIc	DN	R	SB	OP	R	AOT
IIId	DN	R	SB	OP	R	SB
IVa	DN	R	AOT	DN	R	AOT
IVb	DN	R	AOT	DN	R	SB
IVc	DN	R	AOT	DN	SB	M
IVd	DN	R	AOT	DN	M	M
IVe	DN	R	SB	DN	R	SB
IVf	DN	R	SB	DN	R	AOT
IVg	DN	R	SB	DN	SB	M
IVh	DN	R	SB	DN	M	M

OP = Operating.
DN = Shutdown.
R = Pump running.
SB = Pump in standby.
AOT = Pump in test (allowable outage time).
M = Maintenance.

Table 4.1.2
State Dependent LOSW Initiating Event Frequencies

States Unit 1 Pumps	Unit 2 Pumps	ESW Unit Initiating Event Frequency/ Reactor-Year
I - Unit 1-Up/2-Up		
R/AOT	R/AOT	1.6E-01
	R/SB	1.4E-02
R/SB	R/AOT	1.2E-02
	R/SB	1.1E-03
II - Unit 1-Up/2 Down		
R/AOT	R/AOT	1.2E-02
	R/SB	1.1E-02
	SB/M	1.4E-02
	M/M	1.6E-01
R/SB	R/AOT	9.7E-04
	R/SB	8.9E-04
	SB/M	1.1E-03
	M/M	1.2E-02
III - Unit 1-Down/2-Up		
R/AOT	R/AOT	2.3E-02
	R/SB	2.1E-02
R/SB	R/AOT	2.6E-02
	R/SB	2.3E-03
IV - Unit 1-Down/2-Down		
R/AOT	R/AOT	2.3E-02
	R/SB	2.1E-02
	SB/M	2.6E-02
	M/M	2.9E-01
R/SB	R/AOT	2.6E-03
	R/SB	2.3E-03
	SB/M	2.9E-03
	M/M	3.2E-01

Table 4.1.3
Sequence Conditional Core Damage Probabilities

Sequences	Conditional Core Damage Probability
Power Operations	
RCP Seal LOCA - P(Seal LOCA)	6.8E-02
Auxiliary Feedwater - P_{AFW}	2.3E-02
Long Term AFW - P_{LAFW}	9.1E-03
Other Sequences - P_{Other}	3.2E-03
Total - P(Operation)	1.03E-01
Shutdown - P(Shutdown)	2.82E-02

Table 4.1.4
Core Damage Frequency Due to Individual Sequences

Sequences	Initiating Event Frequency λ*RT	Sequence Probability-P	Core Damage Frequency CDF/R-YR
Seal LOCA - P(SL)	1.3E-03	6.8E-02	8.8E-05
AFW - P_{AFW}	1.3E-03	2.3E-02	3.0E-05
Long Term - P_{LAFW}	1.3E-03	9.1E-03	1.2E-05
Other - P_{Other}	1.3E-03	3.2E-03	4.2E-06
Total Power Operation - P(Power Operation)	1.3E-03	1.03E-01	1.3E-04
Shutdown - P(Shutdown)	7.1E-04	2.82E-02	2.0E-05
TOTAL			1.5E-04

Table 4.1.5
Core Damage Frequency - Summary

States	Initiating Event Frequency λ*RT	Sequence Probability P	Core Damage Frequency CDF/RYR
I + II	1.30E-03	1.03E-01	1.3E-04
III + IV	7.1E-04	2.82E-02	2.0E-05
TOTAL			1.5E-04

Table 4.1.6
Failure Mode Classification

Failure Mode	Relative Contribution to Initiating Frequency
Intake structure unavailable	35%
Loss of electrical power supply	35%
Loss of ESW pumps	20%
Other	10%

Table 4.1.7
CDF Reduction For Alternatives

Alternative	ΔCDF
1. No Action	N/A
2. Additional Crosstie	1.60E-05
3. Electrical Power Cross Connection	1.4E-05
4. Separate Intake Structure	9.13E-05
5. Technical Specifications Modifications and Procedures	2.55E-05
6. Independent RCP Seal Cooling	7.82E-05
7. Combination of Alt. 6 + Alt. 5	9.10E-05

4.2 Dose Consequence Analysis

For purposes of this study, consequences are measured in person-rem and benefits in person-rem averted. Once the core damage frequency (CDF) and changes in CDF due to a potential resolution alternative have been calculated (Section 4.1), the next step is to calculate the corresponding consequences in person-rem, and hence, benefits in person-rem averted. The reactor safety study (WASH-1400) first attempted to evaluate containment performance for a number of accident sequences. As part of that attempt a set of radioactive release parameters was developed corresponding to specific containment failure modes. More recently, the NRC has documented in NUREG-1150 a detailed assessment of the risk associated with five nuclear power plants. This study (NUREG-1150) represents the most updated analytical framework for the assessment of containment performance including source terms and off-site consequences. It was decided to use NUREG-1150 as the basis for the evaluation of the seven two-unit sites of this issue. A more detailed description of these calculations and their application to this study is given in Reference 3. The consequence model specific to the Zion site was used as the starting point of the consequence assessment of the seven sites of this issue because of the availability of its detailed modeling and evaluation in the NUREG-1150 effort. The multi-unit sites evaluated in the GI-130 study would be expected to produce average consequences smaller than those calculated for the Zion site because of their location and respective population densities within their evacuation zones. For this reason, adjustments were made to the Zion consequences as discussed in detail in Reference 3, and summarized in the following paragraph.

A comparison of the Zion-based results was made with those of the Surry and Sequoyah plants, and it was concluded that the consequences of an ESW induced core-damage at a large, dry containment plant, typical of the GI-130 plants, to be 47% of the total consequences for Zion, or 8.0E+06 person-rem. It should be noted that this is for power operation only and without taking containment systems recovery into consideration. When recovery actions are taken into consideration this number is modified to 5.5E+06 person-rem.

A calculation of the consequences associated with shutdown operations was also performed. While the use of power operation release categories for consequence calculations at shutdown may appear to overestimate consequences, Reference 3 indicates that the person-rem consequences are relatively insensitive to the source term. This is because of interdiction criteria and because of the relatively high contribution of long-lived isotopes to the long term dose. The total consequences for shutdown operations were calculated in NUREG/CR-5526 (Ref. 3) to be 3.1E+06 person-rem. Hence, the overall benefit for each alternative considered in terms of averted consequences in person-rem may be estimated by multiplying the power consequences with the power ΔCDF and the shutdown consequences with the shutdown ΔCDF, adding the two products and multiplying by 30 years, the assumed lifetime of the average GI-130 plant. Hence:

$$\text{Total Benefit} = 30 \times (\Delta CDF_{power} \times 5.5E{+}06 + \Delta CDF_{SD} \times 3.1.E{+}06).$$

Table 4.2.1 shows the benefits (or consequences reduction) in person-rem that was calculated for each proposed alternative.

Table 4.2.1
Benefits of Proposed Alternatives (Person-Rem)

Alternative	Low Estimate	Best Estimate	High Estimate
1. No Action	---	---	---
2. Additional Crosstie	739	2,635	4,951
3. Electrical Cross-Connection	645	2,349	4,467
4. Separate Intake Structure	3,992	14,324	27,004
5. Technical Specifications Modifications	1,150	4,141	7,825
6. Independent RCP Seal Cooling	3,510	12,870	24,570
7. Combination of Alternatives 6 and 5	4,063	14,821	28,211

4.3 Cost Analysis

To calculate costs for the various alternative backfits, several sources were consulted (Ref. 3). Some cost estimates were derived from an NRC-sponsored research report (Ref. 9). Another source was the computer printout for the Energy Economic Data Base (EEDB) and supporting documents (Ref. 10). Still another source was various discussions with utilities.

An initial overall assumption was that the backfits can be accomplished outside of the critical path. Consultation with utility personnel confirmed that this should be possible. Otherwise, the direct costs will rise substantially, at the rate of $400K for each day that replacement power is needed.

For each resolution alternative, the costs noted in Subsections 4.3.1-4.3.7 were considered.

4.3.1 Direct Costs

This cost category includes factory purchases, installation and onsite labor and materials, but excludes indirect costs (e.g., engineering, administrative, etc.). It is given in the first column of Table 4.3.1 as a best estimate.

Table 4.3.2 shows the best estimate and the range of estimates in the direct cost. Alternative 5 (technical specification modifications including procedures and crosstie testing) shows a zero in the direct cost because this item was already included in Column 4 (technical specification costs) of Table 4.3.1.

4.3.2 Indirect Costs

The indirect costs are usually a certain fraction of the direct cost. As recommended in NUREG/CR-4627 (Ref. 9), 30% was used (the range is from 25% to 33% for engineering and quality assurance costs for in-place structures). Column 2 of Table 4.3.1 includes this cost component.

4.3.3 Operating and Maintenance Costs

Usually, these costs annually equal 3% of total "overnight" costs. Overnight costs represent the sum of total direct and indirect costs assuming that the modification was completed overnight (e.g., excluding the time costs of capital). To arrive at the total operating and maintenance (O&M) cost, the annual value was integrated and discounted over the remaining plant life

(30 years). Alternative 5 (modify Technical Specifications) was assumed not to involve any O&M costs. Column 3 of Table 4.3.1 includes this cost component. In calculating O&M costs, a 5 per cent discount rate was assumed, consistent with the KRC recommended practice.

4.3.4 Technical Specifications Costs

Each alternative involves modifying technical specifications to a certain extent. According to NUREG/CR-4627 (Ref. 9), these costs are $18K per reactor for a simple case and $35K per reactor for a complicated or controversial one. It was assumed that each alternative will result in a simple technical specification change. Neither choice includes the cost of a public hearing. The fourth column of costs in Table 4.3.1 includes this component of cost.

4.3.5 NRC Costs

NRC costs include the development and implementation costs. The development costs should be about $11K/reactor for a simple case and $21K/reactor for a complicated one. Neither case includes the cost of a public hearing. The former figure was chosen here. Operating costs would be incurred after the resolution's implementation and they would cover ensuring compliance with the new requirements. The operating costs have to be integrated and discounted, since they are recurring. The implementation and operating costs were estimated at $50K per reactor. Total NRC costs would then be $11K + 50K = $61K per reactor. Column 5 of Table 4.3.1 includes the NRC costs. For a technical specification and procedures change, the total NRC costs would be $21K per reactor (Ref. 9).

4.3.6 Averted Onsite Costs

Averted onsite costs are taken into account as cost offsets (Table 4.3.3) to the calculated cost of the proposed resolution alternatives, consistent with NRC policy. Table 4.3.4 lists the averted consequences. It can be seen that the onsite personnel exposure per accident will be low, compared to the offsite exposure, and other onsite consequences, so this component was not considered further. The numbers are from NUREG/CR-3568 (Ref. 11) as best estimate numbers. Averted onsite exposure would be added to the offsite person-rem exposure as part of the benefits, but the effect is negligibly small. For cleanup and replacement power, the integrated and discounted costs is then multiplied by the ΔCDF to arrive at the offset cost of each alternative. The cleanup and replacement power costs were calculated as follows:

$$u = (C_c + C_r)\,\frac{1}{r^2}\,(1-e^{-r\Delta t})\,(1-e^{-rm}) \quad \text{(Ref. 11)}.$$

where: u = integrated and discounted cost

C_c = cost of cleanup (\$100M/yr)
C_r = cost of replacement power (\$400K/day)
r = discount rate (0.05/yr)
Δt = remaining plant life (30 yr)
m = duration of cleanup/power replacement (10 yr)

Table 4.3.1 shows components of the total cost and the net cost for the best estimate case (the costs are per reactor). The net cost is the total cost minus the cost offset (from Table 4.3.3). If the net cost is negative, the alternative is cost-beneficial regardless of the cost benefit ratio. It should be noted that each column in Table 4.3.1 subsumes the cost item in the previous column and includes an additional indicated cost component. For instance, column "include indirect cost" includes the direct cost and the indirect costs of an alternative.

4.3.7 Range of cost Estimates

Table 4.3.5 presents the range of estimates obtained for the total cost (corresponding to Column 5 of Table 4.3.1) and the net cost (corresponding to Column 6 of Table 4.3.1). The low values were calculated by taking the lowest estimates in the data of various cost components (mainly direct costs) and carrying the computation through to the final number. The high values were calculated by taking the highest estimates in the data of the various cost components and carrying the computation though to the final number.

Table 4.3.1

Best Estimate Costs of Proposed Alternatives, ($ Per Reactor)

Column Number	1	2	3	4	5	6
Alternatives	Direct Cost	Include Indirect Cost	Include O&M Cost	Include Tech. Spec. Cost	Include NRC Cost- Total Cost	Include Onsite Conseq. Offset Net Cost
1. No Action	---	---	---	---	----	----
2. Additional Crosstie	$557K	$724K	$1.05M	$1.08M	$1.14M	$627K
3. Electrical Cross-Connection	$50K	$65K	$94K	$128K	$189K	-$246K
4. Separate Intake Structure	$29M	$38M	$55M	$55M	$55.1M	$52.3M
5. Technical Spec. Modifications	$0	$0	$0	$83K	$104K	-$684K
6. High Pressure Pump for RCP Seals	$5.9M	$7.7M	$11M	$11M	$11.1M	$8.8M
6a. Firewater for Thermal Barrier Cooling	$200K	$260K	$378K	$412K	$473K	-$1.9M

Table 4.3.2
Direct Cost Estimates ($ Per Reactor)

Alternatives		Low Estimate	Best Estimate	High Estimate
1.	No Action	----	----	----
2.	Additional Crosstie	250K	557K	1M
3.	Electrical Cross-Connection	50K	50K	50K
4.	Separate Intake Structure	7M	29M	38M
5.	Technical Specifications Modifications (see text)	0	0	0
6.	High Pressure Pump for RCP Seals	1M	5.9M	15M
6a.	Firewater for Thermal Barrier Cooling	127K	200K	273K

Table 4.3.3
Cost Offsets for Proposed Alternatives ($ per Reactor)

Alternatives		Cost Offset ($)
1.	No Action	----
2.	Additional Crosstie	513K
3.	Electrical Cross-Connection	435K
4.	Separate Intake Structure	2.75M
5.	Technical Specifications Modifications	788K
6.	Independent RCP Seal Cooling	2.34M
7.	Combination of Alternatives 5 & 6	2.73M

Table 4.3.4
Onsite Consequences

Type		Amount
Occupational Doses:		
-Immediate:	1,000 Person-Rem	
-Long Term:	20,000 Person-Rem	
Total	21,000 Person-Rem x 30 yr x $1,000/p-rem =	$6.3E+08 yr
Replacement Power		$1.8E+10 yr
Cleanup		$1.2E+10 yr
Total Onsite Consequences		$3.0E+10 yr*

*This number to be multiplied by Δ CDF for each alternative.

Table 4.3.5

Range of Estimates for the Total Cost and the Net Cost ($)

Alternatives	Total Cost			Net Cost		
	Low Estimate	Best Estimate	High Estimate	Low Estimate	Best Estimate	High Estimate
1. No Action	---	---	---	---	---	---
2. Additional Crosstie	550K	1.14M	2M	37K	627K	1.5M
3. Electrical Cross-Connection	173K	189K	205K	-262K	-246K	-230K
4. Separate Intake Structure	14M	55.1M	72M	11M	52.3M	69M
Technical Specifications Modifications	48K	104K	171K	-740K	-684K	-617K
6. High Pressure Pump for RCP Seal Cooling	2M	11.1M	29M	1.2M	8.8M	28.2M
6a. Firewater for Thermal Barrier Cooling	318K	473K	624K	-2M	-1.9M	-1.7M

5. VALUE/IMPACT ANALYSIS

The value/impact (V/I) methodology in analyzing the various alternatives examined under this study is based on the requirements of the backfit rule (10 CFR Part 50.109) and related implementing guidance contained in References 11, 12, and 13. One of the primary considerations here is the derivation of cost/benefit ratios for each alternative evaluated in terms of cost in $ per person-rem averted, which may be compared to a guideline such as $1,000/person-rem. This quantitative guideline is one of the elements considered in the decision-making process. Deterministic considerations on the merits of a proposed alternative resolution are also a part of the decision with respect to a given alternative (Chapter 6). In the following sections a description of each alternative and the results of a value/impact assessment are presented. Table 5.1 summarizes the results of this assessment for the various alternatives analyzed.

5.1 Alternative 1 - No Action

Under this alternative there would be no new regulatory requirements. Consistent with existing regulations, this alternative does not preclude a licensee, or an applicant for an operating license, from proposing to the NRC staff design changes intended to enhance the reliability/operability of the Essential Service Water System and its components on a plant-specific basis. Table 5.1 summarizes the results of this assessment for the various alternative analyzed.

5.2 Alternative 2 - Install Additional Crosstie

The ESW systems of the seven multi-unit sites analyzed under GI-130 are cross-connected through pipe connections and isolation valves. This arrangement allows the operator of one unit to utilize the ESW cooling capacity of the other unit. In most cases, the crosstie isolation valves can be remotely operated. A hardware failure to open the isolation valves, should the need arise, could result in adverse conditions. A parallel cross-connection could reduce the possibility of this kind of failure, and in addition would allow for more flexible maintenance options. The effects of the isolation valve failures on the CDF were not large due to the relatively low observed isolation valve failure rates indicating that other hardware components are more significant in reducing the overall system unavailability. The core damage frequency reduction of this alternative was estimated to be 1.6 E-05/RY.

The cost-benefit ratio for this alternative was calculated to be $433/person-rem, or $238/person-rem taking into account averted onsite costs.

5.3 Alternative 3 - Provide Electrical Power Cross-Connection

One of the observed contributors to the unavailability of the ESW system is related to the reliability of the electrical power supply and control system. Based on the data reported in Reference 3, the loss of the electrical power supply due to various causes was relatively high; however, the recovery times associated with these events indicate a relatively faster average recovery observed during losses of the ESW system.

In general, the electrical power supplies to the ESW trains are separated and have no cross-connection capability, i.e., Train A ESW pump cannot be powered from electrical Train B. This alternative therefore investigated the implementation of a cross-connection between the electrical trains of the unit with respect to the operation of the two ESW pumps (Trains A and B). The cross-connection of electrical power supply of other electrical components, such as MOVs was not considered as part of this alternative because of their less significant potential to risk contribution as observed in the operational data. It is envisioned that the electrical power cross-connection would be an exclusively manual operation. However, the possibility of adverse interactions between electrical trains A and B, such as the inadvertent transfer of faults from one train to the other, and hence, the loss of both trains, make this alternative of questionable value. Even if this contribution of possible adverse interactions between trains is set aside, the CDF reduction is not significant due to the relatively fast recovery observed during losses of electrical power. The cost-benefit ratio without taking into account the potential adverse interactions for this alternative was calculated to be $80/person-rem, and, if the averted onsite costs are taken into account, the net cost becomes negative, i.e. resulting in a net savings.

5.4 Alternative 4 - Provide Separate Intake Structure

A review of the failure modes of the intake structure indicates that one of the observed ESW failure mechanisms is the failure of certain intake components (such as travelling screens or strainers). This type of failure within the intake structure produces a general stopping or restricting the flow of cooling water to the plant. A separate intake structure, either located on the same body of water or using a different water source, would make a backup cooling capability available.

The intake structure is usually a single structure divided into separate bays by concrete walls. There are a number of screens installed to prevent the intake blockage by large foreign objects. The collapse or plugging of these screens may occur as a common mode failure due to the common inlet and/or common water source. The whole intake structure could also be affected by events such as flooding or freezing.

The alternative considered here is a completely separate intake structure serving as a redundant intake source of ESW. It may be located on the same water source, but on a physically separate location. An alternate design, which would provide additional independence/diversity, would be to install the additional intake structure on a physically separate water source. Naturally, there are sites where this would not be feasible.

The separate intake structure alternative includes the structure, screens and the associated motors, valves and piping. A swing ESW pump would be made available to either unit with redundant electrical power supplies. This arrangement is intended to reduce the probability of two failure mechanisms; one involving electrical supply failures, and the other involving operating failures of the ESW pumps. The additional ESW pump would be a swing pump serving either unit depending on the current needs of both units. This combination of a separate intake structure and additional swing pump with redundant electrical power supplies would affect a large fraction of the initiating event frequency related to the failure mechanisms involving the intake, the ESW pumps, and their power supplies.

The calculated reduction in CDF associated with this alternative was 9.13E-05/RY. The respective cost-benefit ratio was calculated to be $3,847/person-rem, and $3,651/person-rem taking into account averted onsite costs.

5.5 Alternative 5 - Modify Technical Specifications Requirements

There are certain operating modes, Modes 5 and 6 (shutdown and refueling modes respectively), that were examined with regard to specific requirements in the Technical Specifications (TS). In these operating modes the reactor is in shutdown condition and the status of its ESW pumps is uncertain. The TS do not require that any of the ESW pumps be operational in these modes. An implicit requirement is imposed on the ESW trains through the explicit requirement to operate the RHR system to remove decay heat.

In essence, the operator of the unit in shutdown may utilize the unit's own ESW pumps to provide the necessary heat removal function, but may just as well decide to use the unit crossties to supply ESW flow from the other unit. In the absence of any requirements on the ESW pumps, both pumps could be maintained or made inoperable at the same time. Although this is not a universal practice, certain modeling assumptions were made based on information gathered from plant sites representing a more conventional practice involving the administrative control of crosstie use, and the ESW pump maintenance schedule. In the basic analytical model it was assumed that the simultaneous shutdown of both ESW pumps could occur only randomly.

The unavailability of the Unit 2 ESW pumps to provide backup for the Unit 1 ESW system may be reduced by imposing an explicit operability requirement on at least one of the ESW pumps of Unit 2 while the latter is in Modes 5 and 6. An additional improvement is the testing of the unit-to-unit crosstie valves to provide greater assurance of operability. Also, this alternative includes additional credit for improvements in emergency procedures for recovering from a LOSW accident. The resulting CDF calculations indicated that the CDF would be reduced by 2.55E-05/RY. The respective cost-benefit ratio for this alternative was determined to be $25/person-rem, and, if the averted onsite costs are taken into account, the net cost becomes negative, i.e. resulting in a net cost savings.

5.6 Alternative 6 - Provide Independent RCP Seal Cooling System

The technical findings reported in Chapter 4 and Reference 3 indicate that the major contributor to the ESW-related component of CDF comes from the failure of the RCP seals following a loss of ESW. Specifically, the RCP seal LOCA sequence contributes about 60% of the total CDF attributable to ESW failures. Hence, if the likelihood of a RCP seal induced LOCA may be reduced, a proportionately significant reduction in CDF may be achieved.

This alternative provides for a dedicated seal cooling system that would continue to provide heat removal capability after a loss-of-ESW event. The cooling requirements of the RCP seals are relatively small, and a single small capacity high pressure pump capable of delivering about 50-100 gpm was judged to be sufficient. The pump may be driven either by an electric motor or, for electrical independence from the point of view of other accident scenarios (such as station blackout), a diesel-driven pump option may also be considered.

The single high pressure pump and diesel would provide flow via the cooling header to the four injection lines (one to each RCP seal). It was assumed that the pump and diesel would not require auxiliary cooling for the lube oil, bearings, etc., as the suction flow or air cooling would be sufficient to provide all their heat removal requirements. It was also assumed that the return flow from the RCP seals would not be recycled. In other words, a once-through cooling cycle would be used with a sufficient water supply to last about 8-10 hours.

It is assumed that a dedicated tank will be installed, with a capacity satisfying 8-10 hours of seal cooling. After this time, added cooling could be provided by other available water supplies, such as the refueling water storage tank.

In modeling the system, the following assumptions were made:

1. single high pressure pump, 50-100 gpm capacity,

2. dedicated water storage tank with capacity to last at least 8-10 hours,

3. ac-independent (non-seismic) diesel-driven pump,

4. no support system cooling required, and

5. once-through RCP seal heat removal.

Other design alternatives may also be considered utilizing arrangements different from that of the high pressure pump injection. One less costly alternative would provide flow through the RCP thermal barrier heat exchangers by connecting the firewater system into the CCW lines. Most firewater systems have one diesel-driven firewater pump which usually is independent of the ESW system.

The CDF reduction for this alternative involving a high pressure seal cooling system was calculated to be 7.82E-05. The respective cost-benefit ratio for this alternative involving a high pressure seal cooling system was calculated to be $862/person-rem, or $684/person-rem if the averted onsite costs are taken into account. The cost-benefit ratio for this alternative involving a connection to the fire water system for thermal barrier cooling was calculated to be $37/person-rem, or, if the averted onsite costs were taken into account, this alternative would result in a net cost savings.

5.7 Alternative 7-Combine Alternatives 5 and 6 (TS Changes and Independent RCP Seal Cooling)

As shown in Table 5.1, most of the analyzed alternatives have favorable cost-benefit ratios (presented as \$/person-rem). In these cost-benefit calculations, it was assumed that each of the alternatives (1 through 6a) was utilized individually and independently from the other alternatives.

For the combination case, the CDF reduction is calculated when two alternatives are combined and utilized together to reduce the risk due to the loss of ESW function. The alternative with the highest ΔCDF and favorable cost/benefit ratio was ranked first and served as the starting basis point. This was Alternative 6 (or 6a), the dedicated cooling system for the RCP seals. When the next alternative was considered, the CDF reduction was calculated from the case where Alternative 6 (or 6a) was already incorporated. The combined CDF reduction resulting from the implementation of alternatives 5 and 6 was calculated to be 9.12×10^{-5}/RY, and the respective cost-benefit ratio of \$756/person-rem, or \$574/person-rem with the averted onsite costs taken into account with a RCP seal cooling system involving a high pressure cooling system. The cost-benefit ratio for this combination of alternatives with a RCP thermal barrier cooling system utilizing the fire water supply was calculated to be \$39/person-rem, and if the averted onsite cost were taken into consideration a net gain would be achieved (i.e., a negative cost of implementation).

5.8 Uncertainty Analysis

This section discusses, the sources and treatment of uncertainty for the GI-130 study. Uncertainty is expressed as a quantitative bounding of the mean value. Uncertainty arises due to the selection of the data base used to determine parameter values, modeling assumptions, and completeness of the analysis.

Although a complete analysis of all data uncertainties was not conducted, uncertainty studies were performed on selected issues that were important to the results. Uncertainty data were gathered, evaluated, and reported in the form of distributions for these selected issues. This data-gathering and reduction was used to gauge the effects of the individual data uncertainty on the final core damage frequency results of the analysis.

The primary areas of uncertainty exist in the determination of the initiating frequency values, modelling and data uncertainties. Each of these particular areas were addressed and the final result combines these issues to present the uncertainty of the core damage frequency. All other parameters were treated as point-estimates.

The results of the uncertainty analysis show a mean value of CDF due to LO3W of 1.49E-4 per reactor-year, with a value of 5% and 95% of 3.99E-5/RY and 3.73E-04/RY, respectively.

5.9 Life Extension Considerations

The regulatory process by which license renewal may be accomplished is currently under development by the NRC. It is envisioned that a license renewal for an additional term of 20 years may be achievable based on satisfying specific requirements still to be established. Hence, for considerations regarding the effect of license renewal on the results of the evaluation of GI-130, a reanalysis of the cost-benefit ratio parameters for each backfit alternative was performed. The results of this reanalysis show that the benefits will increase by factor of 1.67, while the costs, both incurred and averted will increase by a factor of about 1.2 for most of the backfit alternatives analyzed.

Table 5.2 summarizes the cost-benefit ratios based on a license renewal of 20 years or a remaining plant life of 50 years. A comparison of these numbers with those listed in Table 5.1 shows that the cost-benefit ratios for all analyzed backfit alternatives are considerably lower for extended plant life of 50 years vis a vis a plant life of 30 years, corresponding to licenses in force currently.

Even though all alternatives listed in Tables 5.1 and 5.2 become more cost-effective with life extension, Alternative No. 4, Separate Intake Structure, still remains appreciably higher than the $1,000/person-rem guideline at a cost-benefit ratio of $2,285/person-rem.

Table 5.1

Best Estimate Cost-Benefit Ratios ($/Person-Rem)

Alternatives	Total Cost/Benefit	Net Cost/Benefit
1. No Action	---	---
2. Additional Crosstie	433	238
3. Electrical Cross-Connection	80	---*
4. Separate Intake Structure	3847	3651
5. Technical Specifications Modifications	25	---*
6. High Pressure RCP Seal Cooling	862	684
6a. Firewater for Thermal Barrier Cooling	37	---*
Combination of 6 and 5	756	574
7a. Combination of 6a and 5	39	---*

*Including averted onsite costs results in a net cost savings.

Table 5.2

Best Estimate Cost-Benefit Ratios ($/Person-Rem) for 20-year License Renewal

Alternatives	Total Cost/Benefit	Net Cost/Benefit
1. No Action	---	---
2. Additional Crosstie	271	133
3. Electrical Cross-Connection	50	---*
4. Separate Intake Structure	2421	2285
5. Technical Specifications Modifications	16	---*
6. High Pressure RCP Seal Cooling	541	412
6a. Firewater for Thermal Barrier Cooling	23	---*
7. Combination of 6 and 5	474	343
Combination of 6a and 5	24	---*

*Including averted onsite costs results in a net cost savings.

6. DECISION RATIONALE

This generic issue was identified as a consequence of the Byron Unit 1 evaluation with respect to its vulnerability to core-damage sequences in the absence of a crosstie from the ESW of Unit 2. This configuration existed because Unit 2 was under construction, and was eventually supplemented by the crosstie between units. There are fourteen units at seven sites having two service water pumps per unit (one per train) with a sharing of one pump between units via a crosstie between them, in a similar manner as currently in the two Byron units. It was decided to focus the attention of this study on these seven two-unit sites because the design of their ESW system was expected to show the most vulnerable configuration to risk-significant sequences. The remaining LWRs will be evaluated under GI-153, "Loss of Essential Service Water in LWRs."

As discussed in Chapter 5, most of the alternatives for reducing the risk associated with this issue would be cost-effective in meeting the $1,000/person-rem guideline. Furthermore, the objective of the GI-130 resolution is that the risk contributions from loss of the ESW system be reduced consistent with the backfit rule's two basic requirements that the improvement be both a substantial increase in protection, and be cost-effective.

A combination of potential improvements consisting of the installation of a dedicated RCP seal cooling system, and improvements in Technical Specifications with respect to ESW system operation, including crosstie testing and improvements in procedures, was shown to be capable of reducing the total CDF by 60% (to 6.1E-05/RY) in a cost-effective manner. Hence, this is deemed to meet the backfit rule.

The overall approach to arriving at the proposed resolution considered both the numerical results of the cost-benefit analysis and the spectrum and type of potential improvements available for potential risk reduction for loss of service water sequences. From the prevention perspective of a LOSW, it would be desirable to choose those alternatives which could reduce the number of occurrences of the LOSW initiators. From the mitigation perspective, it would be desirable to choose those alternatives which would help to reduce the consequences of a LOSW. The proposed resolution (Alternative 7) was selected to achieve some balancing of both these views; that is, the improvements in technical specifications would assist on the prevention side, while the improved emergency procedures and backup seal cooling would provide a blend of both prevention and mitigation capabilities.

The BNL analysis (Ref. 3) shows that after implementation of Alternative 7 there remains a residual component of CDF of 6.1E-05/RY due to ESW loss which, on the face of it, would tend to indicate the need for additional risk reduction.

We have reviewed this aspect of our evaluation of GI-130 and have concluded that additional improvements beyond Alternative 7 cannot be justified at this time based on the following considerations:

1. When the possibility of additional corrective measures (beyond Alternative 7) was considered, the resulting reduction in CDF was either too small (i.e., approached diminishing returns), or the cost/benefit ratio too high to be consistent with the backfit rule. The examination for added corrective measures focused on those systems which are dependent on ESW, and which performed a role in several of the more dominant event sequences. For example, the alternative of including a recommendation for a design change to make the Auxiliary Feedwater System (AFWS) independent of ESW cooling did produce a modest CDF reduction(CDF was reduced from 6.1E-05/RY to 4.8E-05/RY). Even further reduction is theoretically possible by removing dependence on ESW of each system and component, one-by-one until virtually complete independence is achieved; this is the ideal maximum reduction in vulnerability to LOSW; however it is judged that going further in this generic, representative plant calculation is pressing the limits of precision beyond what is warranted for plant-specific application to these 14 units. In addition, such an alternative (AFWS upgrade), would be applicable only to some of the plant sites evaluated under this issue; three of the seven sites are known to have already AFW systems independent of ESW cooling. In another case, Alternative 4, involving the installation of a separate intake structure and a swing pump to be shared by the two units, was determined to be capable of providing a substantial risk reduction, but was estimated to be not cost-effective.

2. As part of the implementation phase of resolving this issue, we recommend that the licensees/applicants of the fourteen plants evaluated under GI-130 perform a review of their respective plant-specific designs vis-a-vis the recommendations of Alternative 7, (combination of Alternatives 5 and 6 as discussed earlier in this chapter and in Chapter 5) and report, pursuant to 10 CFR 50.54(f), whether and how these recommendations would be implemented.

   This licensee/applicant effort would take into consideration the existing plant-specific design features, which, in some cases, would be different from those assumed in the generic model used in the evaluation of this issue. Hence, as a result of this effort, it is expected that individual licensees/applicants will submit a description of the measures taken as a result of the resolution of this generic issue, considering producing at least a comparable CDF reduction as has been calculated for the Alternative 7 combination in the GI-130 generic calculations. The results of some plant-specific PRA evaluations reported by EPRI in Reference 14 supports the view that plant-specific designs incorporating features recommended by the resolution of this generic issue would result in significant reductions in CDF. For some plants, the licensee or applicant may find it desirable or necessary to propose other design features, such as providing AFWS cooling independent of ESW, to improve on the assurance that the risk due to loss of ESW will result in a small fraction of the total risk for their individual plants.

3. A number of generic safety issues related to GI-130 have been in various stages of resolution, including some that have already been resolved. Their impact on GI-130 is as follows:

   o GI-23, "Reactor Coolant Pump Seal Failures" - This generic safety issue addresses the same possible improvements as Alternative 6 and, in part, Alternative 7 of GI-130. The evaluation of GI-23 has been completed and a proposed resolution has been reported (Ref 7).

An objective of the proposed resolution of GI-23 is to reduce the risk of severe accidents associated with RCP seal failure by reducing the probability of seal failures thus making it a relatively small contributor to total core-damage frequency. The proposed means of doing so entail the installation of a separate and independent cooling system for the RCP seals. Hence, implementation of the proposed GI-23 resolution could provide a substantial portion of the proposed GI-130 resolution. As such, the proposed resolution of GI-130 will be coordinated with the resolution of GI-23. (See Chapter 7, Implementation)

- GI-51, "Improving the Reliability of Open-Cycle Service Water Systems" - The resolution of this generic safety issue has been reported in August 1989 (Ref. 15) and its implementation began with the issuance of Generic Letter 89-13 (Ref. 16), and Supplement 1 (Ref. 17). The GI-51 implementation entails the implementation of a series of surveillance, control and test recommendations to ensure that the ESW systems of all nuclear power plants meet applicable licensing guidelines.

  During the review of the operational experience data for GI-130, credit was taken for corrective measures as a result of the GI-51 resolution by excluding those events that involved fouling of the ESW (sediment, biofouling, corrosion, etc.). Hence, there is no direct impact of GI-51 on GI-130.

- GI-153, "Loss of Essential Service Water in LWRs" is under prioritization review and expected to be assigned NRC staff resources (Ref. 18) for its resolution. Its purpose is to assess this issue for all LWRs not already covered by GI-130. Insights gained by the evaluation of generic safety issue 153 are expected to be useful in confirming and/or supplementing the technical findings of GI-130.

On the basis of the considerations discussed in Items 1-3 above and the technical findings of this study, including the value/impact analysis of Chapter 5, it is concluded that the combination of Alternatives 5 and 6, namely, the augmentation of technical specifications and procedures along with the installation of an independent RCP seal cooling backup system are the appropriate risk reduction measures that are recommended. These measures provide a substantial increase in overall protection of the public health and safety, and are cost-effective.

Of interest to the decision process on this generic issue are the insights and views available in related PRA documentation in the open literature. Although still not finalized, the preliminary PRA work available in NUREG-1150, "Severe Accidents Risks: An Assessment for Five U.S. Nuclear Power Plants" (plus supporting documentation) is a source of extensive risk analyses information one might turn to for an understanding of ESW vulnerabilities. An examination of the NUREG-1150 documentation of the three PWRs that were studied indicates that the analyst considered that the ESW redundancy for two of the three PWRs was large enough that a complete loss of ESW as an event-initiator was deemed not credible (eight pumps available in Sequoyah, Unit 1). None of the five plants in the NUREG-1150 study is a GI-130 plant; however, it is worthwhile to note that one of the PWRs (Zion) identified the service water contribution to risk to be substantial (approximately 1.5E-4/RY). This contribution for Zion was approximately 42% of the total core damage frequency due to all causes.

Another PRA work available in the open literature is NSAC--148, "Service Water Systems and Nuclear Plant Safety," dated May 1990. Although it is only a compilation of earlier PRA results for six plants performed by the industry, it is useful to note that a greater appreciation of the service water system's contribution to plant risk has moved the industry to initiate a program to improve service water performance. The limited guidance available in NSAC-148 is a step in the right direction. The wide range of core damage frequencies (due to LOSW) over the six plants studied suggests large variability in plant-specific ESW configurations. The average CDF due to LOSW for the six plants was 6.55E-05/RY, with a range of 2.33E-04/RY-to-"negligible" contribution. While many details of these six PRAs are not included in NSAC-148 and, therefore, must be considered to be used only with a great caution, the overall message that the service water system provides an important safety function which could be a substantial contributor to overall plant risk tends to lend added credence to the GI-130 conclusions.

7. IMPLEMENTATION

The staff proposes to implement the resolution of Generic Issue 130 by issuing a generic letter, under 10 CFR 50.54(f), to the licenses and applicants of the fourteen plants involved in this evaluation. The content of the generic letter will address both the preventive and mitigative aspects of the proposed resolution as discussed in Chapter 6. The implementation phase of Generic Issue 130 will be closely coordinated with that of Generic Issue 23, which deals with the RCP seal reliability for both normai operation and accident conditions. Specific guidance for resolving that generic issue is given in proposed Regulatory Guide DG-1008. While awaiting completion of public review and comment of Regulatory Guide DG-1008, the backup seal cooling portion of Alternative 7 (see Chapter 6) may be deferred. The reason for allowing the deferral of this additional protection relates to the earlier development and promulgation of 10 CFR 50.63, (Station Blackout Rule), which was based on an assumption regarding the magnitude of RCP seal leakage during a station blackout event. While it was left to GI-23 to validate that assumption, the enclosed GI-130 is also based on a seal LOCA model very similar to GI-23...but different from the leakage assumption in 10 CFR 50.63

8. REFERENCES

1. Memorandum from T. P. Speis to H. L. Thompson, "Safety Evaluation Report Related to the LCO Relaxation Program for the Byron Generating Station," January 15, 1986.

2. Cho, N. Z. et al., "Analysis of Allowed Outage Times at the Byron Generating Station," NUREG/CR-4404, June 1986.

3. P. Kohut, et al., "Analysis of Risk Reduction Measures Applied to Shared Essential Service Water Systems at Multi-unit Sites," NUREG/CR-5526, BNL, June 1990.

4. Lam, P. et al., " Operating Experience Feedback Report - Service Water System Failure and Degradation", NUREG-1275, Volume 3, November 1988.

5. U.S. Nuclear Regulatory Commission Report, 10 CFR Part 50, "Safety Goals for the Operations of Nuclear Power Plant," Policy Statement, dated August, 1986.

6. Memorandum from Edward L. Jordan (CRGR) to Eric S. Beckjord (RES) "Implementation of the Safety Goals," September 6, 1990.

7. Memorandum from C. J. Heltemes, Jr. to F. P. Gillespie, et al., "Proposed Resolution of GI-23, "Reactor Coolant Pump Seal Failures," June 20, 1990.

8. "Analysis of Core Damage Frequency from Internal Events: Expert Judgement Elicitation," NUREG/CR-4550, Volume 2, April 1989.

9. E. Claiborne et al., "Generic Cost Estimates," NUREG/CR-4627, Rev. 1, Science Engineering Associates, February 1989.

10. "Complete CONCISE Printout for Model 148-PWR-ME (A3-ME) Energy Economic Date Base (EEDB) Phase IX Update," volume 2 of 9, Plant Set 2 of 3, 7697.900, DOE Report 871102, (Philadelphia, PA; Oak Ridge, TN; United Engineers and Constructors, Inc.), 1987.

11. S. W. Heaberlin et al., "A Handbook for Value-Impact Assessment, NUREG/CR-3568, PNL, December 1983.

12. Memorandum from E. S. Beckjord to Distribution, RES Office Letter No. 2, "Procedures for Obtaining Regulatory Impact Analysis Review and Support," November 18, 1988.

13. Memorandum from E. S. Beckjord to Distribution, RES Office Letter No. 3, "Procedure and Guidance for the Resolution of Generic Issues", May 10, 1988.

14. "Service Water Systems and Nuclear Plant Safety" NSAC/148, May 1990, prepared by Pickard, Lowe and Garrick, Inc. for Nuclear Safety Analysis Center and Electric Power Research Institute.

15. Memorandum from E. S. Beckjord to J. M. Taylor, "Closeout of GI-51, "Improving the Reliability of Open-Cycle Service Water System'", August 10, 1989.

16. Generic Letter 89-13, "Service water System Problems Affecting Safety-Related Equipment", July 18, 1989.

17. Generic Letter 89-13, Supplement 1, "Service Water System problems Affecting Safety-Related Equipment", April 4, 1990.

18. Memorandum from K. Kniel to C. E. Ader, "Request for Prioritization of New Generic Safety Issue 'Loss of Essential Service Water in LWRs'," May 2, 1990.

ENCLOSURE 2

DRAFT GENERIC LETTER

TO: Licensees and Applicants of the Following Pressurized Water Reactor Nuclear Power Plants.

1. Braidwood Units 1 & 2.
2. Byron Units 1 & 2.
3. Catawba Units 1 & 2.
4. Comanche Peak Units 1 & 2.
5. Cook Units 1 & 2.
6. Diablo Canyon Units 1 & 2.
7. McGuire Units 1 & 2.

SUBJECT: Request for Action Related to the Resolution of Generic Issue 130, "Essential Service Water System Failures at Multi-Unit Sites," Pursuant to 10 CFR 50.54 (f) - Generic Letter 90-XX.

As a result of the technical resolution of Generic Issue 130, "Essential Service Water System Failures at Multi-Unit Sites," the NRC has concluded that certain corrective measures to increase the availability of the Essential Service Water (ESW) system and/or mitigate the effects of its loss are recommended as cost-justified safety enhancements under the backfit rule (10 CFR 50.109).

The ESW system is important in maintaining plant safety during power operation, shutdown, and accident conditions. As part of our evaluation of Loss of Essential Service Water (LOSW), extensive analyses of this issue were performed at the Brookhaven National Laboratory (BNL). The technical findings of this effort at BNL are reported in NUREG/CR-5526 "Analysis of Risk Reduction Measures Applied to Shared Essential Service Water Systems at Multi-unit Sites." In addition, the NRC staff performed a regulatory analysis to determine the efficacy of certain corrective measures analyzed by BNL. The staff's regulatory analysis is contained in NUREG-1421., "Regulatory Analysis for the Resolution of Generic Issue 130: Essential Service Water System Failures at Multi-unit Sites." Based on the results of our evaluation of this generic safety issue, we have determined that the following recommendations should be implemented to resolve this issue:

1. Modification of Technical Specifications and Procedures. Under this recommendation, the technical specification changes contained in Enclosure 1 to this generic letter, should be implemented. In addition, specific improvements in emergency procedures should address accident management of a LOSW using existing design features for recovery. Procedural improvements in the following areas are recommended:

   o procedures to operate and maintain HPI pump integrity in the event of loss of RCP seals as a result of ESWS failure.

   o procedures to test and manipulate the crosstie during loss of ESWS accident.

The recommendations to incorporate technical specification (TS) requirements for the resolution of GI-130 are considered to be consistent with the Commission's Policy Statement on Technical Specification Improvements. This policy statement captures existing requirements under Criterion 3 (Mitigation of Design-Basis Accidents or Transients) or under the provisions to retain requirements that operating experience and probabilistic risk assessment show to be important to the public health and safety. Specifically, General Design Criteria 14, 44, 45 and 46 of 10 CFR 50, Appendix A, in conjunction with the probabilistic risk assessment performed under GI-130, form the technical bases for the recommended TS improvements.

2. Installation of an Independent RCP Seal Cooling System. Under this recommendation, a dedicated Reactor Coolant Pump (RCP) seal cooling system should be installed to provide heat removal capability to the RCP seals after loss of ESW. The specific features of an acceptable backup, dedicated RCP seal cooling system are as follows:

   - single high pressure pump, 50-100 gpm capacity,
   - dedicated water storage tank with capacity to last at least 8-10 hours,
   - ac-independent (non-seismic) pump,
   - no support system cooling required, and
   - once-through RCP seal heat removal.

Limited plant-specific information obtained through the existing literature (FSAR's, etc), site visits, or discussions with licensees have indicated that a number of the units covered by GI-130 already have plant-unique features which could be responsive to the generic resolution sought by the NRC staff. At this time, rather than attempting to perform a series of PRAs tailored to each of the 14 units, this Generic Letter will allow each licensee or applicant to identify the plant-specific feature (if any) which could be credited with departing from the generic (representative) base case plant configuration modelled in NUREG/CR-5526. In addition, other design alternatives may also be considered utilizing arrangements different from that of the high pressure pump injection. One such alternative would provide flow through the RCP thermal barrier heat exchangers by connecting the firewater system into the CCW lines. Most firewater systems have one diesel-driven firewater pump which usually is independent of the ESW system.

Generic Issue 23, "Reactor Coolant Pump Seal Failures" deals with this recommendation also, and specific guidance for resolving that generic issue is given in proposed Regulatory Guide DG-1008. While awaiting completion of public review and comment of Regulatory Guide DG-1008, item 2 of the foregoing requirements may be deferred. The reason for allowing the deferral of item 2 relates to the earlier development and promulgation of 10 CFR 50.63 (Station Blackout Rule), which was based on an assumption regarding the magnitude of RCP seal leakage during a station blackout event While it was left to GI-23 to validate that assumption, the resolution of GI-130 is also based on a RCP seal failure LOCA model very similar to that of GI-23...but different from the leakage assumption in 10 CFR 50.63.

Each licensee and applicant addressed in this letter should review the aforementioned recommendations and determine applicability to their respective facilities. To determine whether any license or construction permit for facilities covered by this request should be modified, suspended or revoked, we require, pursuant to Section 182 of the Atomic Energy Act and 10 CFR 50.54 (f), that you provide the NRC, within 180 days of the date of this letter, a certification as to whether you will implement the aforementioned recommendations and, if so, that you provide a schedule for implementation and the basis for the schedule. In addition, you should supply a summary description of your intended course of action for each of the subject recommendations. If you do not intend to implement these recommendations, you should provide the reasons why you do not intend to do so.

The recommendations of this letter should be fully implemented by the end of the first refueling outage that starts 6 months or later from the date of this letter. Your response should also indicate whether you will implement these recommendations in accordance with the recommended schedule. If you are not going to comply with the recommended schedule, your response should contain an alternate schedule and the reasons for it.

This information shall be submitted to the NRC, signed under oath and affirmation. Each licensee or applicant should retain supporting documentation consistent with the records retention program for their facility.

A justification for this information request has been prepared in accordance with the requirements of 10 CFR 50.54(f). This justification analysis concludes that the information requested is justified in view of the safety significance of Generic Issue 130. This justification analysis is contained in Enclosure 2.

The actions proposed by the NRC staff in the technical resolution of GI-130 represent a new interpretation of existing regulatory guidelines for some licensees, and this request is considered a backfit in accordance with NRC procedures. This backfit is a cost-justified safety enhancement. Therefore, a backfit analysis of the type described in 10 CFR 50.109 (a)(3) and 10 CFR 50.109(c) was performed, and a determination was made that there will be a substantial increase in overall protection of the public health and safety and that the costs are justified in view of this increased protection. The backfit analysis is contained in Enclosure 3.

This request is covered by Office of Management and Budget Clearance Number _____ which expires _________. The estimated average burden hours is 100 person-hours per licensee response, including assessment of the new recommendations, searching data sources, gathering and analyzing the data, and the required reports. These estimated average burden hours pertain only to these identified response-related matters and do not include the time for actual implementation of the requested actions. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch, Division of Information Support Services, Office of Information Resources Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555; and to the Paperwork Reduction Project ( ), office of Management and Budget, Washington, DC 20503.

A list of recently issued NRC generic letters is enclosed for your information.

If you have any questions on this matter, please contact your project manager.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor
Regulation

Enclosures:
1. Draft Technical Specifications 3/4.7.4
2. Justification Analysis - 10 CFR 50.54(f)
3. Backfit Analysis
4. List of Recently Issues NRC Generic Letters

ENCLOSURE 1

DRAFT TECHNICAL SPECIFICATIONS

PLANT SYSTEMS

3/4.7.4 ESSENTIAL SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two indpendent service water loops per unit and the crosstie between the service water systems of each unit (as applicable) shall be operable. In addition, the crosstie shall be capable of being opened [from the main control room] as a flow path between the two units.

APPLICABILITY: MODES 1,2,3, and 4.

ACTION:

A. Both units in modes 1,2,3, or 4.

1. With one service water loop per unit OPERABLE, restore at least two loops per unit to OPERABLE status within 72 hours or, for the unit with the inoperable service water loop be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With one [or both] of the crosstie valve(s) INOPERABLE and not capable of being opened [from the control room], within 72 hours restore the valve(s) to OPERABLE status or open the affected valve(s), and maintain the affected valve(s) open; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

B. One unit in Modes 1,2,3, or 4 and one unit in Mode 5 or 6.

1. Verify that at least one pump in the shutdown unit is OPERABLE and available to provide service water to the operating unit. If neither service water pump in the shutdown unit is OPERABLE, restone at least one pump to OPERABLE status within 72 hours, or place the operating unit in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With one service water loop in the operating unit INOPERABLE, restore two loops in the operating unit to

PLANT SYSTEMS

OPERABLE status within 72 hours or be in at least STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. With one [or both] of the crosstie valve(s) INOPERABLE not capable of being opened [from the control room], within 72 hours restore the valve(s) to OPERABLE status or open the affected valve(s), and maintain the affected valve(s) open; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 Two service water loops per unit shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 92 days by cycling crosstie valves and/or verifying that valves are locked open with power removed; and

c. At least once per 18 months during shutdown, by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on a _______ test signal,

2) Each Service Water System pump starts automatically on a _______ test signal, and

3) Each crosstie valve is cycled or is locked open with power removed.

PLANT SYSTEMS

BASES

3/4.7.4 ESSENTIAL SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

In the event of a total loss of service water in one unit of a two-unit site where backup cooling capacity is available via a crosstie between the two units, the OPERABILITY of the unit crosstie along with a service water pump in the shutdown unit ensures the availability of sufficient redundant cooling capacity for the oeprating unit. These limiting conditions will ensure a significant risk reduction as indicated by the analyses of a loss of service water system accident. The surveillance requirements ensure the short and long-term operability of the service water system and crosstie between the two units. The service water system crosstie between the two units consists of appropriate piping, valves and instrumentation cross-connecting the discharge of the service water pumps of the two units. By operating the crosstie, the supply of additional redundant cooling capacity from one unit is available to the service water system of the other unit.

ENCLOSURE 2

10 CFR 50.54(f) JUSTIFICATION ANALYSIS
FOR GENERIC LETTER ON GENERIC ISSUE 130

10 CFR 50.54(f) requires that "... the NRC must prepare the reason or reasons for each information request prior to issuance to ensure that the burden to be imposed on respondents is justified in view of the potential safety significance of the issue to be addressed in the requested information." Further, Revision 4 of the Charter of the Committee to Review Generic Requirements (CRGR), dated April 1989 specifies that, at a minimum, such an evaluation shall include:

a. A problem statement that describes the need for the information in terms of potential safety benefit,

b. The licensee actions required and the cost to develop a response to the information request, and

c. An anticipated schedule for NRC use of the information.

The staff's 10 CFR 50.54(f) evaluation of the information request addressing the above elements follows:

a. A problem statement that describes the need for the information in terms of potential safety benefit.

   The recommended resolution of Generic Issue 130 (GI-130), "Essential Service Water System Failures at Multi-Unit Sites," applies to fourteen reactor units at seven sites and indicates that Essential Service Water System (ESWS) failures at these plants are a significant contributor to the overall plant risk. As a consequence of these technical findings, and based on the cost/benefit analyses performed, the staff is recommending that these fourteen plants should: (1) modify technical specifications and procedures for the ESWS, and (2) install a dedicated, backup RCP seal cooling system to provide seal cooling for 8-10 hours following loss of ESW.

Generic Issue 23, "Reactor Coolant Pump Seal Failures" deals with recommendation No. 2 above also, and specific guidance for resolving that generic issue is given in proposed Regulatory Guide DG-1008. While awaiting completion of public review and comment of proposed Regulatory Guide DG-1008, recommendation No. 2 above may be deferred.

The reason for allowing this deferral of recommendation No. 2 relates to the earlier development and promulgation of 10 CFR 50.63 (Blackout Rule), which was based on an assumption regarding the magnitude of RCP seal leakage during a station blackout event. While it was left to GI-23 to validate that assumption, the resolution of GI-130 is also based on a RCP seal failure LOCA model very similar to that of GI-23, but different from the leakage assumption in the Blackout Rule.

The estimated benefit from implementation of the two recommendations is a reduction in the core damage frequency per reactor-year, and a reduction in the associated risk of off-site radioactive releases due to ESW failure. The risk reduction to the public (per plant) is estimated to be 14,821 person-rem (best-estimate numbers used) and supports the conclusion that the proposed items provide a substantial increase in the overall protection of the public health and safety. Also, the direct and indirect costs of implementation are justified in view of this increased protection.

As discussed in NUREG-1421, when considered individually, most of the alternatives analyzed for reducing the risk associated with this issue would be cost-effective in meeting the $1,000/person-rem guideline. The objective of the GI-130 resolution is that the risk contributions from loss of the ESW system be reduced consistent with the backfit rule's two basic requirements that the corrective alternatives be both substantial and cost-effective.

A combination of potential improvements consisting of the installation of a dedicated RCP seal cooling system, and improvements in Technical Specifications and procedures, was shown to be capable of reducing the CDF due to loss of ESW (1.5E-04/RY) by 60% in a cost-effective manner. Hence, this is deemed to meet the backfit rule.

The overall approach to arriving at the recommended resolution considered both the numerical results of the cost-benefit analysis and the spectrum and type of potential improvements available for potential risk reduction for loss of service water sequences. Desirable for the prevent perspective of a LOSW would be those alternatives which could reduce the number of occurrences of the LOSW initiators. Desirable from the mitigation perspective would be those alternatives which would help to reduce the consequences of a LOSW. Although there is uncertainty in the cost/benefit analysis due to the range of core damage uncertainty resulting from initiating event frequencies, seal failure modelling, release consequences, and recovery models, the proposed resolution (identified as Alternative 7 in NUREG-1421) is concluded to achieve some balance of both these views. That is, the improvements in technical specifications would assist on the prevention side, while the improved emergency procedures and backup seal cooling would provide a blend of both prevention and mitigation capabilities.

The conclusion of our analysis is that a substantial increase in the protection of the public health and safety will be derived from backfitting of the ESW system improvements, and that the backfit is justified in view of the favorable cost/benefit ratios. Hence, in view of the safety significance of the recommended resolution of GI-130, the issuance of generic letter under 10 CFR 50.54(f) is justified. (See also item b below).

b. The licensee actions required and the cost to develop a response to the information request.

All the recipient licensees/applicants of this generic letter would be requested to review the recommended backfits of GI-130 and determine the degree to which these backfits are applicable to their respective facilities. They would also be asked to submit either a commitment to implement the NRC staff recommendations within the recommended schedule, or provide alternative resolutions and/or alternative implementation schedules along with a discussion of the reasons for proposed alternative resolutions and related implementation schedules.

We estimate that the cost of reviewing and evaluating the contents of this generic letter and preparing a response will cost no more than $5,000 per licensee/applicant. It is expected that the cost may vary from site-to-site depending on the degree to which the recommended TS and procedures apply to individual cases. This cost is insignificant compared to the expected costs of the recommended cost-justified backfits (see cost estimates presented in NUREG-1421), which represent a substantial safety improvement.

c. An anticipated schedule for the NRC use of the information.

We expect that most, if not all, responses to this generic letter would be submitted within the schedule recommended by the NRC staff, and that NRC staff review of the responses will be completed within 180 days from their receipt.

ENCLOSURE 3

BACKFIT ANALYSIS (REF. 10 CFR 50.109)

FOR

GENERIC ISSUE 130

A. 1 INTRODUCTION

This enclosure presents the backfit analysis for Generic Issue 130 (GI-130), "Essential Service water System Failures at Multi-Unit Sites." The technical findings for GI-130 are presented in NUREG/CR-5526. The regulatory analysis is presented in NUREG-1421. The studies apply to fourteen reactor units at seven sites and indicate that Essential Service Water System (ESWS) failures at these plants are a significant contributor to the overall plant risk. As a consequence of these technical findings, and based on the cost/benefit analyses performed, the staff is proposing that these fourteen plants should: (1) modify Technical Specifications and procedures for the ESWS, and (2) install a dedicated, backup RCP seal cooling system to provide seal cooling for 8-10 hours following loss of ESW.

Generic Issue 23, "Reactor Coolant Pump Seal Failures" deals with recommendation No. 2 above also, and specific guidance for resolving that generic issue is given in proposed Regulatory Guide DG-1008. While awaiting completion of public review anc comment of proposed Regulatory Guide DG-1008, recommendation No. 2 above may be deferred. The reason for allowing this deferral of recommendation No. 2 relates to the earlier development and promulgation of 10 CFR 50.63 (Blackout Rule), which was based on an assumption regarding the magnitude of RCP seal leakage during a station blackout event. While it was left to GI-23 to validate that assumption, the resolution of

GI-130 is also based on a RCP seal failure LOCA model very similar to that of GI-23, but different from the leakage assumption in the Blackout Rule.

The estimated benefit from implementation of the two recommended items is a reduction in the core damage frequency per reactor-year, and a reduction in the associated risk of off-site radioactive releases due to ESW failure. The risk reduction to the public (per plant) is estimated to be 14,821 person-rem (best-estimate numbers used) and supports the conclusion that the proposed items provide a substantial increase in the overall protection of the public health and safety. Also the direct and indirect costs of implementation are justified in view of this increased protection.

As discussed in NUREG-1421, when considered individually, most of the alternatives analyzed for reducing the risk associated with this issue would be cost-effective in meeting the $1,000/person-rem guideline. The objective of the GI-130 resolution is that the risk contributions from loss of the ESW system be reduced consistent with the backfit rule's two basic requirements that the corrective alternatives be both substantial and cost-effective.

A combination of potential improvements consisting of the installation of a dedicated RCP seal cooling system, and improvements in Technical Specifications and procedures, was shown to be capable of reducing the total CDF by 60% to 6.1E-05/RY in a cost-effective manner. Hence, this is deemed to meet the backfit rule.

The overall approach to arriving at the proposed resolution considered both the numerical results of the cost-benefit analysis and the spectrum and type of potential improvements available for potential risk reduction for loss of service water sequences. Desirable for the prevention perspective of a LOSW would be those alternatives which could reduce the number of occurrences of the LOSW initiators. Desirable from the mitigation perspective would be those alternatives which would help to reduce the consequences of a LOSW. Although there is uncertainty in the cost/benefit analysis due to the range of core damage uncertainty resulting from initiating event frequencies, seal failure modelling, release consequences, and recovery models, the proposed resolution (identified as Alternative 7 in NUREG-1421) is concluded to achieve some balance of both these views. That is, the improvements in technical specifications would assist on the prevention side, while the improved emergency procedures and backup seal cooling would provide a blend of both.

The conclusion of this backfit analysis is that a substantial increase in the protection of the public health and safety will be derived from backfitting of the ESW system improvements, and that the backfit is justified in view of the favorable cost/benefit ratios. In the following sections of this backfit analysis, the nine factors stipulated by 10 CFR 50.109(c) to be used in the determination of backfitting are addressed.

A2 ANALYSIS OF 10 CFR 50.109(c) FACTOR FOR "ALTERNATIVE 7"

A.2.1 Objective

The objective of Alternative 7 of the proposed backfit is to improve the performance of the ESW system by providing a blend of both prevention and mitigation capabilities. This backfit will be applicable to all the PWR plants (14 Units) covered by the GI-130 issue.

A.2.2 Licensee Activities

To implement "Alternative 7", each licensee would (1) provide an independent system for RCP seal cooling, and (2) modify Technical Specifications by incorporating a periodic test of the existing ESW crosstie between units, and by imposing an explicit operability requirement on Unit 1 that at least one of the other unit's (Unit 2) ESW pumps be available while Unit 2 is in Mode 5 or 6; implement emergency procedures for recovering from a LOSW. However, as discussed earlier, deferral of Part (1) entailing the installation of a backup system for RCP seal cooling will be allowed pending resolution of GI-23.

A.2.3 Public Risk Reduction

Backfitting in accordance with the provision of the proposed alternative will yield a reduction in public risk incident from the accidental off-site release of radioactive materials of 14,821 person-rem (best-estimate) for each of the 14 plants covered by this study and with an average remaining life of 30 years. The proposed alternative will also reduce the CDF/RY by about 60 percent from 1.52E-04 to 6.08E-05.

A.2.4 Occupational Exposure

As indicated in NUREG-1421, the radiological operational exposure is negligible and therefore the implementation of Alternative 7 will not result in any increase in the radiological exposure to facility employees.

A.2.5 Installation costs

The best-estimate total costs per reactor associated with Alternative 7 (combination of Alternatives 5 and 6) are 11.2 million dollars.

The total cost could be substantially reduced if the fire water supply were used to allow backup thermal barrier cooling instead of a dedicated high pressure pump for RCP seals, ie., the combination of Alternatives 5 and 6a. The best-estimate total cost per reactor for this combination (Alternatives 5 and 6a) is $577K.

2.6 Potential Safety Impact

A number of generic safety issues related to GI-130 have been in various stages of resolution, including some that have already been resolved. Their relation to GI-130 is as follows:

- GI-23, "Reactor Coolant Pump Seal Failures" - This generic safety issue addresses the same possible improvements as Alternative 6 and, in part, Alternative 7 of GI-130. The evaluation of GI-23 has been completed and a proposed resolution has been reported.

  An objective of the proposed resolution of GI-23 is to reduce the risk of severe accidents associated with RCP seal failure by reducing the probability of seal failure, thus making it a relatively small contributor to total core-damage frequency. The proposed means of doing so entail the installation of a separate and independent cooling system for the RCP seals. Hence, implementation of the proposed GI-23 resolution could provide a substantial portion of the proposed GI-130 resolution. As such, the proposed resolution of GI-130 is coordinated with the resolution of GI-23, by allowing the installation of a backup RCP sealing cooling system to be deferred pending resolution of GI-23.

- GI-51, "Improving the Reliability of Open-Cycle Service Water Systems" - The resolution of this generic safety issue has been reported in August 1989 and its implementation began with the issuance of Generic Letter 89-13 and Supplement 1. The GI-51 implementation entails the implementation of a series of surveillance, control and test requirements to ensure that the ESW systems of all nuclear power plants are in compliance with all applicable licensing requirements.

During the review of the operational experience data of GI-130, credit was taken for corrective measures as a result of the GI-51 resolution by excluding those events that involved biofouling of the ESW. Hence, there is no direct impact of GI-51 on GI-130.

- o GI-153, "Loss of Essential Service Water in LWRs" is under prioritization review and expected to be assigned NRC staff resources for its resolution. It's purpose is to assess this issue for all LWRs not already covered by GI-130. Insights gained by the evaluation of generic safety issue 153 are expected to be useful in confirming and/or supplementing the technical findings of GI-130.

Of interest to the decision process on this generic issue are the insights and reviews available in related PRA documentation in the open literature. The PRA work available in NUREG-1150," Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" (plus supporting documentation) is a source of extensive risk analyses information one might turn to for an understanding of ESW vulnerabilities. An examination of the NUREG-1150 documentation of the three PWR's that were studied indicates that the analyst considered that the ESW redundancy for two of the three PWRs was large enough that a complete loss of ESW as an event-initiator was deemed not credible (eight pumps available in Sequoyah, Units 1&2). None of the five plants in the NUREG-1150 study is a GI-130 plant; however, it is worthwhile to note that one of the PWRs (Zion) identified the service water contribution to risk to be substantial (approximately 1.5E-04/RY). This contribution for Zion was approximately 42% of the total core damage frequency due to all causes.

Another PRA work available in the open literature is NSAC-148, "Service Water Systems and Nuclear Plant Safety," dated May 1990. Although only a compilation of earlier PRA results for six plants performed by the industry, it is useful to note that a greater appreciation of the service water system's contribution to plant risk has moved the industry to initiate a program to improve service water performance. The limited guidance available in NSAC-148 is a step in the right direction. The wide range of core damage frequencies (due to LOSW) over the six plants studied suggests the large variability in plant-specific ESW configurations. The average CDF due to LOSW for the six plants was 6.55E-05/RY, with a range of 2.33E-04/RY-to-"negligible" contribution. While many details of these six PRAs are not included in NSAC-148 and, therefore, must be considered to be used only with great caution, the overall message that the service water system provides an important safety function which could be substantial contributor to overall plant risk tends to lend added credence to the GI-130 conclusions.

A.2.7 NRC Cost

Implementation of Alternatives 7 is estimated at $61,000 (best estimate), and $100,000 (best estimate), depending on which seal cooling method is selected (per unit). These estimates assume minimal resources for review of the generic letter responses.

A.2.8 Facility Difference

Alternative 7 is applicable to all fourteen plants covered by this study regardless of age or design. Other PWR and BWR plants which are not included under the resolution of GI-130 will be evaluated under GI-153, "Loss of Essential Service Water in LWRs."

A.4.9 Terms of Requirement

The ESW system resolution is the final resolution of GI-130; it is not an interim measure.

ENCLOSURE 3

NUREG/CR-5526
BNL-NUREG-52225

Analysis of Risk Reduction Measures Applied to Shared Essential Service Water Systems at Multi-Unit Sites

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ABSTRACT

This report summarizes a study performed by Brookhaven National Laboratory for the U.S. Nuclear Regulatory Commission in support of the resolution of NRC Generic Issue 130. GI-130 is concerned with the potential core damage vulnerability resulting from failure of the emergency service water (ESW) system in selected multiplant units. These multiplant units are all twin pressurized water reactor designs that have only two ESW pumps per unit (one per train) backed up by a unit to unit crosstie capability. This generic issue applies to seven U.S. sites (14 plants).

The study established and analyzed the core damage vulnerability and identified potential improvements for the ESW system. It obtained generic estimates of the risk reduction potential and cost effectiveness of each potential improvement. The analysis also investigated the cost/benefit aspects of selected combinations of potential improvements.

EXECUTIVE SUMMARY

This report summarizes a study performed by Brookhaven National Laboratory for the U.S. NRC in support of the resolution of NRC Generic Issue 130. Generic Issue 130 is concerned with the core melt vulnerability caused by the failure of the Essential Service Water (ESW) system in a selected class of multi-unit pressurized water reactor plant sites that have only two ESW pumps per unit (one per train) backed up by a unit to unit crosstie capability. This generic issue applies to seven U.S. sites (14 plants).

The main objectives of this study were to develop a generic model that essentially enveloped the designs of the seven sites, to establish the overall core damage vulnerability based upon this generic model, to identify potential improvements for the ESW systems and to obtain generic estimates of each identified improvement's risk reduction potential and cost effectiveness. The specific design arrangements of each of the multi-unit plants were surveyed and classified with regard to the applicability of the potential improvements.

A generic initiating event frequency was established for the loss of ESW (LOSW) event and was extrapolated to multi-unit system and plant configurations. The LOSW initiating frequency for the most common operating configuration (both units at power and each unit has one ESW train in the run mode and one in the standby mode) was calculated as 1.1×10^{-3}/reactor year including both direct and indirect causes.

The core damage model that was constructed was done in a way that allowed investigation of different operating modes as well as ESW system arrangements. Sequence-specific recovery actions and potential operator actions were also considered. A time- and leak-rate dependent reactor coolant pump (RCP) seal leakage model was developed (based upon already completed work in NUREG-1150) and a time dependent ESW system recovery model was also developed. The total core damage frequency was calculated as 1.52×10^{-4}/reactor year.

Based on the identified failure modes, a number of different potential improvements were considered. The most effective option identified was a dedicated RCP seal cooling system that was independent of service water. This feature was shown to be capable of reducing the total service water-related CDF by about a factor of two with apparent favorable cost/benefit aspects.

When considered in combination, another cost-effective alternative was shown to be a dedicated RCP seal cooling system coupled with stricter Technical Specification controls on the service water system. This particular combination was shown to be capable of reducing the CDF to 6.1×10^{-5}/reactor year on a cost effective basis.

The costs for each option considered were obtained from industry sources and NRC handbooks. Each estimated cost had a range associated with it. The benefits were obtained by calculating averted consequences, assuming an "average" consequence model. The average consequence model was constructed by considering loss of ESW consequences for Zion, Surry and Sequoyah and then adapting these numbers to an average GI-130 plant by considering the containment strength and site characteristics of the GI-130 plants versus the reference plants. A range of consequences was also obtained. A best-estimate cost-benefit ratio for each improvement was derived and range of cost-benefit ratios was also calculated.

The most significant findings of the this study are:

- By quantifying the enveloping model for these seven sites, the service water-related CDF was determined to be 1.52×10^{-4}/reactor year. This is considered to be essentially upper bound. Any of the seven sites that have additional design features/enhancements would exhibit a lower service water related contribution to the overall plant CDF.

- A unique combination of potential improvements, namely, the installation of a dedicated RCP seal cooling system and stricter/more explicit Technical Specifications (with respect to ESW system operation including crosstie testing and improvements in procedures) was shown to be capable of reducing the service water-related CDF by ~60% to 6.1×10^{-5}/year. The cost/benefit aspect of this combination was shown to be well below the $1000/person rem ratio normally applied in such cases as indicated in Table 1.

Table 1
Option Cost/Benefit Values

Option	Total $/Person Rem (Point Estimate)
a) Dedicated cooling for RCP Seals (Firewater)	37
b) Changing Technical Specifications*	53
c) Combination of a) and b)	39

*Includes additional TS requirements, testing of the crosstie valves, and improvements in procedures.

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1. INTRODUCTION

1.1 Scope/Objective

Generic Issue 130, "Essential Service Water System Failures at Multiplant Sites," is concerned with the core damage vulnerability associated with the failure of the essential service water (ESW) system in selected multi-unit pressurized water reactor plants. This specific group of plants (7 sites -14 plants) is characterized by having only two ESW pumps per unit and relying on additional backup capability through a crosstie to the second unit. This issue evolved as a result of the BNL review[1] of the Byron Unit 1 Limiting Conditions for Operation (LCO) Relaxation Program. In the initial absence of the availability of the Byron Unit 2 ESW system via crosstie between the two units, the core damage vulnerability for Unit 1 was found to be significant. The insights derived from the BNL PRA study to support extending the allowable out-of-service time of the ESW systems of Byron 1 and 2 indicated that the contribution to core damage frequency due to the unavailability of the ESW system could be significantly higher than was previously expected. Specific measures were taken at that time to address Byron Unit 1. This study was commissioned to investigate the above described multi-unit service water system design to identify any significant vulnerabilities.

In general, ESW systems are highly plant-specific in terms of plant equipment, crosstie capability, operational and functional requirements. The ESW system typically supports most if not all of the frontline safety systems that are required for safe shutdown of the plant. The ESW system in most plants also provides cooling water to non-safety related components and systems during normal plant operations. Only one train is typically required to operate in order to satisfy all safety-related cooling requirements. In the event of a LOCA or other emergency mode of operation, the ESW pump supply lines to any non-safety related equipment are normally isolated by automatic closure of isolation valves.

In many plants, crossties exist between the ESW systems of one unit and another unit and thereby provide defense-in-depth for events that are beyond the design basis. Since the crossties provide accident mitigation capabilities that are beyond the design basis, the Technical Specifications for these plants have typically not placed limiting conditions for operation on the operability of these crossties. The Technical Specifications usually address only the operability requirements for systems in the context of the single failure criterion as imposed by regulatory requirements and consistent with the assumptions of the safety analysis.

The specific goals of the study were to (a) develop a generic loss of service water (LOSW) initiator frequency, (b) obtain a quantitative perspective of the potential vulnerability of the characteristic multi-unit service water design described above, (c) determine the underlying causes of ESW system failures, (d) develop potential solutions, (e) determine the risk-reduction potential, and (f) establish the cost/benefit aspect of each of the potential solutions.

1.2 Methodology

The overall approach of the project included the following elements:

- The specific design arrangements of all seven multi-unit plants were surveyed and classified.
- A loss-of-ESW initiator was calculated using all available information based on operating experience.
- Core damage frequency due to the loss of ESW was calculated utilizing a previously developed core damage model (this is documented in Reference 1) modified accordingly to fit the needs of this study. The effect of recovery actions was also considered.
- Sensitivity studies were conducted to examine the risk reduction potential of various improvements.
- An estimate of the conditional risk-based offsite consequences was made based on typical sites and utilizing existing calculations.
- A risk-based value-impact evaluation was performed for each identified potential improvement.

1.3 Organization of the Report

Section 2 provides a brief survey of the various shared service water system designs at the subject seven multiplant pressurized water reactor (PWR) sites. A short discussion is given on system components as well as operational, test, and maintenance philosophy. Section 3 summarizes the results of a licensing event report (LER) survey of the operating history of the ESW systems at all PWRs. Complete and/or partial losses of the ESW system functions were classified and evaluated with regard to their effects and potential consequences. Section 3 also contains the details and results of the initiator frequency calculations. The core damage frequency (CDF) calculations along with the time-dependent recovery and seal LOCA model are discussed in Section 4. The effects of the potential improvements on the CDF are analyzed in Section 5. The risk-based cost/benefit estimates are presented in Section 6 for the potential improvements. Section 7 summarizes the results obtained and the most important conclusions. Appendix A describes the details of each operating event included in the data used to derive the initiating frequency. Appendix B presents the results of the full ESW LER survey which includes the documented events resulting in partial degradation of ESW systems. A simplified heatup analysis of the component cooling water system is presented in Appendix C. Appendix D includes a fault tree model of the ESW system for a typical multi-unit system configuration. Appendix E presents a detailed discussion of the cost estimates.

1.4 References

1. Cho, N. Z. et al., "Analysis of Allowed Outage Times at the Byron Generating Station," NUREG/CR-4404, June 1986.

2. SURVEY OF ESW SYSTEMS

Essential service water systems are designed to provide cooling water to various heat loads in both the safety and non-safety portions of the power plants. The primary safety function of the ESW system is to supply adequate cooling water to the safety-related systems and components that are required for the safe shutdown of the plant and/or to mitigate the consequences of an accident. In addition it may also provide cooling water to safety and non-safety components during the normal operation of the plant.

The ESW systems are designed to have either open or closed-cycle arrangements. The open-cycle design is essentially a once-through cooling method where the water is pumped from the ultimate heat sink through heat exchangers and discharged back to the heat sink. The closed-cycle design circulates the cooling water in a closed loop and the heat is removed in a cooling tower using air as the cooling medium. Any water loss (evaporation in the cooling tower) is made up from an outside water source. However, the amount of makeup water is much less than the water requirements in the open-cycle. Therefore, this arrangement is preferred at locations with limited water supply.

The system is generally designed to operate in an open-cycle taking water at ambient temperatures from the ultimate heat sink which is normally a river, pond, lake, or ocean. The cooling water is pumped through a number of heat exchangers removing heat from the various plant loads and then is discharged back to the ultimate heat sink rejecting the heat to the outside environment. The primary components of the ESW systems are the a) heat sink and intake section, b) ESW pumps with motors and strainers, c) heat exchangers with associated piping and valves, and d) the discharge section and piping. There are numerous different designs for the ESW systems among the operating power plants taking into account the specific features of each plants' ultimate heat sink and the design arrangements of the plant itself. In addition, the design and operating arrangements vary greatly making general classifications rather difficult.

The previously listed main components of the ESW systems may differ in capabilities, physical arrangement, their number and types and the variety of heat loads to be supplied. In a typical PWR plant, the ESW system may consist of a number of subsystems. The heat sink is usually a large body of water at ambient temperatures such as a lake, ocean or river. In some cases an alternate heat sink is also provided, such as the combination of a river and a dedicated emergency cooling pond requiring additional intake structures, valves and piping. The discharge section and piping returns the ESW flow to the ultimate heat sink through a large underwater opening.

In some plants, chemical additives are injected into the cooling water to control biological fouling and is accomplished with a chemical injection system located at or near the intake structure(s).

The intake structure itself usually consists of a number of bays each with a large underwater opening with a fixed screen. In addition, traveling screens are also provided to prevent large objects from being pumped into the system. The traveling screens are usually equipped with a cleaning system with specially dedicated pumps (screen wash pumps). The intake structure may also be the location of the ESW pumps and the associated strainers. The number and capacity of the pumps vary greatly from plant to plant and in this particular case has served as one of the important distinguishing characteristics of the ESW systems. The pumps generally supply two redundant loops and in case of a multi-unit arrangement may also serve the cooling requirements of the other unit through various cross connections.

The two redundant cooling loops remove heat from the various components through the respective heat exchangers. The primary reactor coolant is usually isolated from the ESW system by an intermediate cooling loop (component cooling water, CCW). In normal operation the main ESW heat load is generally the CCW heat exchangers, however, additional cooling requirements may be imposed by the containment fan coolers, air conditioning units, and various pump bearing and lube oil coolers.

During the cooldown to cold shutdown and the ensuing shutdown period the residual heat generated by the reactor core has to be removed by the ESW system (again through the intermediate cooling loop, CCW) and constitutes the major heat load in that mode. During accident conditions, other safety-related equipment is also in operation such as the station emergency diesel-generators, possibly the containment spray heat exchangers, and various air conditioning and ventilation systems requiring ESW cooling flow. This indicates that the success criteria of the ESW systems depends largely on the operational mode of the plant and has to be specifically evaluated for each plant. In general, the ESW system is required to operate in all modes of reactor plant operation, since it is used to reject heat to the ultimate heat sink.

The heat removal process is of vital importance and the loss of the ESW system leads to the loss of other vital safety systems (e.g., RHR, emergency diesel generators, containment fan units, etc.) designed to prevent or mitigate the consequences of an accident.

The main purpose of this study is to investigate the effects of the loss of the ESW system as defined previously at multi-unit PWR sites and identify any potential improvements for these systems which would effectively reduce the risk from the operation of these plants in a cost-effective manner. The seven plant sites under considerations in GI-130 are specifically indicated in Table 2.1.

The important design characteristics used to classify the ESW systems were a) the number of ESW pumps dedicated to each redundant and independent train and b) the existence of any independent and separate water supply and/or intake structure. Numerous other design features may be used for classification, but with regard to the specific improvements considered later in this study, these two seemed to be the most important ones. At the multiplant sites the basic design of the ESW systems consists of two independent ESW cooling trains, each capable of 100% heat removal capacity. The success criterion for the highest heat load is always 1-out-of-2 ESW trains. An additional backup or redundancy is provided by the unit crosstie.

In the most common arrangement, two 100% ESW pumps provide the required cooling water supply to two independent and separate cooling loops (Trains A and B). During normal operation the typical setup is that one pump is running and supplies the coolant flow and the other is on standby. Allowed outage times for service and maintenance of the standby pump are generally 72 hours. There are generally no explicit TS requirements on the ESW pumps of one unit with regard to the operational mode of the other unit (especially in Mode 5 and 6 - cold shutdown and refueling mode respectively). The cross-connections may allow the operators to supply the cooling water requirements from the other units' ESW pumps.

Another important design consideration is the possibility of a redundant ultimate heat sink. This is important from the point of view that common cause problems or human errors (especially at the intake structure or with the ultimate heat sink) could cause the loss or degradation of the ESW system. In such cases, additional backup or swing pumps may not be very useful, since a separate intake or water source may be required. The existence of a separate emergency cooling pond capable of providing the emergency service water supply and serve as the alternate ultimate heat sink is also indicated in Table 2.1.

Typical ESW system arrangement composites for the GI-130 plants are shown in Figure 2.1 indicating the most widely used design with two 100% capacity ESW pumps providing the required cooling water flow to two independent trains. Various heat loads are indicated, which may be significantly different from plant to plant. The cross-connection to the other unit is also shown. This line is generally equipped with a motor operated valve that may be remotely operated by the control room operator as required.

The ESW pumps are electrically driven vertical (or in some cases horizontal) centrifugal pumps. The vertical shaft arrangement allows the direct pumping of the cooling water from the intake structure at the required volume and head. These pumps occasionally experience vibrational problems. The electric motors driving the pumps are powered by separate electrical buses, each from their respective safety trains and the electrical ac source is backed up by the station emergency diesel generators.

The basic testing and maintenance policies are established in conformance with the Technical Specifications. One of the ESW trains is normally in operation and the standby ESW pumps are required to be tested periodically (monthly or quarterly). Allowed outage times (AOT) for the maintenance of one train is usually restricted to 72 hours. At multi-unit sites, there are no special testing requirements on the isolation MOVs of the crossties between the units. It is also important to note that at these multi-unit sites, there are no specific maintenance or AOT restrictions on any of the ESW trains of one unit in certain modes of operation (Modes 5 and 6 - cold shutdown and refueling mode).

The failure or degradation of the ESW system may be recognized by the control room operators through numerous system indicators and/or monitors. The traveling screens are usually provided with differential pressure measuring devices and these may activate the screen wash system upon reaching a preset differential indicating the clogging of the screens. Similar indication is available for the strainers at the discharge of the ESW pumps. The intake water level is also monitored to insure adequate suction to the pumps. The ESW pumps are generally monitored for vibration, temperature, overload, etc. The loss of the discharge pressure at the headers is usually used to automatically start the standby pump. The various heat loads are usually individually monitored for pressure and temperature. Trends in differential pressures across heat exchangers are sometimes used to predict clogging or fouling and may serve as a basis for preventive maintenance programs. Temperature indications, automatic trips, etc., at various devices (diesel lube oil coolers, pump oils, bearings, etc.) may also serve to indicate the degradation or loss of the ESW system.

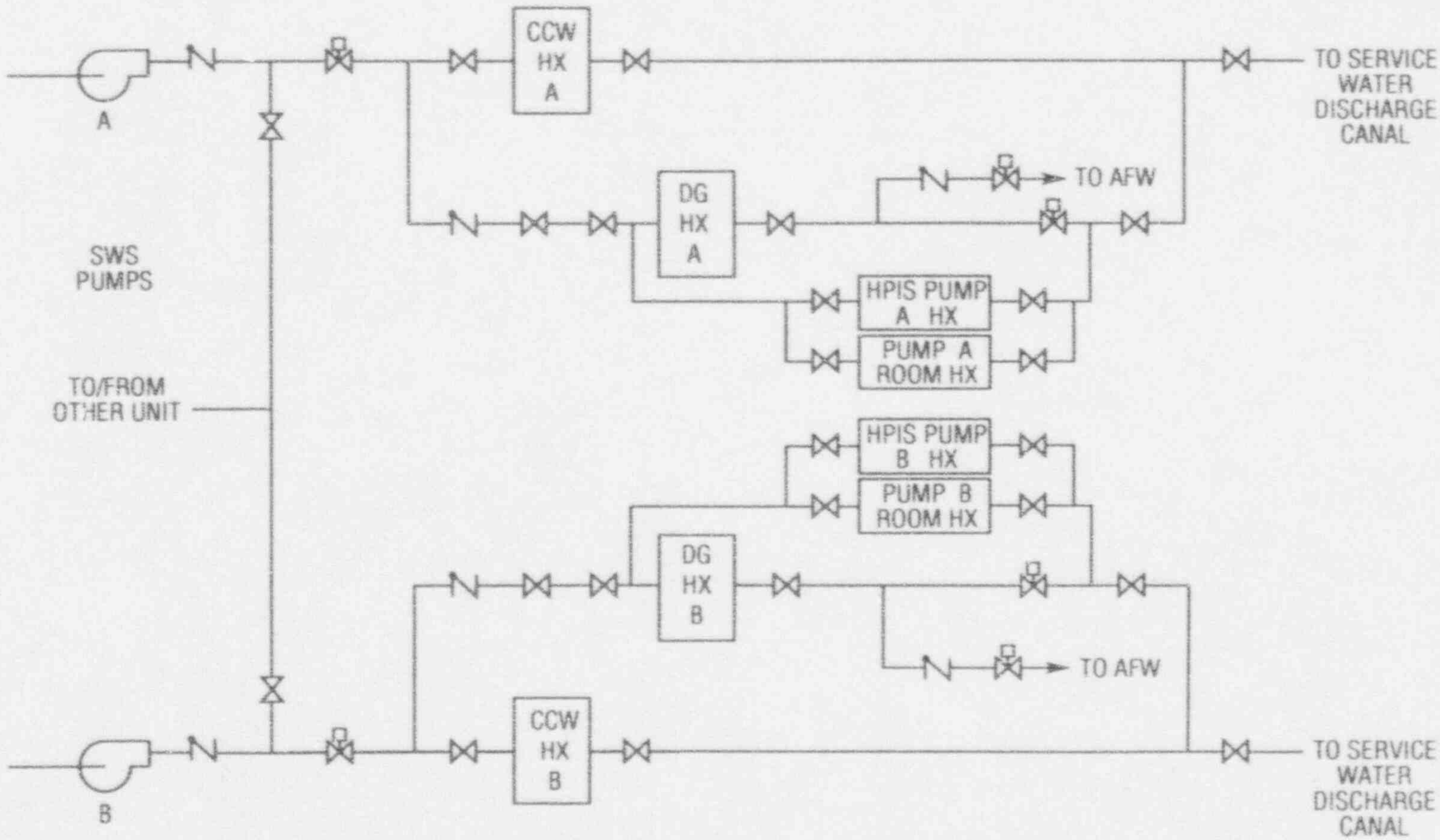

Figure 2.1 Typical ESW system design - 2 pumps/2 trains - cross connection to other unit.

Table 2.1
Classification of Service Water Systems for Multi-Unit PWR Sites Considered in GI-130

Multi-Unit Plants	Service Water Pumps/1 Unit	Success Criterion for Highest Load	Separate Water Source (Or Intake) Each Unit
Braidwood 1 & 2	2	1 of 2	
Byron 1 & 2	2	1 of 2	
Catawba 1 & 2	2	1 of 2	Yes
Comanche Peak 1 & 2	2	1 of 2	
Cook 1 & 2	2	1 of 2	
Diablo Canyon 1 & 2	2	1 of 2	
McGuire 1 & 2	2	1 of 2	Yes

3. SURVEY OF OPERATING EXPERIENCE FOR ESW SYSTEMS: INITIATOR FREQUENCIES

3.1 Survey of Operating Experience

Operating experience regarding essential service water systems is embedded in various extensive data bases which include events dating back to the 1970's. BNL has performed a search for events involving the ESW system at PWRs by using the RECON[1] data base and the NPE operating events listing.[2] The available information consists mostly of LER submittals and in the case of the NPE system, additional component, engineering and failure reports are listed. The data bases have been systematically searched for system and component failures involving the ESW systems. All operational events have been collected and reviewed.

Even though this study concentrates on PWR plants, the BWR data bases were also reviewed, since some of the ESW system arrangements are similar. However, events occurring at BWRs are not included in the data base for the initiating frequencies.

The actual configuration may vary greatly between ESW systems and plants. The typical design generally consists of an intake structure, service water pumps, number of heat exchangers and the associated valves and piping.

For the purposes of the present study the most important failures are those which lead to the total loss of the ESW system of one unit. Based on this, the failure events may be classified as (a) failures involving the partial and/or total loss of the system and (b) failures resulting in a partial degradation involving one or several components of the ESW system, but not disabling or leading to a significant reduction in heat removal capability.

There are a number of different components in the ESW system that may fail to perform their intended function in a variety of ways. However, the review of the operating events have indicated that there are specific dominant failure modes for the ESW system as a result of failures of certain components.

a. Partial or complete loss of ESW system function (at one unit) - the failures involving the traveling screens and/or other common cause problems at the intake structure leading to the partial or complete loss of the water supply are one of the most important failure modes. The ESW pumps and their electrical power supply are the other important contributors to the partial or total loss of the ESW system. A less frequent class of events involves operational or procedural failures of the ESW pumps.

b. Degradation of the ESW system - biological fouling, sediment deposition resulting in blocked heat exchangers and strainers, as well as corrosion of pipe walls, tubes and subsequent leakages are the most dominant failure modes. Additionally, mechanical and electrical problems with the operation of the ESW pumps also contribute to the degradation of the system.

In the following subsections, the collected operating events leading to partial or complete loss of the ESW function coupled with significant reduction in heat removal capability are discussed. The report concentrates on events that belong to Group A of the above classification. Events characterized as leading to ESW system degradations (Group B) have somewhat different failure mechanisms primarily dominated by biological fouling of the ESW system.

A NRC sponsored research program to resolve Generic Issue 51, "Improving the Reliability of Open-Cycle Service Water System," was previously established addressing and studying the effects of fouling on plant safety. The program included biofouling as well as fouling due to mud, silt and corrosion products.

The research program was recently completed and a generic letter was issued[4] that identified and recommended a number of specific actions to be taken by the affected plant licensees.

As a consequence of these proposed actions (assumed to be fully implemented within the context of this study), the expected frequency of potential degradations of the ESW system due to events in Group B will be significantly reduced. The contribution to the core damage frequency from these events is not the subject of this study and based on the results of the research program addressing GI-51[5] may not be significant after all corrective actions identified in Generic Letter 89-13 are fully implemented.

A more detailed description of the events in Group A is given in Appendix A.

3.1.1 Events Involving the Partial or Complete Loss of The ESW System

Table 3.1 summarizes the comprehensive review and evaluation of the operating events resulting in the complete loss of ESW system function. There were a total of 12 operating events observed over the review period from the 1970's to 1988 for all the PWRs. The table lists the plants, the corresponding LER number, and a brief description of the event. In the following, some of the events representing typical failure modes are described in more detail.

By examining the sequence of events which lead to the complete loss of the ESW function the following typical failure modes have been identified.

1. Failures involving the unavailability of the intake structure. This generally involved weather or flooding related events where the traveling screens or the ESW pumps became unavailable (Events A.1, A.2, A.8, and A.10) - ~35% of the total.

2. Loss of electrical power supply. In these cases the electrical power supply to the operating train was lost and generally no redundant source was available due to procedural or maintenance errors (Events A.4, A.6, A.7, A.9, and A.12) - ~35% of the total.

3. Loss of ESW pumps. There were a number of cases when mechanical or design deficiencies of the ESW pumps resulted in a complete loss of the system function (Events A.3 and A.5) - ~20% of the total.

4. Other events not in these categories may also cause the complete loss of the ESW system function and may significantly contribute to the system unavailability (Event A.11) - ~10% of the total.

Events at Salem 1 (Event A 1) and Farley 1 (Event A.2) are typical of weather and flood related causes 'c Salem 1, a winter storm shut down the system. The traveling screens were iced over and the water flow to the ESW pumps was severely restricted. At Farley 1, the intake source is a river and the water level was high after heavy rainfall. The intake structure was flooded through holes and other penetrations to the extent that eventually a switchover to the standby emergency cooling pond was necessary. This event was included as a failure in our statistics, since standby ponds are generally not available at other plants.

A typical event representing the second failure mode is the operating event at Palisades. During a planned maintenance of a switchyard breaker the offsite power supply was interrupted. Operating personnel did not recognize that no operable ESW pump would be supplied by the available diesels, and all ESW cooling was lost as a result.

Mechanical failures of the ESW pumps also represent an important class of failure modes. At the San Onofre 1 plant one pump shaft sheared due to vibration. The standby pump automatically started but it's discharge valve failed mechanically. An auxiliary pump was also started manually but lost suction due to inadequate priming. The ESW heat removal function was reestablished by utilizing the screen wash pumps. Effectively, for about 15 minutes the plant was without ESW flow (see Appendix A).

For each class of failure mode an average recovery time can be estimated by examining the time evolution of each event (Table 3.1). The first class of failure modes involving the intake structure usually requires long recovery times which may extend from two hours to a few days.

The recovery from a loss of electrical power supply (Failure Mode 2) requires less time and experience indicates a time period of ~1 hour to be representative. This relatively long recovery time may be explained by noting that the recognition of the problem itself usually took longer than the corrective actions in almost all events. For the other two failure modes the recovery times are somewhat more unpredictable, but generally it took at least 1-2 or in a few cases up to 10-15 hours to recover the loss of the ESW function.

In Section 4, a time-dependent recovery model is developed where the ESW recovery as a function of time is calculated based on these recovery data.

It is interesting to note that biological fouling and/or sediment depositions were not the basic underlying cause of any of the events listed in Table 3.1, even though these are the dominant causes of partial degradations. This may be understood by recognizing that the time involved in these cases (fouling or deposition till total loss of function) is relatively long and the system degradation is gradual allowing ample time for diagnosis and corrective actions.

3.1.2 Events Involving Degradation of the ESW System

All events related to the partial degradation of the ESW system were reviewed and are listed in Appendix B. For each event the plant, the reference LER number, and a brief description is given.

A review of the operating experience indicates that the ESW system is vulnerable to a large number of failure mechanisms as indicated by the large number of events listed in Appendix B. The primary failure mechanism may be identified as biological fouling and/or sediment deposition. This is an indication that the quality of the cooling water may not be very well controlled, since it is an open-cycle design. The affected components are various heat exchangers and strainers which become clogged and restrict the flow of the cooling water. Pipe wall corrosion was also an important contributor to component/system unavailability especially affecting the containment fan coil units at various plants.

Mechanical and electrical failures of the ESW pumps are also numerous, which could lead to the complete loss of the ESW function. The general conclusion drawn from this survey is that the essential service water system is vulnerable to various failure mechanisms, the expected failure frequency of the component causing system degradation is relatively high and the impact on other safety systems is significant.

These class of events are dominated by fouling as was discussed earlier. The resolution of GI-51 addresses most of the problems in this group which may have significant safety impact.[4,5]

3.2 Initiating Frequency - Loss of ESW Function

The search process for operating events that are representative of the complete loss of ESW function was discussed in the previous section. In this section more details are given about the event analysis and the estimation of the initiating frequency.

The relevant events are listed in Table 3.1. All events were included to estimate the probability of the loss of ESW function, even if the given plant was not part of a multi-unit arrangement. The review and the selected events span the PWR population only and no BWR events were included in the statistics.

3.2.1 Data Reduction

The total number of events involving the partial or complete loss of the ESW function was 12. The total number of PWR plant years was calculated as 666.7 years.[3] Since the ESW system is in continuous operation in all modes (with the possible exception when all fuel is removed from the core) actual plant calendar time was used to estimate the ESW system years.

A more detailed examination of the loss of SW events listed in Table 3.1 indicates that a number of events occurred in Modes 5 and 6 (shutdown) and some of them may not have been a complete loss in terms of the total stoppage of the SW flow.

The operational difference of the SW system between at power (Modes 1-4) and shutdown operation is primarily the actual heat load and equipment affected by the loss of SW. In addition, the actual administrative requirements imposed by the TS may also differ and make the two operational modes more distinct.

The heat load on the SW system is generally much larger during shutdown than in operation due to the decay heat removal function. The loss of decay heat removal function is a significant event that could potentially also lead to core uncovery. In essence, the operational need for the SW system is equally important in both operational modes (at power or shutdown) and in both cases the loss of SW function may lead to a scenario with potential core damage.

Although the SW system must operate regardless of the plant mode of operation and its loss may lead to potential core uncovery scenarios, the TS requirements do not explicitly recognize this and generally the operability of the SW system during shutdown is imposed implicitly through a requirement on the RHR decay heat removal function. In addition, during plant shutdown the system operations in general, supporting equipment availability, and potential human interactions with testing and maintenance activities are different from at power operations and events that occur under these conditions may not be adequately covered by normal power operating mode requirements.

In order to take into account these effects, the data base was divided into two major categories. The first includes events that occurred during power operations (A.3, A.2, A.10, and A.11) and the second contains all shutdown events (A.1, A.4, A.5, A.6, A.7, A.8, A.9, and A.12). This classification may further be refined to differentiate between partial and complete loss of ESW function. The resulting event classification listed in Table 3.2 indicates that the complete loss of ESW function predominantly occurred in shutdown, but a significant number of partial losses also occurred during power operations. It is important to recognize that all of the events listed as partial losses were such that the ESW system was declared inoperable. This condition or inoperability reflects the fact that the ESW system was unable to satisfy the heat removal requirements for a given design basis condition (usually a LOCA combined with cooldown).

This implies that the TS requirements were not met, but some heat removal capability still remained which generally was not determined. The actual reduction in heat removal capability may have been significant, but in some cases sufficient capacity may have remained and the plant was able to recover without any damage to the safety equipment.

The initiating frequency represents the frequency of expected events where total or complete loss of ESW system function results. In this respect partial loss events are not directly applicable, since potentially some heat removal capacity may have remained. However, the actual cooling capacity still av:ilable was not determined for most of the events and may not have been sufficient to remove the remaining heat load.

In order to take into account the uncertainties involved with partial loss of ESW events a weight or applicability factor was introduced to express the probability of extrapolating a partial error mode to complete loss of ESW. That is

$$I_{Complete\ LOSW}(\text{Corrected Total}) \sim W_p * I_{Partial\ LOSW} + I_{Complete\ LOSW}(\text{Actual}),$$

where I is the number of events.

The weight factor W_p expresses the probability that given a partial loss of ESW the extent of the degradation is such that it will correspond to a loss of sufficient cooling. In other words, a certain fraction of partial LOSW events (W_p) results in a complete LOSW. This weight factor may be established for both at power and shutdown modes of operation. The present estimate for these weight factors are:

$W_p(\text{at power}) = 0.05$

$W_p(\text{shutdown}) = .1$

It was assumed that during operation the loss of ~90% of the ESW system flow may result in a significant reduction of the heat removal capability from safety equipment. If the heat removal capability reduction due to partial losses is assumed to be random and uniform (that is a partial LOSW resulting in 90% reduction in heat removal capability is just as likely as any other level say 60 or 40%), then this results in an approximate estimate of the weight factor as ~.1 (1-.9). This factor may be further decreased to account for potential operator actions that could help to reduce the actual heat load on the ESW, such as shedding loads or throttling certain equipment flow. This and other reduction effects may reduce the weight factor by about a factor of two leading to the value of .05 during power operations. This estimate is primarily based on engineering judgment and may be further refined as more data becomes available.

The value of the weight factor for shutdown operation mode is judged to be higher by about a factor of two $W_p(\text{shutdown}) = .1$ for the following considerations. The SW system must operate continuously both during shutdown and operating modes. The TS explicitly controls the availability of the redundant train or components only during the operating modes and only implicitly (through RHR operability) during shutdown. Therefore, it is conceivable and is an actual practice that one SW train may be out of service for relatively extended periods in Modes 5 and 6 (shutdown).

In addition, during shutdown there are other actions such as multiplicity of testing and maintenance activities that complicate the recognition/diagnosis of an actual LOSW event and the potential ability to appropriately respond may also be curtailed due to the various and numerous other actions involved in shutdown.

The total number of events that led to the complete loss of ESW function may be obtained as

$$I_{Complete\ LOSW}(\text{Corrected Total}) = I_{Complete\ LOSW}(\text{Actual}) + W_P\ I_{Partial\ LOSW}$$

At power:

$$I_{LOSW}(\text{Power}) = 1 + .05*3$$

Shutdown (one event A.12 will be treated separately; number of events = 7-1=6):

$$I_{LOSW}(\text{Shutdown}) = 6 + .1*1 = 6.1$$

The one complete loss of ESW event that occurred during power operation (A.3 - San Onofre 1) has to be examined in more detail to determine its applicability with regard to the plants under consideration in GI-130.

The actual event (see Appendix A) involved the failure of the normal standby pump and also the consequent failure of a second auxiliary standby pump. This led to the complete loss of ESW system flow for about 15 minutes, when a third standby system was activated to reestablish cooling water flow.

The important fact to recognize is the layers of redundancy available at this plant and the actual equipment performing the required function in comparison with the multi-unit plants of GI-130. These multi-unit plants generally have two standby pumps to the normally running SW pump.

One additional standby pump on each unit serves as the first level of backup source. The second level of redundancy on one unit is provided by a crosstie connection to the other unit and relies on the standby SW pump of that unit. The main difference between this particular arrangement versus San Onofre Unit 1 is that the second standby SW pump is not exclusively dedicated for the unit, and in addition, the unit crossties have to be opened to provide backup SW flow. However, none of these differences was judged to be significant enough to neglect the San Onofre 1 event.

There is one important characteristic of the event that has to be taken into account. The second standby pump on San Onofre 1 is a specially designed suction lift pump requiring continuous priming. In effect, this priming action was lost due to the loss of vacuum provided through an air compressor and station air supply. In this sense, the SW pumps on the multi-units are different, since those are all electric motor driven pumps.

The failure rate of a typical electric motor driven vertical pump is on the order of Q_{SW}(Electric/Fail-to-Start) ~ $3\text{x}10^{-3}$/demand. There is no data available of the reliability of the priming system, hence a similarity argument is used to estimate the reliability of this type of pump.

The priming system is somewhat less complicated than the controls and associated supports of a turbine driven pump. The failure rate for this type of design is about Q(Turbine/Fail-to-Start) ≈ $3\text{x}10^{-2}$/demand or about a factor 10 larger than the electric motor driven design. The actual failure rate is considered and judged to be in-between these two bounds and based on engineering judgment a factor of five was estimated.

This implies that for multi-unit arrangements the failure of the second set of standby equipment (including the pump and crosstie) is about five times more reliable. This may be expressed as an applicability factor W = 1/5 = .2 and the initiating events expressing complete loss of ESW becomes:

$$I_{LOSW} = W*I_{Complete\ LOSW} + W_P*I_{Partial\ LOSW}$$

$$I_{LOSW} = .2*1 + .05*3 = .35 \text{ \# of Events.}$$

In order to evaluate the initiating frequency for LOSW events, the total operating years for all SW systems must be determined. Based on industry records[3] the total operating SW system years for all PWRs was calculated as 666.7 years. This includes both at power and shutdown modes of operation.

Corresponding to the classification of the initiating events this time period is also divided as:

At power T_P = 486.7 years

Shutdown T_{SH} = 180.0 years

where the total shutdown times were calculated from Reference 6.

The modelling in this study differentiates between not only the operating modes (at power versus shutdown), but also introduces additional states with regard to the actual status of the SW pumps. For example, one state is when the unit is at power and one SW train/pump is operating and the other is on standby. Another state would be when the standby pump is placed in test or maintenance mode. This will be discussed in more detail in Section 4. For the present purpose it is assumed that most of the LOSW events at power occurred with the usual arrangement of the SW trains, that is, one operating and the other on standby. This will be noted as - R/SB (running/standby). The initiating frequency at power is obtained as

$$\lambda_{ESW}(\text{Power, R/SB}) = \frac{I_{Complete}(\text{Power, R/SB})}{T_{(Power)}} = \frac{.35}{486.7} = 7.2\text{x}10^{-4}/\text{RYR}$$

where RYR = Reactor Year. Assuming an underlying lognormal distribution and accounting for the uncertainties in the data, an error factor of five is estimated based on engineering judgment that would result in the mean value of the initiating frequency at power as,

$$\lambda_{ESW}^{Mean}(\text{Power, R/SB}) = 1\text{x}10^{-3}/\text{RYR}.$$

The shutdown events were further classified as indicated in Table 3.2b with regard to the status of the unit ESW trains. This enabled the calculation of initiating frequencies for the specific status in shutdown with one train in maintenance as,

$$\lambda_{ESW}(\text{Shutdown, R/M}) = \frac{\text{I(\# events when one train was in maintenance)}}{\text{T(total SW time with one train in maintenance in shutdown)}}$$

$$\lambda_{ESW}(\text{Shutdown, R/M}) = \frac{4}{22.2} = 1.8\text{x}10^{-1}/\text{RYR}$$

and for shutdown, with one train in standby

$$\lambda_{ESW}(\text{Shutdown, R/SB}) = \frac{N_{Complete} + W_P\, N_{Partial}}{T(R/SB)} = \frac{2 + .1*1}{136.5} = 1.5\text{x}10^{-2}/\text{RYR}$$

The respective time estimates were derived by assuming that on the average one train is maintained for about one week in a shutdown. Section 4.3 contains a more detailed discussion of the relative timing considerations for each state. The mean value of the initiating frequency in shutdown is obtained as:

$$\lambda_{ESW}(\text{Shutdown, R/M}) = 2.9\text{x}10^{-1}/\text{RYR}$$

$$\lambda_{ESW}^{Mean}(\text{Shutdown, R/SB}) = 2.5\text{x}10^{-2}/\text{RYR}$$

In summary, three initiating frequencies were established based on the operating history of the ESW systems. The initiators were determined with respect to the specific status of the ESW trains. However, the data allowed the determination of only a limited number of configurations and for other arrangements additional assumptions will be required (see section 4).

The final initiating frequencies at power were:

$$\lambda_{ESW}(\text{R/SB}) = 1\text{x}10^{-3}/\text{RYR}$$

and in shutdown:

$$\lambda_{ESW}(\text{R/M}) = 2.9\text{x}10^{-1}/\text{RYR}$$

$$\lambda_{ESW}(\text{R/SB}) = 2.5\text{x}10^{-2}/\text{RYR}$$

There are important areas of any risk assessment analysis which require considerable attention regarding the establishment of the initiating frequencies and modelling. The treatment of support state dependencies and external events are such cases. In the present study a simplified approach was adopted with respect to these areas by incorporating the effects arising from these potential accident sequences in the loss of ESW initiating frequency.

The ESW system may directly be linked to an initiator, that is the initiating event may cause the malfunction or the loss of the ESW system. The correct treatment of these occurrences may be accomplished in different ways. The technique used in this report corresponds to an event tree transfer method. In this case, it was assumed that in the event tree describing the accident progression of an initiator, which is linked to the ESW, a top event representing the ESW success function is explicitly included. At that point, the event is transferred to the ESW event tree as an initiating event due to the failure of the ESW.

A significant number of events listed in Table 3.1 involve the degradation or the loss of a support system that directly or indirectly caused the loss of the ESW function (Events A4, A6, A7, and A12). By including these events in the statistics and the initiating frequency, the core damage calculations may be simplified, i.e., no separate support state calculations are required.

In addition, there are events (Events A.1 and A.2) which are normally treated in the external event categories. The presently employed core damage model (described in the next section) does not explicitly treat external events. Therefore, the inclusion of these two events in the initiating frequency serves as a simplified approach to account for external accident sequences that may cause the loss of the ESW function. In this respect the initiating frequency used in this study consists of two components. The first one corresponds to events which were due directly to the loss of ESW components and the other was where the ESW system was impaired through the loss or malfunction of a support system or external causes. This represents the "linked-initiator" approach.

The inclusion of the indirect events in the initiating frequency basically corresponds to a simple transfer technique in the event tree modelling. This essentially means that in the progression of other accident sequences, whenever the event representing the failure of the ESW system occurs, the sequence is transferred to the ESW event tree, i.e., the initiating frequency is increased by the fraction of the indirect events.

It is important to point out that the initiating frequency discussed above refers to an operating single unit. The design arrangements of single vs. multi-units are different and vary from site to site. For other operating modes and multi-unit arrangements, the initiating frequency was derived using appropriately modified single unit data (λ_{ESW}) and is discussed in Section 4.2.1.

The initiating frequency derived and represented above may be viewed as data applicable to PWRs in general, but may not reflect the specific experiences of any individual plant.

An independent study[7] performed by the Office for Analysis and Evaluation of Operational Data (AEOD) of the USNRC, completed a comprehensive review and evaluation of ESW system failures and degradations in the time period of 1980 through 1987. The main conclusion of Reference 4 with regard to the total loss of ESW function may be quoted as:

"The frequencies of service water system failures and degradations as observed in operating events are relatively high: 1.5×10^{-2} per reactor year for system failures..."

3.2.2 Station Blackout Considerations

An important accident sequence class, called the station blackout (SBO) accident is directly related to the loss of ESW system accident sequences. In this scenario, the unit ac electrical power supply is lost including offsite (grid) and onsite supply provided by the station diesels. As a consequence, all electric power supply to the ESW pump motors is also lost leading to the complete loss of ESW for units with SW pumps driven by electric motors.

The data base includes 1 actual event (A12) that corresponds to a SBO sequence. In this section considerations are given to include this particular accident scenario in the loss of ESW sequences. The consequences of the SBO scenarios are also considered with regard to reactor coolant pump seal failures in Generic Issue 23.

The potential improvements considered for GI-23 may be similar to improvements analyzed in this study. However, the full implementation of certain or various combinations of the potential improvements or recommended actions is presently uncertain and hence all aspects of the potential loss of ESW scenarios will be considered in this report.

The impact of possible recommended actions to resolve GI-23 must be considered with respect to their implementation and their effect on the potential recommended improvements for the plants under consideration in GI-130.

In effect, the inclusion of SBO scenarios may be accomplished by increasing the initiating frequency by a certain fraction corresponding to the frequency of SBO occurrences.

The frequency of SBO during power operations was established as a combination of operating history data with unavailability estimates. The two step procedure is described in Appendix B of Reference 8 where the SBO initiating frequency was determined. This two step semi-analytical method is used since Reference 8 could not identify actual SBO occurrences. First loss of all offsite power data was reviewed and based on the numerous occurrences a loss of offsite ac power frequency was calculated. This frequency was modified by the probability that the station diesels fail to provide backup ac power. For a generic PWR site which is most similar to the seven sites under consideration the final initiating frequency for loss of all ac power was calculated as $\lambda_{SBO} = 1 \times 10^{-4}$/RYR.

It was assumed for the purposes of this study, that this frequency reflects a unit at power, $\lambda_{SBO} = \lambda_{SBO}(\text{Power})$. During shutdown, the determination of the frequency of SBO is somewhat more complex if the semi-analytical procedure is used, due to the multiplicity of testing and maintenance actions affecting the availability of the station diesels as well as the offsite power supply. In order to circumvent these difficulties the following simple procedure was utilized. There has been one uniquely identified SBO event at Vogtle Unit 1[9]

(A12) lasting for ~36 minutes. A construction truck backed into a power pole causing the offsite power loss. One diesel generator was down for maintenance and the other started, but tripped. This one event may be used to estimate the frequency of SBO in shutdown conditions together with the total accumulated shutdown time ~180 years. That is

$$\lambda_{SBO}(\text{Shutdown}) = \frac{I_{SBO}(SH)}{T(SH)} = \frac{1}{180} = 5\text{x}10^{-3}/\text{RYR}$$

and

$$\lambda_{SBO}^{Mean}(SH) = 8\text{x}10^{-3}/\text{RYR}$$

This frequency is divided between two states with regard to the SW train availability:

a. One SW train operating other SW train in standby - (R/SB)

$$\lambda_{SBO}(R/SB) = \lambda_{SB}(SH) * \frac{T(SH\text{-}R/SB)}{T(SH)}$$

$$\lambda_{SBO}(R/SB) = 8\text{x}10^{-3}\ \frac{136.5}{180} = 6.9\text{x}10^{-3}/\text{RYR}$$

where T(SH) = T(SH-R/SB) + T(SH-R/M) and T(SH-R/SB) indicate the total accumulated shutdown time when one SW train is operating and the other is on standby.

b. One SW train operating one SW train in maintenance - (R/M)

$$\lambda_{SBO}(R/M) = \lambda_{SBO}(SH) * \frac{T(SH\text{-}R/M)}{T(SH)}$$

$$\lambda_{SBO}(R/M) = 8\text{x}10^{-3} * \frac{22.2}{180} = 1.1\text{x}10^{-3}/\text{RYR}$$

The total initiating frequency is obtained by adding the λ_{SBO} contributions to the previous total that is:

$$\lambda_{LOSW}(R/SB) = \lambda_{ESW}(R/SB) + \lambda_{SBO}(R/SB)$$

$$\lambda_{LOSW}(R/SB) = 1\text{x}10^{-3} + 1\text{x}10^{-4} = 1.1\text{x}10^{-3}/\text{RYR}$$

In shutdown,

$$\lambda_{LOSW}(R/SB) = 2.5\text{x}10^{-2} + 6.9\text{x}10^{-3} = 3.2\text{x}10^{-2}/\text{RYR}$$

$$\lambda_{LOSW}(R/M) = 2.9\text{x}10^{-1} + 1.1\text{x}10^{-3} = 2.91\text{x}10^{-1}/\text{RYR}$$

3.2.3 Failure Mode Classification

Based on the statistics listed in Table 3.1, a generalized classification of the various failure modes affecting the operation of the ESW may be derived. Table 3.2 shows the most important failure modes along with the relative contribution of each of these classes.

The data seems to indicate that no single component may be selected as the dominant contributor, but rather a combination of various components and their supports. In this sense the intake structure, the ESW pumps and their electrical power supply are probably the most dominant contributors to the loss of ESW system function.

This also suggests that any potential improvement considered specifically for the ESW system must address more than one particular failure mode in order to be effective. However, other improvements not directly involved with the ESW system and consequently not addressing these failure modes may also be considered to prevent the degradation or loss of equipment supported by the ESW (such as reactor coolant pump seals).

3.3 References

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5. D.A. Neitzel, K.I. Johnson, "Technical Findings Document for Generic Issue 51: Improving the Reliability of Open-Cycle Service Water Systems," NUREG/CR-5210, August 1988.

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6. Licensed Operating Reactors, NUREG-0020.

7. P. Lam and E. Leeds, "Service Water System Failures and Degradations in Light Water Reactors," Preliminary Case Study Report, NUREG-1275, Volume 3, 1988.

8. "Evaluation of Station Blackout Accidents at Nuclear Power Plants," Appendix B, NUREG-1032.

9. Nucleonics Week, March 22, 1990.

Table 3.1
Total Loss of Service Water System Function

Event	Plant	Reference	Recovery	Description
A.1	Salem 1	NPE/PWR-2 VIII-110	Few days*	Winter storm shuts down the ESW system. Traveling screens blocked by ice.
A.2	Farley 1	NPE/PWR-2 VIII-155	~3 Days	Flooding of the intake structure.
A.3	San Onofre 1	LER-206/80-06	~45 Min.	One ESW pump shaft sheared due to excessive vibration, the discharge valve of the standby pump didn't open and the auxiliary pump lost suction.
A.4	Palisades	LER-255/84-01	~1 Hour	Offsite power removed, no operable service water pump supplied by the operating diesel.
A.5	Oconee 1	LER-269/86-11	---	Loss of LPSW suction due to inadequate design.
A.6	Salem 1	LER-272/84-14	~1 Hour	Vital bus 1A failed, bus 1B in maintenance, bus 1C didn't energize, loss of ESW system.
A.7	Salem 1	LER-272/82-15	~1 Hour	Vital bus 1A tripped, operating ESW train is lost, other train in maintenance.
A.8	Crystal River	LER-302/86-02	>2 Hours*	All ESW pumps are shut down, two divers drowned.
A.9	Calvert Cliffs	LER-318/82-54	~30 Min.	Power was lost on a 4k V bus resulting in the loss of ESW pump on the operating loop. Other train in maintenance.
A.10	San Onofre 2 & 3	LER-361/83-72	>2 Hours*	Traveling screens were damaged, CCW heat exchangers clogged.
A.11	Catawba 1	LER-413/85-68	~45 Min.	Both ESW trains declared inoperable due to torque switch problems on the discharge valves.
A.12	Vogtle 1	March 20, 1990	36 Min.	Loss of all safety ac power in cold shutdown. Offsite power wa lost due to a truck accident. The emergency diesel-generator tripped upon start.

*Estimated.

Table 3.2
Failure Mode Classification

Failure Mode	Relative Contribution to Initiating Frequency
Intake structure unavailable	~35%
Loss of electrical power supply	~35%
Loss of ESW pumps	~20%
Other	~10%

Table 3.2a
Operating History of ESW Systems

Mode of Operations	Complete Loss	#	Partial Loss	#
At Power	A.3	1	A.2, A.10, A.11	3
Shutdown	A.1,A.4,A.5 A.6,A.7,A.9,A.12	7	A.8	1

Table 3.2b
Classification of Shutdown Events with Regard
to ESW Train Status

ESW Train Status	Complete Loss of Events	Total #	Partial Loss of Events	Total #
R/SB Train A - Operating Train B - Standby	A.1, A.5	2	A.8	1
R/M Train A - Operating Train B - In Maintenance	A.4, A.6 A.7, A.9, A.12	5		

4. CORE DAMAGE FREQUENCY - MODELLING AND CALCULATIONS

4.1 Introduction

The core damage vulnerability caused by the failure of the ESW system may be estimated by developing an appropriate risk model. This would imply the development of a full scale PRA model including initiating frequency categories, full event tree/fault tree analysis and incorporation of support system dependencies. In addition, the various operating configurations of the multiplants taken together must also be considered and the plant PRA model would have to be appropriately modified for each separate operating state.

In the present study the following approach was used to approximate this complex modelling problem noting that GI-130 is concerned with those accident sequences which involve the failure of the ESW system. First, an already existing PRA model[1] was selected which was previously developed to specifically analyze the ESW system of a single unit. The sequences representing the loss of ESW were modified to reflect the new initiating event frequency established based on operating experience (Section 3). Then, in order to account for the various operational configurations in a multiplant situation, a conditional core damage frequency (CCDF) was derived for each sate by using system unavailability analyses (fault trees) reflecting the actual ESW system arrangements for that state. The total core damage frequency was then calculated as the sum of the fractional time-weighted averages of these CCDFs.

The modelling is discussed in more detail in the remainder of this section and the calculational results are also presented.

4.2 Loss of ESW Function Accident Sequence

The primary concern of GI-130 is to establish whether or not the loss of the ESW system in the selected multiplant units would significantly increase their core damage vulnerability and the risk to public health and safety. In such a scenario the ESW system on one plant is lost and alternate cooling capacity from the ESW system of the other unit may also be unavailable either through random component failures, maintenance or common mode failures.

The heat removal function provided by the ESW system is essential to all phases of operation of the plant, i.e., during normal operation as well as in shutdown modes. For this reason the actual operating configuration of the ESW system and the plant itself has to be considered in order to analyze the ESW system unavailability/capability and the progression of a potential accident.

Since it was impractical to develop separate PRA models for each operating configuration, an approximation technique was used to analyze the consequences of the loss of ESW function in each operating mode.

The basic approach was to select a base model reflecting the operation of a single plant with two independent trains of ESW either at power or in shutdown. The previously developed and analyzed PRA model for Byron Unit 1[1] was selected for this purpose. The main advantage of the Byron model was that the loss of ESW event is explicitly included along with an appropriate event tree which models the accident sequence in detail. In the next section a short discussion is given of this model as implemented and used in the present analysis.

From the single plant model the various plant configurations were derived by a comparative unavailability analysis discussed in Section 4.2.2. The original model from Ref.1 was modified to include the effects of short and long-term recovery actions. In addition, the probability of a reactor coolant pump seal loss-of-coolant accident (RCP seal LOCA) was established based on a recently developed seal failure model contained in NUREG-1150. These modifications to the single plant base model are described in Section 4.2.3.

4.2.1 Loss of ESW Function: Base Model

In the present study the loss of ESW accident sequences were analyzed using a previously developed model for Byron Unit 1.[1] Reference 1 gives a full account of the complete PRA model and therefore, in the following only a brief summary is given with regard to the loss of ESW accident sequence.

In NUREG/CR-4404 (Ref.1) the operation of Byron Unit 1 was modelled assuming that Unit 2 was not available (as was the case prior to the completion of Unit 2). The ESW system supplies cooling water to the safety-related equipment from the essential service water cooling towers. The system consists of two redundant, independent, 100% capacity closed cooling trains and each is capable of providing full heat removal capacity in all operating modes. The service water pumps, one per train, are powered from separate emergency power sources.

The loss of ESW function poses a serious threat because the heat removal function essential to equipment required for normal operation as well as for shutdown becomes unavailable. The affected equipment are numerous and generally include the loss of heat removal from the RCP seals, various pumps such as the high pressure injection, charging, RHR, the loss of the RHR and/or CCW heat exchangers and the lube oil cooling to the station diesels among others.

Following a loss of ESW event, the plant operator is likely to shut down the reactor coolant pumps to prevent damage to the RCP motor, bearing and seals upon loss of cooling and start recovery actions. By tripping the RCPs, the reactor automatically trips and the auxiliary feedwater system (AFW) is initiated to provide heat removal through the steam generators.

The RCP seals are generally cooled indirectly by the service water system via the component cooling water system (CCW). In this cooling arrangement, the heat capacity of the CCW may provide additional time to cool the seals. A simplified analysis investigating the expected heatup rate of the CCW system is presented in Appendix C. The results are largely dependent on plant-specific arrangements, and the available time before any damage to the CCW pumps is expected to be in the range of ~30 minutes with an average heat load. Once the cooling to the seals is lost, failure of the seal mechanism may occur resulting in a seal LOCA. The actual leakage through the seals may vary from 20-500 gpm depending on the actual failure mechanism.

The high pressure makeup capability to the primary system is also assumed to be lost, since the HP injection and charging pumps are generally dependent on ESW cooling. In this case the operator may attempt to reduce primary system pressure through steam dump, in order to allow the use of the low pressure injection system. However, the RHR pumps are also expected to fail due to lack of ESW cooling capability resulting ultimately in core damage.

If the RCP seals do not fail and the primary coolant inventory is maintained, the operator can remove heat through the steam generators using the AFW system. The reactor can then be brought to hot standby conditions and may be kept there for as long as there is enough water to supply the AFW pumps. The available time, in this case, is 5-15 hours, which may be sufficient for most cases to recover the ESW system or establish cooling through some alternate method. Core damage may occur if there were a failure of the AFW system during this period or if there were a failure to establish alternate cooling prior to the exhaustion of the AFW system available inventory.

The event tree representing this accident sequence, as developed in Ref.1, is shown in Figure 4.1. The two most important top events, S2 and L-1 represent the probability of a small LOCA (S2), and the unavailability of the auxiliary feedwater system (L-1). There are a number of assumptions built into this particular Byron model such as a) the unavailability of the high pressure injection pumps after a loss of ESW event and b) the probability of the RCP seal LOCA was fixed at 0.5. These assumptions were re-examined and modified for this study using the recovery model discussed in Section 4.2.3.

The AFW system (as modelled in Reference 1) includes a dedicated diesel-driven AFW pump which requires ESW cooling for proper operation. However, based on engineering calculations, it was shown to be able to operate for ~5 hours without any heat removal function upon loss of SW. In this respect, the arrangement is similar to other plants, where steam turbine-driven AFW pumps are installed requiring no auxiliary cooling or additional electrical power.

The final results of the single plant analysis may be summarized by examining the dominant cut sets given in Table 4.1. Each of these cut sets in Table 4.1 represents a particular accident sequence identifying the initiator I(LOSW) along with the probability of the major component that failed to perform its safety function. The actual details of the calculations and numerical data are given in Ref.1.

For the purpose of this study, two groups of cut sets were closely examined. The first contained sequences which are directly initiated by the total loss of the ESW event (Sequences 1 through 7). The most dominant sequence (Sequence 1) essentially represents the contribution of the RCP seal LOCA event to the core damage frequency. The other sequences in this group describe failures in the auxiliary feedwater system, however, their contribution to CDF is substantially lower.

The other group of events indicated at I(Other)*P(Other)*P(ESW) are initiated by the failure of support systems wherein a particular failure or unavailability causes the loss of the ESW system that eventually leads to core damage and an increase of the calculated CDF.

The logic sequences listed in Table 4.1 serve as the basis of this study. All of these sequences were reevaluated with respect to the initiator frequency as established in Section 3. The failure probabilities of each sequence (P(Failure)) were also reevaluated to establish and incorporate time-dependent behavior and recovery actions.

There are certain accident sequences which were not accounted for in the base model (from Reference 1) that are related to loss of room cooling. In particular, the main control room air conditioning equipment relies on the ESW for heat removal. In addition, at some plants, the switchgear room may also be cooled utilizing the ESW system. The analysis of the consequences of the potential loss of cooling in these areas is complex and was beyond the scope of the present study. However, based on engineering judgement and the following considerations, these effects are not considered to significantly contribute to the core damage frequency.

The most dominant accident sequence is due to a relatively large RCP seal LOCA without ESW recovery. The availability of the switchgear room would not affect this particular sequence, since the ECCS pumps (HPI or RHR pumps) cannot be operated to mitigate the RCP seal LOCA due to the unavailability of ESW cooling. The instrumentation and control equipment located in the main control room could potentially degrade over time upon loss of cooling. However, the most dominant failure mechanisms for long-term failure of the ESW are due to external events, SW pump malfunctions and intake structure problems. These failures are relatively easy to diagnose and the potential recovery actions must be accomplished locally not necessarily relying on instrumentation and control located in the control room. In addition, the degradation of the various control equipment is expected to occur over a long time period of 4-10 hours that may allow for "ad hoc" air-conditioning arrangements of air circulation paths to be devised.

An additional consideration is the applicability of the Reference 1 model (based on Byron Unit 1) to the other sites under consideration for GI-130. Three particular features has to be considered:

a. Closed cycle ESW vs. open cycle: The present model was appropriately modified to take into account that the majority of these plants have an open cycle ESW design by including large intake structures with associated pumps and equipment in the model.

b. Diesel vs. turbine-driven AFW pump: The limitation of the running time of the diesel-driven AFW pump (~5 hours) was removed in order to reflect the turbine driven AFW pumps found in the other designs.

c. HPI pump unavailability due to direct ESW cooling: The HPI pump lube oil coolers at Byron are cooled directly by ESW. The HPI pumps at the other multiplant sites are generally cooled using the CCW system which may provide some additional time due to its own heat capacity which may not be present at Byron Unit 1. It will be shown that the time period between the occurrence of the loss of ESW event and the start up of the HPI pumps (upon seal LOCA and decreasing primary system pressure) is on the order of ~2 hours. At the end of 2 hours, the additional time due to the heat capacity of the CCW system is not available since the CCW system itself would heat up much sooner (~30-45 minutes) preventing the operation of the CCW pumps. In this respect, it is therefore believed that there is no major difference between the seven sites and the model based on Byron Unit 1 is believed to be applicable to the other plants.

The analysis in Reference 1 demonstrated the importance of the loss of ESW event and that the consequent RCP seal LOCA accident may become one of the dominant contributors to the total CDF.

4.2.2 Core Damage Frequency Modelling

The modelling of the loss of ESW accident sequence is rather complex due to the variability of the operating configurations of the various multi-units. Regardless of the operating mode, the heat removal function provided by the ESW system is always required. This implies that at least one ESW train must always operate either using the ESW pump associated with that train or, depending on the operational mode, the unit crossties may be substituted to get ESW flow from the other unit.

The possible operating configurations are summarized in Table 4.2. The basic assumption is that Unit 1 requires ESW cooling in its operating mode by having one ESW pump running and the other in either standby or in a scheduled outage for test. In effect, the study analyzes the LOSW events for Unit 1 and the ESW system of Unit 2 is regarded as a potential backup for Unit 1. The other columns indicate the various possible ESW system configurations for the other unit (Unit 2) depending on its operating mode. An additional consideration in State III and IV is the possibility of using the unit crossties to supply ESW for Unit 1. In these cases both ESW pumps of Unit 1 may be intentionally placed in an inoperable status preventing quick recovery if a loss of ESW event should occur. In States III and IV, when both pumps of Unit 1 are shut down, it is assumed to be not required with the possible exception when the unit crossties are utilized.

The loss of ESW accident sequence as analyzed in Reference 1 essentially represents single unit operation where the ESW system of the other unit including the crosstie is not available.

The initiating frequency of the LOSW sequence in Reference 1 was established differently from the method described in Section 3. In this respect, the CDF model was renormalized using the initiating frequency determined in Section 3 which takes into account the experience base covering the direct and indirect loss of the ESW system. The LOSW initiating frequency was determined using data based on experience for single unit operations. However, for multi-unit operations the operating configurations may be different thus requiring the determination of their respective initiating frequencies.

The initiating frequencies derived in Section 3 are assumed to be valid for any single unit and are not specifically limited to multi-unit sites. In general, single unit ESW systems for PWRs have built-in redundancy such as two independent SW trains and more specifically three full capacity pumps. The actual design arrangement at the multi-units is very similar with the exception that the third pump is available through the unit crosstie and this pump is not fully dedicated to the first unit. These differences do not appreciably change or degrade the availability of the total ESW system at the multi-unit sites as compared to single units when both units are at power and the ESW systems are in the run/standby mode (one pump runs and the other is on standby in both ESW systems). Therefore, it was assumed that the initiating frequencies as established in Section 3 based on single unit data are directly applicable as multi-unit loss of ESW system frequencies, that is

$$\lambda(\text{Single Unit, R/SB}) \approx \lambda(\text{Multi-Unit, R/SB})$$

where R/SB indicates run/standby mode. The λ(Multi-Unit, R/SB) indicates the frequency of the loss of the ESW system to a single unit of the multi-unit plant i.e., the unit has lost its ESW system and for some reason the other unit is unable to supply back-up via the unit to unit crosstie. It does not represent the loss of ESW at both units.

To derive initiating frequencies for other possible multi-unit operating modes not covered by the single unit data, an approximation method was used. Effectively, the combination of the single unit data experience with an analytical technique was used. In this approach a multi-unit ESW system fault tree was developed from the existing model of Reference 1. This upgraded multi-unit model basically represents the unavailability of the second unit to supply SW to the first unit, given the failure of ESW on the first unit.

The multi-unit fault tree (provided in Appendix D) was used to predict the primary failure mechanisms and their magnitude. It represents the failure to supply ESW flow from the second unit. The primary failure mechanisms are 1) failure of an operating SW train, 2) failure of a standby SW train, and 3) crosstie failure through operator error.

The failure probabilities related to ESW train operations are listed in Table 4.3 . The sum of the appropriate failure probabilities gives the total failure probability of Unit 2 to provide SW flow to Unit 1 upon demand in a particular state and is noted as

P(2/1-R/SB) = Unit 2 fails to provide SW flow to Unit 1 - Unit 2 ESW system is in run/standby.

The P(2/1-State) (State could be R/AOT, R/SB, etc.) may be calculated with the use of the fault tree for the different operating configurations. Table 4.3a lists the results including considerations for two different success criteria. In State I when both units are at power, the success criterion is that two ESW pumps are required on the second unit given the loss of ESW system on the first unit (one ESW pump to serve Unit 2 and the standby Unit 2 ESW pump would remove heat from Unit 1). In the other states (II, III, IV) only one SW pump is sufficient. In this case, Unit 2 is in cold shutdown and one of its respective ESW pumps is removing decay heat. If a LOSW event occurs at Unit 1, the unit will go on hot standby status. The SW requirement is much lower in this mode vs. shutdown mode since decay heat is removed through the secondary side utilizing the auxiliary feedwater system and steam generator. This SW requirement may be satisfied by opening the unit crosstie and diverting SW flow from the one operating Unit 2 SW pump. These pumps have sufficient reserve capacity to handle the additional heat load requirement.

The following notation is introduced to facilitate further discussion. The initiating frequency for a given configuration of the ESW system is noted as

$$\lambda(R/SB,\ R/SB) = \lambda\ (\text{Unit 1 ESW pumps in run/standby, Unit 2 ESW pumps in run/standby})$$

The notation R/AOT (allowable outage time) will mean that the corresponding ESW pumps are in run/test condition (one running and the other is in test or maintenance).

Two important relations are introduced with respect to single- and multi-unit operations.

In a single-unit arrangement of a multi-unit design, there is one train in operation and the other is either in standby or testing. The initiating frequency of a LOSW for a run/standby (R/SB) configuration (without accounting for the unit crosstie) may be obtained from a run/test (R/AOT) condition as

$$\lambda(R/SB) = \lambda(R/AOT) * P(FS + FR + \beta)$$

Where $P(FS + FR + \beta)$ represents the failure of a standby ESW train and consists of the sum of the three relevant failure probabilities as listed in Table 4.3. The ESW standby train could fail to start (FS), could fail to run (FR), and there is a common mode factor through the coupling of the bays in the intake structure. The above relation may be rearranged as

$$\lambda(R/AOT) = \lambda(R/SB) * \frac{1}{P(FS + FR + \beta)} \qquad (1)$$

Considering the actual two-unit configuration (both units at power) we get the following relations for Unit 1 loss of ESW frequency (with crosstie):

$$\lambda(R/SB,\ R/SB) = \lambda(R/SB) * P(2/1 - R/SB)$$

The first term on the right hand side is the frequency of loss of ESW at Unit 1 of the multi-unit site (without crosstie) and the second is the probability that Unit 2 is unable to supply SW flow to Unit 1.

The other important relationship is obtained by rearranging the above equation to get

$$\lambda(R/SB) = \lambda(R/SB,\ R/SB) \frac{1}{P(2/1 - R/SB)} \qquad (2)$$

These two equations (1 and 2) may be used to derive and express the state-dependent initiating frequencies in terms of the basic data derived in Section 3. The operating data provided a basis to establish $\lambda(R/SB,\ R/SB)$, hence all other initiating frequencies (during power operations) are expressed as a function of $\lambda(R/SB, R/SB)$.

In State I with success criterion I the following results:

<u>State Ia</u>: Unit 1: R/AOT, Unit 2: R/AOT

$$\lambda(R/AOT, R/AOT) = \lambda(R/AOT) * P(2/1 - R/AOT)$$

Utilizing Equations (1) and (2)

$$\lambda(R/AOT, R/AOT) = \frac{\lambda(R/SB, R/SB)}{P(FS + FR + \beta)} * \frac{P(2/1 - R/AOT)}{P(2/1 - R/SB)}$$

<u>State Ib</u>: Unit 1: R/AOT, Unit 2: R/SB

$$\begin{aligned}\lambda(R/AOT, R/SB) &= \lambda(R/AOT) * P(2/1 - R/SB) \\ &= \lambda(R/SB, R/SB) * \frac{1}{P(FS + FR + \beta)} * \frac{P(2/1 - R/SB)}{P(2/1 - R/SB)}\end{aligned}$$

<u>State Ic</u>: Unit 1: R/SB, Unit 2: R/AOT

$$\lambda(R/SB, R/AOT) = \lambda(R/SB, R/SB) * \frac{P(2/1 - R/AOT)}{P(2/1 - R/SB)}$$

<u>State Id</u>: Unit 1: R/SB, Unit 2: R/SB

$$\lambda(R/SB, R/SB) = \lambda(R/SB, R/SB)$$

It is clearly seen that the value based on operating experience $\lambda(R/SB, R/SB)$ is combined with analytical results representing the different modes of operation.

The relations in State II (Unit 1 at power, Unit 2 in shutdown) are somewhat more complicated due to the different success criterion. Considering the initiating frequency with regard to the different success criterion we obtain

$$\begin{aligned}\lambda(R/SB, R/SB)_{-2} &= \lambda(R/SB) * P(2/1 - R/SB)_2 \\ \lambda(R/SB, R/SB)_{-1} &= \lambda(R/SB) * P(2/1 - R/SB)_1\end{aligned} \qquad (3)$$

where the subscript 2 refers to a success criterion when both (2) RSW pumps are required on the second unit and 1 is when one pump is sufficient. This leads to

$$\lambda(R/SB, R/SB)_1 = \lambda(R/SB, R/SB)_2 * \frac{P(2/1 - R/SB)_1}{P(2/1 - R/SB)_2}$$

The actual value of $P(2/1 - R/SB)$ with the different success criteria are given in Table 4.3a ($9x10^{-2}$ and $7.3x10^{-2}$).

State IIa: Unit 1: R/AOT, Unit 2: R/AOT

$$\lambda(R/AOT,\ R/AOT)_1 = \lambda(R/AOT) * P(2/1 - R/AOT)_1$$

$$= \lambda(R/SB)\ \frac{1}{P(FS + FR + \beta)} * P(2/1 - R/AOT)_1$$

$$= \lambda(R/SB,\ R/SB)_1\ \frac{1}{P(2/1 - R/SB)_1} * \frac{1}{P(FS + FR + \beta)} * P(2/1 - R/AOT)_1$$

$$= \lambda(R/SB,\ R/SB)_2\ \frac{P(2/1 - R/SB)_1}{P(2/1 - R/SB)_2} * \frac{1}{P(2/1 - R/SB)_1} * \frac{P(2/1 - R/AOT)_1}{P(FS + FR + \beta)}$$

$$= \lambda(R/SB,\ R/SB)_2 * \frac{1}{P(FS + FR + \beta)} * \frac{P(2/1 - R/AOT)_1}{P(2/1 - R/SB)_2}$$

where Equation (1), (2) and (3) were used to get the final expressions. For the other states only the final result is given, since the operations are identical.

State IIb: Unit 1: R/AOT, Unit 2: R/SB

$$\lambda(R/AOT,\ R/SB)_1 = \lambda(R/SB,\ R/SB)_2 * \frac{1}{P(FS + FR + \beta)} * \frac{P(2/1 - R/SB)_1}{P(2/1 - R/SB)_2}$$

State IIc: Unit 1: R/AOT, Unit 2: SB/M

$$\lambda(R/AOT,\ SB/M)_1 = \lambda(R/SB,\ R/SB)_2\ \frac{1}{P(FS + FR + \beta)}\ \frac{P(2/1 - SB/M)_1}{P(2/1 - R/SB)_2}$$

State IId: Unit 1: R/AOT, Unit 2: M/M

$$\lambda(R/AOT,\ M/M) = \lambda(R/SB,\ R/SB)_2\ \frac{1}{P(FS + FR + \beta)}\ \frac{P(2/1 - M/M)_1}{P(2/1 - R/SB)_2}$$

State IIe: Unit 1: R/SB, Unit 2: R/AOT

$$\lambda(R/SB,\ R/AOT)_1 = \lambda(R/SB,\ R/SB)_2\ \frac{P(2/1 - R/AOT)_1}{P(2/1 - R/SB)_2}$$

State IIf: Unit 1: R/SB, Unit 2: R/SB

$$\lambda(R/SB,\ R/SB)_1 = \lambda(R/SB,\ R/SB)_2 \frac{P(2/1 - R/SB)_1}{P(2/1 - R/SB)_2}$$

State IIg: Unit 1: R/SB, Unit 2: SB/M

$$\lambda(R/SB,\ SB/M)_1 = \lambda(R/SB,\ R/SB)_2 \frac{P(2/1 - SB/M)_1}{P(2/1 - R/SB)_2}$$

State IIh: Unit 1: R/SB, Unit 2: M/M

$$\lambda(R/SB,\ M/M)_1 = \lambda(R/SB,\ R/SB)_2 \frac{P(2/1 - M/M)_1}{P(2/1 - R/SB)_2}$$

The conditions during shutdown operations are much simpler due to the state-specific initiating frequencies. In this case, the initiating frequency corresponds to the single unit of the multi-unit site and only the availability of the second unit must be considered that is in shutdown

$$\lambda(R/AOT,\ R/\ AOT) = \lambda(R/AOT) * P(2/1 - R/AOT)$$

where $\lambda(R/AOT)$ is derived in Section 3 (similarly at power $\lambda(R/SB)$ is also available). Therefore, in shutdown states the value of the initiating frequency for a given state is multiplied with the failure probability of the second unit failing to provide ESW flow to get the state-specific initiating frequency. Table 4.3b lists the base initiating frequency for each state (from Section 3) and the corresponding value of the probability multiplier as derived previously. For example, for State IIa:

$$\lambda(R/AOT,\ R/AOT)_1 = \lambda(R/SB,\ R/SB)_2 * \frac{1}{P(FS + FR + \beta)} * \frac{P(2/1 - R/AOT)_1}{P(2/1 - R/SB)_2}$$

$$= \text{Base Initiating Frequency} * \text{Modifier}$$

where

Base Initiating Frequency $= \lambda(R/SB,\ R/SB)_2$,

$$\text{Modifier} = \frac{1}{P(FS + FR + \beta)} \frac{P(2/1 - R/AOT)_1}{P(2/1 - R/SB)_2}$$

Table 4.3b lists the final initiating frequency values for each state as used in this study. An important point to emphasize is that the particular model used in this study incorporated a set of success criteria shown in Table 4.3a (see notes in Table 4.3a). If both units are operating, it was assumed that upon a loss of ESW event (Plant A) both ESW pumps of the other unit (Plant B) must be available in order to shut down both plants. For some multi-units this assumption may be somewhat conservative, since it is conceivable that the reserve capacity of one ESW pump may allow the units to reach hot standby conditions even using only one ESW pump. However, it is unlikely that cold shutdown conditions can be established without recovering additional cooling capacity.

If Plant B is in a shutdown mode, the success criterion was that the cooling capacity of one Plant B ESW pump may be sufficient to bring Plant A down to safe shutdown conditions given the loss of ESW at Plant A. This may not be trivial, since residual heat still has to be removed from Plant B. However, the use of AFW and at later stages feed and bleed operation may extend the time available before core uncovery to at least 10-15 hours or more. This time period should be sufficient to recover additional cooling capacity or the heat removal requirements of the unaffected unit could be lowered to such a level where the capacity of one ESW pump would be sufficient for both units.

The core damage model used in Reference 1 and adopted with modification in this study refers to an operating plant analyzing abnormal events which may prevent normal shutdown. In States III and IV one or both units of the plant are in the shutdown mode, and require the operation of the ESW system, since residual heat has to be removed. In this respect the base model is not applicable, since the accident sequences leading to core damage in shutdown may be different from sequences in the operational mode. One of the most dominant sequences in the operational mode, reactor coolant pump seal LOCA, is not fully applicable in the shutdown mode.

However, the loss of ESW event in this mode could also be very serious, since it directly leads to the loss of the RHR heat exchangers and cooling for the RHR pumps. In addition, the HP injection and charging pumps may also be affected by loss of cooling to their lube oil or bearing components and consequently losing the makeup capability to the reactor coolant system. Alternate heat removal from the primary system using the AFW system and steam generator and/or the feed and bleed operation may not be available due to maintenance or other activities. On the other hand, there may be substantially more time available for the recovery of the ESW system. Depending upon the length of time at shutdown conditions, the time available before the primary system heats up and boils could be anywhere between ~2-10 hours. A simplified shutdown risk model was developed to account for these specific characteristics and is discussed in Section 4.2.3.6.

For each of the states listed in Table 4.2, a conditional core damage frequency (CCDF) can be calculated using the above discussed model. In essence, each sequence of the original base case was renormalized by the respective configuration-dependent initiating frequency. The sum of the renormalized sequences represents the conditional core damage frequency of that particular state. The CCDF may be obtained by weighing the state-dependent initiating frequency (λ_1 (state)) with the corresponding sequence failure probability P(Sequence). The total CDF is calculated by considering the relative time fractions of each state (RT) and multiplying by the state CCDFs to get

$$CDF = \sum_i \lambda_i(\text{State}) * P_i(\text{Sequence}) * RT_i = \sum_i CCDF_i * RT_i .$$

The various state-dependent initiators (λ_i's) are determined in this section. The sequence failure probabilities are discussed in the next section (4.2.3) and the relative time fractions and their actual determination are discussed in Section 4.3.

4.2.3 Sequence Failure Probabilities/Recovery Model

Table 4.1 lists the dominant failure sequences in a LOSW accident scenario. The various sequences may be grouped into the following logical classes considering the status of the unit and the time dependence of the accident:

a) RCP seal LOCA (P_{SL})
b) Failure of the AFW system - short term (P_{AFW})
c) Loss of water supply and consequent failure of the AFW system - long term (P_{LAFW})
d) Other sequences (P_{Other})
e) Shutdown scenario (P_{SH})

The model in Reference 1 included logical classes a, b, and d without considering time-dependent behavior. In addition, the probability of RCP seal LOCA was arbitrarily fixed at .5 without any physical consideration of the failure mechanisms and timing involved.

In the following sections, logical and numerical models are developed to quantify the sequence failure probabilities. In the first step, a simple recovery model is determined for the ESW system based on the operating experience.

4.2.3.1 ESW Recovery Model

The ESW system recovery may be established by considering the data in Table 3.1 regarding each event. The data suggest that there are, on the average, three characteristic regions of recovery. The first (which lasts about 1 hour) is when a large fraction of the ESW events (~65-70% of the total) would be recovered. This time period primarily reflects events caused by operator error or misjudgment and involves, in general, support system failures.

The second time period (which lasts ~4-5 hours) involves more problematical hardware or other failures and at the end of ~5 hours about 90-95% of all events would be recovered. The last group of events are such that recovery may take a relatively long time and generally involve some serious hardware problem. It is estimated that by the end of 24 hours about only 1% of the ESW events would not be able to recover.

Two events (A.1 and A.2) took exceptionally long times to recover estimated at 10-15 hours and a few days respectively. However, these recovery times are not representative of actual events for the following reasons.

In case of A.1, severe weather caused icing and consequent malfunctioning of the ESW system. The reactor was not fueled at this time, hence there was no great urgency to correct the problem. The flooding of the intake structure (A.2) is actually anticipated at much higher flood levels and the plant is able to switch over and rely on the dedicated SW pond for all safety related cooling requirement. This again indicates that recovery may have been delayed, since all safety requirements were met. The actual recovery times are judged to be somewhat shorter given the loss of safety related heat removal, but still could take considerable time.

Based on these data and engineering judgement, the average non-recovery fractions of the ESW system (SWR) were established as listed in Table 4.4. The specific times listed in Table 4.4 may be used to construct the time-dependence of the ESW non-recovery using simple linear interpolation for times other than the listed ones.

4.2.3.2 RCP Seal LOCA

4.2.3.2.1 Recovery Theory

In Reference 1 the probability of a seal LOCA given the loss of ESW was assumed to be $P_{SL} = .5$. The reference further stated that that value of P_{SL} may not even be conservative, since the average repair time of the ESW system was judged to be ~20 hours and by that time core uncovery may occur even at relatively low RCP seal leak rates. BNL determined that this model was simply too crude for the present analysis.

In order to establish a more realistic value for the seal LOCA probability, a simple recovery model with updated seal failure probabilities was incorporated in this study. The RCP seal failure probabilities are based on the model developed in NUREG-1150[2] which gives the probability of a leak as a function of the leak rate and elapsed time after the loss of cooling or onset of the accident. The NUREG-1150 model, which is essentially based on expert judgement, is shown in Table 4.5 (Table 5.4.2 of Reference 2) and includes the probability of RCP seal failure as a function of time and leak rate for both the present O-ring (old) and a proposed new O-ring design. The calculations are based on the present O-ring data.

The time dependence of the seal LOCA occurrence is also based on the NUREG-1150 model. There is a relatively small probability of a seal LOCA occurring in the first 1.5 hours after the loss of ESW and seal cooling (less than ~5% according to the expert opinion).* Therefore, it was assumed that initially the RCP seals remain intact upon loss of cooling and the integrity of the seal is maintained during the first 1.5 hours. Beginning at 1.5 hours, the seals are assumed to leak with a probability of unity. The size of the leak is determined probabilistically as shown in Table 4.5.

If there is an RCP seal LOCA after a loss of ESW event occurs, the operating and/or emergency procedures require the operator to shut the RCPs down to reduce the primary system pressure which decreases the coolant loss. The resulting pressure drop would automatically start the HP injection pumps to

*Verbal communication, J. Jackson of U.S. Nuclear Regulatory Commission, February 7, 1990.

provide makeup capability. Since the HP injection and/or charging pumps require direct (or indirect) ESW cooling, they could become inoperable after a few minutes of operation without the ESW flow.

In order to incorporate the probability information contained in Table 4.5 into the sequence models, it is necessary to calculate a time of the initial seal failure and a core uncovery time for each possible failure scenario. Therefore, it is necessary to define a series of individual scenarios which identify the time of seal failure, the initial leak rate, the progression of leak rate, and the probability of the scenario. The data in Table 4.5 were used to develop these scenarios.

The various seal leak scenarios were collapsed to seven groups as indicated in Table 4.6. The table contains information about the initial leak rate, any increases in the leak rate, the time at which the leak rate increases and the probability (P_{SL}).

The total core damage probability contribution from $P_{SL}(t)dt$ may be obtained if the probability of non-recovery of the ESW is also considered. If there were no recovery, then the core damage probability function is simply $P_{SL}(t)dt$ and the total is $\int_o^T P_{SL}(t)dt$.

However, there is a finite probability of ESW recovery, and one has to include only those cases that were not recovered by time t, i.e., $NR(t)*P_{SL}(t)dt$. Here, NR(t) is the integrated or total non-recovery function indicating the fraction of total events unable to recover by time t.

Therefore, the total core damage probability is just:

$$P(\text{Seal LOCA}) = \int_o^T NR(t)*P_{SL}(t)dt.$$

The non-recovery function represents the non-recovery of systems and is better described by using an explicit event tree of the function.

The potential recovery actions are essentially related to the preservation or recovery of the high pressure injection function and the recovery of the ESW system itself. An event tree depicting the progress of the recovery actions is shown in Figure 4.2. Other specific recovery actions, such as the establishment of ad hoc alternative RCP seal injection and/or cooling, or temporary feed and bleed operation of the CCW system or establishing an alternate cooling source for the HPI, charging and RHR go well beyond the scope of this study and are therefore not considered. However,the addition of a dedicated RCP seal cooling system has been included as a potential upgrade and is analyzed in Section 5.

The recovery event tree describes the possible actions and outcomes of the loss of ESW accident at any given time into the accident. Each of the top events are time-dependent and therefore, the end states of the event tree are also time-dependent quantities. The first top event establishes the fraction of events where the injection capability is either lost or preserved through operator action. Next, the question of restoration of the injection capability is asked given the initial loss of this function. The ESW cooling capability is examined next, whether it is available to remove heat from the affected equipment. The final top event is the probability that the leak rate of the seal LOCA, is such that it leads to core uncovery.

The following is a brief description of the sequences:

- Sequence 1: HP injection capability is preserved, ESW is recovered at time t, no core damage.
- Sequence 2: HP injection capability is preserved, ESW is not recovered at time t, core damage.
- Sequence 3: HP injection capability is initially lost, but recovered by time t, ESW system is also recovered.
- Sequence 4: HP injection capability is recovered, but the ESW system is unable to provide cooling capacity, core damage.
- Sequence 5: HP injection capability is lost and is not recoverable, core damage. The recovery of the ESW system is irrelevant in this case.

Sequences 3 and 4 assume that the HP injection function may be recovered given its initial loss. This may be explained in the following manner.

Given the loss of the ESW function and consequent RCP seal LOCA event, the HPI pumps are started to provide injection flow. Due to the lack of heat removal, the bearing lube oil of the HPI pumps heats up causing some damage to the pumps. In these sequences (Sequences 3 and 4), the operator fails to recognize the event and/or alarm signals and does not act to prevent the damage. Implicitly, it is also assumed that the limited injection capability provided by the charging pumps is lost, since these pumps, in general, also require heat removal by the ESW system.

In our judgement, the recovery of some injection capability seems possible in certain cases over the relatively long time period of 24 hours. It is unrealistic to assume that all HPI or charging pumps are damaged exactly in the same manner or at the same time. Thus, major damage to subsequently failing pumps could be limited by operator intervention.

During the many discussions with plant operating personnel that we had as part of this project, it was clearly indicated that the operational staff are highly sensitized to prevent major ECCS equipment damage, even if that may mean the temporary loss of injection. Therefore, it is reasonable to assume that some of the two HPI pumps (or three at some plants) and three charging pumps will have limited or repairable damage.

The total conditional core damage probability due to RCP seal LOCAs may be calculated as the sum of the seven scenarios (see Table 4.6) weighted with the respective non-recovery probabilities, i.e.,

$$P(\text{Seal LOCA}) = \sum_{i=1}^{7} NR_i(t) * P_{SL_i}(t_{SL}) = \sum_{i=1}^{7} NR_i(t_{SL} + \lambda_i) * P_{SL_i}(t_{SL})$$

where

- λ_i = core uncovery time associated with the ith scenario
- t_{SL} = time when seal LOCA occurs
- t = time associated with loss of ESW accident (t=0 accide ~ initiated)
- NR_i = non-recovery probability of the ESW system by time t
- P_{SLi} = probability of the ith seal LOCA scenario

The actual timing of the events such as the occurrence of the RCP seal LOCA and potential core uncovery is a complex thermal-hydraulic problem. In this study, the following simple approach was chosen to estimate the time evolution associated with the RCP seal LOCA scenario.

In Appendix C a short discussion is given with regard to the expected heatup rate of the CCW system after the loss of ESW event. The primary conclusions are that the CCW system will rapidly heat up to the spent fuel pool temperature (which is generally between ~120-140°F). This initial time period may last about ~15-20 minutes. The heat up rate will then considerably slow down due to the reverse heat transfer process that begins. Heat is then transferred to the spent fuel pool from the CCW slowly heating the pool.

The limiting factors with respect to the RCP seal cooling are a) RCP motor bearing and seal cooling (limit ~180°F), b) CCW motor/pump cooling (limit ~120-150°F), and c) spent fuel recirculation cooling pump (limit ~120-150°F). It is clear that after the CCW system heats up to the spent fuel pool temperature the most limiting condition is the loss of cooling to the CCW pumps/motors. The increase in the CCW pump cooling temperatures would limit its operation to a relatively short time. Based on engineering judgement, it is assumed that after about 15-20 minutes of operation with the elevated cooling temperatures the operator would turn these pumps off to prevent major damage.

In the last phase with CCW circulation stopped the RCP seal cooling temperatures again rapidly increases to 180°F or above. The sum of the two time periods a) initial heat up to spent fuel pool temperature and b) loss of CCW circulation results in ~30-40 minutes before the total loss of cooling to the RCP seals.

The time evolution is described in Table 4.6a and the core uncovery times λ_i as given in NUREG-1150 for each cf these scenarios as well as the times $t_{SL} + \lambda_i$ are also listed in Table 4.6b. The value of $t_{SL} + \lambda_i$ is the time available for ESW recovery before core damage and is discussed in the next section.

4.2.3.2.2 Seal LOCA Recovery Model Quantification

The time dependence of the ESW non-recovery function [ESW(NREC)] may be associated with the core uncovery times to determine the numerical value of ESW (NREC) at the time potential uncovery. Table 4.7 summarizes the probability of each leak path/scenario, the core uncovery times ($t_{SL} + \lambda_i$) and the respective non-recovery probabilities of the ESW system. The value of ESW (NREC) was established using linear interpolation in Table 4.4 to the appropriate time.

In Table 4.7b the recovery values associated with the HP injection system are shown. In the seal LOCA scenario initiated by the loss of ESW, the primary system pressure drops and the HP injection pumps are eventually initiated to provide makeup. However, these pumps require cooling that is either directly or indirectly supplied by the ESW system. The available time before the HP pumps are damaged may be estimated by using the calculational results of Reference 4. Various sizes of small LOCA scenarios were analyzed and the reference indicated that depending on the initial flow rate, the amount of time before the HP pumps start up may range from 2-15 minutes. Once the pumps start, it is assumed that about 2-5 minutes will elapse before substantial damage occurs to the pumps.

The probability of losing the injection capability is entirely determined by operator action. The recommended nominal HEP values from Table 20-3 of Reference 3 were used and are reproduced in Table 4.7c. These HEP values represent diagnostic error probabilities as a function of available time.

One of the most important aspects of the loss-of-ESW event and a consequent RCP seal LOCA is the potential unavailability of the HPI system. The sequence of events unfolds in the following manner.

For about 1.5 hours after the total loss of seal cooling, the RCP seals remain intact (according to the expert opinion model). Once the RCP seal failure occurs (at time 1.5 hours into the loss of seal cooling), the primary system pressure drops and the HPI system starts up providing injection flow. The available time for operator action to secure the HPI pumps is limited.

However, the operator has ample time in the first 2.0 hours to diagnose the loss of ESW event and is expected to be working on it at this time if the ESW system is not yet recovered. Generally, there are numerous indicators, such as ESW coolant or lube oil temperature and/or loss of ESW coolant flow to lube oil heat exchangers alarms, warning the operators of the abnormal conditions at each affected piece of equipment. Therefore, it is quite reasonable to assume that the operator in the first 2 hours is able to diagnose the conditions and could consult the procedures about the particular actions to be taken at each affected piece of equipment.

It was also apparent during our discussions with the operators that they place a great emphasis on saving any ECCS equipment given abnormal conditions. The main conclusion is that by the time the RCP seal LOCA occurs the operator should be fully aware of the lack of cooling to some of the ECCS equipment and in spite of the need for injection, will in most likelihood, terminate the operation of the HPI pumps.

Even though the emergency procedures may not be explicitly clear about this aspect of the LOCA operations, there are references to this coupling between the primary injection function and support system operation. As an example, portions of a generic Westinghouse "Reactor Trip or Safety Injection, E-0" emergency procedure are included as Table 4.7d. Step 8 of the procedure reminds the operator to check if the HPI pumps are running. At Step 10 the operation of the ESW system is checked. Although the instructions are not very clear with respect to a total loss of the ESW system and its relation to the potentially affected ECCS equipment, it clearly calls the operator's attention to this coupled problem.

The numerical values of the human error rates incorporated in the present model were established using Reference 3 which is the normally accepted industrial standard for selecting time dependent HEPs. Table 4.7c (reproduced from Reference 3) lists the recommended values and was used to select the appropriate HEPs.

The crucial feature of our model is the credit given for the first initial time period (about 2.0 hours, .5 hour for CCW heat up and 1.5 hours for RCP seal damage) when there is the condition of loss of ESW but no RCP seal LOCA. If this credit were not included, i.e., essentially assuming that the operator would start to diagnose the problem only when the HPI pumps start up, the recommended HEP from Table 4.7c would be between .1-.5. The actual HEP values used in this study to represent operator errors connected to the unavailability of the HPI function is shown in Table 4.7b (fourth column) and are $1x10^{-3}$ and $5x10^{-4}$.

A stress factor of 10 was applied to the nominal HEP values due to the nature of these events which are unusual in the sense that the operator has to terminate HP injection when the primary system pressure is decreasing, in order to preserve HP injection capability for a later time when the ESW cooling function is recovered.

Table 4.7b indicates the estimated time before the HP pumps would be damaged (third column) and the nominal HEP values used in each time period associated with the available time for recognition of the problem and operator action. In the final column, the non-recovery fractions of the HP injection capability are listed. These values are based on engineering judgement and discussion with plant operators.

The estimation of the average repair time in such emergency situations depends on spare parts availability or interchangeability between the pumps, repair experience, etc. Based on the projection by plant personnel, the repair, depending on the extent of the damage, may last anywhere between 2-24 hours or even as long as 2-3 days in the worst cases.

The numbers used in this study are based on actual discussions with plant maintenance personnel and our engineering judgement. The model assumes that ~5% of the lost injection capability could be recovered in the time period of 1.5-4.5 hours and ~30% in 4.5-24 hours. These numbers do not reflect experience since there are no data available and only engineering judgment as well as operator estimation is possible.

Table 4.8 lists the results of the calculations for each sequence (see Figure 4.2). The quantifications of the seal LOCA probability and the associated non-recovery factors are contained in Tables 4.7a-c.

The sum of the sequences, P(Seal LOCA) = 6.8×10^{-2} represents the conditional core damage probability over 24 hours given an RCP seal LOCA event due to the loss of the ESW system.

The most dominant contribution arises from a small fraction of those cases where the leak rate is relatively large (~1000 gpm) and the ESW system is unable to recover before core damage occurs (Sequence 2, leak path 1000 and 240/1000). These two sequences contribute about 87% of the total.

4.2.3.3 <u>Failure of AFW System - Short Term</u>

The sequences relating to the failure of the AFW are considered next. In this case the RCPs are shut down and heat removal through the steam generators is initiated using the AFW system. The failure of this system prevents heat removal resulting in a heat up of the primary coolant which eventually leads to the opening of the power operated pressure relief valves (PORV) discharging coolant from the RC system. This sequence is essentially a small LOCA through the PORVs.

The core damage contribution from the failure of the AFW may be written as

$$CDF = \lambda * (\bar{P}_{SL} * AFW * P_{NR}) = \lambda * P_{AFW}$$

where

λ = initiating frequency
$\bar{P}_{SL}$ = $1 - P_{SL}$
P_{SL} = probability of core damage due to seal LOCA
AFW = probability of AFW failure through random fault
P_{NR} = non-recovery probability of AFW system

A simple time-independent recovery model was constructed based on the available time of about one to two hours. This time period was determined using the results of deterministic LOCA calculations documented in Reference 4. The recovery model basically consists of a number of recovery factors (P_{NR}) which are established based on the particular failure mode appearing in the cut set and the time available.

The respective operator/recovery event tree is shown in Figure 4.3. The first top event is similar to the seal LOCA event and expresses the probability that the operator is unable to stop the HP pumps in order to prevent major damage to the pumps and consequently the primary system makeup capability is lost. The next top event examines the probability of recovering the components which leads to the loss of the AFW system. This is represented by the various failure modes of the AFW diesel-driven pump (Byron) such as fails-to-start, fails-to-run or the pump may be in maintenance. (For steam turbine-driven AFW pumps the failure modes are similar and the recovery factors would not change substantially.)

The AFW system includes one electric motor driven pump in addition to the diesel driven one and one electrically driven startup pump. However, the motors require direct ESW cooling and are assumed to be failed due to the lack of ESW flow. Certain recovery factors are applied to each of these failure modes which were established based on technical and engineering evaluation. The last top event refers to the probability that the ESW system may be recovered before the core is damaged. It was assumed that the available time is about 1 hour and based on data from Table 3.1, where approximate recovery times are indicated, the probability of recovering the ESW system was established at $P_{ESW} = .54$ (non-recovery = $1 - P_{ESW}$).

The calculation of the non-recovery factors P_{NR} for each failure mode proceeds by summing the probabilities of failures that lead to core damage, i.e., for pump fails-to-start (see Figure 4.3).

$$P_{NR}(\text{pump-fails-to-start}) = \sum_i SEQ_i$$

$$\begin{aligned} P_{NR} &= OP\text{-}1 * (1\text{-}x) * \overline{OP\text{-}3} + \overline{OP\text{-}1} * (x) * \overline{OP\text{-}3} + OP\text{-}1 * (1\text{-}x) \\ &= .5 * (1-.2) * .46 + .5 * (.2) * .46 + .5 * (1-0.2) = .63 \\ &= P_{NR}\ (\text{AFW pump-fails-to-start}). \end{aligned}$$

The resulting non-recovery factors are listed in Table 4.9 for each sequence. The numbers seem to indicate that recovery is most effective in reducing the CDF contribution of the AFW pump fails-to-start sequence.

The core damage contribution represented by P_{AFW}, that is, the probability of core damage due to the failure of the AFW system is listed in Table 4.10 for each failure mode. The sum of the sequences, $P_{AFW} = 2.25\times10^{-2}$ is the conditional CDF which must be multiplied with the respective loss of ESW initiating frequency. The most dominant failure mode is when the AFW pump fails to start with a contribution of ~60%.

4.2.3.4 Failure of AFW System - Long Term

The loss of ESW accident scenario may result in a relatively long term sequence where SW cooling may not be recovered for a long period of time. Even if the reactor is brought to hot standby conditions utilizing the AFW system and steam generator, there is a requirement for long term water supply to the secondary side (that is, to the AFW system).

The question of long term water supply to the AFW is somewhat complicated due to the variety of options and alternatives available to the plant operators and varies from plant to plant. For the plants under consideration in GI-130, the available water resources are varied and depending on the plant may provide sufficient water supply up to 12-20 hours of AFW operation. This may even be extended in some cases with temporary connections to other water sources.

The total available time was estimated to be between 20-24 hours before core uncovery. This is based on the available water supply lasting on the average 15-18 hours and an additional 3 hours for core uncovery after the AFW system runs out of water. An additional ½-1 hour credit may also be taken into account with regard to the availability of the CCW system at the start of the accident. During the initial ½-1 hour the CCW system slowly heats up (spent fuel pool provides an additional heat sink) and the plant may reduce power or could initiate regular shutdown using the secondary side with the main feedwater system still operating. This provides an additional time period where the AFW system would not be placed in operation.

The core damage contribution from the long term failure of the AFW system may be written as

$$CDF = \lambda * \overline{P_{SL}} * \overline{P_{AFW}} * AFWR = \lambda * P_{LAFW}$$

where

$\overline{P_{AFW}}$ = $1-P_{AFW}$ (see Section 4.2.3.3)

$\overline{P_{SL}}$ = $1-P_{SL}$ (see Section 4.2.3.2)

AFWR = non-recovery of ESW system by the time of core uncovery - 20-24 hours

The numerical value of AFWR ~0.01 from Table 4.4 and the final P_{LAFW} is

$$P_{LAFW} = \overline{P_{SL}} * \overline{P_{AFW}} * AFWR = (9.32x10^{-1}) * (9.78x10^{-1}) * .01$$

$$P_{LAFW} = 9.1x10^{-3}$$

4.2.3.5 Residual Sequences

The model in Reference 1 includes a number of minor sequences with relatively insignificant contribution to the CDF. These sequences primarily represent accident scenarios initiated by events other than the loss of ESW, but contain the failure of the ESW through random faults.

The core damage contribution is

$$CDF = \lambda * \overline{P_{SL}} * P = \lambda * P_{Other}$$

where

$\overline{P_{SL}}$ = $1-P_{SL}$

P = conditional core damage probability due to "other" sequences.

The base probability number or the normalized sum of P_{Other} in Reference 1 is $P_{Other} = 3.4x10^{-3}$, which was used to quantify P_{Other}

$$P_{Other} = \overline{P}_{SL} * P_o = (9.32x10^{-1})(3.4x10^{-3}) = 3.1x10^{-3}.$$

4.2.3.6 Shutdown Model

The risk associated with the operation of the plant depends on the particular mode of operation at each unit. The potential accident sequences and the response of the safety systems is also strongly coupled to the reactor operational mode. In this respect, when the reactor is at power, certain accident sequences with significant contribution to the CDF may not be applicable at all or lose their safety significance when the unit is placed in shutdown mode of operation.

In order to account for the differences between the at power and shutdown mode of operation, a separate simplified shutdown model was developed. This was required, since the base model as developed reflected full power operations and its associated risk through the respective accident sequences.

The following simplifications were made during the shutdown model development. First, low power operations, especially associated with ascending and decreasing power to and from full power were excluded due to the short relative time periods associated with this particular mode. On the average, the total shutdown time in one year is about 100 days. This may consist of three separate shutdown periods. The average length of ascending/decreasing to/from full power is about 10-20 hours, hence the total low power time is about .5% of the total shutdown time (3(shutdown) * (20+20) hours = 120; 120/100 * 24 ~ .05).

This particular assumption removes the RCP seal LOCA as the most important accident sequence as compared to the full power model. It is further assumed that during shutdown, when the primary system pressure and temperature is considerably reduced from full power operation, the probability of a seal LOCA upon loss of thermal barrier cooling is reduced by at least one or two orders of magnitude. Therefore, the RCP seal LOCA sequence was not included in the shutdown model.

The most important accident scenario is associated with residual decay heat removal (RHR) performed by the RHR system. The decay heat is removed by the RHR and through the CCW system is transferred over to the ESW system and the ultimate heat sink. The loss of ESW system disrupts the heat removal process from the reactor core and could eventually result in core uncovery.

The core damage contribution due to this particular mode of operation may be written as follows:

$$CDF(Shutdown) = \lambda * P_{SH}(t) * P_{NR}(t_R/t)$$

where

λ = loss of ESW initiating frequency

$P_{SH}(t)$ = probability to be in shutdown for time t

$P_{NR}(t_R/t)$ = probability of ESW non-recovery given recovery time t_R at time t in shutdown

$t_R(t)$ = core uncovery time at time t in shutdown.

The first term $P_{SH}(t)$ expresses the probability that the unit is in shutdown mode for the time period of t (t=0 corresponds to beginning of the shutdown period). The core uncovery time t_R depends mainly on two factors in shutdown. First, decay power produced in the reactor is gradually decreasing with increasing length of the shutdown period. The second factor is the water inventory above the reactor core that may greatly vary depending the particular shutdown activity.

The second term $P_{NR}(t_R/t)$ expresses the probability that at time t in the shutdown given the core uncovery time t_R the ESW system was unable to recover.

In the following subsections, these two factors are developed regarding shutdown operations and their associated time scales and the non-recovery factors based on the core uncovery times.

4.2.3.6.1 Shutdown Operations

There are three distinct shutdown activities with varying levels of water inventory: a) refueling, b) drained, and c) non-drained maintenance.

In the refueling operations the reactor head is removed and the refueling cavity is filled with water. During drained maintenance, the primary system is drained to the mid-loop level greatly reducing the available water inventory. In the third mode of operation (non-drained), the primary system is full. The average fuel cycle length is about 18-24 months, and once the unit is in refueling it may be in this mode for about 83 days/refueling (see Reference 5 for data). This results on the average ~50 days in each year when the unit is in shutdown.

$$T_{RF}(\text{Refueling Period}) = \frac{\text{1 Year}}{\text{Fuel Cycle Length}} * \text{Refueling Time}$$

$$= \frac{\text{365 Days}}{\text{600 Days}} * \text{83 Days} \simeq \text{50 Days/Yr (1200 hr/yr)}$$

Drained maintenance occurs about once a year for ~40 days.

$$T_D(\text{Drained Maintenance}) \simeq \text{40 days/yr (960 hr/yr)}$$

Non-drained maintenance is somewhat less frequent:

$$T_{ND}(\text{Non-Drained Maintenance}) \simeq \text{10 days/yr (240 less/yr)}$$

The total shutdown time is:

$$T_{SH} = T_{RF} + T_D + T_{ND} = 50 + 40 + 10 = 100 \text{ days/yr}$$

The time data may easily be converted to fractional probabilities that is

$$P_{RF} = \frac{T_{RF}}{T_{SH}} = \frac{50}{100} = .5$$

$$P_D = \frac{T_D}{T_{SH}} = \frac{40}{100} = .4$$

$$P_{ND} = \frac{T_{ND}}{T_{SH}} = \frac{10}{100} = .1$$

4.2.3.6.2 Core Uncovery Times/Inventory

In this section the time dependence of the core uncovery time t_R is determined based on the available water inventory in the primary system. If the decay heat removal is degraded, the RCS could heat up to the point of core uncovery by boiling away the water inventory. The decay heat generation is dependent on the length of the shutdown and must also be considered.

Two major processes were considered. The water inventory is initially assumed to be at ~100°F and must be heated up to 212°F. At that point, there is an additional energy requirement to boil the water to the core mid-plane which is equivalent to core uncovery.

In the initial step the water volume is estimated and the total energy input that is required to uncover the core is calculated.

Refueling

The reactor head is removed most of the time during refueling and the water volume in the refueling cavity is also available for cooling. The estimated volume of water is ~500,000 gallons at 100°F.

The specific volume of water is $V = 0.161\ ft^3/lbm$ and the total mass of water is $M = 4.2x10^6/lbm$.

The heat up process is expressed as an energy requirement

$$Q_{Heat} = [h_f(212) - h_f(100)] * M = (180.17 - 68.04) * 4.2x10^6$$

$$Q_{Boil} = h_{fg} * M = 970.3 * 4.2x10^6$$

$$Q = Q_{Heat} + Q_{Boil} = 4.7x10^8 + 4.1x10^9 \text{ Btu}$$

$$Q_{RF} = 4.5x10^9 \text{ Btu}$$

Drained Maintenance

In this operation the RCS is drained to hot leg mid-plane. The following water volumes are estimated using Reference 5.

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.

The water mass in the different regions is:

Mass (hot leg centerline to core mid-plane)
= 1790/0.016
= $1.11x10^5$ lbm
Mass (top of core to bottom of core)
= 1114/0.0161
= $6.92x10^4$ lbm
Mass (below the core)
= 1050/0.0161
= $6.52x10^4$ 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 the water from 100°F up 212°F is

$[h_f(212°F, 1 atm) - h_f(100°F, 1 atm)] *$
$(1.1*10^5$ 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

$Q_D = 2.37*10^7 + 1.08*10^8 = 1.32*10^8$ Btu

Non-Drained Maintenance

The total water volume above the core is assumed to be 70,000 gallons or ~9350 ft^3. Adding the additional water volumes below the core (1050 ft^3) and the core region (1114 ft^3) and subtracting hot leg center line/mid-plane column (1790 ft^3)

V = 9.350 + 1050 + 1114 - 1790 = 9.724 ft^3 = $6.0x10^5$ lbm

The heat up and boiling Btu requirements are

$$Q_{Heat} = (180.17 - 68.04) * 6.0x10^5 = 6.77x10^7 \text{ Btu}$$

$$Q_{Boil} = 970.3 * (9724 - 1114 - 1050) = 4.56x19^8 \text{ Btu}$$

The total

$$Q_{ND} = Q_{Heat} + Q_{Boil} = 5.2x10^8 \text{ Btu}$$

The following equation from Reference 5 expresses the decay power as a function of time, τ(sec.), after shutdown and the time, T_0(sec.), that the plant had been operating before shutdown.

$$P(\tau) = P_0 * 0.1[(\tau - T_0 + 10)^{-0.2} - (\tau + 10)^{-0.2} + 0.87(\tau + 2*10^7)^{-0.2}$$
$$- 0.087(\tau - T_0 + 2*10^7)^{-0.2}]$$

where P_0 is the power of the reactor. 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 loss of ESW 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.

Table 4.11 lists the uncovery times t_R (as calculated using the above equation) for the different modes of operation. The first column indicates the time period or length of shutdown.

4.2.3.6.3 <u>Shutdown CCDF Probability</u>

The conditional CDF in shutdown may be obtained by combining the previous information

$$P(\text{Shutdown}) = P_{SH}(t) * P_{NR}(t_R/t)$$

The probability of being in shutdown for time period of t is assumed to be uniform over the total shutdown interval and may be written as (for drained maintenance):

$$P_{SH}(t) = P_{DR} * \frac{\Delta t}{T_{DR}}$$

where P_{DR} is the probability of being in drained shutdown and $\Delta t/T_{DR}$ expresses the probability of being in time interval Δt. Table 4.12 lists the values of P_{SH} for the different modes of operation. In each case (drained and non-drained), the probability for the first time interval ($\Delta t/T$) was reduced to account for the following. In the initial 20-50 hours of shutdown operation, the RHR may not be placed in operation and the unit is on hot standby. In addition, maintenance and test activities usually start 2-3 days into shutdown reducing the risk of losing ESW.

The next columns list the core uncovery times and the associated non-recovery probabilities of the ESW system $P_{NR}(t_R/t)$. The final column contains the $P_{SH}(t)$ for each time period in each particular mode. The sum of these fractional $P_{SH}(t)$ gives the total conditional core damage probability in shutdown

$$P(\text{Shutdown}) = 2.82 \times 10^{-2}$$

4.3 Relative Time Fractions

In this study the average CDF due to the loss-of-ESW event was obtained by first calculating an initiating frequency for each particular state of the multi-unit configuration (see Table 4.3b). Each state dependent initiating frequency was multiplied with the respective state failure probabilities and averaged with their relative time fraction gives the total CDF.

The relative time fractions (RT_i) essentially represent the average relative length of the time period of the specific multi-plant operating configuration together with the arrangement of the ESW systems. For example, $RT_i = .5$ would indicate that the two units are in this particular operating configuration for 50% of the time. In the following, a brief discussion is given on the assumptions and data used in deriving the relative time fractions.

The average time for the different operating configurations was established based on PWR operating data.[6] For single units the average operating time when the reactor operates, but not necessarily at 100% power, is found to be $P_1(\text{operating}) \approx 73\%$ and consequently the average time in shutdown is $P_1(\text{shutdown}) \approx 27\%$ of the calendar year.

The following notation is used to indicate the operational status of the plants. If only one unit is operating, it will be noted as "P_1 (operating)" or in shutdown "P_1 (shutdown)". The relative operational status of two plants is indicated as follows; when both are operating "P_2 (operating/operating)" or if one is shut down then "P_2 (operating/shutdown)" will indicate the relative status of the multi-units.

If the assumption is made that each unit operates independently, then the average time length of the multi-unit configuration ma· simply be calculated in the following manner.

State I (see Table 4.2) represents the configuration when both units are operating and the corresponding time fraction may be expressed as the probability of two independent units operating simultaneously, i.e.,

$$P_2(\text{operating/operating}) = P_1(\text{operating}) * P_1(\text{operating}) = .73 * .73 = .53$$

similarly for State II and State III

$$P_2(\text{operating/shutdown}) = P_1(\text{operating}) * P_1(\text{shutdown}) = .2$$

$$P_2(\text{shutdown/operations}) = P_1(\text{shutdown}) * P_1(\text{operating}) = .2$$

and State IV

$$P_2(\text{shutdown/shutdown}) = P_1(\text{shutdown}) * P_1(\text{shutdown}) = .07$$

The assumption of independent multi-unit operations may somewhat overestimate the time fraction values for State IV, since the owners of multi-units try to avoid simultaneous shutdowns. However, the predicted average time, 7%, is relatively low and lowering it further would not significantly impact the contribution of this state to the total CDF.

The average time fractions of the ESW system configurations must be also taken into account, once the state time fractions are determined. In this case the following additional assumptions were made:

State I: During operating periods, two maintenance or test related outages of the ESW pumps or trains occur (on the average) each lasting about 72 hours.

State II: The average outage time for maintenance or test operations of the ESW trains is about one week and is assumed to occur once during each shutdown. The ESW flow requirement may be satisfied through the unit crossties utilizing the ESW pumps of the other units. It is assumed that the crossties are used 10% of the time in the shutdown period.

State III: All the assumptions listed for State II are equally applicable with the exception that the crossties are assumed used only 5% of the time.

In State II, III and IV with the Unit 1 ESW pumps in SB/M or M/M conditions the ESW is either not required or served by the other unit. This is simply modelled by including these cases in the respective "mirror" cases, that is, the condition (SB/M, R/AOT) is included in (R/AOT, SB/M) or (M/M, R/AOT) in (R/AOT, M/M).

In Table 4.13 the relative time fractions of each of the states are listed that were calculated based on the above assumptions. The most dominant time fraction is State Ib, which is the normal operating arrangement (i.e., both plants at power and one ESW train of each plant running with the other in standby).

4.4 Core Damage Frequency Calculations

The CDF calculation proceeded in two steps. First, configuration-dependent initiator frequencies were developed in Section 4.2.2. Each initiator value reflects not only those failures that directly disable the ESW system, but also the indirect failure modes (linked initiator model) which eventually result in the malfunction of the ESW heat removal function.

The state dependent initiating frequencies were then multiplied with the relative time fractions to get the yearly normalized loss of ESW frequency at each state. The CDF may be written as

$$CDF = \sum_i \lambda_i * P_i(\text{Sequence}) * RT_i$$

Where λ_i is the state dependent initiating frequency (given that the unit is in this state for the full year) and RT_i is the relative time fraction of the ith state. The above described first step corresponds to the calculation of $\lambda_i * RT_i$ and the results are listed in Table 4.14.

The $\lambda_i * RT_i$ values may be summed for all those states where the conditional core damage probability is identical. States I and II represent Unit 1 at full power and States III and IV correspond to the shutdown mode. This reflects the fact that P_i(Sequence) is identical for each substate in States I and II that is $P_{Ia} = P_{Ib} = \ldots = P_{IIh}$. Table 4.14 contains in the last column the summed values for the operating configurations (States I and II) and the shutdown mode States III and IV.

The sequence failure or conditional core damage probabilities are summarized in Table 4.15 for all the sequences. The sum of all the sequences during operation results in P(Operation) = 1.03-01 which reflects the probability of core damage given a loss of ESW during full power operations. The corresponding value for shutdown is P(Shutdown) = 2.82-02. The most dominant contributor of all the sequences (including shutdown) is the RCP seal LOCA at P(Seal LOCA) = 6.8-02 which is ~65% of P(Operation).

The core damage calculations are summarized in Tables 4.16a and 4.16b. The initiating frequencies ($\lambda * RT$) for the states are taken from Table 4.14 and the state failure probabilities (P) from Table 4.15. The combination of $\lambda * RT * P$ results in the core damage frequency listed in Table 4.16a. The last column gives the CDF for the operating states CDF(Operation) = 1.3-04 and the shutdown CDF(Shutdown) = 2.0-05 indicating the dominance of the operating mode. The total CDF due to loss of ESW events in all operational modes is

$$\text{CDF(Loss of ESW)} = 1.5 \times 10^{-4}/\text{RYR}.$$

Table 4.16b gives the CDF in terms of the individual sequences in the last column. The most dominant sequence is the RCP seal LOCA CDF(Seal LOCA) = 8.8-05 about ~60% of the total. A complete list of the CDF by each state is given in Table 4.17.

In summary, the CDF contribution due to the direct or indirect loss of ESW events is very significant indicating the importance of ensuring the reliable operation of the heat removal system together with the ultimate heat sink.

4.5 References

1. Cho, N.Z. et al., "Analysis of Allowed Outage Times at the Byron Generating Station," NUREG/CR-4404, June 1986.

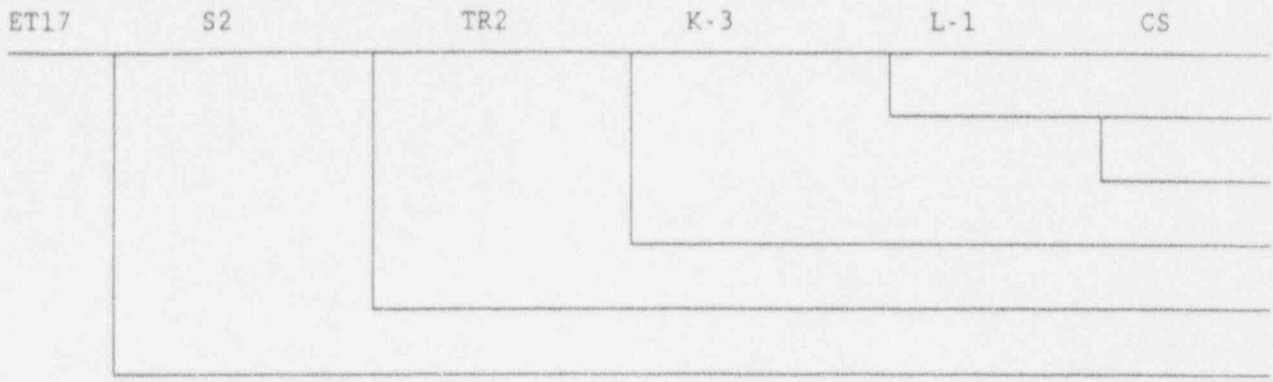
2. "Analysis of Core Damage Frequency From Internal Events: Expert Judgment Elicitation," NUREG/CR-4550, Volume 2, April 1989.

3. Swain, A.D., Guttmann, H.E., "Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications, Final Report," NUREG/CR-1278, August 1983.

4. Dearing, J.F. et al., "Dominant Accident Sequences in Oconee-I PWR," NUREG/CR-4140, April 1985.

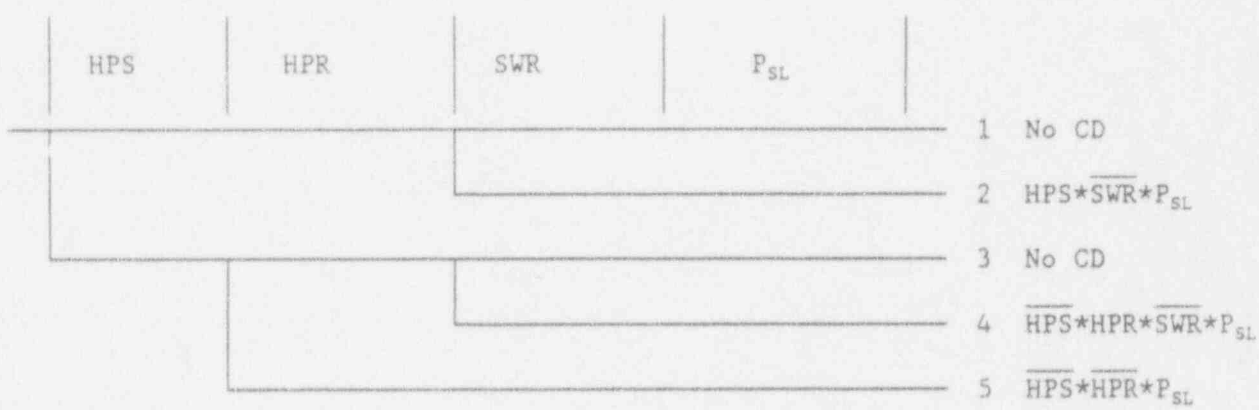
5. Chu, T-L. et al., "Improved Reliability of Residual Heat Removal Capability in PWRs as Related to Resolution of Generic Issue 99," NUREG/CR-5015, May 1988.

6. "Operating Units Status Report," NUREG-0020.



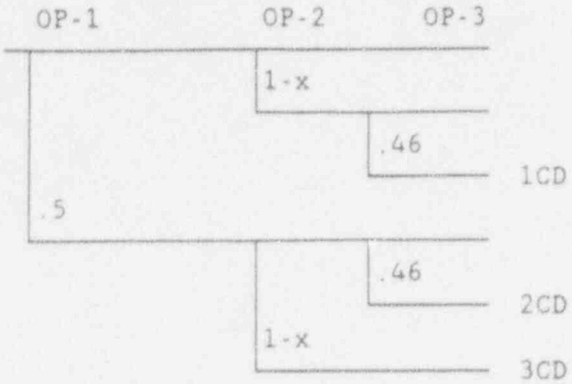
S2: Small LPCA
TR2: Steam Generator Tube Rupture
K-3: Reactor Trip
L-1: AFW
CS: Containment Spray

Figure 4.1. Event tree for loss of essential service water initiator.



HPS - HP injection function is secured initially maybe used when ESW recovers.
HPR - HP injection function is recovered given its initial loss.
SWR - ESW cooling function is recovered.
P_{SL} - Probability of a seal LOCA with a specific leak rate leading to core uncovery at time t.

Figure 4.2. Seal LOCA recovery event tree.



OP-1 - Operator turns HP pumps off.
OP-2 - AFW system recovered.
OP-3 - ESW system recovered.

X=P(Pump fail-to-start, recovery) = .2
P(Pump fail-to-run, recovery) = .05
P(Pump in maintenance, recovery) = .03

Figure 4.3. Recovery event tree small LOCA through PORV, auxiliary feedwater sequences.

Table 4.1
Dominant Cut Sets of Core Damage From Reference 1

1.	I(LOSW) * P(Seal LOCA) +
2.	I(LOSW) * P(Failure of Auxiliary Feedwater Pump to Start) +
3.	I(LOSW) * P(Failure of Auxiliary Feedwater Pump to Run) +
4.	I(LOSW) * P(Auxiliary Feedwater Pump in Maintenance) +
5.	I(LOSW) * P(Auxiliary Feedwater Valves SX178D Fails) +
6.	I(LOSW) * P(Auxiliary Feedwater Valves SX173D Fails) +
7.	I(Other) * P(Other) * P(ESW System Failure)

Table 4.2
Conditional Status of Multi-Plants

Status	Unit 1	Unit 1 ESW Pump 1	Unit 1 ESW Pump 2	Unit 1	Unit 2 ESW Pump 1	Unit 2 ESW Pump 2
Ia	OP	R	AOT	OP	R	AOT
Ib	OP	R	AOT	OP	R	SB
Ic	OP	R	SB	OP	R	AOT
Id	OP	R	SB	OP	R	SB
IIa	OP	R	AOT	DN	R	AOT
IIb	OP	R	AOT	DN	R	SB
IIc	OP	R	AOT	DN	SB	M
IId	OP	R	AOT	DN	M	M
IIe	OP	R	SB	DN	R	AOT
IIf	OP	R	SB	DN	R	SB
IIg	OP	R	SB	DN	SB	M
IIh	OP	R	SB	DN	M	M
IIIa	DN	R	AOT	OP	R	AOT
IIIb	DN	R	AOT	OP	R	SB
IIIc	DN	R	SB	OP	R	AOT
IIId	DN	R	SB	OP	R	SB
IVa	DN	R	AOT	DN	R	AOT
IVb	DN	R	AOT	DN	R	SB
IVc	DN	R	AOT	DN	SB	M
IVd	DN	R	AOT	DN	M	M
IVe	DN	R	SB	DN	R	SB
IVf	DN	R	SB	DN	R	AOT
IVg	DN	R	SB	DN	SB	M
IVh	DN	R	SB	DN	M	M

OP - Operating.
DN - Shutdown.
R - Pump running.
SB - Pump in standby.
AOT - Pump in test (allowable outage time).
M - Pump in maintenance.

Note: The states, when the Unit 1 ESW pumps are in SB/M or M/M conditions and the unit crosstie is utilized to supply SW are included in IIc, d, g, h, and IVc, d, g, and h respectively.

Table 4.3
Failure Probabilities

ESW Train	P
FS (Standby train fails-to-start)	9.96×10^{-3}
FR (Running train fails-to-run)	6.7×10^{-3}
XT (Crosstie failure)	1.31×10^{-2}
β (Intake common mode factor)	6×10^{-2}

Table 4.3a
Numerical Values of P(2/1 - State)

	P(2/1 - State)
Success Criterion I:	
P(2/1 - R/AOT)	1.0
P(2/1 - R/SB)	9×10^{-2}
Success Criterion II:	
P(2/1 - R/AOT)	8×10^{-2}
P(2/1 - R/SB)	7.3×10^{-2}
P(2/1 - SB/M)	9×10^{-2}
P(2/1 - M/M)	1.0

Success Criterion I: Two ESW pumps required on second unit.
II: One ESW pump required on second unit.

Pump 1/Pump 2

R/AOT - Run/In Test
R/SB - Run/Standby
SB/M - Standby/Maintenance
M/M - Maintenance/Maintenance

Table 4.3b
State Dependent Initiating Frequency

States Unit 1 Pumps	Unit 2 Pumps	Base Initiator	Modifier	ESW Unit Initiating Frequency/ Reactor Year
I - Unit 1-Up/2-Up				
R/AOT	R/AOT	*	1.4+02	1.6-01
	R/SB	*	1.3+01	1.4-02
R/SB	R/AOT	*	1.1+01	1.2-02
	R/SB	1.1-03	1.0	1.1-03
II - Unit 1-Up/2-Down				
R/AOT	R/AOT	*	1.1+01	1.2-02
	R/SB	*	1.0+01	1.1-02
	SB/M	*	1.3+01	1.4-02
	M/M	*	1.4+02	1.6-01
R/SB	R/AOT	*	8.8-01	9.7-04
	R/SB	*	8.1-01	8.9-04
	SB/M	*	1.0	1.1-03
	M/M	*	1.1+01	1.2-02
III - Unit 1-Down/2-Up				
R/AOT	R/AOT	2.9-01	8.0-02	2.3-02
	R/SB	2.9-01	7.3-02	2.1-02
R/SB	R/AOT	3.2-02	8.0-02	2.6-03
	R/SB	3.2-02	7.3-02	2.3-03
IV - Unit 1-Down/2-Down				
R/AOT	R/AOT	2.9-01	8.0-02	2.3-02
	R/SB	2.9-01	7.3-02	2.1-02
	SB/M	2.9-01	9.0-02	2.6-02
	M/M	2.9-01	1.0	2.9-01
R/SB	R/AOT	3.2-02	8.0-02	2.6-02
	R/SB	3.2-02	7.3-02	2.3-02
	SB/M	3.2-02	9.0-02	2.9-02
	M/M	3.2-02	1.0	3.2-02

*Same as I(R/SB, R/SB) = 1.1-03.

Table 4.4
ESW System Non-Recovery Times

Time (Hour)	Fraction of ESW Systems Non-Recovered SWR
1.5	.36
4.5	.05
24	.01

Table 4.5
Aggregated RCP Seal LOCA Probabilities for a Westinghouse Four Loop Plant

Leak Rate (gpm)	Old O-Rings - Time (h)					New O-Rings - Time (h)				
	1.5	2.5	3.5	4.5	5.5	1.5	3.6	3.5	4.5	5.5
84	.302	.286	.271	.271(.255)	.271(.239)[a]	.810	.809	.809	.807	.805
244/245[b]	.148	.038	.053	.051(.067)	.049(.081)	.014	.016	.017	.0198	.020
313	-	-	-	-	-	.010	.010	.010	.010	.010
433	.011	.012	.028	9.9E-3	9.3E-3	6.0E-4	6.0E-4	6.0E-4	6.0E-4	6.0E-3
480	1.3E-3	1.3E-3	1.3E-3	1.3E-3	1.3E-3	-	-	-	-	-
543	-	-	-	-	-	2.6E-3	2.6E-3	2.6E-3	2.6E-3	2.6E-3
688/698/728	1.2E-3	1.2E-3	1.1E-3	1.1E-3	.146	.146	.146	.146	.146	.146
796	-	-	-	-	-	2.7E-3	2.7E-3	2.7E-3	2.7E-3	2.7E-3
1000/1026	.530	.659	.659	.665	.666	8.3E-3	8.3E-3	8.3E-3	8.3E-3	8.3E-3
1230	1.6E-6	1.6E-3	1.6E-3	1.6E-3	1.6E-3	-	-	-	-	--
1920	4.2E-3	4.2E-3	4.2E-3	4.2E-3	4.2E-3	4.2E-3	4.2E-3	4.2E-3	4.2E-3	4.2E-3

[a]Parentheses denote calculations which change if no depressurization is assumed. All other probabilities are for depressurized conditions.

[b]Similar leak rates have been lumped together.

These values are the probabilities of being at a particular leak rate at a particular time.

Table 4.6
RCP Seal LOCA Failure Sequences

Leak Path (GPM)	Time to Transfer (HR)	Probability P_{SL}
1000*	1½	.5298
240-1000	2½	.1253
240*	1½	.049
433-1000	2½	.0051
1920*	1½	.0042
433*	1½	.0042
84*	1½	.2704

*Constant leak rate.

Table 4.6a
Time Evolution of LOSW and RCP Seal LOCA

Time, t	
0	Loss of ESW
0-15 Min.	Loss of charging pumps
15 Min.	CCW heats up to spent fuel pool
30 Min.	CCW pumps turned off
~30 Min.	Loss of all cooling to RCP seals
t_{SL}=2.0 Hr.	RCP seal LOCA
$t_{SL}+\lambda_i^*$	Core uncovery for ith scenario

λ_i = Core uncovery times for ith scenario.

Table 4.6b
Core Uncovery Times

Leak Path (GPM)	Core Uncovery Times λ_i (HR)	$t_{SL} + \lambda_i (= 2. + \lambda_i)$ (HR)
1000	2.1	4.1
240-1000	2.2	4.2
240	8.1	10.1
433-1000	2.1	4.1
1920	0.75	2.75
433	4.4	6.4
84	19.0	21.0

Table 4.7a
RCP Seal LOCA Probability
ESW Non-Recovery

Leak Path (GPM)	Seal LOCA Probability P_{SLi}	Core Uncovery $t_{SL} + \lambda_i = t$	ESW Non-Recovery SWR
1000	.5298	4.1	9.1-02
240-1000	.1253	4.2	9.1-02
240	.049	10.1	3.9-02
433-1000	.0051	4.1	9.1-02
1920	.0042	2.75	2.3-01
433	.0042	6.4	4.6-02
84	.2704	21.0	1.6-02

Table 4.7b
HP Injection Non-Recovery

Leak Path (GPM)	Uncovery Time t(HR)	Time Available Before HP Pump Damage (MIN)	Probability of Losing Injection Capability $\overline{HPS}$	Probability of HPI Non-Recovery $\overline{HPR}$
1000	4.1	~10-15	$1x10^{-3}$	.95
240-1000	4.2	~10-15	$1x10^{-3}$	.95
240	10.1	~15-20	$5x10^{-4}$	.70
433-1000	4.1	~10-15	$1x10^{-3}$	.95
1920	2.75	~10-15	$1x10^{-3}$	.95
433	6.4	~15-20	$5x10^{-4}$	.70
84	21.0	~15-20	$5x10^{-4}$	.70

Table 4.7c
Nominal Model of Estimated HEPs and EFs for Diagnosis Within
Time T by Control Room Personnel of Abnormal Events

Time After LOSW (Minutes)	Human Error Probability (HEP) for Diagnosing the Event	Error Factors
1	1.0	---
10	.1	10
20	.01	10
30	.001	10
60	.0001	30
1500	.00001	30

Table 4.7d
Reactor Trip or Safety Injection*

Purpose

This procedure provides actions to verify proper response of the automatic protection systems following manual or automatic actuation of a reactor trip or safety injection, to assess plant conditions, and to identify the appropriate recovery procedure.

Symptoms or Entry Conditions

1. Any symptom that requires a manual reactor trip listed in Attachment A, if one has not occurred.

2. The following are symptoms of a reactor trip:

   a. Any reactor trip annunciator lit.
   b. Rapid decrease in neutron level indicated by nuclear instrumentation.
   c. All shutdown and control rods are fully inserted. Rod bottom lights are lit.

3. Any symptom that requires a manual reactor trip and safety injection listed in Attachment B, if one has not occurred.

4. The following are symptoms of a reactor trip and safety injection:

   a. Any SI annunciator or status lamp lit.
   b. ECCS pumps in service.

Table 4.7d (Continued)

Step	Action/Expected Response	Response Not Obtained
6	Verify Containment Isolation Phase A Actuation:	
	a. Phase A components - ALL STATUS PANEL LIGHTS LIT • TRAIN A • TRAIN B	a. Manually actuate 'T' signal for BOTH trains. Align equipment as necessary by status panels.
7	Verify EFW Pumps Running:	
	a. Motor-driven pump - RUNNING	a. Manually start pump.
	b. Turbine-driven pump - RUNNING • MS-V127 - OPEN • MS-V128 - OPEN • MS-V393 - OPEN • MS-V394 - OPEN • MS-V395 - OPEN • TRIP VALVE MS-V129 - OPEN	b. Manually open at least one steam supply path or reset trip valve as necessary.
8	Verify ECCS Pumps Running: • CCPs - TRAIN A AND B • SI pumps - TRAIN A AND B • RHR pumps - TRAIN A AND B	Manually start pumps.
9	Verify PCCW Pumps - RUNNING: a. Loop A - ONE PUMP RUNNING b. Loop B - ONE PUMP RUNNING c. Thermal barrier cooling pumps - AT LEAST ONE PUMP RUNNING	Manually start pumps.

Table 4.7d (Continued)

Step	Action/Expected Response	Response Not Obtained
10	Verify Ultimate Heat Sink Operation:	Manually start SW pumps or actuate TA as necessary.
	a. Train A - RUNNING	
	• One SW Pump - or - • One CT pump <u>AND</u> CT fan in TA mode as necessary	
	b. Train B - RUNNING	
	• One SW pump - or - • One CT pump <u>AND</u> CT fan in TA mode as necessary	
11	Verify SW Cooling to Diesels:	
	a. Train A cooling established:	
	1. SW-V16 - OPEN 2. Flow indicated - GREATER THAN 1700 GPM	1. Manually or locally open SW-V16.
	b. Train B cooling established:	
	1. SW-V18 - OPEN 2. Flow indicated - GREATER THAN 1700 GPM	1. Manually or locally open SW-V18.

*Procedure No.E-0/Revision No.00, Date: 5/16/86

Table 4.8
Core Damage Probability Due to Seal LOCA

Leak Path (GPM)	Sequence 3 $HPS*\overline{SWR}*P_{SLi}$	Sequence 4 $\overline{HPS}*HPR*\overline{SWR}*P_{SLi}$	Sequence 5 $\overline{HPS}*\overline{HPR}*P_{SLi}$	Total
1000	4.8-02	2.4-06	5.0-04	
240-1000	1.1-02	5.7-07	1.2-04	
240	1.9-03	2.9-07	1.7-05	
433-1000	4.6-04	2.3-08	4.8-06	
1920	9.7-04	4.8-08	3.9-06	
433	1.9-04	2.9-08	1.5-06	
84	4.3-03	6.5-07	9.5-05	
Total	6.7-02	4.0-06	7.4-04	6.8-02

P(Seal LOCA) = 6.8-02

Table 4.9
Loss of ESW Accident Sequence Non-Recovery Factors

Sequence	Non-Recovery Factor
1 (AFW pump fails to start)	.63
2 (AFW pump fails to run)	.98
3 (AFW pump in maintenance)	.99

Table 4.10
Conditional Core Damage Probability Auxiliary Feedwater System P_{AFW}

Failure Mode	P_{NR}	AFW	P_{SL}	$P_{AFW} = P_{SL}*AFW*P_{NR}$
AFW FS	.63	2.33-02	9.61-01	1.41-02
AFW FR	.98	6.45-03	9.61-01	6.07-03
AFW PM	.99	2.40-03	9.61-01	2.28-03
Total, P_{AFW}				2.25-02

Notes: AFW FS - AFW pump fail to start. AFW FR - AFW pump fail to run.
AFW PM - AFW pump in maintenance.

Table 4.11
Core Uncovery Times in Shutdown

Length of Shutdown, Hr.	Core Uncovery Times, Hr.
Refueling	
0-83 days	>>24 hours
Drained Maintenance	
0-200	2.5
200-400	4.5
400-600	5.7
600-800	7
800-1000	8
Non-Drained Maintenance	
0-100	6
100-200	14
200-250	16

Table 4.12
Conditional Core Damage in Shutdown

Length of Shutdown, Hr.	Probability of Shutdown Mode	Probability of Being in Time $\Delta t/T$	Average Time to Core Uncovery t_R	ESW Non-Recovery $P_{NR}(t_R/t)$	
Refueling	P_{RF}				$P_{RF}*\Delta t/T*P_{NR}$
0-83 days	.5	1	>>24	~0	Negligible
Drained Maintenance	P_{DR}				$P_{DR}*\Delta t/T*P_{NR}$
0-200	.4	.075	2.5	.26	7.8-03
200-400	.4	.23	4.5	.05	4.6-03
400-600	.4	.23	5.7	.05	4.6-03
600-800	.4	.23	7	.04	3.7-03
800-1000	.4	.23	8	.04	3.7-03
Non-Drained Maintenance	P_{ND}				$P_{ND}*\Delta t/T*P_{NR}$
0-100	.1	.1	6	.05	5.0-04
100-200	.1	.6	14	.03	1.8-03
200-250	.1	.3	16	.03	9.0-04
Total P(Shutdown)					2.8-02

Table 4.13
Relative Time Fractions for Operational Configurations

States Unit 1 Pumps	Unit 2 Pumps	Relative Time Fractions RT_i
I - Unit 1-Up/2-Up		
R/AOT	R/AOT	3.0-04
	R/SB	1.2-02
R/SB	R/AOT	1.2-02
	R/SB	5.1-01
II - Unit 1-Up/2-Down		
R/AOT	R/AOT	5.4-04
	R/SB	3.3-03
	SB/M	1.2-04
	M/M	1.5-04
R/SB	R/AOT	2.4-02
	R/SB	1.5-01
	SB/M	4.0-02
	M/M	6.4-03
III - Unit 1-Down/2-Up		
R/AOT	R/AOT	5.4-04
	R/SB	3.3-03
R/SB	R/AOT	2.4-02
	R/SB	1.5-01
IV - Unit 1-Down/2-Down		
R/AOT	R/AOT	2.0-04
	R/SB	1.3-03
	SB/M	1.1-04
	M/M	1.8-05
R/SB	R/AOT	8.7-03
	R/SB	5.4-02
	SB/M	4.8-03
	M/M	7.9-04

Table 4.14
State Dependent Initiating Frequency

States Unit 1 Pumps	Unit 2 Pumps	ESW Unit Initiating Frequency-λ	Relative Time Fraction-RT	λ*RT	Total
I - Unit 1-Up/2-Up					
R/AOT	R/AOT	1.6-01	3.0-04	4.8-05	
	R/SB	1.4-02	1.2-02	1.7-04	
R/SB	R/AOT	1.2-02	1.2-02	1.5-04	
	R/SB	1.1-03	5.1-01	5.6-04	
II - Unit 1-Up/2-Down					
R/AOT	R/AOT	1.2-02	5.4-04	6.9-06	
	R/SB	1.1-02	3.3-03	3.9-05	
	SB/M	1.4-02	9.2-04	1.3-05	
	M/M	1.6-01	1.5-04	2.4-05	
R/SB	R/AOT	9.7-04	2.4-02	2.4-05	
	R/SB	8.9-04	1.5-01	1.3-04	
	SB/M	1.1-03	4.0-02	4.4-05	
	M/M	1.2-02	6.4-03	7.8-05	
Total/Operating States					1.3-03
III - Unit 1-Down/2-Up					
R/AOT	R/AOT	2.3-02	5.4-04	1.2-05	
	R/SB	2.1-02	3.3-03	6.9-05	
R/SB	R/AOT	2.6-03	2.4-02	6.2-05	
	R/SB	2.3-03	1.5-01	3.5-04	
IV - Unit 1-Down/2-Down					
R/AOT	R/AOT	2.3-02	2.0-04	4.6-06	
	R/SB	2.1-02	1.3-03	2.7-05	
	SB/M	2.6-02	1.1-04	2.9-06	
	M/M	2.9-01	1.8-05	5.2-06	
R/SB	R/AOT	2.6-03	8.7-03	2.3-05	
	R/SB	2.3-03	5.4-02	1.2-04	
	SB/M	2.9-03	4.8-03	1.4-05	
	M/M	3.2-02	7.8-04	2.5-05	
Total/Shutdown States					7.1-04

Table 4.15
Sequence Failure Probabilities

Sequences	P
<u>Full Power Operations</u>	
RCP Seal LOCA - P(Seal LOCA)	6.8-02
Auxiliary Feedwater - P_{AFW}	2.3-02
Long Term AFW - P_{LAFW}	9.1-03
Other Sequences - P_{Other}	3.2-03
Total - P(Operation)	1.03-01
Shutdown - P(Shutdown)	2.82-02

Table 4.16a
Core Damage Frequency - Summary

States	Initiating Frequency λ*RT	Sequence Failure Probability - P	Core Damage Frequency CDF/RYR
I + II	1.3-03	1.03-01	1.3-04
III + IV	7.1-04	2.82-02	2.0-05
Total			1.5-04

Table 4.16b
Core Damage Frequency by Sequences

Sequences	Initiating Frequency λ*RT	Sequence Failure Probability-P	Core Damage Frequency CDF/RYR
Seal LOCA - P(SL)	1.3-03	6.8-02	8.8-05
AFW - P_{AFW}	1.3-03	2.3-02	3.0-05
Long Term - P_{LAFW}	1.3-03	9.1-03	1.2-05
Other - P_{Other}	1.3-03	3.2-03	4.2-06
Total Operation - P(Operation)	1.3-03	1.03-01	1.3-04
Shutdown - P(Shutdown)	7.1-04	2.82-02	2.0-05
Total			1.5-04

Table 4.17
Core Damage Frequency

States Unit 1 Pumps	Unit 2 Pumps	ESW Unit Initiating Frequency λ*RT	Sequence Failure Probability-P	Core Damage Frequency CDF/RYR	Total
I - Unit 1-Up/2-Up					
R/AOT	R/AOT	4.8-05	1.03-01	4.9-06	
	R/SB	1.7-04	1.03-01	1.8-05	
R/SB	R/AOT	1.5-04	1.03-01	1.5-05	
	R/SB	5.6-04	1.03-01	5.8-05	
II - Unit 1-Up/2-Down					
R/AOT	R/AOT	6.9-06	1.03-01	7.1-07	
	R/SB	3.9-05	1.03-01	4.0-06	
	SB/M	1.3-05	1.03-01	1.3-06	
	M/M	2.4-05	1.03-01	2.5-06	
R/SB	R/AOT	2.4-05	1.03-01	2.5-06	
	R/SB	1.3-04	1.03-01	1.3-05	
	SB/M	4.4-05	1.03-01	4.5-06	
	M/M	7.8-05	1.03-01	8.0-06	
Total Operations					1.3-04
III - Unit 1-Down/2-Up					
R/AOT	R/AOT	1.2-05	2.82-02	3.4-07	
	R/SB	6.9-05	2.82-02	1.9-06	
R/SB	R/AOT	6.2-05	2.82-02	1.7-06	
	R/SB	3.5-04	2.82-02	9.9-06	
IV - Unit 1-Down/2-Down					
R/AOT	R/AOT	4.6-06	2.82-02	1.3-07	
	R/SB	2.7-05	2.82-02	7.6-07	
	SB/M	2.9-06	2.82-02	8.2-08	
	M/M	5.2-06	2.82-02	1.5-07	
R/SB	R/AOT	2.3-05	2.82-02	6.5-07	
	R/SB	1.2-04	2.82-02	3.4-06	
	SB/M	1.4-05	2.82-02	3.9-07	
	M/M	2.5-04	2.82-02	7.1-06	
Total Shutdown					2.0-05
Total					1.5-04

5. EFFECTS OF POTENTIAL IMPROVEMENTS ON CORE DAMAGE FREQUENCY

In order to reduce the core damage frequency due to loss-of-ESW events, numerous options appear to be available. From these options, however, potential improvements within the perspective of actually getting implementation are rather limited. In the present section, those improvements are discussed which have been deemed to be implementable without excessive difficulties.

The potential options for improvements were selected by considering the dominant failure modes of the ESW system listed in Table 3.2. In addition, the dominant accident sequences were also included to consider potential reduction in the CDF. It was apparent that there is no single dominant failure mechanism represented in the initiating frequency. Therefore, a number of different options must be considered targeted to each or combinations of particular failure modes to reduce significantly the initiating frequency of loss of ESW. The approximate importance of the failure modes is indicated in Table 3.2 and is based on actual experience. This offers a simple systematic method to incorporate the effects of the potential improvements into the present core damage model.

In essence, the base initiator frequency was modified to take into account the effects of the particular option under consideration. The first step was to determine the fraction of the initiator frequency that could be improved by the considered option. This was basically established using the data listed in Table 3.2.

In the next step, the relative change in the ESW system component reliability with and without the improvement provides an indication of the potential reduction in the core damage frequency. The targeted component of the initiator frequency is reduced by the same ratio as the relative change in the calculated unavailability of the ESW system component, i.e.,

$$\lambda_{ESW}(\text{improved}) = (1-f)\lambda_{ESW}(\text{base}) + f^{*}\lambda_{ESW}(\text{base})\ \frac{P(\text{ESW-improved})}{P(\text{ESW-base})}\ ,$$

where f represents the fraction of the base initiating frequency that may be reduced by the given improvement. The P(ESW-Improved) and P(ESW-Base) are the unavailability of the respective ESW system component with and without the considered improvement.

The potential option selected to reduce the dominant accident sequence contribution to the CDF is connected to the RCP seal LOCA scenario due to its dominant contribution to the total CDF. The following potential improvements were analyzed regarding their effects on the core damage frequency:

1. Additional Crosstie - Reducing the possibility of the malfunction of the unit cross-connections.
2. Electrical Dependency - Increasing the redundancy of the electrical power supplies.
3. Separate Intake Structure or Bay With an Additional Swing ESW Pump - Increasing the redundancy of the ultimate heat sink or source of cooling and increasing the availability of the ESW pumps.

4. Changing Technical Specification requirements.
5. Installation of a dedicated RCP seal cooling system.

The first four options were selected based on considerations regarding "prevention" of the LOSW failure mechanisms. A particular operating mode, State IId and h, when both ESW pumps of the shut down plant are inoperable, is a special concern, since there are no explicit Technical Specification (TS) requirements on the ESW system in this operating mode. Therefore, an option of imposing additional TS requirements was also analyzed regarding its effect and CDF reduction potential.

On the "mitigation" side, the most dominant contribution to the CDF arises from the failure of the RCP seal upon loss of seal cooling due to the unavailability of the ESW. It was therefore proposed that the installation of a dedicated RCP seal cooling system which would cool the seals in the event of loss of ESW, be evaluated as a potential plant improvement.

5.1 Additional Crosstie - Option 1

The ESW systems of the multi-unit plants currently being analyzed herein are cross-connected through pipe connections and isolation valves. This arrangement allows the operator of one plant to utilize the ESW cooling capacity of the other plant under most circumstances. In most cases, the crosstie isolation valves can be remotely operated. A hardware failure to open the isolation valves, should the need arise, could result in adverse conditions. Naturally, a parallel cross-connection could reduce the possibility of this kind of failure and in addition would allow for more flexible maintenance options. The results of this analysis are shown in Table 5.1. The effects on the CDF were minimal due to the relatively low isolation valve failure rates indicating that other hardware components are relatively more significant in reducing the overall system unavailability. The core damage frequency was reduced from CDF(Base) = 1.52-04/RYR to CDF(Option 1) = 1.36-04/ryr with a ΔCDF(Option 1) = 1.60-05/ryr.

5.2 Electrical Dependency - Option 2

One of the important contributors to the unavailability of the ESW system is related to the dependability of the electrical power supply and control system. Based on the data listed in Table 3.1, the loss of the electrical power supply due to various causes was relatively high. However, the recovery times associated with these events indicate a generally faster average recovery of the ESW system.

In general, the electrical power supplies to the ESW trains are separated and have no cross-connection capability, i.e., Train A ESW pump cannot be powered from electrical Train B (or Diesel B). This option therefore investigated the implementation of crossties between the electrical trains of the unit with respect to the operation of the two ESW pumps (Trains A and B). The cross-connection of electrical power supply of other electrical components, such as MOVs was not considered as part of this option. It is envisioned that the electrical cross-connection option would be an exclusively manual operation and meet all applicable current NRC Standard Review Plan acceptance criteria.

The average recovery time associated with electrical power supply failures was established using events A.4, A.6, A.7, A.9 and A.12. The non-recovery factors are listed in Table 5.2a along with the SW non-recovery factors used in the base case. The conditional core damage probabilities (P) may be recalculated with the electrical recovery data to derive a set of P's related to the electrical events. The CDF then may be written as

$$CDF = \sum\lambda(Electrical) * P(Electrical) + \sum\lambda P(Others)$$

The set of P(Electrical) for the various sequences is listed in Table 5.2b and compared with the base case. The initiating frequency λ(Electrical) may be written as $f*\lambda_{ESW}$ and further reduction is achieved by reducing the fractional contribution of the electrical failure mechanism that is $f \rightarrow f^1$ where f^1 is established using the procedure described previously.

The effects on the CDF are indicated in Table 5.1 (Option 2). The CDF reduction is rather minor due to the existence of fast recovery actions. The ΔCDF(Option 2) = 1.4-05/ryr CDF(Option 2) = 1.4-04/ryr CDF(Base) = 1.5-04/ryr.

5.3 Separate Intake Structure - Option 3

The classification of the failure modes (Table 3.2) indicates that one of the important ESW failure mechanisms is the failure of certain components such as travelling screens, strainers, or the intake structure in general stopping or restricting the flow of cooling water to the plant. A separate intake structure either based on the same body of water or using a different water source would make alternate cooling capability available.

The intake structure is usually a single structure divided into separate bays by concrete walls. There are a number of screens installed to prevent the intake of large foreign objects. The collapse or plugging of these screens may occur as a common mode failure due to the common inlet and/or common water source. The whole intake structure could also be affected by external events such as flooding or weather-related effects such as freezing.

The option considered here is a totally separated intake structure serving as a redundant intake source of ESW water. It may be located on the same water source in a physically separate location. An alternate design, which would provide even more redundancy, would be to install the additional intake structure on a physically separate water source. Naturally, there are locations where this would be uneconomical and/or simply not feasible.

The additional intake option includes the structure, screens and the associated motors, valves and piping. In addition, a swing ESW pump would be made available to both units with redundant (Unit 1/Unit 2) electrical power supplies. This arrangement is targeted to reduce two failure mechanisms, one representing failures due to electrical supply problems and the other due to the operating problems of the ESW pumps. The additional ESW pump would be a swing pump serving either unit depending on their current needs.

The combination of a separate intake structure and additional swing pump with redundant electrical power supply would affect a large fraction of the initiating frequency related to the failure mechanisms involving the intake, the ESW pumps and their power supply. In addition, the common mode failure of the intakes (β) would also be reduced from β = 0.6 to 0.3, along with SW pump-related failures (i.e. failure to start or run). The electrical redundancy of the swing pump reduces the conditional core damage probabilities associated with the electrical failures due to the different characteristics of the electrical recovery.

The result of the calculations is shown in Table 5.1 (Option 3) indicating a significant reduction of the CDF from CDF(Base) = 1.52-04/ryr to CDF(Option 3) = 6.07-05/ryr and ΔCDF(Option 3) = 9.13-05/ryr.

5.4 Changing Technical Specification Requirements - Option 4

There are certain operating modes, Modes 5 and 6, that have to be carefully examined with regard to specific requirements in the Technical Specifications. In these operating modes the reactor is in shutdown condition and the status of its ESW pumps is uncertain. The TS do not require that any of the ESW pumps be operational in these modes. An implicit requirement is imposed on the ESW trains through the explicit requirement to operate the RHR system to remove decay heat.

In essence, the operator of the unit in shutdown may utilize its own ESW pumps to provide the necessary heat removal function, but may just as well decide to use the crosstie to supply ESW flow from the other unit. In the absence of any requirements on the ESW pumps, both pumps could be removed for maintenance or made inoperable at the same time. It is recognized that this is not a universal practice and the model takes this into account by placing a limitation on crosstie use of 10, 10 and 5% in States II, III and IV respectively. This reflects the assumption that crosstie use is administratively limited and not a general practice, but it is not ruled out by the TS.

Unit 1 at full power and Unit 2 in either Mode 5 or 6 is represented in our analysis as States IId and IIh. The base case assumed that in States II and III the crossties are utilized 10% the time. The actual status of the pumps was established through the assumption that the ESW pumps are maintained according to a certain schedule and the maintenance periods are placed randomly through the shutdown. This was based on information gathered from the plant operators and represents a general or conventional practice.

Once the crosstie is placed in service, the status of the ESW pumps of the affected unit is uncertain. In the base model, it was assumed that the simultaneous shutdown of both ESW pumps could occur only randomly. This is reflected in the relative time fraction values. States IId and IIh represent the case when both ESW pumps of Unit 2 are unavailable and the crosstie is being utilized.

The uncertainty or potential unavailability of the Unit 2 ESW pumps to provide backup for the Unit 1 ESW system may be significantly reduced by imposing an explicit operability requirement on at least one of the ESW pumps of Unit 2 while in Modes 5 and 6. Essentially, States IId and IIh would then be eliminated (the relative time fractions RT(IId,h) are negligible). The resulting CDF calculations indicated that the CDF would be reduced to CDF(Tech.Spec.) = $1.44*10^{-4}$/ryr (CDF(Base) = $1.52x10^{-4}$/ryr) or a reduction of ΔCDF = $8.4x10^{-6}$/ryr.

Two other improvements were also considered as part of Option 4 due to their similarity and logical dependence. One of the additional improvements would require the testing of the unit crosstie valves once in each fuel cycle to insure their operability. This may be done remotely for motor-operated valves or by a local test crew. This would greatly reduce the hardware error associated with the crosstie valves and increase the awareness of the operators to the importance of these valves.

The other option is coupled to the human interface in the context of a loss of ESW accident. Specific improvements in emergency procedures would increase the potential for proper diagnosing of the various couplings between the loss of ESW accident and the numerous safety equipment. In this regard, the potential heatup of the CCW system and the possible heat rejection path to the spent fuel pool should be examined. The loss of cooling to RCP seals and the various safety pumps especially the potential loss of HPI function must be carefully considered. In addition, the procedure to insure long term water supply to the AFW system should be formalized.

The CDF calculations for both of these options indicated a reduction in CDF by ΔCDF(Crosstie Testing) = 2.48-06/ryr and ΔCDF(Procedure Improvement) = 1.21-05/ryr. The associated cost/benefit ratios were C/B(Crosstie) ~ 100 and C/B(Procedure) was completely offset by the onsite cost values. Both of these numbers indicate advantageous C/B values (below 1000), hence the combination of these three separate options was considered as Option 4 due to their logical similarity. (See Section 6 for further details.)

The results of the CDF calculations for Option 4 including TS changes, crosstie testing and procedure improvement are presented in Table 5.1. The CDF is reduced to CDF(Option 4) = $1.26x10^{-4}$/ryr (CDF(Base) = $1.52x10^{-4}$/ryr) with a ΔCDF(Option 4) = $2.55x10^{-5}$/ryr.

5.5 Dedicated RCP Seal Cooling System - Option 5

5.5.1 RCP Seal Cooling System

The analysis in Section 4 indicated that the major contribution to the service-water-related CDF arises from the failure of the RCP seals after a loss of ESW event. The RCP seal LOCA sequence contributed ~60% of the total CDF (CDF(RCP Seal) = $8.8x10^{-5}$/ryr and CDF(Total) = $1.5x10^{-4}$/ryr. If this failure could be reduced, then a significant reduction in CDF may be achieved.

The suggested option is to install a dedicated seal cooling system that would provide heat removal capability after a loss-of-ESW event. The cooling requirements of the RCP seals are relatively modest and a single small capacity high pressure pump capable of delivering about 50-100 gpm was judged to be fully sufficient. The pump may be driven either by an electric motor or, if ac electrical independence is desired from the point of view of other accident scenarios (such as blackout), then a diesel-driven pump option may also be considered. However, for the purposes of this study, this was not discussed or analyzed in detail.

The single high pressure pump would provide flow via the cooling header to the four injection lines (one to each RCP seal). It was assumed that the pump would not require auxiliary cooling for the lube oil, bearings, etc., as the suction flow would be sufficient to provide all heat removal requirements. It was also assumed that the return flow from the RCP seals would not be re-cycled. In other words a once through cooling cycle would be used with a sufficient water source to last for a considerable time.

The cooling water requirement could be provided, for example, from the refueling water storage tank which could provide sufficient water to last more than 24 hours. If a dedicated tank were to be installed, it was assumed that its capacity would satisfy ~10-15 hours of seal cooling.

In order to model the system, the following assumptions were made:

1. single high pressure pump, 50-100 gpm capacity,
2. dedicated water storage tank with capacity to last at least 10 hours,
3. ac-independent (non-seismic) pump,
4. no support system cooling required, and
5. once-through RCP seal heat removal.

Other design alternatives may also be considered utilizing arrangements different from the high pressure pump option. One typical alternative would provide flow through the RCP thermal barrier heat exchangers by connecting the firewater system into the CCW lines. Most firewater systems have one diesel-driven firewater pump which usually is independent of the ESW system. Both of these design alternatives (dedicated high pressure cooling system or backup through the connection to the firewater system via the CCW) are considered to be similar with regard to their CDF reduction potential and therefore only one option was modelled. The failure probabilities of the two systems (high pressure cooling pump vs. diesel driven firewater pump and associated valves control) are judged to be at the same order of magnitude.

5.5.2 RCP Seal Cooling System Modelling

The CDF contribution was modelled using the cut set representation of the base model. In this case the following terms are present:

$$CDF = \lambda_{ESW} * P(Cooling) * P(Seal) + \text{Other Sequences.}$$

The term λ_{ESW} is the frequency of the loss of ESW event as established in Section 3. The second term P(Cooling) represents the potential failure of the proposed cooling function. The P(Cooling) term may be approximated using a similarity argument. The auxiliary feedwater system basically performs a very similar function. It is always on standby and, given a loss of feedwater or other transient, the system should start and provide cooling flow. The major difference is that the AFW system usually has three trains, two electrical motor driven and the third either turbine or diesel driven. Another important difference is that the AFW pumps have larger capacities (~400-500 gpm) and the associated valves, piping and control is much more complicated than that envisioned for the simple seal cooling system.

Based on these arguments, it was judged that the unavailability of a single train of the AFW system would provide an upper bound for the one-train RCP seal cooling system arrangement either the high pressure cooling/pump or the firewater backup. Reference 1 includes a comprehensive evaluation of the various AFW systems at U.S. Nuclear Power Plants. Based on actual statistical data, the one train unavailability was established at P(Cooling) = $4x10^{-2}$. The corresponding data from Reference 1 is reproduced here as Table 5.3 for convenience. The first two columns of data represent the single train failure probabilities for two and three train AFW systems respectively. The average of these two data points (last column) was used in this study to represent the average unavailability of the proposed dedicated seal cooling system.

In order to establish the P(Seal LOCA), the failure probability of the RCP seals given the loss of ESW cooling, the method described in Section 4 was used. The seal LOCA probability is 6.8-02.

The core damage frequency for a particular state and configuration may be calculated as:

$$CDF(State) = \lambda_{ESW}*P(Cooling)*P(Seal\ LOCA) + Other\ Sequences$$
$$CDF_{Old} = \lambda_{ESW}*4*10^{-2}*6.8\text{-}02 + Other\ Sequences$$

The initiator, λ_{ESW}, was renormalized using the procedure described in Section 4 for each state and configuration.

The results of the calculations are shown in Table 5.1. The CDF reduction is significant (about a factor of ~2). The analysis in Section 4 indicates that the RCP seal LOCA contribution to the total CDF is the most dominant, ~60%. If Option 5 were to be utilized, the contributions would be reduced to ~5% (CDF(Seal LOCA/Option 5) = $3.48x10^{-6}$/ryr (CDF (Seal LOCA/Base Case) = $8.8*10^{-5}$ryr) and CDF(Total/Option 5) = $7.38x10^{-5}$/ryr). This indicates that further reductions in the RCP seal LOCA contribution would not significantly change the total CDF.

5.6 Combination of the Proposed Options

The value impact analysis of the previously discussed options (presented in Section 6) evaluated each option with regard to its cost/benefit effectiveness. It was shown (in Tables 6.19) that most of the proposed options have favorable cost/benefit aspects (presented as $/person-rem). In these cost/-benefit calculations, it was assumed that each option was utilized individually and independently from the other potential improvements.

In this section the CDF reduction is calculated when a number of options are combined and utilized together to reduce the risk due to the loss of ESW function.

In order to evaluate the effectiveness of each option, as they are combined together, the following simple procedure was used. First the options were ranked based on the effectiveness or the magnitude of the CDF reduction achievable by the particular option. Table 5.4 presents the list of the options with the corresponding ΔCDFs.

The first option with the highest ΔCDF and favorable cost/benefit ratio was the first selected and served as the basis point. This was Option 5, the dedicated cooling system for the RCP seals. The next option was considered and the CDF reduction was calculated from the case where Option 5 was already incorporated. The ΔCDF corresponding to this option (Option X) was calculated as

ΔCDF(Option "X") = CDF(with Option 5) - CDF (Option 5 + Option X).

The new ΔCDF of this option provided a base for a cost/benefit evaluation. If the cost/benefit ratio indicated a favorable ratio then the option was kept and the next option was considered. In this manner a number of different combinations were analyzed and options with disadvantageous cost/benefit aspects were discarded.

The final selection with advantageous cost/benefit aspects is shown in Table 5.5. The results indicate that the most effeccive option is the use of a dedicated RCP seal cooling system (Option 5).

Options such as TS changes, electrical cross-connections, are very cost effective and therefore have advantageous cost/benefit ratios. Section 6 will discuss the value impact analysis in more detail and present the corresponding cost/benefit values. The utilization of Options 5 and 4 would result in a ~60% reduction of the CDF. The CDF is reduced from CDF(Base = 1.52×10^{-4}/ryr to CDF(Option 5) = 7.38×10^{-5}/ryr (ΔCDF(Option 5) = 7.82×10^{-5}/ryr) by the dedicated RCP cooling system. The addition of Option 4 results in a CDF(Options 5 + 4) = 6.08×10^{-5}/ryr or an additional ΔCDF(Option 4) = 1.30×10^{-5}/ryr.

5.7 References

1. G.E. Bozoki and I.A. Papazoglou, "Probabilistic Evaluation of Limiting Conditions of Operations Outage Times for Auxiliary Feedwater Systems," Brookhaven National Laboratory, Technical Report, April 1985.

Table 5.1
Core Damage Frequency Reduction

	Total CDF	ΔCDF
Base Case	1.52-04	---
Option 1	1.36-04	1.60-05
Option 2	1.38-04	1.37-05
Option 3	6.07-05	9.13-05
Option 4	1.26-04	2.55-05
Option 5	7.38-05	7.82-05

Table 5.2a
ESW Non-Recovery Fractions

	ESW Non-Recovery Fraction	
Time, Hour	Electrical Failures	All Failures
1.5	.02	.36
2.5	.015	.26
4.5	.01	.05
24	Negligible	.01

Table 5.2b
Conditional Core Damage Probabilities

Sequence	Electrical Failures	Base Case
P(Seal LOCA)	7.0-03	6.8-02
P_{AFW}	2.3-02	2.3-02
P_{LAFW}	9.7-03	9.1-03
P_{Other}	3.3-03	3.2-03
$P_{Shutdown}$	3.4-03	2.8-02

Table 5.3
Failure Probabilities of Single AFW Trains

Probability	Based on Two Train System, P(2)	Based on Three Train System, P(3)	Average
Failure Probability to Start and Run an AFW Train	7.2×10^{-2}	3.4×10^{-2}	4.2×10^{-2}

Table 5.4
CDF Reduction Effectiveness

Option	ΔCDF
3	9.13-05
5	7.82-05
4	2.55-05
1	1.60-05
2	1.37-05

Table 5.5
Combination of Options

Option	CDF	ΔCDF
Base	1.52-04	---
Base + 5		7.82-05
Base + 5 + 4		1.30-05
Total (with 5+4 implemented)	6.08-05	

6. VALUE IMPACT ANALYSIS

Once the change in core damage frequency has been calculated for each mitigative option (Section 5 and Table 6.1), a value impact analysis is required to determine if implementation would be cost effective. An option is considered to be cost effective if the benefits (i.e., averted consequences) outweigh the costs. The benefits are expressed in person-rem averted, and the costs in dollars. If the dollars spent divided by averted person-rem exceed $1,000/person-rem, the option is not effective. Otherwise, it is cost effective. Note that in this study, shorthand for scientific notation is used, e.g., 2.3+5 means $2.3*10^5$.

6.1 Calculation of Benefits

As discussed above, benefits are expressed in averted person-rem. An important question then is how to calculate the person-rem associated with the calculated core damage frequency. The Reactor Safety Study (RSS)[18] was the first attempt to assess containment performance for a range of potential accident sequences. The RSS also developed a set of radioactive release characteristics associated with various containment failure modes. A number of other GIs have used the probabilities of containment failure and magnitudes of fission product release given in the RSS as the basis for assessing off-site consequences. However, the NRC staff has sponsored an extensive reassessment of the risk from five nuclear power plants (two from the RSS). This study (NUREG-1150) represents the most updated assessment of containment performance, source terms, and off-site consequences, and it was decided to use this as the basis for our evaluation of the GI-130 plants. The Zion consequence model[1] developed for NUREG-1150 was, therefore, used as the basis of our assessment. The consequences of a loss of service water accident were taken to be those for the SE sequence in Reference 1 (small LOCA with no high pressure injection, no containment fans and no sprays). The consequence model is specific to the Zion site. While the average population density from 0-500 miles around this plant is around 100 people/sq.mi. (close to the average U.S. population density of 80/sq.mi.),[2] it is much higher at certain distances (see Table 6.2) and in certain directions from the plant (around 1,000 people/sq.mi. in the 30-50 mile ring around the plant). On the other hand, the plant is situated on Lake Michigan, so large areas to the east of the plant are unpopulated. Although some of the multiplant sites encompassed within this study are not very far from major population centers, on the average, accident consequences should be smaller than the ones calculated for the Zion plant.

Table 6.3 shows how the Zion sequence SE (which we use here for loss of service water scenarios) is distributed into release bins. The bins represent source term release categories, with certain fractions of representative isotopes released for each bin. The bins are characterized by considering the important features of the accident from the containment behavior standpoint (see Table 6.4).

There are uncertainties associated with calculation of severe accident risk. These are explored by consideration of major uncertainty issues (areas of uncertainty where our knowledge is limited and where impact is believed to be significant). The issues are parametrized ("levels" are defined) and the parameter space is sampled, in this case using Latin Hypercube Sampling (LHS). For instance, one of the issues is severity of the direct heating effect, which can have several levels associated with it. In the Zion report,[1] the LHS was used to sample 100 points for each source term bin, in terms of the above mentioned uncertainties. The consequences of releases thus calculated were grouped together in "clusters," resulting in 30 clusters for Zion. Therefore, each cluster may have elements of one or more bins, which, by exploring various uncertainty issues, have consequences of similar magnitude for that particular "level" of issues considered. The person-rem consequences of each cluster, per accident, are shown in Table 6.5.

Table 6.6 shows the clusters that the six bins associated with the SE sequence fall into, and the consequences of the clusters and the bins. The average consequences of each bin are calculated, where the probability of each cluster occurring is given by the ratio of number of bin elements in the cluster and total number of samplings of the particular bin (100).

In order to arrive at the total average consequences for sequence SE, the bin consequences were added and weighted according to the probabilities in Table 6.3. The bin consequences and total consequences are given in Table 6.7, along with a short description for each bin.

In addition to Zion, we have looked at consequences of accidents at some other plants. Surry and Sequoyah are discussed in NUREG-1150,[3] and may be more representative of the plants in question, at least as far as the offsite consequences model is concerned. Surry is slightly smaller than the average plant, which was assumed to be 1170 MWe (Surry's output is 781 MWe for either reactor). This can be accommodated by simply multiplying the consequences in NUREG-1150 by the power ratio. Sequoyah is a 1148 MWe ice-containment design (two such reactors onsite). Three of the plants impacted by this study (D.C. Cook, Catawba and McGuire) have ice containments of the Sequoyah type, while the other four have large, dry containments of the Zion type.

For Surry we approximated the loss of ESW by choosing sequence T1(SL)_D1CF1. This is loss of ac power, with RCP seal LOCA, unavailability of the HPI system, containment spray injection mode and inside spray recirculation system. This leads to plant damage state SNNN: LOCA with high RCS pressure prior to vessel failure, no discharge of RWST water into containment, no containment heat removal and no injection or recirculation of containment spray water. Sequence T1(SL)_D1CF1 has a frequency of 6.6×10^{-6}/yr, thus contributing 26% to the total CDF. Damage state SNNN contributes 42% to the risk of latent fatalities. Based on the latter number and adjusting for reactor size, the person-rem entry in Table 6.8 was calculated.

Similarly, in the Sequoyah PRA the plant damage state SNNY (seal LOCA with no core cooling in either injection or recirculation phase, no containment heat removal, no containment sprays and with ac power available) was judged to represent the loss of ESW initiator. This state contributes 62.5% to latent fatality risk.

The numbers for total consequences in Tables 6.7 and 6.8 should be multiplied by 30 years (average remaining life of affected plants) and ΔCDF (change in core damage frequency due to the mitigative options considered) to obtain the benefits of the proposed actions. It can be seen that the numbers are all in the vicinity of $1.0*10^7$ person-rem (Zion number is a little higher, but that is expected). This number is a rough indicator of consequences to be expected from this type of accident. The consequences calculated for Zion and Sequoyah are next used to arrive at a more refined estimate of consequences of a GI-130 plant, as is shown below.

From Table 6.7, the mean Zion consequences are $1.7*10^7$ person-rem, given a core melt from loss of service water. This number will be used as a starting point for calculating the consequences of the large dry containment plants affected by GI-130 (Comanche Peak, Byron, Braidwood, and Diablo Canyon). It should also be noted that, at this point, we are considering only accidents at full power operation. In addition, no recovery actions after core damage to limit off-site consequences (i.e., restoration of CHRS) are considered. Adjustments will be made in later sections to include shutdown accidents and recovery actions in our consequence analysis.

For the three plants with ice-condenser containments, we will use Sequoyah numbers as a starting point. If the limits given in Table 6.8 are assumed to represent the 5th percentile and the 95th percentile of a lognormal distribution then the calculated mean is 5.07+6 person-rem for Sequoyah.

Table 6.9a shows the "relative safety margins" for those large dry containment plants affected by GI-130. The "relative safety margin" is calculated by first taking the following ratio of plant parameters important to containment performance in an accident:

$$\frac{\text{Containment Design Pressure (psig)} * \text{Containment Free Volume (ft}^3\text{)}}{\text{Reactor Thermal Power (MWth)}}$$

The above ratio yields a containment performance indicator. Next, this number is divided by the containment performance indicator of a representative plant (Zion in this case) to give the relative safety margin displayed in Table 6.9a. This number represents the relative strength of a containment compared to the representative plant. We can see that the Byron and Braidwood containments have higher performance indicators than that of Zion (relative safety margin >1.00), while the Comanche Peak and Diablo Canyon containments have lower performance indicators (relative safety margin <1.00). The average relative safety margin of the four plants is 1.04; therefore, it is assumed that the average GI-130 plant containment will behave in a similar manner to the Zion containment. This information will be combined with data in Table 6.10a and Table 6.7 to give the expected consequences of SW induced core melt at an average GI-130 site.

Table 6.9b displays the relative safety margins of the ice condenser plants compared to Sequoyah as the reference plant. The average margin for the GI-130 ice-condenser containments is 1.18 indicating an average GI-130 ice condenser containment is stronger than Sequoyah's.

Tables 6.10a and 6.10b display information taken from the SNL site study.[4] This shows the mean latent cancers (which are proportional to total person-rem consequences) of the sites of interest (GI-130 sites and the two reference sites, Zion and Sequoyah) for three release categories, SST1, SST2, and SST3. SST1 represents a core melt with loss of all installed safety features and an early failure of the containment. SST2 represents a core melt with containment systems operating and containment failure by hydrogen combustion or failure to isolate. SST3 is a core melt with containment systems operating and late containment failure by basemat melt-through. We will be looking mostly at the SST1 column (since containment systems are assumed to fail); however, the other two columns would lead to essentially the same conclusion as to the magnitude of consequences for an average GI-130 site relative to the consequences of the two reference plant sites.

Note that the results in Tables 6.10a and 6.10b assume a reference 1,120 MWe reactor at all the sites. The consequence calculations are conditional on the various source terms occurring; therefore, the study does not discriminate among various containment designs, relative containment strengths or reactor sizes, but are meant to compare site characteristics only (chiefly based on the actual population densities). It should be noted that all the reactors under consideration are close to the 1,120 MWe reference.

Based on Table 6.10a, we conclude that an average large, dry containment GI-130 site would produce consequences that are 0.47 those of Zion. The range in relative consequences is 0.16 for the Comanche Peak site to 0.80 for the Braidwood site. Since, according to Table 6.9a, the average GI-130 large, dry containment strength is very close to that of Zion (see discussion above), then we will take the consequences of a service water induced core melt at an average GI-130 large, dry containment plant to be 47% of the total consequences for Zion shown in Table 6.7, or $8.0*10^6$ person-rem. Again, this is for power operation only and without taking containment systems recovery into consideration.

Similarly, according to Table 6.10b, an average GI-130 ice condenser containment site will have consequences that are 1.44 times the consequences at the Sequoyah site (the range in this coefficient is from 1.15 for the Catawba site to 1.92 for the D.C. Cook site). According to Table 6.9b, the average strength of a GI-130 ice condenser containment is 1.18 times the strength of the Sequoyah containment. Therefore, simplistically, the expected consequences of a loss of service-water induced core melt at an average GI-130 ice containment plant are 1.44/1.18 = 1.22 times the mean consequences at Sequoyah, calculated above, or $6.2*10^6$ person-rem. This number is close (within the uncertainties of this analysis) to the consequences calculated for the average dry containment plant ($8*10^6$ person-rem). Since the dry containment number is based on a more detailed analysis for Zion, we will take this number ($8*10^6$ person-rem) to be representative of all the GI-130 plants. This figure will now be modified to take into account the recovery actions.

6.1.1 Full Power Consequence Analysis With Recovery of Containment Systems

Given core melt after a loss of service water accident, a certain fraction of subsequent containment failures could be prevented by recovery of the ESW system on a timely basis. We will now adjust our consequence model to account for such recovery actions.

Three types of containment failures have been identified--namely, early failures (at or shortly after vessel breach), late failures and basemat melt-through (BMT). Table 6.3 shows the distribution of SW-induced core melt scenarios into bins (for Zion). Table 6.7 explains the bin characteristics as derived from Table 6.4. Therefore, bins 1 and 16 are early over pressure failures due to various phenomena, while bins 9 and 10 are late failures. Bin 6 is an isolation failure, so it will be grouped with early failures for our purposes. Bin 13 represents the BMT. It is assumed that once the vessel breach is complete, neither the early failures nor the BMT failures are recoverable. Early failures are not recoverable at this point because there is no time, by definition. BMT is not recoverable, because if coolable geometry is not established right away (by sufficient water in reactor cavity), the core debris can potentially continue to attack concrete even if water is poured on top at some later time. Combining the BMT and the early failures means that about 46% of would-be containment failures (conditioned on the core melt) can only be recovered in the time period between the core melt and the vessel breach. The other 54% of containment failures can be recovered at any time after the core melt.

In order to establish a time scale for this analysis we first look at the analysis in Section 4. The ESW non-recovery factors are given such that at 4.5 hours after the initiator, the non-recovery factor (NR) equals 0.05, and at 24 hours, it is 0.01. For this analysis, we assume a linear interpolation between these two points. According to Tables 4.6b and 4.8 the dominant time to uncovery for seal-LOCA sequences is about 4.1 hours, but with the 21 hour sequence also contributing. The time to uncovery for early AFW sequences is somewhat longer. Late AFW sequences and 'other' are assumed to lead to no containment failure because the time is sufficient for almost perfect recovery. Based on Table 4.16b, these represent about 12% of the total CDF. Therefore, about 88% of total CDF is caused by sequences that will be included in the containment recovery analysis. Based on the above discussion, 5.5 hours on the average will elapse between the initiator and the core uncovery (i.e., core melt, because uncovery is to the mid-plane of the core). Another 1.5 hours will pass until the vessel breach (VB). Therefore, VB and early containment failures occur at seven hours after accident initiation. Late containment failure will occur, on the average, 24 hours after VB, or at 31 hours after loss of ESW. The range is 12-48 hours after VB. We will consider only 12 and 24 hours after VB for late failures, as our sensitivity analysis.

One point to consider in recovery of late containment failures is the generation of aerosols after VB. These aerosols may plug the containment spray nozzles or the containment fan cooler units (unless the filters can be by-passed). Potential degradation of containment heat removal systems (CHRS) was therefore examined by sensitivity analysis in which degradation of CHRS was given no chance, 50% chance and 100% chance of occurring. Degradation of CHRS will, of course, prevent containment recovery.

To summarize, the times of interest are:

5.5 hours: core melt
7 hours: vessel breach, early containment failure, BMT
19 hours. late failure, or
31 hours: late failure

The slope of the non-recovery curve in the time period of interest is:

$$s = \frac{0.05 - 0.01}{24 - 4.5} = 2.05*10^{-3}$$

At 5.5 hours (core melt), the non-recovery factor is:

$$NR_{CM} = 0.05 - (5.5 - 4.5) * s = 0.048 \quad .$$

At 7 hours (VB and early containment failure):

$$NR_{EF} = 0.05 - (7 - 4.5) * s = 0.045 \quad .$$

At 19 hours (late failure):

$$NR_{LF,19} = 0.020 \quad .$$

At 31 hours (late failure), the non-recovered factor is assumed zero,

$$NR_{LF,31} = 0.0 \quad .$$

Therefore, the early failure non-recovery fraction given core melt is:

$$EFNRF = \frac{0.045}{0.048} * 0.88 = 0.825,$$

where the factor of 0.88 signifies the fraction of core melts that lead to containment failure (as discussed above).

The late failure non-recovered fraction (assuming late failure 12 hours after VB or 19 hours after the initiator) is:

$$LFNRF_{19} = \frac{0.020}{0.048} * 0.88 = 0.37 \quad .$$

Assuming late failures occur 24 hours after VB (31 hours after the initiator):

$$LFNRF_{31} = 0.0 \quad .$$

Remembering that bins 1, 6, 13, and 16 should be counted in early failures for this analysis, and using data from Table 6.3 (for bin distribution fractions) and Table 6.7 (for person-rem consequences for each bin), the expected consequences at Zion due to early containment failures from this type of accident are:

$$CQ_{EF,Z} = EFNRF * (0.27*3.63*10^{7} + 0.16*3.63*10^{4} + 0.0021*1.7*10^{7}$$
$$+ 0.03*3.0*10^{7}) = 8.86 + 6 \text{ person-rem} \ .$$

For late failures at 12 hours and no aerosol plugging, the consequences are:

$$CQ_{LF,Z,12,0} = LFNRF_{19} * (0.3*1.2*10^{7} + 0.24*1.05*10^{7}) = 2.24 + 6 \text{ person-rem} \ .$$

For late failures at 12 hours and 100% chance of aerosol plugging, the consequences are:

$$CQ_{LF,Z,12,100} = 0.88 * (0.3*1.2*10^{7} + 0.24*1.05*10^{7}) = 5.39 + 6 \text{ person-rem} \ .$$

For late failures at '2 hours and 50% chance of aerosol plugging, the consequences are:

$$CQ_{LF,Z,12,50} = (2.24*10^{6} + 5.39*10^{6})/2 = 3.82 + 6 \text{ person-rem.}$$

For late failures at 24 hours and no chance of plugging, the consequences are:

$$CQ_{LF,Z,24,0} = 0.0 \ .$$

For late failures at 24 hours and 100% chance of plugging, the consequences are:

$$CQ_{LF,Z,24,100} = CQ_{LF,Z,12,100} = 5.39 + 6 \text{ person-rem.}$$

And, in case of 50% chance of plugging, the consequences are the average of the two extremes:

$$CQ_{LF,Z,24,50} = (5.39*10^{6} + 0)/2 = 2.70 + 6 \text{ person-rem.}$$

Summing up the expected early and late consequences, we arrive at Table 6.11a for Zion.

Table 6.11b shows the expected consequences with recovery at the average GI-130 plant for power operation. These values are calculated by applying the factor of 0.47 (discussed above) to the Zion numbers in Table 6.11a.

At this point our best estimate is that late failures will occur 24 hours after VB, and there will be a 50% chance of plugging by aerosols. Therefore, from Table 6.11b, the expected consequences at a GI-130 plant, assuming power operation with recovery, will be 5.5×10^{6} person-rem.

6.1.2 Calculation of Consequences at Shutdown With Containment Recovery

As shown in Section 4, the reactor will spend approximately 100 days per year in shutdown, or 27% of the time. As stated in Section 4, about 40% of that time will be spent draining the reactor to the mid-loop, or in the mid-loop condition, 50% will be spent refueling or fuel shuffling and 10% in a non-drained state, assumed prior to power operation.

It is assumed that in the last phase, the time for recovery will be long and the containment hatch will be closed. Any accident in this phase will lead to release category PWR7 (following the methodology and the nomenclature of Reference 2). This is a release where all containment systems are working, the containment is isolated and it never fails.

In the second phase (refueling), the technical specifications require that the containment equipment hatch be closed when the fuel is shuffled. The time available for recovery will be long since the refueling cavity will be flooded. Therefore, any accident in this phase will also lead to release category PWR7.

In the first phase (drained and draining), the time to recovery will be short (the time to core uncovery would be on the order of 5.5 hours, the same as in the power operation recovery model). The equipment hatch is assumed to be open, absent technical specifications to the contrary. In this case, if there is no recovery of service water, the release category would be PWR2 (containment systems not working and containment open). In case of recovery between core melt and vessel breach, the release category would be PWR2B (containment open but containment sprays working to scrub certain fission products, thus reducing the offsite consequences). If the recovery occurs after vessel breach, we assume it is too late to reduce the consequences (since the containment is open), therefore, the release category stays as PWR2.

It should be noted that our assumption of equipment hatch being open in Phase 2 with a probability of one is conservative in view of resolution of Generic Issue 99 (RHR reliability). Plants now have procedures to start closing the equipment hatch after certain abnormal occurrences (it takes one to two hours to close the hatch and bolt it down). Therefore, this probability of open containment should be reduced to somewhere below 1.0, but it's not clear what this number should be. At this point, we leave this probability equal to 1.0 and note that shutdown consequences have a minor (~10%) effect on our cost-benefit analysis.

We relate the PWR release categories to bins from the Zion NUREG-1150 study, and calculate the bin consequences as before, for power operation.

Using power operation bins for consequence calculations at shutdown may seem overly conservative (the same approach was used in Reference 2, except that WASH-1400 source terms were used there). However, as shown in Reference 2, the person-rem consequences are relatively insensitive to the source term, because of interdiction criteria and because of the relatively high contribution of long lived isotopes to the long term dose (there was not much difference in consequences between a release two days after shutdown and one five days after shutdown).

Based on the bin description in Table 6.4 and Table 6.7, bin 6 corresponds to PWR2 release category, bin 7 to PWR2B release category, bin 14 to PWR7 release category and bin 15 to no containment failure. Table 6.12 shows this relationship, and the calculated bin consequences.

If the core melt in the first phase occurs at an average time of 5.5 hours after the initiator (see Table 4.11), the non-recovery probability at that time will be 0.048 (see calculations in Section 6.1.1). The vessel breach occurs at seven hours, when the non-recovery probability is 0.045. Therefore, in Phase 1 (which is 40% of shutdown time), PWR 2 releases will occur in 0.045/0.048 of the cases, and PWR2B releases will occur in (0.048-0.045)/0.048 = 0.003/0.048 of the cases. Remembering that an average GI-130 plant has consequences that are 47% the consequences at Zion, the expected Phase 1 consequences contribution will be:

$$0.4 * (\frac{3}{48} * 5.04 * 10^6 + \frac{45}{48} * 1.7 * 10^7) * 0.47 = 3.06 + 6 \text{ person-rem.}$$

Phases 2 and 3 occur 60% of the time in shutdown, and the consequences correspond to bins 14 and 15 in Table 6.12. Therefore, consequences contribution for Phases 2 and 3 at an average GI-130 plant will be:

$$0.6 * 0.47 * 3.63 * 10^4 = 1.02 + 4 \text{ person-rem.}$$

Adding the contributions for all three phases, we arrive at expected shutdown consequences of 3.07+6 person-rem.

6.1.3 Total Benefits

The total benefits of implementing an option over an assumed 30-year lifetime of the plant are calculated by combining the power consequences with the power ΔCDF and the shutdown consequences with the shutdown ΔCDF, adding the two results and multiplying by 30 years. Thus:

$$BEN = 30 * (\Delta CDF_{power} * 5.5 * 10^6 + \Delta CDF_{sd} * 3.1 * 10^6).$$

The total expected consequences due to SW losses at an average GI-130 plant will be the base case (from which consequences will be reduced by applying various improvement options). The total power CDF is 1.3-4/yr and the total shutdown CDF is 2.0-5/yr. Therefore, the total 30 year consequences will be:

$$30 * (1.3 * 10^{-4} * 5.5 * 10^6 + 2.0 * 10^{-5} * 3.1 * 10^6) = 23,310 \text{ person-rem.}$$

Table 6.13 shows the benefits (in person-rem) of each improvement option. This information is calculated by using data in Table 6.1 and data on consequences for power and shutdown operation, as in formula for BEN, above. For our best estimate, we use 5.5+6 person-rem for consequences from a power operation accident and 3.1+6 person-rem for consequences from a shutdown accident.

For our highest estimate, we take the highest number from Table 6.11b (100% plugging at an average GI-130 plant) and adjust for Braidwood site (which should have the highest consequences). This will give us the highest estimate for power operation. Thus:

$$CON_{hi,power} = 6.67*10^6 * \frac{0.8}{0.47} * \frac{1.04}{1.12} = 1.05 + 7 \text{ person-rem}$$

where 0.8 is the ratio of latent cancers at Braidwood to latent cancers at Zion (Table 6.10a), 0.47 is the same ratio for an average GI-130 site, 1.12 is the relative safety margin for Braidwood containment (see Table 6.9a) and 1.04 is the relative safety margin of a GI-130 average plant with large, dry containment.

For our high estimate of consequences at shutdown, we simply apply our best estimate, 3.1+6 person-rem, to the Braidwood reactor site, thus:

$$CON_{hi,sd} = 3.1*10^6 * \frac{0.8}{0.47} * \frac{1.04}{1.12} = 4.9 + 6 \text{ person-rem}$$

We combine these two numbers with data from Table 6.1 to arrive at high limit on benefits for a proposed option:

$$BEN_{hi} = 30 * (\Delta CDF_{power} * 1.05*10^7 + \Delta CDF_{sd} * 4.9*10^6)$$

Similarly, for our low limits, we start at lowest consequences at power operation, from Table 6.11b (no plugging, 24 hour span for late containment failures) and apply this number to the Comanche Peak reactor site (which should have the lowest consequences of the GI-130 plants). Again, using information from Tables 6.9a and 6.10a, the power operation low limit consequences are:

$$CON_{\ell o,power} = 4.16*10^6 * \frac{0.16}{0.47} * \frac{1.04}{0.96} = 1.5 + 6 \text{ person-rem}$$

For shutdown operation, the low limit on consequences is:

$$CON_{\ell o,sd} = 3.1*10^6 * \frac{0.16}{0.47} * \frac{1.04}{0.96} = 1.1 + 6 \text{ person-rem}$$

Then the low limit on benefits for a proposed option will be:

$$BEN_{\ell o} = 30 * (\Delta CDF_{power} * 1.5*10^6 + \Delta CDF_{sd} * 1.1*10^6)$$

6.2 Calculation of Costs

6.2.1 Onsite Consequences

Onsite consequences are taken into account as offsets to the calculated cost of proposed options. Table 6.14 lists the onsite consequences considered in this study. It can be seen that the onsite personnel exposure per accident will be low, compared to the offsite exposure, and compared to other onsite consequences, so this component was not considered further. (The numbers are from NUREG/CR-3568[5] as representative best estimate numbers). Averted onsite exposure would be added to the offsite person-rem exposure as part of the benefits. For cleanup and replacement power, the integrated and discounted cost is $\$3.0*10^{10}$yr. This number is multiplied by the total ΔCDF to arrive at the offset cost of each option. The total ΔCDF is simply the sum of the power ΔCDF and the shutdown ΔCDF. This can be compared to $\$1.0*10^{10}$yr estimated in NUREG/CR-3568[5]). The cleanup and replacement power costs in Table 6.14 were calculated from:[5]

$$u = (C_c + C_r) \frac{1}{r^2} (1-e^{-r\Delta t}) (1-e^{-rm}) \qquad (6.1)$$

where: u = integrated and discounted cost as explained above
C_c = cost of cleanup (\$100M/yr)
C_r = cost of replacement power (\$400K/day)
r = discount rate (NRC recommended[7] = 0.05/yr)
Δt = remaining plant life (30 yr)
m = duration of cleanup/power replacement (10 yr)

The onsite consequences are deducted from the cost of each option. Table 6.15 shows cost offsets for proposed improvements.

6.2.2 Cost Calculations

In order to arrive at costs for improvement options, several sources were consulted. Two of the sources for cost estimates were from two GI-130 plants. Some cost estimates were derived from an NRC-sponsored research report.[6] Another source[7] was the computer printout for the Energy Economic Data Base (EEDB) and supporting documents.[8,9,10,11] This computerized data base shows "green-field" costs of systems in a nuclear (or coal-fired) plant. In other words the costs are for a plant that is under construction. The cost of a backfit (i.e., installing a component in an already finished plant) is greater than the green-field cost due to such factors as problems in access and handling, congestion and interference, possible radiation environment and manageability problems when many different tasks are going on concurrently during an outage. Labor productivity factors[11] are used to adjust the green-field costs from the EEDB.

For the option of providing an ac-independent charging pump for RCP seals, or for providing firewater to cool the thermal barriers, several reports that dealt with resolution of Generic Issue 44 (station blackout) and Generic Issue 23 (RCP seals) were looked at.[12,13,14,15,16] In addition, informal input from the two utility sources on these costs was included in our data.

For some components of Option 4 improvement (updating tech specs, procedures, and cross-tie testing), an additional source of cost estimates was Reference 17. The result of this process is that there are 2-3 cost estimates for most improvement options. The best estimate was determined as the average of the individual estimates for an option. A high and a low limit was obtained for the uncertainty analysis. The cost estimate sources and bases (mainly for direct costs) are discussed in more detail in Appendix E.

It should be noted that the overall assumption is that the backfits can be accomplished out of the critical path, and utility personnel confirm that this should be possible. Otherwise, the direct costs will rise substantially, at the rate of $400K for each day that replacement power is needed.

Table 6.16 shows components of the total cost and the net cost for the best estimate case (the costs are per reactor). The net cost is the total cost minus the cost offset (from Table 6.15). If the net cost is negative, the option is beneficial regardless of the cost benefit ratio. It should be noted that a cost item in Table 6.16 subsumes the cost item in the previous column and includes an additional indicated cost component. For instance column 'include indirect cost' includes the direct cost and the indirect costs of an option.

For each improvement option the following costs are considered.

6.2.2.1 Direct Cost of Installed Option

This cost includes factory purchases, installation and onsite labor and materials, but excludes indirect costs (e.g., engineering, administrative, etc.). It is given in the first column of Table 6.16 for the best estimate.

Table 6.17 shows the best estimate and the range of estimates in the direct cost. Option 4 (updating tech. specs. and procedures and crosstie testing) shows a zero in the direct cost because this item is placed in Column 4 (tech. spec. costs) of Table 6.16.

6.2.2.2 Indirect Costs

As mentioned above, the indirect costs have to be considered separately. Usually, they are taken to be a certain fraction of the direct cost. As per recommendations in NUREG/CR-4627 we used 30% (the range is from 25% to 33% for engineering and quality assurance costs for in-place structures). Column 2 of Table 6.16 includes this component.

6.2.2.3 Operating and Maintenance Costs

It is usually recommended that these costs annually equal 3% of total "overnight" costs. Overnight costs represent the sum of total direct and indirect costs assuming that the modification was completed overnight (e.g., excluding the time costs of capital). This value is used here. In order to arrive at the total operating and maintenance (O&M) cost, the annual value has to be integrated and discounted over the remaining plant life (30 years). Option 4 (changing Technical Specifications) was assumed not to involve any O&M costs. Column 3 of Table 6.16 includes this component. In calculating O&M costs we assumed 0% real escalation rate (over and above inflation) and 5% discount rate.

6.2.2.4 Technical Specification Costs

Each option involves modifying Technical Specifications to a certain extent. According to NUREG/CR-4627, these costs are \$18K/reactor for a simple case and \$35K/reactor for a complicated or controversial one. It was assumed that each option will result in a simple Technical Specification change. (Neither choice includes the cost of a public hearing.) Column 4 of Table 6.16 includes this component of cost. A utility estimates this cost as \$50K per reactor. An average of the NRC and utility estimates is taken for our analysis.

6.2.2.5 NRC Costs

NRC costs include the development and implementation costs. The development costs should be about \$11K/reactor for a simple case and \$21K/reactor for a complicated one (neither case includes the cost of a public hearing). The former figure was chosen here. Operating costs would be incurred after the rule's implementation and they would ensure compliance with the new rule. The operating costs have to be integrated and discounted, since they are recurring. The implementation and operating costs are estimated at \$50K/reactor as per recommendation of the NRC staff. Total NRC costs would then be \$11K + 50K = \$61K per reactor. Column 5 of Table 6.16 includes the NRC costs. For a technical specification change, the total NRC costs would be \$11K per reactor.[6]

6.2.2.6 Cost Offsets

As mentioned earlier, the integrated and discounted onsite consequences of loss of ESW accidents were treated as cost offsets (negative costs), rather than benefits. The resulting cost offsets are shown in Table 6.15 and were subtracted from the total cost of the proposed action (Column 5 of Table 6.16) to arrive at the net cost of such action (Column 6 of Table 6.16). If the net cost was negative, then the proposed action was cost effective even without cost/person-rem consideration, because the averted onsite costs already exceeded the total cost of the mitigative option.

6.3 Cost Uncertainty

Table 6.18 presents the low limit and the high limit in the total cost (corresponding to Column 5 of Table 6.16) and the net cost (corresponding to Column 6 of Table 6.16). The low limits were calculated by taking the lowest estimates in our data of various cost components (mainly direct costs) and carrying the computation through to the final number. The high limits were done in a likewise fashion.

6.4 Cost Benefit Analysis

Table 6.19 displays the best estimate cost-benefit ratios for each improvement option, as well as for the combination of options. The data in this table was calculated by combining information from Table 6.16 (Columns 5 and 6) and Table 6.13 (Column 1). Therefore, Column 2 of Table 6.19 is the best estimate total cost-benefit ratio in \$/person-rem, which doesn't include the cost offset due to averted onsite consequences. Column 1 gives the best estimate net cost-benefit ratio which includes this cost offset. If the net cost is negative, then no net cost-benefit ratio is shown, and the option is beneficial regardless of the absolute value of the cost benefit ratio. The cost benefit ratios can be compared to the value of $1000/person-rem as the limit for cost effectiveness. On this basis, the swing pump option is the only one which is not beneficial, regardless of which cost benefit ratio is taken. The pump itself is only a small fraction of the cost of this option (~25%). The majority of the cost is due to construction activities related to the intake structure, piping, etc. It can be seen that the combination of options is also cost effective as are the steps in the combination. The combination with firewater is more beneficial than the one with high pressure seal injection.

Table 6.20 shows the uncertainty range in the two cost-benefit ratios. The low values are calculated by taking the low limit on cost in Table 6.18 and dividing by the high limit on benefit in Table 6.13, and vice versa for the high values in Table 6.20. The options whose uncertainty range falls within the cost-effectiveness limit are 2 (electrical cross-connection), 4 (updating tech specs, procedures and crosstie testing), and 5a (firewater for thermal barriers), as is the case for the combination involving firewater cooling of thermal barriers.

6.5 Conclusions

Benefits and costs for mitigative options for multiplant ESW systems have been calculated. The resulting costs per person-rem averted have been compared to the standard cutoff of $1,000/person-rem. Also, calculated negative net costs imply cost effectiveness. On this basis, the cost-effective options are: 1) an additional crosstie, 2) electrical connection, 4) changing Technical Specifications procedures and a crosstie test, and 5) adding non-ac dependent high pressure pumps or firewater connections for the RCP seals.

There are uncertainties in calculating the benefits and the costs of the various options. The uncertainties in the benefits pertain to the appropriate site model, plant type and the consequence model. These are also uncertainties in ΔCDF values which have not been considered in this study.

On the cost side, the uncertainties arise in the direct cost of each option and in the other cost components. In addition, costs will vary among various regions of the country and from plant to plant. Site characteristics will affect the cost (e.g., oceanside plants vs. inland plants, seismic characteristics of the site, etc.). Also, some plants have a closed cooling system. Therefore, the conclusions should be used as general guidance.

6.6 References

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Table 6.1
Core Damage Frequency Reduction

Option	Improvement Alternative	ΔCDF (yr^{-1}) Power Operation	ΔCDF (yr^{-1}) Shutdown
1.	Additional Crosstie	1.45-5	2.61-6
2.	Electrical Dependency	1.39-5	6.0-7
3.	Swing Pump + Intake + El. Connection	8.05-5	1.12-5
4.	Changing Tech. Specs., Procedures, X-tie Test	2.36-5	2.66-6
5.	High Pressure for RCP Seals or Fire Water for Thermal Barriers	7.80-5	0.
Combination, 5 + 4:			
5.	Charging High Pressure Pump for RCP Seals or Fire Water for Thermal Barriers	7.80-5	0.
4.	Updated Tech. Specs., Procedures, X-tie Test	1.03-5	2.7-6
Total Combination		8.83-5	2.7-6

Table 6.2
Zion 1980 Population

Distance (mi)	Population
0-1	3164
1-2	9501
2-3	10803
3-4	15128
4-5	19450
5-10	183382
10-15	111847
15-20	239112
20-25	393187
25-30	480563
30-40	2914014
40-50	2774059
50-60	1400717
60-70	1419040
70-85	669778
85-100	1341699
100-150	5498307
150-200	7455088
200-350	28959226
350-500	25814193

Table 6.3
Distribution of SE Sequences Into Bins

Bin Number	Probability
1	0.03
6	0.0021
9	0.30
10	0.24
13	0.16
16	0.27

Table 6.4
Characteristics of the Source Term Bins

Sequence and Containment Status	Source Term Bins 1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
<u>Containment Failure</u>																			
Rupture Before Core Melt					*														
Early Overpressure	*	*	*	*												*	*	*	*
Late Overpressure-Rupture								*	*										
Late Overpressure-Leakage										*									
Melt-Through													*	*					
Leak or Isolation Failure						*	*												
Containment Bypass-SGTR											*								
Containment Bypass-Dry												*							
No Failure															*				
<u>Containment Spray System</u>																			
(See Note 1) Operates		*		*			*	*						*			*		*
Fails	*		*			*			*				*			*		*	
<u>Primary System Pressure</u>																			
High	*	*														*	*		
Moderate			*	*														*	*
Low																			
<u>Containment Pressure</u>																			
(See Note 2) High													*						
Low														*					
<u>Water Available to Cavity</u>																			
Yes		*		*				*											
No	*		*		*	*	*		*							*	*	*	*
<u>Direct Heating Effect</u>																			
None	*	*	*	*															
Significant																*	*	*	*

*The characteristic is required for the bin. Characteristics not marked are not determinant of the bin and any combination may apply.

<u>Note 1</u>: The spray question is also dependent on timing. Critical time frames are different for different bins.
<u>Note 2</u>: This is only used as a discriminator for Bins 13 and 14. Obviously, containment pressure is high in many other bins and these are not checked.

Table 6.5
Estimated Mean Consequences for Zion Releases*

Source Term Cluster	Person-Rem (<50 Miles)
1	4.04+07
2	4.22+7
3	4.37+7
4	4.42+7
5	3.88+7
6	3.07+7
7	3.24+7
8	2.89+7
9	3.43+7
10	2.85+7
11	3.01+7
12	2.31+7
13	2.80+7
14	2.44+7
15	1.95+7
16	1.95+7
17	1.96+7
18	1.47+7
19	1.53+7
20	1.19+7
21	1.13+7
22	8.26+6
23	8.85+6
24	3.19+6
25	4.92+6
26	1.91+6
27	1.91+6
28	1.15+6
29	6.77+5
30	3.63+4

*Evacuation assumptions: 95% within 10 miles evacuate at 1.1 meters per second after a delay of 1.8 hours from start of core melt. 5% within 10 miles do not participate and follow normal activities for one day. Beyond 10 miles, all individuals follow normal activities for one day.

Table 6.6
Calculation of Bin Consequences From Cluster Consequences

Bin #	Person-Rem	Cluster	Number of Elements
Bin 1	4.22+7	2	5
	4.37+7	3	2
PR = 3.0+7	4.42+7	4	7
	3.88+7	5	2
	3.07+7	6	16
	3.24+7	7	3
	2.89+7	8	5
	3.43+7	9	10
	2.85+7	10	14
	3.01+7	11	5
	2.80+7	13	11
	2.44+7	14	5
	1.95+7	15	8
	1.96+7	17	3
	1.47+7	18	3
	1.19+7	20	1
Bin 6	3.43+7	9	6
	2.85+7	10	3
PR = 1.7+7	3.01+7	11	15
	2.80+7	13	2
	1.95+7	15	8
	1.96+7	17	2
	1.47+7	18	17
	1.19+7	20	28
	8.26+6	22	9
	3.19+6	24	6
	1.91+6	26	4
Bin 9	2.31+7	12	7
PR = 1.2+7	1.95+7	16	15
	1.53+7	19	22
	1.13+7	21	10
	8.85+6	23	21
	4.92+6	25	17
	1.91+6	27	5
	1.15+6	28	1
	6.77+5	29	1
	3.63+4	30	1
Bin 10	2.31+7	12	5
PR = 1.05+7	1.95+7	16	13
	1.53+7	19	15

Table 6.6 (Continued)

Bin #	Person-Rem	Cluster	Number of Elements
	1.13+7	21	12
	8.85+6	23	21
	4.92+6	25	23
	1.91+6	27	7
	1.15+6	28	1
	6.77+5	29	2
	3.63+4	30	1
<u>Bin 13</u> PR = 3.63+4	3.63+4	30	100
<u>Bin 16</u>	4.04+7	1	40
PR = 3.63+7	4.22+7	2	110
	4.37+7	3	14
	4.42+7	4	9
	3.88+7	5	12
	3.07+7	6	9
	3.24+7	7	13
	2.89+7	8	4
	3.43+7	9	6
	2.85+7	10	9
	3.01+7	11	5
	2.80+7	13	3
	2.44+7	14	1

PR = Person-Rem.

Table 6.7
Average Consequences for Core Melt
Following Loss of Service Water at Zion

Bin	Bin Description	Consequences (person-rem)
1	Early overpressure failure	3.0+7
6	Leak or isolation failure	1.7+7
9	Late overpressure-rupture	1.2+7
10	Late overpressure-leakage	1.05+7
13	Basemat melt through	3.63+4
16	Early overpressure-DCH	3.63+7
	SE Sequence, Average	1.7+7

Table 6.8
Person-Rem Consequences for Surry SNNN
and Sequoyah SNNNY States

Plant State	Person-Rem
Surry SNNN	3.0+6-1.05+7
Sequoyah SNNNY	1.4+6-1.2+7

Table 6.9a
Relative Safety Margins of GI-130
Large, Dry Containment Plants, Compared to Zion

Plant	Relative Safety Margin
1. Comanche Peak	0.96
2. Byron	1.12
3. Braidwood	1.12
4. Diablo Canyon	0.97

Table 6.9b
Relative Safety Margins of GI-130
Ice Condenser Containment Plants, Compared to Sequoyah

Plant	Relative Safety Margin
1. D.C. Cook	1.05
2. Catawba	1.25
3. McGuire	1.25

Table 6.10a
Siting Comparison of GI-130 Large, Dry
Containment Plant Sites to the Zion Site

	Mean Latent Cancers		
Site	SST1*	SST2	SST3
1. Comanche Peak	640	49	0.2
2. Byron	2500	190	0.7
3. Braidwood	3200	240	0.9
4. Diablo Canyon	1200	98	0.4
Reference Zion	4000	330	1.2

*SST1: Refers to a set of "siting source terms" that were developed in NUREG-0773[19] to represent a range of potential severe accident source terms for LWRs. The source terms were based on the release categories developed as part of the Reactor Safety Study and supplemented by additional calculations.

Table 6.10b
Siting Comparison of GI-130
Ice Containment Plant Sites to the Sequoyah Site

Site	Mean Latent Cancers		
	SST1	SST2	SST3
1. D.C. Cook	2500	120	0.4
2. Catawba	1500	110	0.4
3. McGuire	1600	130	0.5
Reference Sequoyah	1300	95	0.3

Table 6.11a
Expected Consequences at Zion, Power Operation With Recovery

Late Failure Time (After VB)	Consequence (Person-Rem)		
	No Plugging	50% Plugging	100% Plugging
12 hours	1.110+7	1.27+7	1.42+7
24 hours	8.86+6	1.16+7	1.42+7

Table 6.11b
Expected Consequences at a GI-130 Site,
Power Operation With Recovery

Late Failure Time (After VB)	Consequence (Person-Rem)		
	No Plugging	50% Plugging	100% Plugging
12 hours	5.22+6	5.97+6	6.67+6
24 hours	4.16+6	5.45+6	6.67+6

Table 6.12
Consequences of Release Categories Important in Shutdown

Bin Number	PWR Release Category	Consequences (Person-Rem)
6	PWR2	1.7+7
7	PWR2B	5.04+6
14	PWR7	3.63+4
15	No Failure	3.63+4

Table 6.13
Benefits of Proposed Options (Person-Rem)

Option	Best Estimate	Low Limit	High Limit
1. Additional Crosstie	2,635	739	4,951
2. Electrical Dependency	2,349	645	4,467
3. Swing Pump + Intake + El. Conn.	14,324	3,992	27,004
4. Tech. Spec., Crosstie Test, Procedures	4,141	1,150	7,825
5. RCP Seals (Charging or Firewater)	12,870	3,510	24,570
Combination 5 + 4			
5. RCP Seals	12,870	3,510	24,570
4. Tech. Spec., Crosstie Test, Procedures	1,951	553	3,641
Total Combination	14,821	4,063	28,211

Table 6.14
Onsite Consequences

Type		Amount
Occupational Doses:		
- Immediate:	1000 Person-Rem	
- Long Term:	20000 Person-Rem	
Total	21000 Person-Rem	$21M
Replacement Power		$1.8+10 yr*
Cleanup		$1.2+10 yr*
Total Onsite Consequences		$3.0+10 yr*

*These numbers to be multiplied by ΔCDF.

Table 6.15
Cost Offsets for Proposed Improvements (Per Reactor)

Option	Cost Offset ($)
1. Additional Crosstie	513K
2. Electrical Dependency	435K
3. Swing Pump + Intake + El. Conn.	2.75M
4. Tech. Spec., Crosstie Test, Procedures	788K
5. RCP Seals (Charging or Firewater)	2.34M
<u>Combination 5 + 4</u>	
5. RCP Seals	2.34M
4. Tech. Spec., Crosstie Test, Procedures	390K
Total Combination	2.73M

Table 6.16
Costs of Proposed Actions, Per Reactor

#	Options	Direct Cost	Include Indirect Cost	Include O&M Cost	Include Tech. Spec. Cost	Include NRC Cost - Total Cost	Include Onsite Conseq. Offset - Net Cost
1.	Additional Crosstie	$557K	$724K	$1.05M	$1.08M	$1.14M	$627K
2.	Electrical Cross Connection	$50K	$65K	$94K	$128K	$189K	-$246K
3.	Swing Pump + Intake + El. Connection	$29M	$38M	$55M	$55M	$55.1M	$52.3M
4.	Changing Tech. Spec., Procedures, Crosstie Test	$0	$0	$0	$83K	$104K	-$684K
5.	High Pressure Pump for RCP Seals	$5.9M	$7.7M	$11M	$11M	$11.1M	$8.8M
5a.	Firewater for Thermal Barrier Cooling	$200K	$260K	$378K	$412K	$473K	-$1.9M

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

1.0 2.8 2.5 3.2 2.2 3.6 2.0 4.0 1.1 1.8 1.25 1.4 1.6

150mm

6"

Table 6.17
Direct Cost Estimates ($) (Per Reactor)

Option		Best Estimate	Low Limit	High Limit
1.	Additional Crosstie	557M	250K	1M
2.	Electrical Cross Connection	50K	50K	50K
3.	Swing Pump + Intake + Electrical Connection	29M	7M	38M
4.	Changing Tech. Specs., Updating Procedures and Crosstie Testing	0	0	0
5.	High Pressure Pump for RCP Seals	5.9M	1M	15M
5a.	Firewater for Thermal Barrier Cooling	200K	127K	273K

Table 6.18
Uncertainty Range for the Total Cost and the Net Cost ($)

Option		Total Cost		Net Cost	
		Low Limit	High Limit	Low Limit	High Limit
1.	Additional Crosstie	550K	2M	37K	1.5M
2.	Electrical Cross Connection	173K	205K	-262K	-230K
3.	Swing Pump + Intake + Electrical Connection	14M	72M	11M	69M
4.	Updating Tech. Specs., Procedures, Crosstie Testing	48K	171K	-740K	-617K
5.	High Pressure Pump for RCP Seals	2M	29M	1.2M	28.2M
5a.	Firewater for Thermal Barrier Cooling	318K	624K	-2M	-1.7M

Table 6.19
Best Estimate Cost-Benefit Ratios ($/Person-Rem)

Option	Net Cost/Benefit	Total Cost/Benefit
1. Additional Crosstie	238	433
2. Electrical Cross Connection	---	80
3. Swing Pump + Intake + Electrical Connection	3651	3847
4. Updating Tech. Specs., Procedures, Crosstie Testing	---	25
5. High Pressure RCP Seal Injection	684	862
5a. Firewater Connection*	---	37
<u>Combination 5 + 4</u>		
5. High Pressure RCP Seal Injection	684	862
4. Updates in Tech. Specs. and Procedures, Testing	---	53
Total	574	756
<u>Combination 5a + 4</u>		
5a. Firewater Cooling	---	37
4. Updates in Tech. Specs. and Procedures, Testing	---	53
Total	---	39

*Option 5a assumed to have a similar impact on CDF as Option 5.

Table 6.20
Uncertainty Range in Cost-Benefit Ratios ($/Person-Rem)*

Option	Net Cost/Benefit		Total Cost/Benefit	
	Low Limit	High Limit	Low Limit	High Limit
1. Additional Crosstie	7.5	2030	111	2706
2. Electrical Cross Connection	---	---	39	318
3. Swing Pump + Intake + Electrical Connection	415	17335	518	18036
4. Updating Tech. Specs., Procedures, Testing	---	---	6	149
5. High Pressure RCP Seal Injection	49	8034	81	8262
5a. Firewater Cooling of Thermal Barriers	---	---	13	178
Combination 5 + 4				
5. High Pressure Seal Injection	49	8034	81	8262
4. Updates in Tech. Specs. and Procedures, Testing	---	---	13	309
Total	30	6891	73	7187
Combination 5a + 4				
5a. Firewater Cooling	---	---	13	178
4. Updates in Tech. Specs. and Procedures, Testing	---	---	13	309
Total	---	---	13	196

*Uncertainties in ΔCDF values have not been included.

7. SUMMARY

This report summarizes a study performed by Brookhaven National Laboratory for the U.S. NRC in support of the resolution of NRC Generic Issue 130. Generic Issue 130 is concerned with the core damage vulnerability associated with the failure of the ESW system in those multiplant units that have only two ESW pumps per unit backed up by a unit to unit crosstie capability.

The main objective of this study was to establish the core damage vulnerability presented by this type of ESW configuration, to identify potential improvements for the ESW systems and to obtain generic estimates of their risk reduction potential and cost effectiveness.

As part of the overall approach, the specific design arrangements of the multiplant units were surveyed and classified with special emphasis on commonality of design features and applicability of the potential improvements. Based on an extensive survey of operating data, experiences and events were identified and were used to establish a generic initiating event frequency The initiator is based on events which either directly or indirectly (through the failures of support systems, i.e., linked initiator approach) caused the loss of the ESW systems.

A core damage model was constructed starting from a previously developed single unit model. All operating configurations and ESW system arrangements were taken into account and an approximation method was used to generate configuration-dependent initiator as well as core damage frequencies. The loss of ESW initiating frequency for the most dominant operating configuration of the two unit arrangement (both units are at power and the respective ESW trains are in run/standby mode) was established as $\lambda_{ESW}(R/SB, R/SB) = 1.1x10^{-3}$/ryr. Relative time fractions of each operating and shutdown configuration were estimated based on PWR operating data.

The effect of sequence-specific recovery actions, including estimation of ESW system recovery times and possible operator actions were also considered and incorporated into the core damage model. A time- and leak-rate-dependent RCP seal LOCA model was developed based upon NUREG-1150 that takes into account the time dependence of the ESW system recovery. A simplified shutdown risk model was also developed to take into account the specific differences as compared to full power

The calculated core damage frequency includes the contributions from all operating configurations of the multiplants with respect to ESW system arrangements. The CDF due to loss of ESW events was calculated as CDF(ESW) = $1.52*10^{-4}$/ryr.

The most dominant sequence contributing to the total CDF was found to be the RCP seal LOCA sequence, CDF(RCP Seal LOCA) = $8.77x10^{-5}$/ryr or about ~60% of the total.

The relatively significant absolute value of the calculated CDF due to loss-of-ESW events indicates that multiplants with this particular ESW design arrangement may have a significant core damage vulnerability. In order to reduce the risk from the operation of these units improvements may have to be considered. A few of the plants may already have one or more of these features (or their equivalent included in the design.

Based on the identified failure modes, a number of different potential improvements were considered; each targeting one or multiple failure mechanisms. One important option (Option 5) was selected based on accident-sequence-specific considerations to eliminate the most dominant contributor (RCP seal LOCA caused by loss of ESW) to the total CDF. Option 5 had the most significant effect in reducing the risk. The total CDF was reduced by a factor of 2. The option that included the installation of a swing ESW pump with a separate intake canal capable of serving either unit also demonstrated a large risk reduction potential (~60%). The other options, i.e., providing electrical cross connections between existing ESW trains and adding Technical Specification requirements, were each capable of reducing the CDF by ~10 and ~20%, respectively. Combinations of the various options were also considered based on their CDF reduction potential and the respective cost/benefit impact.

The cost/benefit aspects of each mitigating option were also evaluated. Costs were derived from diverse sources including the industry (including two GI-130 sites) and the NRC.

The benefit calculations involved obtaining person-rem estimates from properly selected plant damage states in selected existing PRAs (NUREG-1150 treatment of Zion, Surry and Sequoyah) and adapting the numbers to GI-130 plants. The person-rem values were multiplied by the change in core damage frequency for a particular option and the remaining plant life (assumed to be 30 years) to arrive at the benefit of each option. Changes in onsite consequences of accidents were treated as cost offsets (negative costs), rather than benefits. These include cleanup costs and replacement power costs (both of duration of 10 years).

For an option to be deemed clearly beneficial, the cost-benefit ratio must be well below $1,000 person-rem, per established NRC guidelines. Negative net costs are always beneficial as they reflect averted onsite costs exceeding the costs of installing the option.

Based on the cost/benefit analysis, it seems that most of the analyzed options are beneficial (cost to benefit ratio negative or much less than $1,000/person-rem). Adding a swing pump with its intake structure was not shown to be beneficial in the best estimate calculation. Table 7.1 compares the costs and benefits (in present day dollars) of the various options considered. If an option has a negative net cost, implementation of that option is beneficial and no cost-benefit ratio is shown.

In summary, the contribution of the loss-of-ESW event to the risk from the operation of the multiplants that have two ESW trains per unit with one pump per train backed up by a unit crosstie capability, was calculated to be a dominant accident sequence. The analysis further demonstrated that there are a number of options available to reduce the CDF contribution due to the ESW accident sequences. Option 5, the addition of a dedicated RCP seal cooling system, had the single most significant CDF reduction potential combined with advantageous cost/benefit aspects.

The combination of the various options using the CDF reduction potential and the cost/benefit aspects as the measure of effectiveness resulted in a unique selection of options presented in Table 7.2. The utilization of Options 5 + 4 reduced the total CDF to $6.08*10^{-5}$/ryr . The cost/benefit ratios for Options 5a and 4 are extremely beneficial and significantly below the 1000 $/person-rem guideline. The cost benefit ratios in Table 7.2 use the total cost instead of the net cost (i.e., the onsite cost offset is not taken into account). The net cost is negative for each entry in the table, and no net cost benefit ratio would, therefore, be shown.

Table 7.1
Comparison of Costs and Benefits of Proposed Actions

#	Option	Benefit (Person-Rem Averted)	Net Cost ($)	$/Person-Rem*
1.	Additional Crosstie	2,635	627K	238
2.	Electrical Cross Connection	2,349	-246K	---
3.	Swing Pump + Intake El. Connection	14,324	52M	3,651
4.	Changing Tech. Specs., Procedures, Crosstie Test	4,141	-684K	---
5.	High Pressure Pump for RCP Seal	12,870	8.8M	684
5a.	Firewater for RCP Thermal Barrier Cooling	12,870	-1.9M	---

*Negative net costs imply cost-effectiveness, negative cost-benefit ratios are meaningless.

Table 7.2
Combination of Options 5a+4 Cost/Benefit

Option	Total $/Person-Rem		
	Low	Best Estimate	High
5a. Firewater for RCP Seal	13	37	178
4. Changing Technical Specifications*	13	53	309
Total 5a + 4	13	39	196

*Includes: a. additional TS requirements,
b. testing of the crosstie valves, and
c. procedure improvements.

APPENDIX A

DESCRIPTION OF OPERATING EVENTS INVOLVING THE TOTAL LOSS OF THE ESW SYSTEM FUNCTION

APPENDIX A: Description of Operating Events Involving the Total Loss of the ESW System Function

A.1 Salem 1 - 1976

Numerous problems were experienced with the plant service water system. The first serious problem was noted in January 1976 when a winter storm shut down the system. Icing due to wind whipped spray and screen wash spray created four inches of ice on the operating deck of the structure making it hazardous to operators and caused the traveling screens in operation to ice-over thereby restricting flow to the pumps. Screens which were out of service froze in their tracks causing shear pins to fail when the screen was started. The eventual buildup of ice and debris resulted in the shutting down of the remaining pumps due to low flow. Some modifications were made to the system, however, the major improvement, a heated protective housing, had not yet been installed.

A.2 Farley 1 - 1978

At ~21:00 hours on January 25th, 600V load centers 1H and 1J, which were located in the river water structure, were de-energized when flooding of this structure occurred. The flooding was the result of high Chattahoochee River levels following heavy rains. The water level in the train A side of the river water structure was ~1 ft. The river level at this time was ~110 feet mean seal level (MSL). The river water pumps were still operable. They set up temporary sump pumps to supplement the permanently installed pumps. The Tech Specs required that load centers 1H and 1J be operable, energized, and aligned to an operable DG.

At 2300 hours a 50% reduction in turbine load was initiated. Power to river water pumps 8A, 9A, 10A was racked out at 2330 hours. At 0007 hours on January 26th, the unit was at 40% reactor power and 430 MWe. At 0040 hours a further load reduction was initiated at 5 MW/min to place the unit in hot standby as required by the Tech Specs. At 0045 hours power to river water pumps 4B and 5B was racked out, and the rate of load reduction was increased to 10 MW/min to have the unit in hot standby within the required one hour. At 0055 hours emergency service water recirculation flow to the pond (ultimate heat sink) was initiated. At 0135 hours the unit was taken off line and at 0136 hours the reactor was manually tripped. The water level in the river water structure train A section reached ~5 feet; train B section reached 2 feet. The river reached a maximum level of ~115 feet MSL at the river water structure. (The river water pumps are protected by design from flooding up to 127 ft msl.)

Water had entered the structure through a hole in each river water pump baseplate and through the gland seal leakoff line on each pump. Additional leakage occurred through compression type cable penetrations of structure.

A.3 San Onofre 1 - LER-206/1980-006

During normal operation, the south salt water cooling pump (SCP) discharge pressure dropped sharply. The north salt water cooling pump (NCP) automatically started on low pressure. However, its discharge POV failed to open. The auxiliary salt water cooling pump (ACP) was then started but flow could not be established. As a result of (1) excessive vibration, the shaft of the (SCP) sheared, (2) mechanical failure, the (NCP) POV did not open, and (3) apparent inadequate prime, the (ACP) lost suction. The POV on the (NCP) was manually opened and the (ACP) regained suction.

The following is a short description of the timing of this event.

	Time	Description
March 10	21:15	Discharge Pressure dropped sharply.
	21:20	Both Trains A and B pumps were running with discharge POVs closed. No saltwater flow.
	21:25	Auxiliary saltwater pump started but lost its prime.
	21:30	Started screen wash pumps, established saltwater flow. maximum temperature at CCW HX is ~82°F.
	21:56	Auxiliary saltwater pump prime reestablished, provides saltwater flow.
	22:10	Secured screen wash pumps.
	00:15	Discharge POV opened.

A.4 Palisades - LER-255/1984-001

On January 8, 1984, the Palisades Nuclear Plant experiences a complete loss of all normal communications links between the plant, the NRC and state/local authorities. The event was precipitated by the need to isolate a faulty switchyard breaker. To accomplish the isolation, it was necessary to interrupt the offsite power supply to the plant. At the time of the event, Palisades was in a refueling outage with all fuel removed from the reactor and one diesel generator inoperable. While operating procedures require two operable diesel generators prior to removing offsite power, the shift supervisor proceeded with the evolution after determining the safety of the fuel would not be jeopardized. In preparing for the evolution, the operators failed to realize that there would be no operable service water pumps supplied by the operating diesel. Consequently, after 50 minutes the diesel overheated due to lack of cooling water and was manually tripped. The resulting loss of onsite ac power caused a loss of all plant telephones and radios for 45 minutes. Onsite power was subsequently re-energized from the switchyard, resulting in the restoration of normal communications.

A.5 Oconee 2 - LER-269/1986-011

On October 1, 1986, with Units 1 and 3 at 100% full power, and Unit 2 shutdown for refueling, a load shed test on Unit 2 was performed. Suction to the low pressure service water (LP) pump was lost about one hour into the test. The loss prime in the condenser circulating water (CCW) siphon flow (or emergency CCW) system was the cause for the loss of the LP pumps. The emergency condenser circulating cooling water (ECCW) system is required to provide water through the main condenser for decay heat removal during loss of all ac power event (station blackout). The immediate corrective action was to analyze the failures that occurred during the load shed test, and shut down Oconee Units 1 and 3. Subsequent corrective actions included redesign of the CCW pump flanges and determination of the design basis of the ECCW system. The root cause of this event is the inadequate design and testing of the ECCW system. This led to a failure of the ECCW system to perform the intended function as described in the final safety analysis report (FSAR).

A.6 Salem 1 - LER-272/1984-014

On June 5, 1984, during a refueling outage, 1A vital bus was de-energized when the 1A vital bus infeed breaker failed to close during breaker testing. Since 1B vital bus was de-energized for inspection at the time, a blackout loading signal started 1A and 1C diesels and opened the 1C vital bus infeed breaker, de-energizing 1C vital bus. 1A diesel loaded, but because the 1C 125V dc bus was de-energized for maintenance, the 1C safeguards equipment cabinet (SEC) was completely de-energized. This prevented 1C diesel from loading. 1C vital bus remained de-energized, resulting in a loss of service water cooling. Numerous control room indicators failed to mid-scale, leading the shift to believe that the 1C vital bus was still energized. As a result, the diesels ran for an extended period of time without cooling water; although, no diesel damage occurred. The root cause of this event was the lack of adequate procedural and/or administrative controls to ensure sufficient electrical systems remained in an operable status during a period when the plant was in a configuration which was not covered by the Tech Specs (i.e., defueled).

A.7 Salem - LER-272/1982-015

Number 1A vital bus tripped resulting in a loss of component cooling water (CCW) and service water (ESW) flows; the redundant CCW and pumps were tagged out for maintenance. All charging pumps, boron injection flow paths, residual heat removal (RHR) loops and diesel generators were declared inoperable due to no CCW or flow. A wire to the TD5 undervoltage relay had shorted to the feeder cubicle door, causing the 1A vital bus infeed breaker to trip without automatic transfer. CCW and flows were restored.

A.8 Crystal River - LER-302/1986-002

On January 10, 1986, Crystal River Unit 3 was in mode 5 during an outage. The intake structure was being cleaned and inspected by two contract divers. At 16:15, one diver failed to reappear following his dive. The second diver attempted to locate and rescue the missing diver but was himself drowned. When the second diver was reported to be in trouble, all seawater pumps taking suction at the intake structure were secured, thus disabling both trains of the decay heat removal system. The body of the second diver was recovered shortly thereafter. The first diver was found to have been drawn into the 48" suction line of the 'A' emergency nuclear services and decay heat seawater system pumps (both pumps were running at the start of the event). All seawater pumps were voluntarily secured and/or disabled in an attempt to prevent loss of life.

A.9 Calvert Cliffs 2 - LER-318/1982-054

At 05:47, during normal shutdown operation in mode 6, power was lost to 24 4kV bus resulting in the loss of 22 saltwater pumps and 22 LPSI pumps, thereby disabling the only operable shutdown cooling loop. Power was restored to 24 4kV bus and shutdown cooling flow restored at 06:05. The redundant shutdown cooling loop was out of service for maintenance. Vendor failure report indicated the cause of the power supply failure to be cracked printed circuit board.

A.10 San Onofre 2 & 3 - LER-361/1983-072, LER-362/1983-041

On July 6, 1983 at 00:30 while Unit 2 was in mode 5 and Unit 3 was in mode 4 operator observed that the Unit 3 circulating water system traveling screen water level differential pressure was off scale indicating clogging of the screens. The screen wash system was actuated to clear the screens of marine debris. The screen wash system failed to clear the screen. The inability to clear the screens resulted in high CCW heat exchangers (Unit 2 train A and Unit 3 trains A and B) differential pressure being alarmed in the control room at 02:10 on July 6, 1983, and at 02:27 SCW flow was reduced to the point that the heat exchangers were declared inoperable. This resulted in exceeding limiting condition for operation (LCO) 3.7.4 for Unit 3, only since the LCO is applicable to modes 1 through 4 and Unit 2 was in mode 5. Exceeding LCO 3.7.4, for Unit 3, resulted in invocation of LCO 3.0.3. Visual inspection of the traveling screens after the incident revealed that several screen panels were dislodged from their housings either before or during this event resulting in marine debris to be carried into the circulating water pump forebay. To preclude concurrent fouling of both trains of CCW heat exchangers during excessive marine debris buildup in a single intake structure, C system operating procedure is being revised.

A.11 Catawba 1 - LER-413/1985-068

On November 25, 1985, the in service test on the nuclear service water (RN) header 1B supply isolation valve was performed. While stroking the valve, it stopped in the intermediate position. Train B of RN was declared inoperable and train A of RN was placed in service. Upon starting RN pump 1A, the discharge isolation valve also stopped in the intermediate position. Train A of RN was declared inoperable and Technical Specification 3.0.3 was entered due to the simultaneous inoperability of both trains of RN. Both trains of RN were inoperable for 43 minutes until the RN header 1B supply isolation valve was opened and train B of RN was declared operable. Investigation revealed that the torque switches for the valves were set at the low end of the allowable tolerance. These setting did not allow the valves to open completely. Therefore, this incident is classified as a design deficiency. Unit 1 was at 45% power.

A.12 Vogtle 1

On March 20, 1990, the Vogtle Electric Generating Plant Unit 1, located in Burke County, Georgia, about 25 miles southeast of Augusta, experienced a loss of all safety (vital) ac power. The plant was in cold shutdown with reactor coolant level lowered to "mid-loop" for various maintenance tasks. Both the containment building personnel hatch and equipment hatch were open. One emergency diesel generator and one reserve auxiliary transformer were out of service for maintenance, with the remaining reserve auxiliary transformer supplying both Unit 1 safety buses. A truck in the low voltage switchyard backed into the support column for an offsite power feed to the reserve auxiliary transformer which was supplying safety power. The insulator broke, a phase-to-ground fault occurred, and the feeder circuit breakers for the safety buses opened. The operable emergency diesel generator started automatically because of the undervoltage condition on the safety bus, but tripped off after about one minute. About 20 minutes later the diesel generator load sequencer was reset, causing the diesel generator to start a second time. The diesel generator operated for about one minute, and tripped off. The diesel generator was restarted in the manual emergency mode 36 minutes after the loss of power. The generator remained on line and provided power to its safety bus. During the 36 minutes without safety bus power, the reactor coolant system temperature rose from about 90°F to 136°F.

APPENDIX B

OPERATING EVENTS INVOLVING THE DEGRADATION OF THE ESSENTIAL SERVICE WATER SYSTEM FUNCTION

APPENDIX B: Operating Events Involving the Degradation of the Essential Service Water System Function

Plant	Reference	Description
San Onofre 1	LER-206/80-01	Pipe support installation error in SW system.
	/80-08	Pipe support corroded on one SW pump.
	/80-31	Discharge valve on pump failed to open automatically.
	/81-09	HX partially blocked, marine growth.
	/82-07	Pressure switch failed, pump discharge valve closed.
	/82-15	Intake structure flooded to dangerous levels, inadequate maintenance procedures.
	/82-22	One pump bearing degraded, other pump out for maintenance, auxiliary SW pump put in service.
	/82-24	Discharge valve opens, reverse flow through pump resulting in damage.
	/84-08	Corrosion of the intake structure.
Haddam Neck	LER-213/83-01	SW leak in fan cooler due to corrosion.
	/83-10	SW filter plugged.
	/86-09	SW flood protectors are ineffective.
Ginna	LER-244/83-01	SW valve failed to open to AFW pump.
Indian Point 2	LER-247/80-16	SW leak in fan cooler coils.
	/81-09	SW pipe wall thinning.
	/81-10	Valve seat problem, reduces pump capacity.
	/81-11	Pipe wall thinning, corrosion.
	/81-21	SW pipe leak.
	/82-13	SW pump vibration excessive.
	/82-26	Impeller wear of three SW pumps.
	/82-31	SW leak in containment.
	/82-33	SW leak in fan coolers.
	/82-37	SW leak in fan coolers.
	/83-07	Strainer plugged.
	/83-10	Pump inoperable, rope tied the impeller.
	/84-11	Leak into the CCW pump.
	/84-21	SW pump discharge valves leak.
	/85-13	SW leak in fan coolers.
	/87-11	SW pumps fail performance tests, vortexing.
Turkey Point 3	LER-250/86-08	SW system design deficiency.
	/86-18	SW system design deficiency.
	/86-24	SW pump inoperable.

Plant	Reference	Description
Turkey Point 4	LER-251/84-18	Strainer removed for longer period as allowed.
	/87-16	SW pump tripped, electrical problems.
	/87-28	Two of three SW pumps are inoperable.
Palisades	LER-255/82-24	SW design problem.
	/86-24	Loss of coolers, SW valve problems.
	/86-36	SW pumps performs below requirements.
H. B. Robinson 2	LER-261/81-19	SW booster pump tripped, bearing and breaker problems.
	/82-13	SW pump failed to restart, blown fuse.
	/83-03	Leak in the CF cooler.
	/83-05	Two of four SW booster pumps lost.
	/83-06	SW pump and its replacement fails, longer in AOT than allowed.
	/83-14	SW leak at CF cooler.
	/83-22	SW leak at CF cooler.
	/83-27	SW leak at CF cooler.
Oconee 1	LER-269/80-02	HPSW inoperable, motor insulation broke down.
	/80-04	HPSW inoperable, motor cooler leakage.
	/80-24	Automatic initiation of HPSW was affected by construction.
	/80-30	Valve failed to close in SW system.
	/81-14	HPSW pumps A and B had no control powers, breakers were open, jockey pump used in place.
	/86-02	Seismic design deficiency in LPSW system.
	/87-04	SW heat exchanger capacity reduced, biological fouling.
Oconee 2	LER-270/80-10	SW valves fail in closed position.
	/81-01	Improper alignment of SW valves.
Salem 1	LER-272/80-22	SW solenoid valve failure isolates CFCU coil.
	/80-23	SW flow reduced to CFCU, inoperable flow transmitter.
	/80-24	Solenoid on SW line failed, no flow to CFCU.
	/80-39	Solenoid on SW line failed, no flow to CFCU.
	/80-49	SW piping leak at charging pump.
	/80-60	SW valve mispositioned, all DG inoperable.
	/81-03	SW pipe leak, CCW HX removed from service.
	/81-10	SW pipe leak, CF coil inoperable.

Plant	Reference	Description
	/81-11	SW pipe leak, CF coil inoperable.
	/81-12	SW hcse leak.
	/81-31	SW pipe leak, CFCU inoperable.
	/81-39	SW pipe leak, CFCU inoperable.
	/81-46	SW valve failure blocks flow to CFCU.
	/81-64	SW pipe leak at CFCU.
	/81-67	SW valve failure reduces flow to CFCU.
	/81-69	SW valve failure reduces flow to CCW HX.
	/81-71	SW flow XMTR line plugged.
	/81-76	SW pipe leak at CFCU.
	/81-77	SW pipe leak at CFCU.
	/81-80	SW pipe leak at CFCU.
	/81-83	SW pipe leak, charging pump operation affected.
	/81-90	SW pipe leak, CCW HX.
	/81-94	SW pipe leak, CFCU.
	/81-96	SW pipe leak, CFCU.
	/81-114	SW pipe leak, CFCU.
	/81-119	SW pipe leak, charging pump.
	/81-121	SW pipe leak, CFCU.
	/82-18	SW valve leaks in containment.
	/32-22	SW flow control valve fails, reduces flow to CFCU.
	/82-24	SW flow control valve fails, reduces flow to CFCU.
	/82-29	SW flow control valve fails, reduces flow to CFCU.
	/82-37	SW flow control valve fails, reduces flow to CFCU.
	/82-41	SW pipe leak, charging pump affected.
	/82-69	SW pipe leak, charging pump affected.
	/82-91	SW leak, CCW HX.
	/83-15	SW valve malfunction, DG inoperable.
	/83-26	SW valve plugged, CFCU inoperable.
	/83-68	SW line freezes - fire OG inoperable.
	/84-06	SW line leak, CFCU.
	/84-08	SW pipe corrosion near CCW HX.
	/84-27	SW pipe leak at CFCU.
	/85-06	SW pipe leak at CFCU.
	/85-08	SW pipe leak at CFCU.
	/86-14	SW valve to turbine lube oil fails, reactor trip.
Surry 1	LER-280/80-54	SW MOV failed to cycle.
	/80-65	SW MOV failed due to marine growth.
	/82-100	Loss of one SW pump due to personnel error.
	/82-124	SW inlet valve to RS HX was inadvertently closed, loss of one train.

Plant	Reference	Description
	/83-42	SW strainer clogged.
	/86-24	SW lines in chillers are clogged.
	/86-30	SW lines clogged, marine growth.
	/86-31	SW strainer malfunction, personnel error.
	/86-34	SW strainer clogged.
	/87-02	SW valve malfunction, chiller affected.
	/87-03	SW valve malfunction, chiller affected.
	/87-05	SW strainer malfunction, chiller affected.
	/87-06	SW low flow to chiller, electrical trouble.
	/87-07	SW leak at chiller.
	/87-08	SW valve malfunction affecting chiller.
	/87-18	SW strainer clogged.
	/87-21	SW strainer clogged.
	/88-07	SW flow problems (manual control).
Surry 2	LER-281/80-28	Check disk missing in SW subsystem.
	/80-37	SW strainer clogged, charging pump affected.
	/81-21	SW strainer clogged, charging pump affected.
	/81-34	SW strainer clogged, charging pump affected.
	/81-47	SW MOV breaker open at CCW HX.
	/81-51	SW MOV failed to close.
	/81-73	SW MOV malfunction.
	/81-76	SW MOV malfunction.
	/82-02	SW check valve failed on booster pump discharge.
	/82-09	SW valve failure, flow obstructed.
	/82-39	SW MOV flooded.
	/82-45	SW MOV breaker open - CCW HX.
	/82-49	SW strainer leaking.
	/82-50	SW strainer clogged, booster pump lost.
	/82-52	SW flow indicator fails, reduces flow.
	/82-54	SW MOV flooded.
	/83-25	SW MOV malfunction.
	/83-26	SW MOV breaker open.
	/83-50	SW strainer clogged.
	/85-02	Improper alignment of SW flow to HX.
	/86-06	SW leak in containment spray HX.
Prairie Island 1	LER-282/83-18	Intake device fails, some SW pumps tripped.
	/83-21	SW isolation MOV failed.
	/85-03	SW valve inadvertently closed.
	/85-16	SW leak at CFCU.
	/87-07	SW booster pump fails due to deposition.
	/87-08	SW booster pump air bound, procedural error.

Plant	Reference	Description
Fort Calhoun 1	LER-285/81-05	Relay problems in the starting circuit.
	/87-01	Three SW pumps are unavailable for two hours.
Indian Point 3	LER-286/81-04	SW supply to non-essential HDR lost, both supply pumps are in maintenance.
	/83-06	Seismic restraining plates removed, possible failure during a DBA.
	/87-07	Pipe snubbers failed.
Oconee 3	LER-287/81-10	SW valve air line break.
	/8[illegible]-08	SW valve failed, CFCU affected.
TMI-1	LER-289/80-15	SW RTD failed.
Zion 1	LER-295/80-18	SW valve failed, AFW pump affected.
	/80-24	SW pump failed to start, electrical switch problem.
	/81-07	SW pipe section made to non-safety specifications.
	/83-32	SW MOV failure.
	/84-04	SW leak at CFCU.
	/85-39	SW crosstie valve between two units cycled, loss of SW, standby pump started.
	/86-01	SW valves inadvertently closed, isolates AFW for three weeks.
Crystal River 3	LER-302/84-11	SW pump discharge check valve stuck open.
	/85-24	Cracked pipe support pedestal at CCW HX.
	/85-35	Design deficiency, fire may affect various pumps.
	/87-20	Design discrepancy in SW system temperatures.
Zion 2	LER-304/80-17	SW pump disabled due to electrical fault at dc bus.
	/80-30	SW MOV failure.
	/81-14	SW MOV failure.
	/81-17	SW valve inoperable, loss of initiating signal.
	/81-36	SW valve inoperable, loss of initiating signal.
	/82-09	SW valves fail, silt deposition.
	/83-29	SW valves fail, electrical problems.
	/83-40	SW valves fail, electrical problems.
	/83-45	SW leak at CFCU.
	/84-13	SW leak, tube degradation.

Plant	Reference	Description
	/85-04	Control valve malfunction.
Kewaunee	LER-305/80-35	SW pump fail to start.
	/81-01	SW valve failure, CFCU inoperable.
	/81-07	SW pump failed to start.
	/82-05	SW pump failed to start.
	/82-33	SW MOV failed to open.
	/83-05	SW MOV malfunction.
	/83-21	SW pump unavailable, strainer tested.
	/83-24	SW pump failed.
	/83-25	Flow indicator failed, SW pump unavailable.
	/83-27	Pipe leak due to corrosion at CCW HX.
	/83-37	SW strainer leaked, SW pump unavailable.
	/84-18	Silt deposition in CFCU coils reduces flow.
	/86-15	SW valve failed in closed position.
Prairie Island	LER-306/80-32	Intake area isolated for one unit, causing a loss of SW pump on the other unit.
Maine Yankee	LER-309/81-07	SW cooling to SCC interrupted due to overload.
	/83-15	SW pump tripped, redundancy reduced.
	/83-17	SW MOV failed to operate.
	/83-33	SW MOV failed to operate.
Salem 2	LER-311/81-04	SW pipe leak at CFCU.
	/81-10	SW pump failed, another in maintenance.
	/81-38	SW leak at CFCU.
	/81-64	SW leak at CFCU.
	/81-90	SW leak at CFCU.
	/81-94	SW leak at CFCU.
	/81-99	Instrument line clogged with silt, valve inoperable.
	/81-114	SW leak at CFCU.
	/81-115	SW leak at CFCU.
	/81-117	Line clogged with silt, limit SW failure on MOV.
	/81-118	SW leak.
	/82-06	Valve stuck closed at CFCU.
	/82-17	Valve stuck closed, line clogged with silt.
	/82-28	SW leak at CFCU.
	/82-35	SW valve inoperable.
	/82-39	SW leak at CFCU.
	/82-40	SW leak at CFCU.
	/82-41	Marine growth reduces flow to/88-08Emergency SW pump unavailable due to test.

Plant	Reference	Description
Catawba 1	LER-413/85-04	Loss of SW to RCP motor, improper airline design.
	/85-26	Loss of suction to SW pumps, incorrect valve operation.
	/85-32	SW intake aligned to standby source, personnel error.
	/86-24	Misalignment of SW intake.
	/86-27	Misalignment of SW intake.
	/86-53	Misalignment of SW intake.
	/86-57	SW MOV torque switches improperly set.
	/87-08	Tornado missile cover missing on SW pipe manways.
	/87-35	Incorrect procedures could prevent SW train operation.
	/87-36	Incorrect crossover supply alignment.
Millstone 3	LER-423/86-56	No flow to SI HX, valve closed.
	/87-01	SW low pressure causes turbine/reactor trip.
Vogtle 1	LER-424/87-03	Incorrect sealant used in penetrations.
Seabrook	LER-443/87-25	Incorrect test monitoring for Sw pump vibrations.
Byron 1	LER-454/86-31	Both SW strainers improperly tested.
Byron 2	LER-455/87-03	SW makeup pumps out of service.
Braidwood 1	LER-456/87-16	Incomplete test of SW systems.
Wolf Creek	LER-482/85-12	SW MOV didn't close properly.
	/85-69	Travelling screens collapse due to plant growth.
	/86-44	SW valve failed to operate.
Callaway 1	LER-483/87-24	SW valve not tested as required.
South Texas 1	LER-498/87-03	SW pump tripped, discharge check valve stuck closed.
	/87-18	SW pipe leak, one train inoperable.
	/88-20	Screen wash booster pump inoperable.
	/88-23	Test on screen wash booster pump performed not as frequently as required.
Palo Verde 1	LER-528/86-14	SW pump failed to start, faulty relays.
	/86-37	SW pump failed to start.

APPENDIX C

SIMPLIFIED HEATUP ANALYSIS OF THE COMPONENT COOLING WATER (CCW) SYSTEM

APPENDIX C: Simplified Heatup Analysis of the Component Cooling Water (CCW) System

The RCP seals are generally cooled indirectly using the CCW system, which transfers the removed heat to the final heat sink through the ESW system. Therefore, the loss of ESW function essentially disables the heat removal function of the CCW. The heat capacity of the CCW system may provide additional time to cool the RCP seals and this will be further investigated below.

The CCW system is a closed cooling loop where a number of pumps circulate cooling water through a variety of heat exchangers that constitute the system heat load. The heat is removed through the CCW/SW heat exchanger completing the loop and heat transfer process. The CCW pumps are generally self-cooled with temperature alarms for increasing temperature. The steady state heat removal rate through the ESW/CCW heat exchanger is

$$q_{REM} = W * \Delta T_{Normal} * C_p$$

where "W" is the CCW flow rate through the heat exchanger (lb/hr), ΔT_{Normal} is the CCW temperature drop and C_p is the specific heat of the cooling water. Given the loss of ESW function, this amount of heat must be initially absorbed by the CCW system. Naturally q represents a certain heat load present, which may be reduced y shedding the various heat loads as conditions may allow.

The heat absorbed in the CCW volume will raise its temperature

$$q_{ABS} = \frac{V_{CCW}}{t} * C_p * \Delta T_{Accident}$$

where V_{CCW} is the water volume, t is the respective time (one hour), and $\Delta T_{Accident}$ is the temperature increase of the CCW system after the initiating of the accident.

By equating the removal and absorbent heat rate

$$W \Delta T_{Normal} C_p = \frac{V_{CCW}}{t} C_p \Delta T_{Accident},$$

the temperature risk per unit time is

$$\frac{\Delta T_{Accident}}{t} = \frac{W \Delta T_{Normal}}{V_{CCW}}$$

for a given heat load configuration. In order to establish the approximate value of the temperature rise, the following typical values were used in the calculations:

$W = 3x10^6$ lb/hr

$\Delta T_{Normal} = 10°F$

$V_{CCW} \approx 30,000$ gallon.

These values are representative, but fairly good approximations to the plants under considerations. With these values the temperature rise is

$$\frac{\Delta T_{Accident}}{t} \approx 120°F/hr,$$

which is substantial considering that the CCW normal operating temperature is ~110°F. The above approximate calculation assumed that the total heat load on the CCW system is

$$q_{Heat\ Load} \approx 30x10^6\ Btu/hr.$$

Normalizing the temperature rise ($\Delta T_{Acc/t}$) to unit heat load would give

$$\frac{\Delta T_{Accident}}{t^*q_{Heat\ Load}} \approx 4°F/Btu$$

The expected heat load from the RCP pumps are ~$1x10^6$Btu/hr per RCP and the self-cooling of the CCW pumps may contribute about ~$1x10^6$Btu/hr. Therefore, with the minimum heat load of about $5x10^6$Btu/hr the temperature rise is

Minimum Heat Load - $\Delta T_{Accident} \approx 20°F/hr$

and

Normal Heat Load - $\Delta T_{Accident} \approx 120°F/hr$.

This seems to indicate that upon loss of ESW, without the shedding of heat loads, the CCW system heats up rapidly requiring the operator to turn the pumps off. Since the system is not designed for natural circulation, one can neglect this process to cool the RCP seals. However, an additional heat transfer mechanism must be considered in connection with the spent fuel pool. The spent fuel is stored in a large capacity pool and the generated decay heat is removed by a spent fuel pool recirculation cooling system.

The heat is transferred over to the CCW system via a heat exchanger. If a loss of ESW accident occurs, it is expected, based on the above analysis that the CCW system temperature would rapidly rise to the temperature level of the spent fuel pool. After reaching this temperature, the heat load of the CCW system would be transferred back to the spent fuel pool (lack of ESW heat exchanger) raising its temperature.

The average operating temperature of the spent fuel pool is between ~120-140°F depending on fuel loads and heat removal design. The average heat load being removed from the pool is about $15\text{-}25x10^6$ Btu/hr constituting the major heat load component on the CCW system. Design calculations indicate that without any heat removal the pool would start boiling after ~10-15 hours.

Given the loss of ESW system, the heat removal would be stopped and in addition, the CCW system would start transferring heat to the pool after reaching its temperature.

Assuming that the additional CCW heat loads are about the same magnitude as the spent fuel decay heat, the time to boiling would be reduced by about a factor of ~2 to ~5-7 hours. This indicates that the limiting factor is not pool heat up but rather the lack of cooling to the various equipment required for this particular heat transfer process.

Specifically, the above described scenario requires the continuous operation of the CCW and spent fuel pool cooling pump. Each of these equipment require some kind of cooling (bearings and motor). The limiting factors or maximum temperatures are:

a. RCP seals/motor bearing - ~180°F
b. CCW pump bearing/motors - ~120-140°F
c. Spent fuel pool pump/motor - ~140°F

Based on these considerations the heat up history of the CCW system may occur as follows:

a. 0-15 minutes: Upon loss of ESW the CCW system temperature rapidly increases to the spent fuel pool temperature. Initial $T_{CCW} \approx 110°F$ with ~1°F/minute rise would reach 130°F in about ~15-20 minutes.

b. 15-30 minutes: The rate of temperature rise would considerably slow down, since the spent fuel pool would act as a heat sink. However, the additional increase would alarm the CCW/spent fuel pool pumps due to high temperature. The operator would most likely terminate the operation of these pumps.

c. 30-45 minutes: RCP motor bearing/seal temperature reaches ~180°F and at that point the RCPs are shutoff and the reactor is scrammed.

In summary, the heat capacity of the CCW system may provide additional time before the loss of cooling to the RCP seals after the loss of ESW. However, the available time is judged to be relatively short depending on plant-specific features and proper operator action. It is considered to be in the order of ~30 minutes with optimum conditions.

APPENDIX D

FAULT TREE MODEL OF A MULTI-UNIT ESW SYSTEM

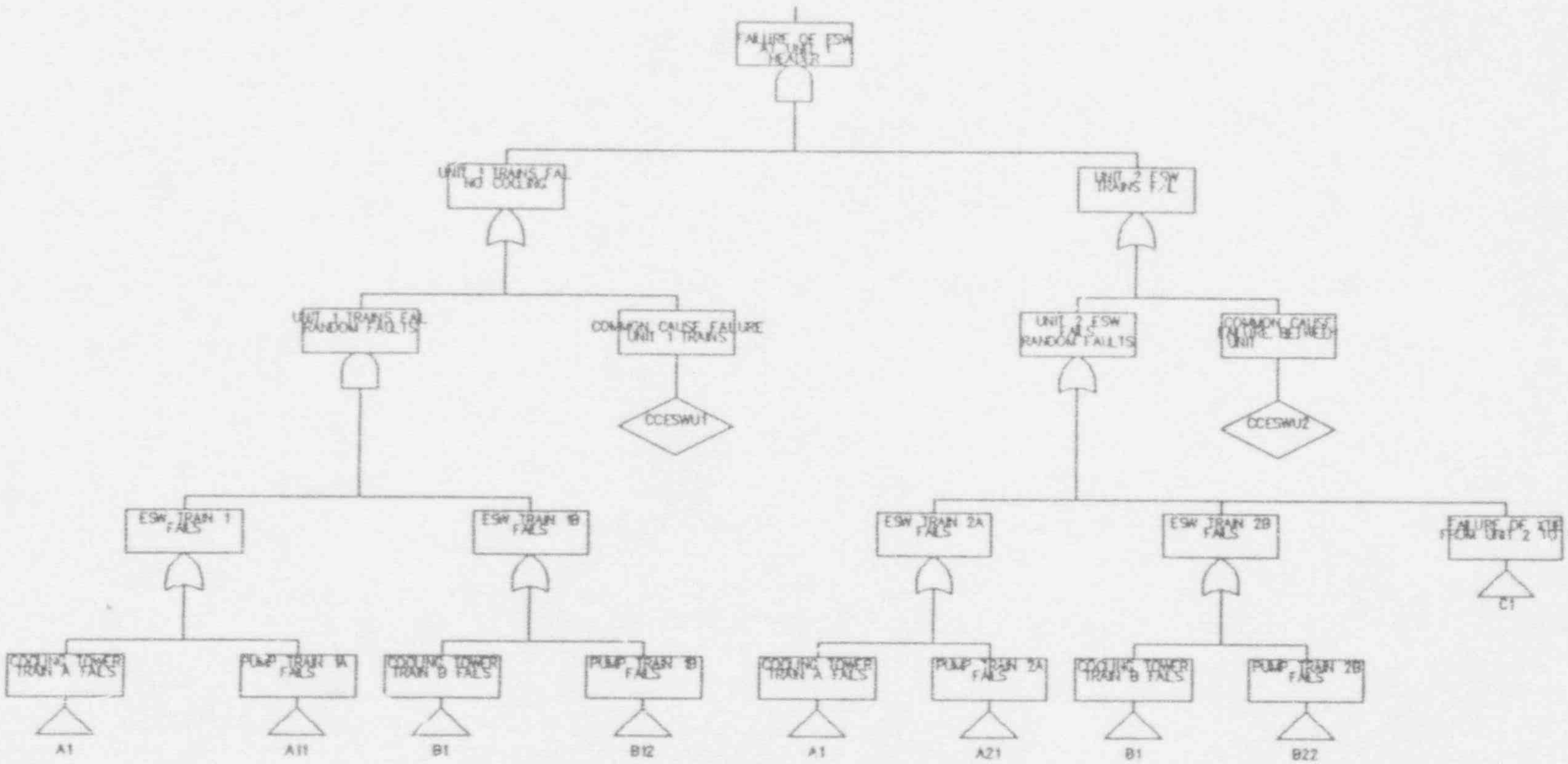


Figure D1. Fault tree model of a multi-unit ESW system.

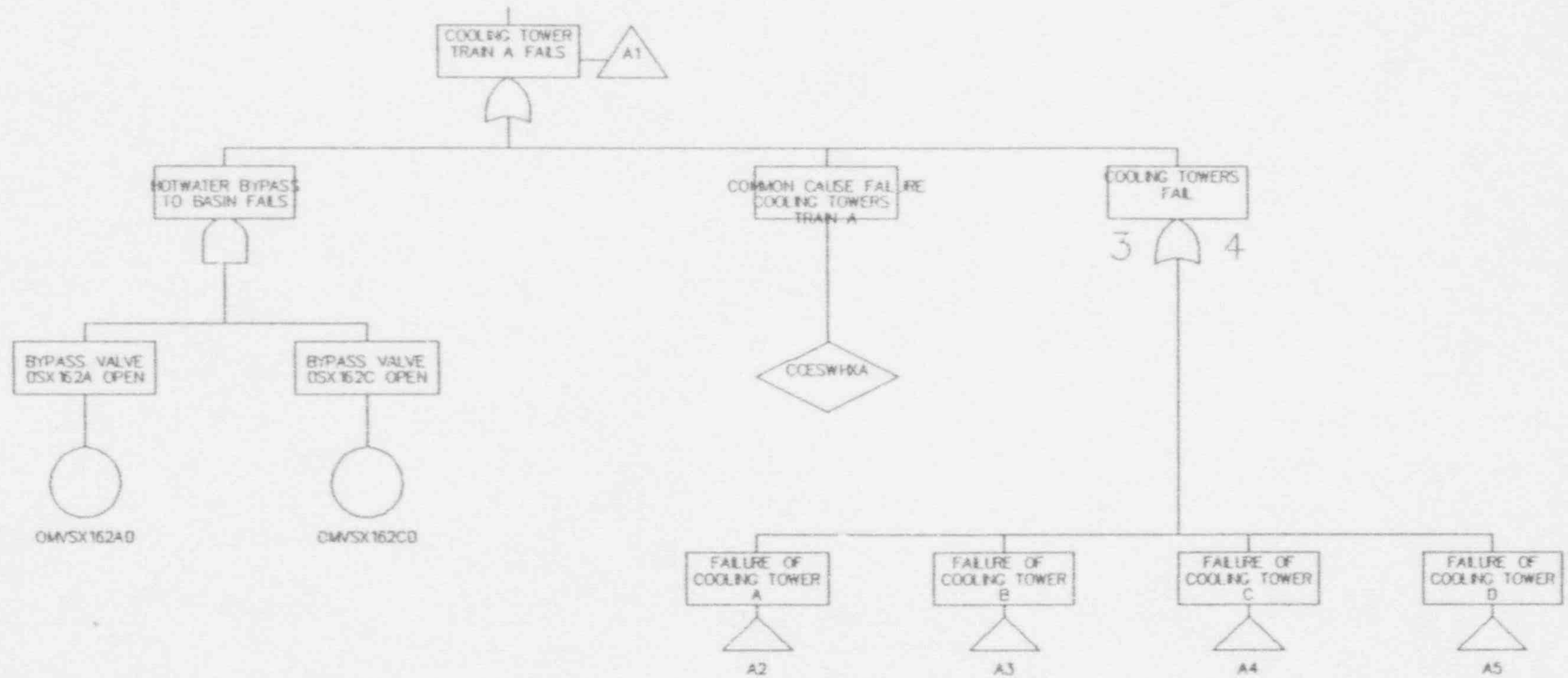
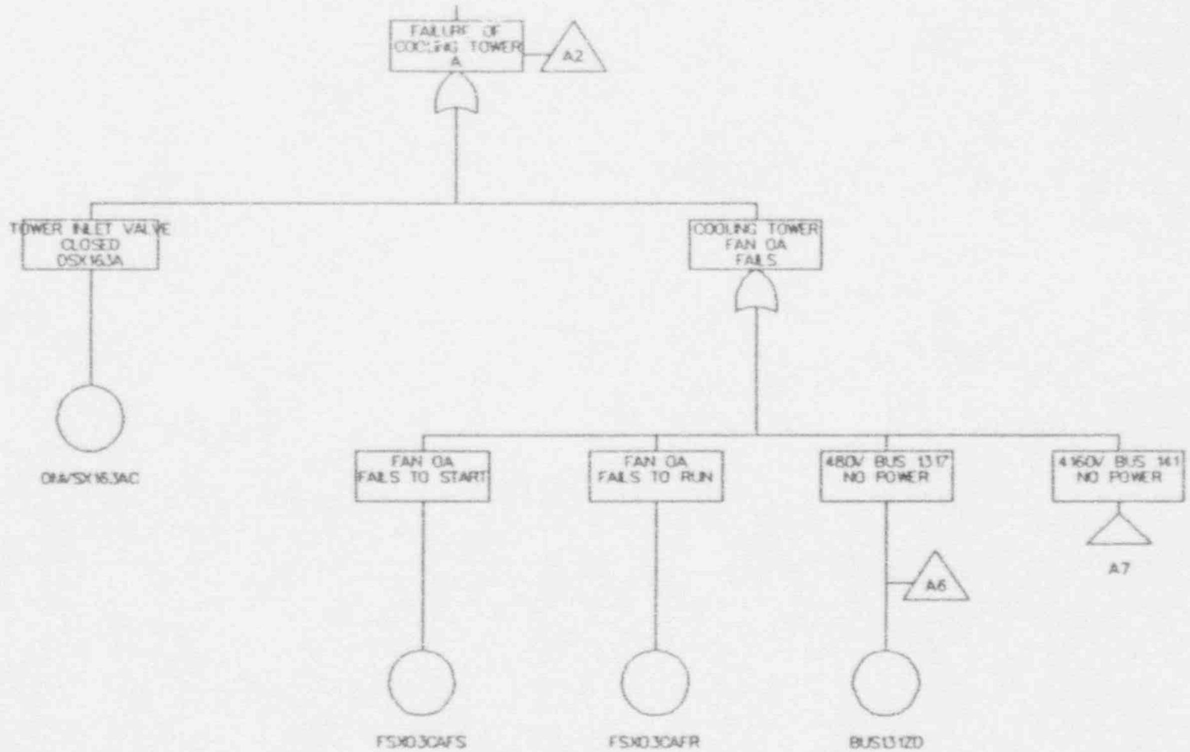


Figure D1. (Continued)



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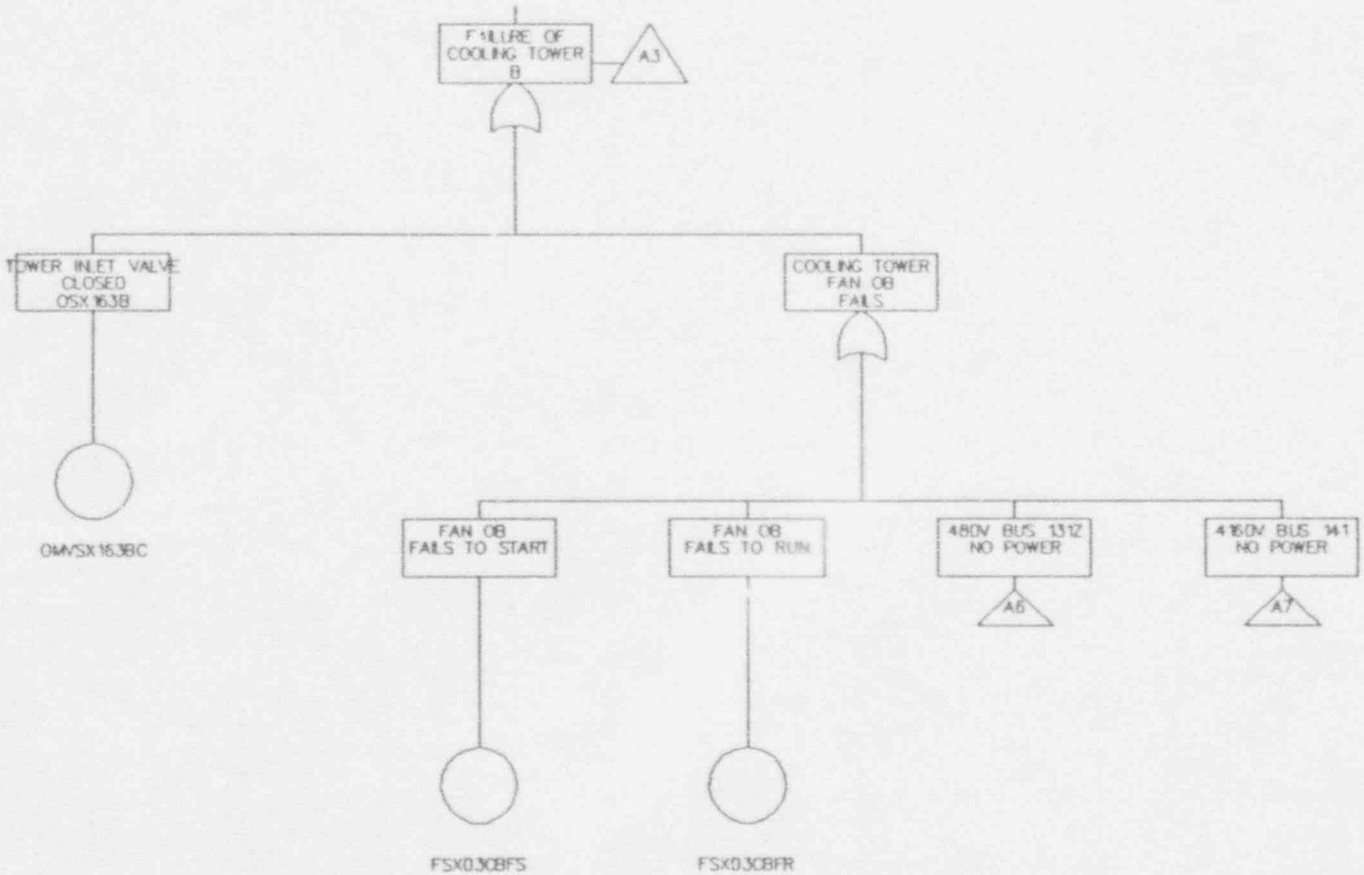
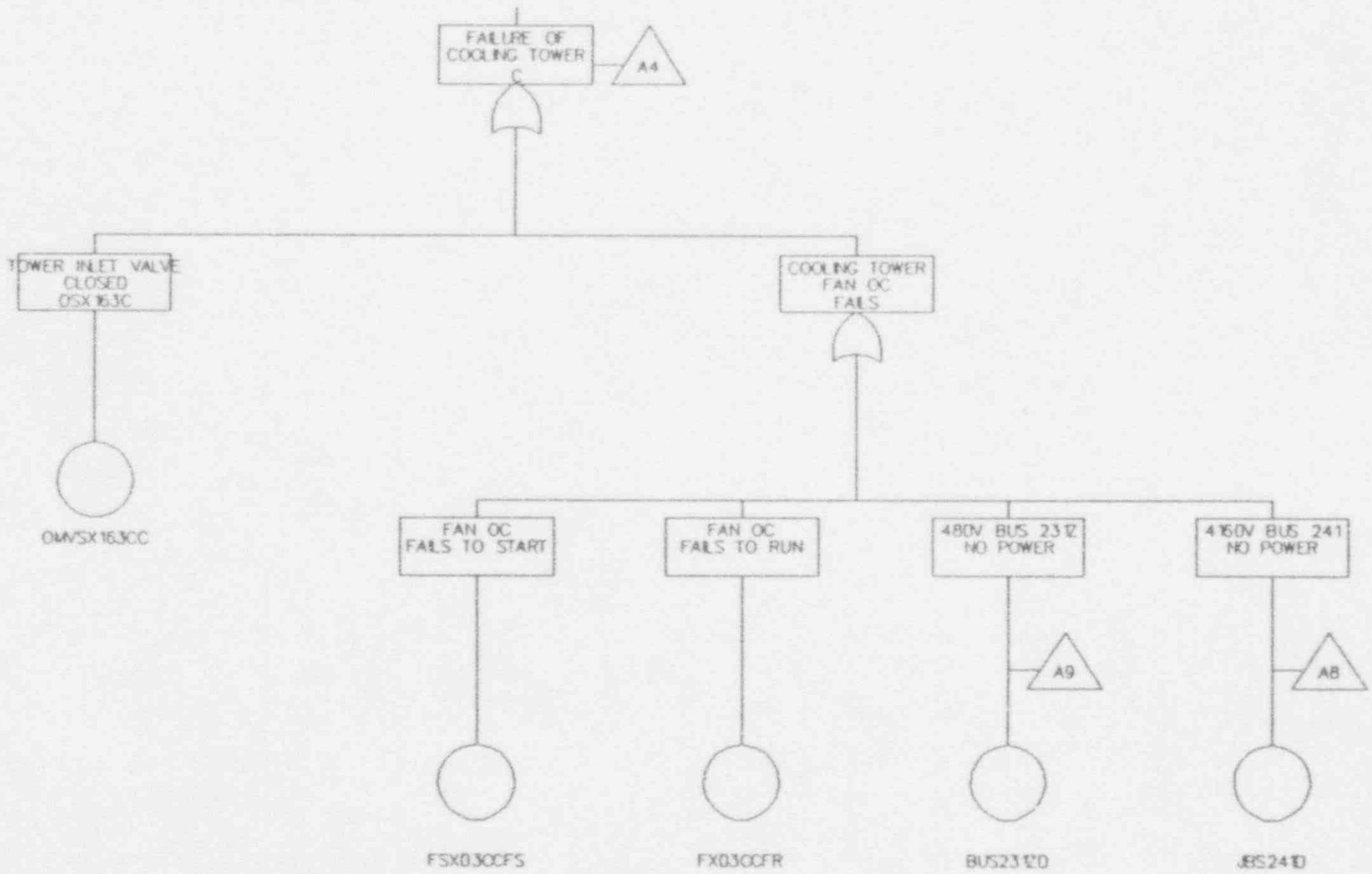


Figure D1. (Continued)


FAILURE OF COOLING TOWER C
A4
TOWER INLET VALVE CLOSED 0SX163C
OMVSX163CC
COOLING TOWER FAN OC FAILS
FAN OC FAILS TO START
FSX030CFS
FAN OC FAILS TO RUN
FX030CFR
480V BUS 2312 NO POWER
A9
BUS2312D
4160V BUS 241 NO POWER
A8
JBS241D

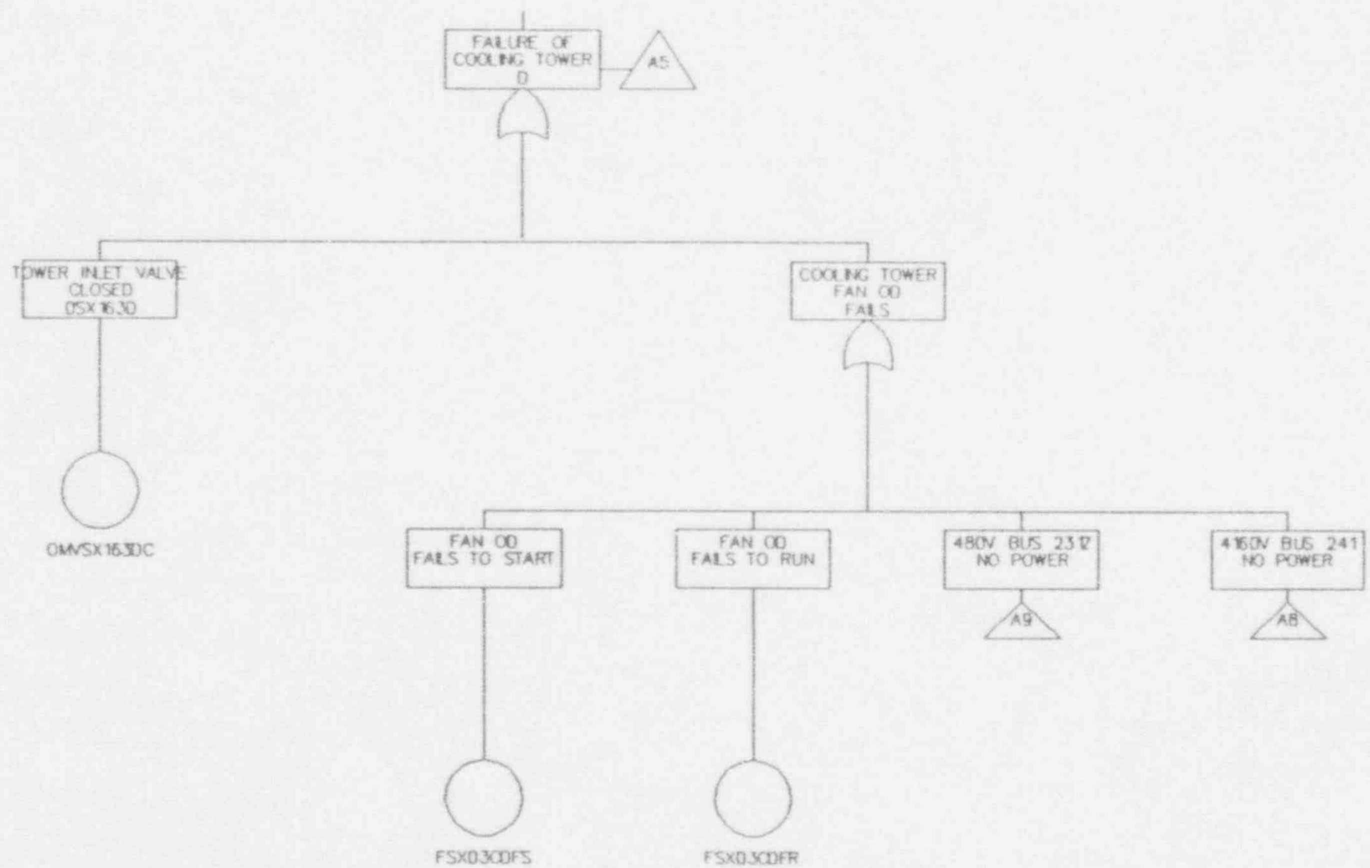
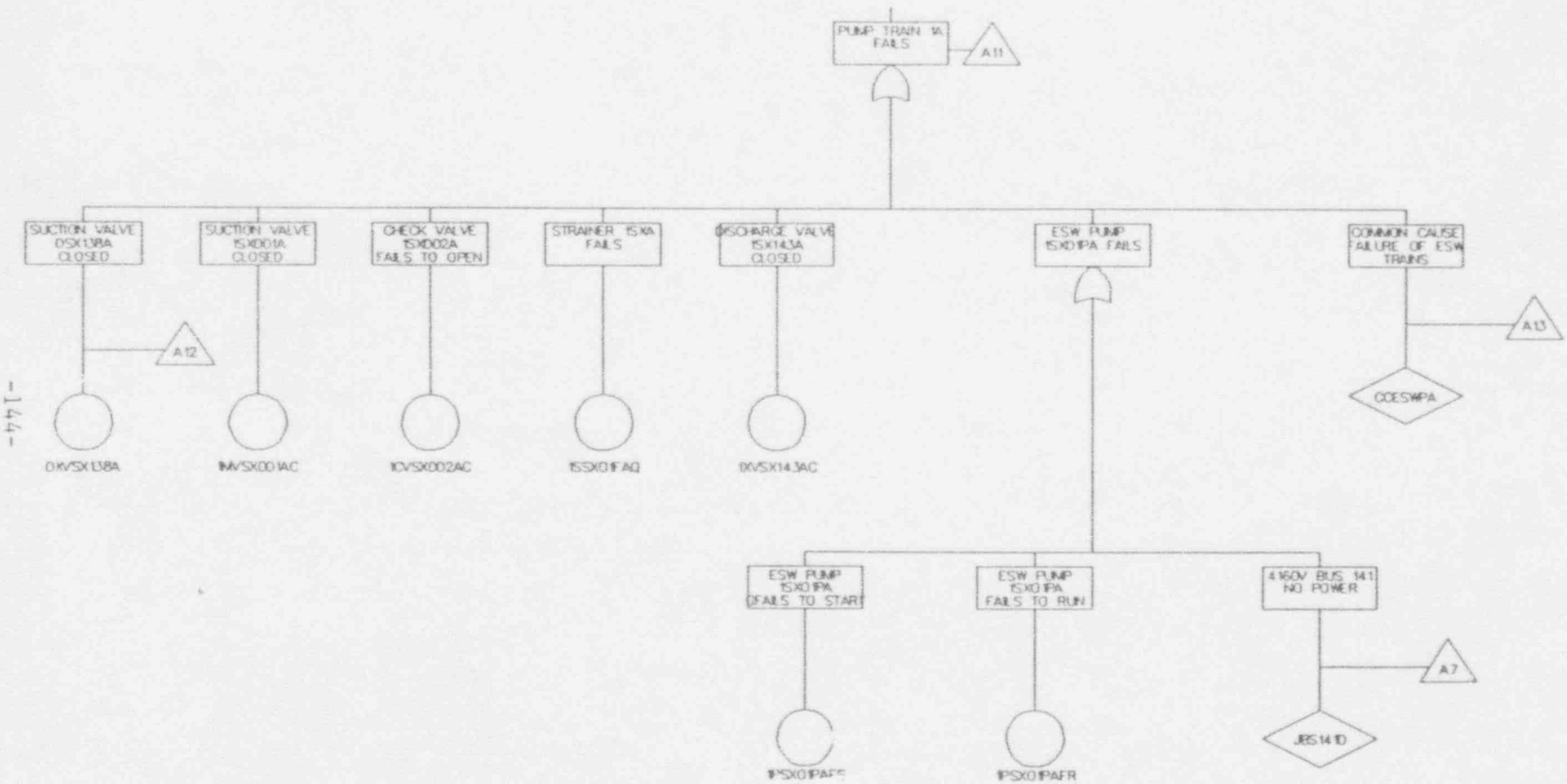


Figure D1. (Continued)


PUMP TRAIN 1A FAILS
A11
SUCTION VALVE 0SX138A CLOSED
SUCTION VALVE 1SX001A CLOSED
CHECK VALVE 1SX002A FAILS TO OPEN
STRAINER 1SXA FAILS
DISCHARGE VALVE 1SX143A CLOSED
ESW PUMP 1SX01PA FAILS
COMMON CAUSE FAILURE OF ESW TRAINS
A12
A13
CCESWPA
0KVSX138A
1MVSX001AC
1CVSX002AC
1SSX01FAQ
1XVSX143AC
ESW PUMP 1SX01PA FAILS TO START
ESW PUMP 1SX01PA FAILS TO RUN
4160V BUS 141 NO POWER
A7
JBS141D
1PSX01PAFS
1PSX01PAFR

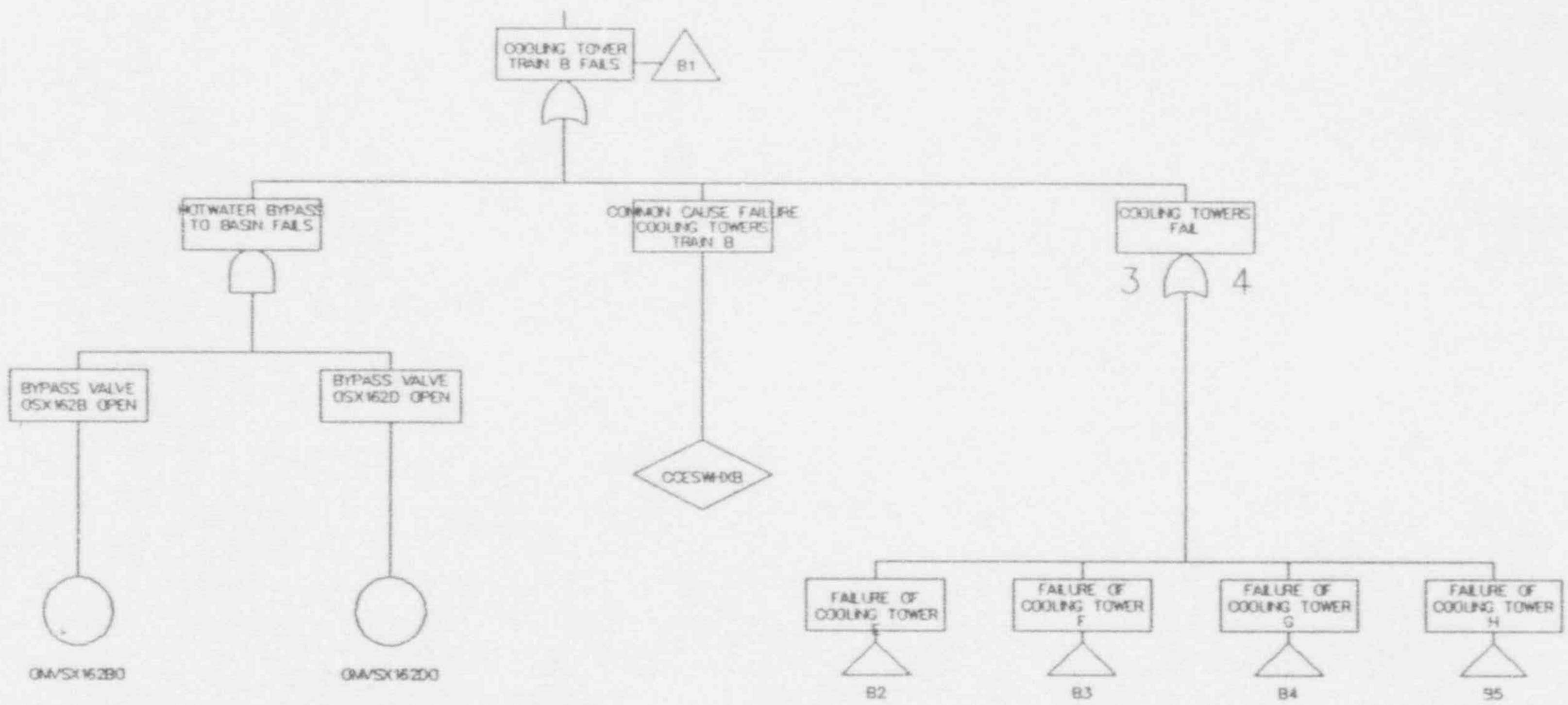


Figure D1. (Continued)

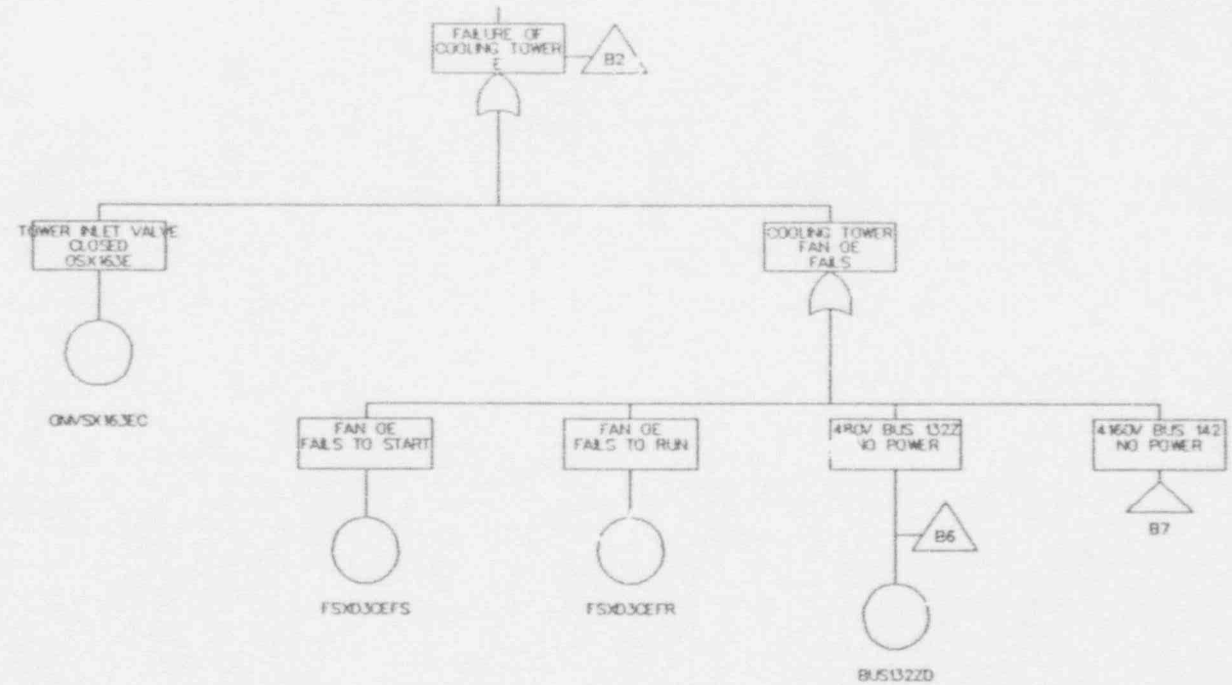


Figure D1. (Continued)

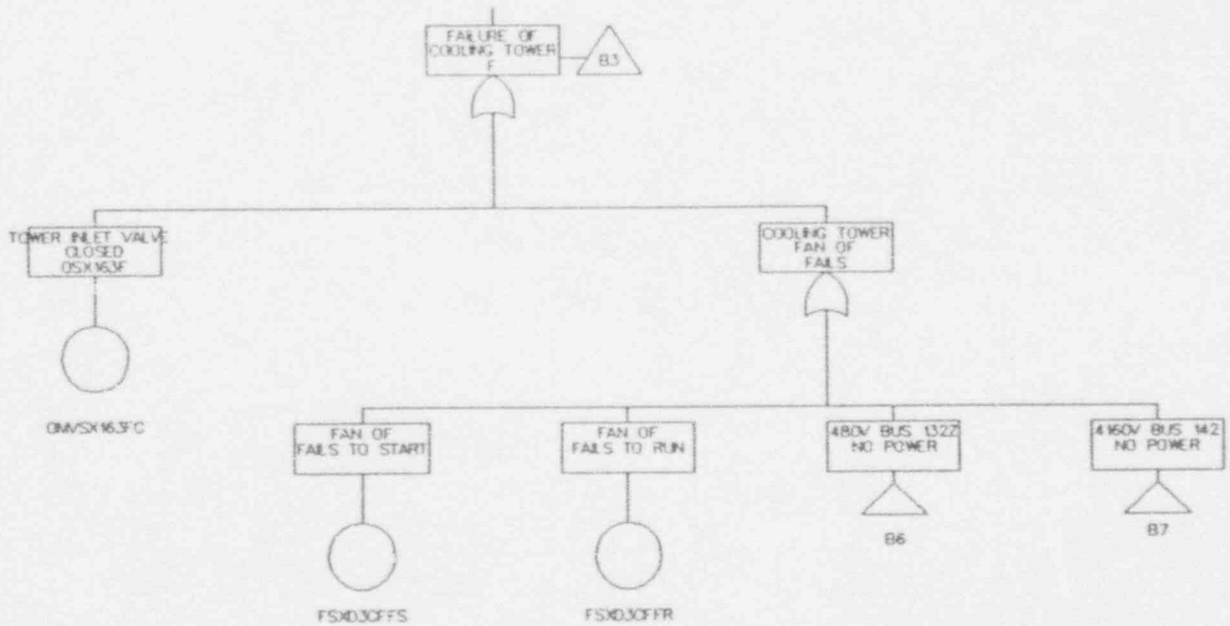


Figure D1. (Continued)

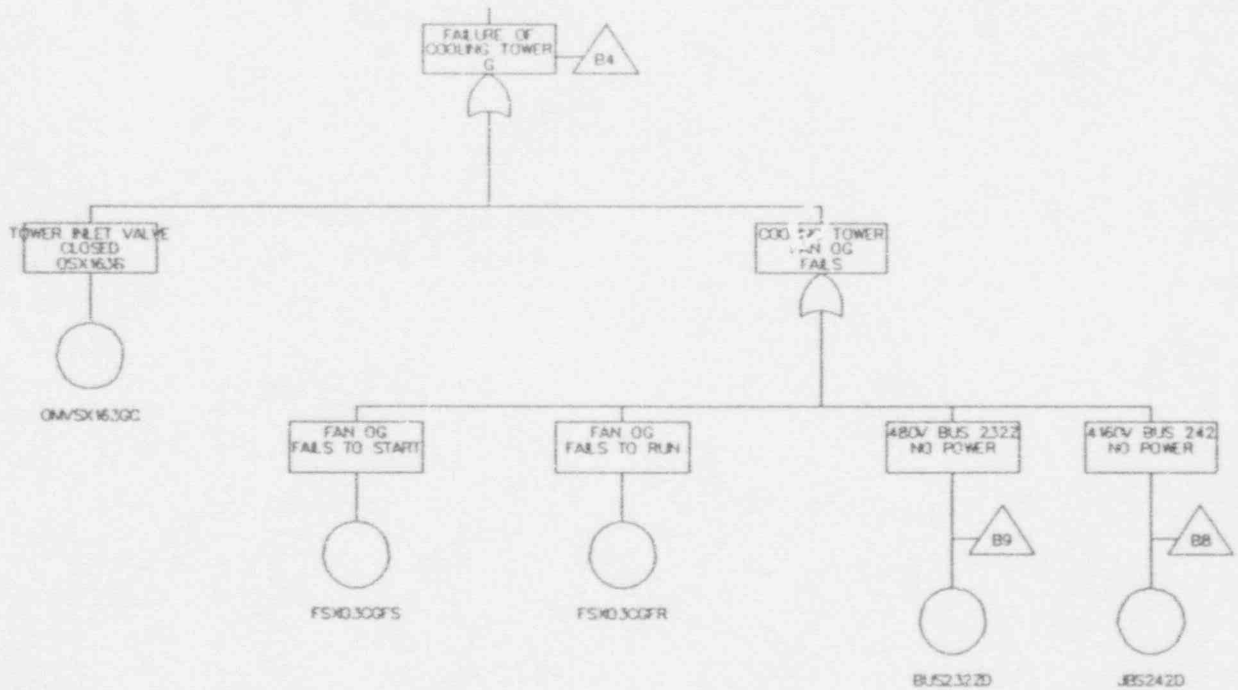


Figure D1. (Continued)

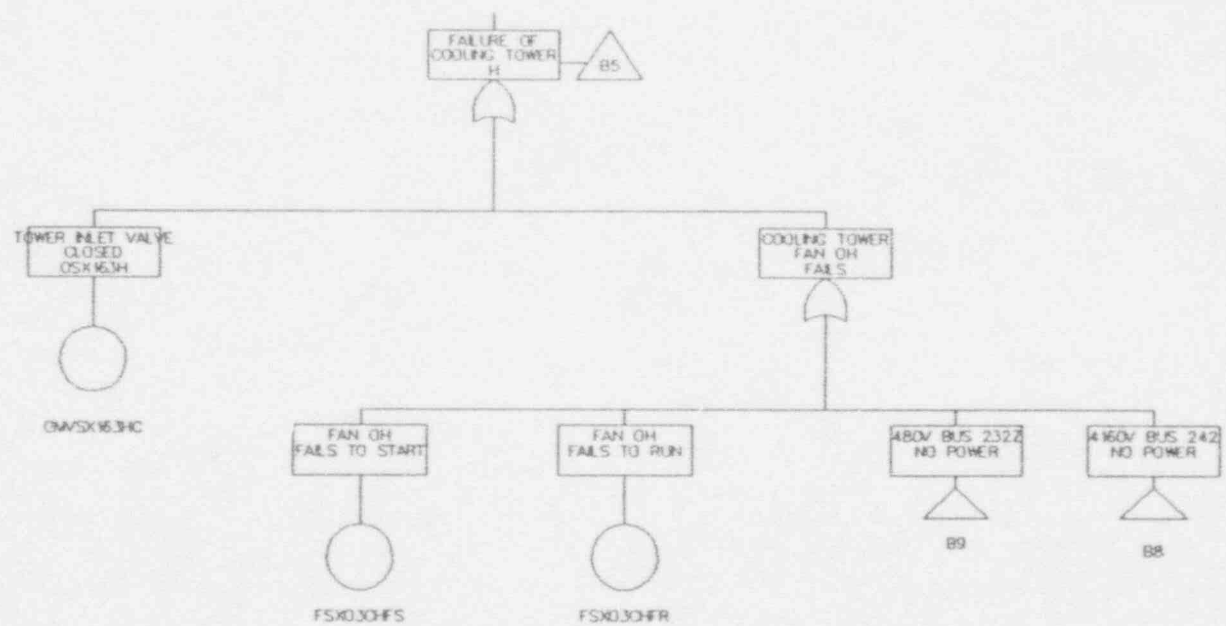


Figure D1. (Continued)

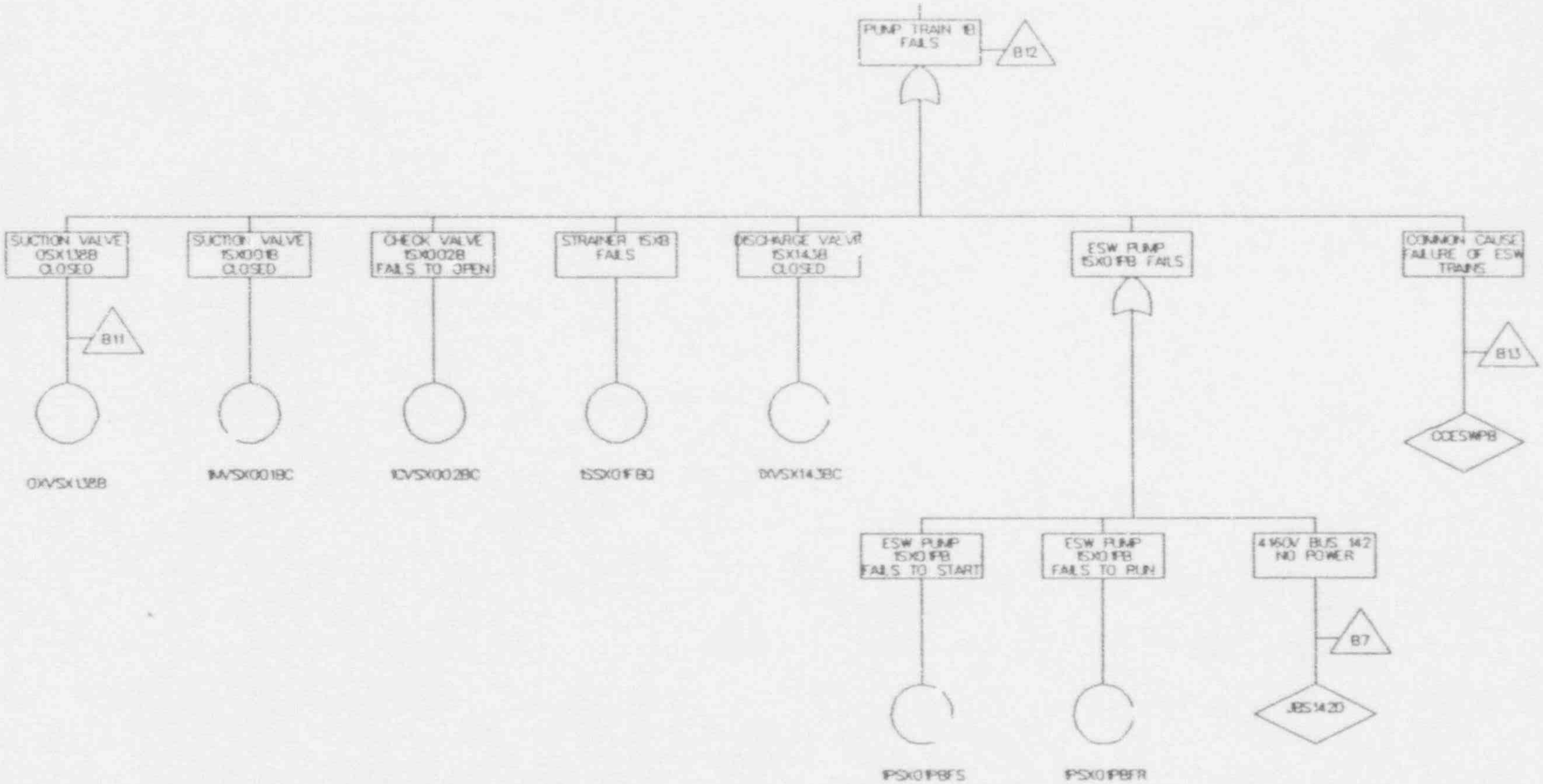


Figure D1. (Continued)

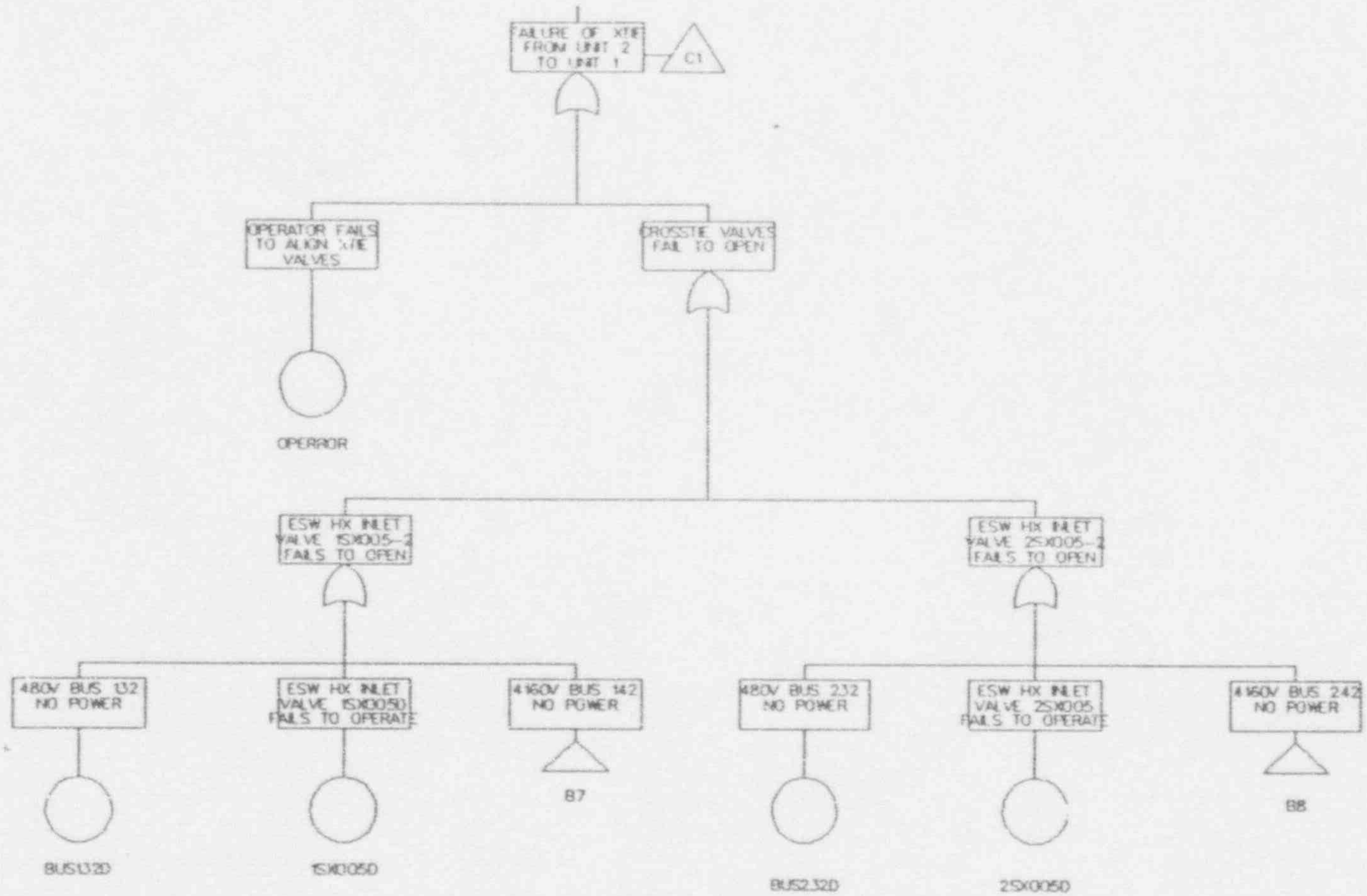


Figure D1. (Continued)

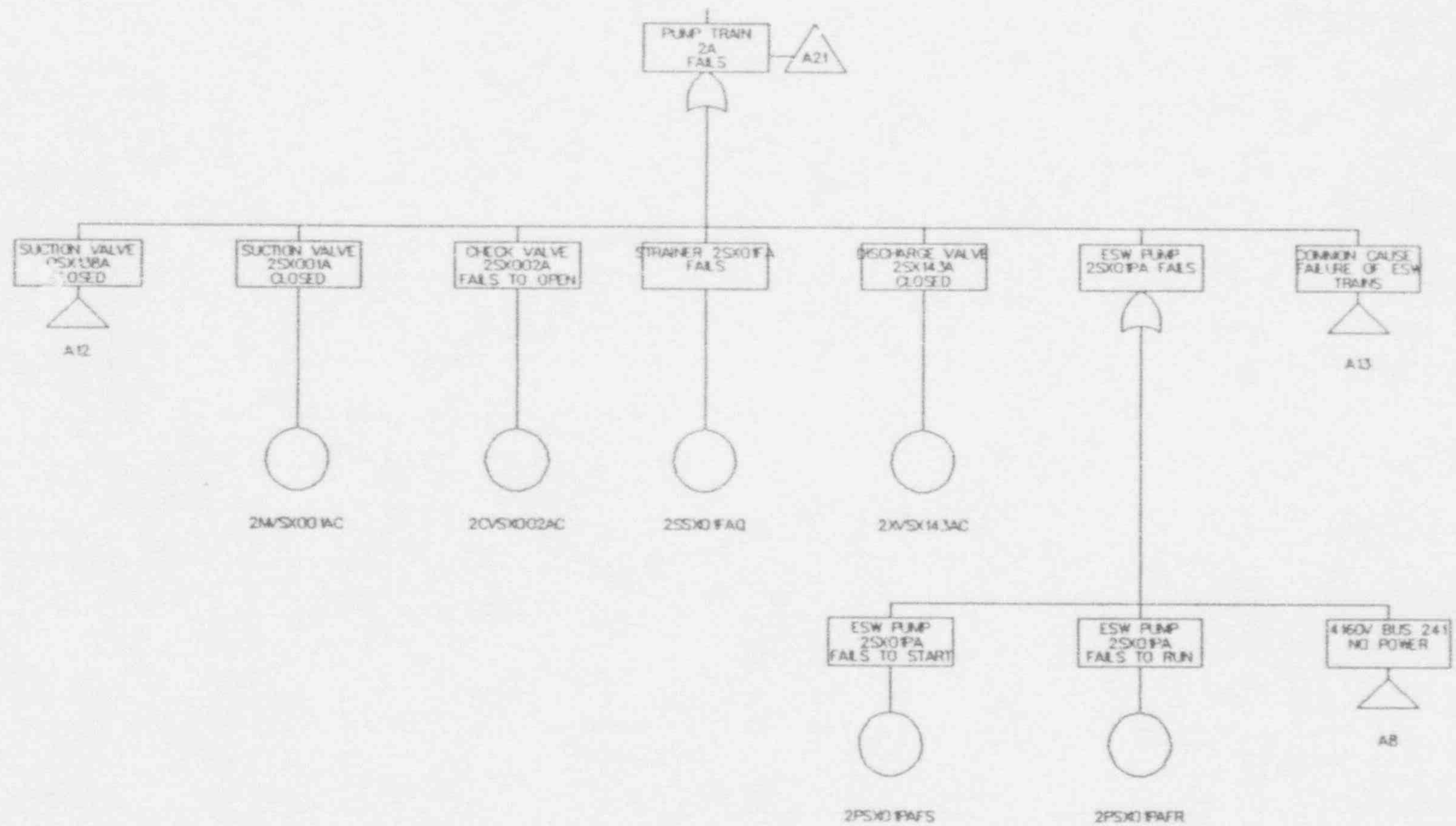


Figure D1. (Continued)

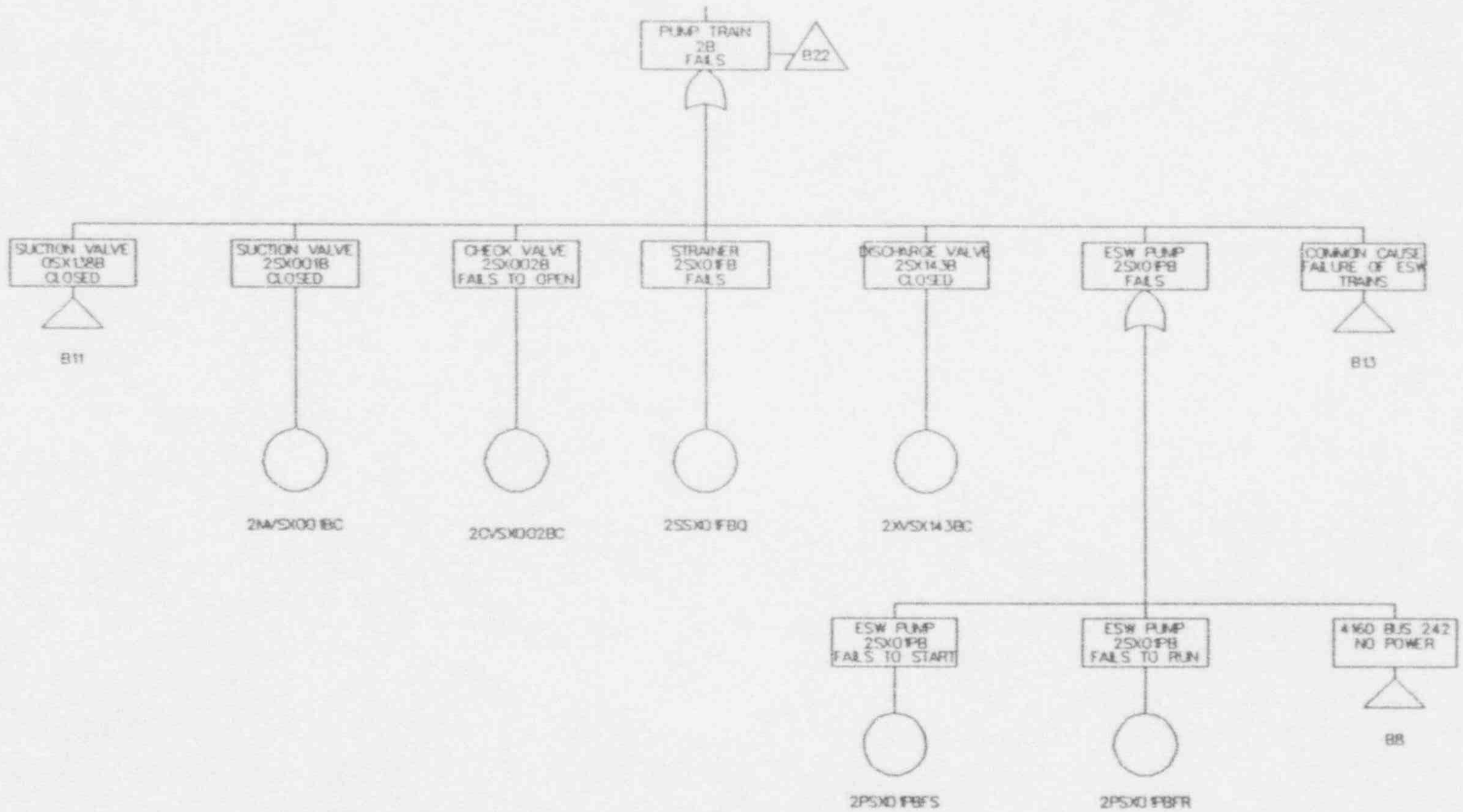


Figure D1. (Continued)

APPENDIX E

DISCUSSION OF COST ESTIMATES

APPENDIX E: Discussion of Cost Estimates

In this appendix, a discussion of source estimates for direct costs is provided (except for Option 4 which involves updates in technical specifications, procedures and testing). See Section 6.2.2 for a more complete referencing of reports that are just mentioned here.

Option 1: Additional Crosstie. Plant A estimated this option to cost $500K (direct cost only). Plant B estimated $2M. This would be a manual bypass valve. The lines were 24 inches, and they were seismically qualified. Two tees would have to be installed and seismically qualified. There would be problems with access and with room to put the valve in.

A third estimate was obtained by consulting NUREG/CR-4627 for green-field piping costs and applying the labor productivity factors. It was estimated that 10 ft. of 24" piping and a 24" valve would be needed.

One reference gives the green-field cost of piping in terms of $/ft in factory materials ($7,500/ft) and site materials ($2,100/ft) and man-hours/ft in labor (800 hr/ft). The labor will be costed at $40/hr. For the valve, the man-hours required would be 243 and the factory cost was $39K. This gives the total green-field cost of the crosstie bypass of $464K. The labor productivity factors applied for this backfit were 0.1 for access and handling (indicating regular nuclear plant environment), 0.4 for congestion and interference (which means severe problems in this area), 0 for radiation and 0.3 for manageability (indicating that this work would be done during a refueling shutdown, with many different activities going on at the same time and round the clock work schedules). The total labor productivity factor is 0.8. Therefore, we multiply $464K by 1.8 to arrive at $835K as the estimated direct cost of this fix. Remembering that this cost is spread over two reactors, the average of the three estimates is $557K per reactor, with a high limit of $1M and a low limit of $250K.

Option 2: Electrical Cross Connection. This would involve acquiring the ability to connect an electrical train to a service water train that is not normally associated with it (e.g., connect electrical train A to service water pump B). This ability would be exercised only in an emergency. Plant A gave us an estimate of $50K for the direct cost. This would involve having available 30-60 ft of jumper cable, a bus transfer device and a breaker box. In an emergency, it would take about ½ day of work to make the necessary electrical connection. This option would not be seriously considered at plant B. Therefore, we have only one estimate for the direct cost of this option, which is $50K.

Option 3: Swing Pump + Intake + Electrical Connection. Plant A gave an estimate of $25M. Of that amount, $3-$5M would be for the pump. About $20M would be for extending the existing intake structure (which is in a man-made lake). One would have to make sure that the electrical system for such an arrangement is acceptable. Plant B gave an estimate of $75M for non-seismic construction and $150M for seismic construction. This plant is on the ocean in a high seismic risk area. The pump has to accommodate the difference between a maximum high tide and a minimum low tide. A vent line for the pump would have to be provided above the water surface. The intake bay is lined with concrete and the pump is situated in a vault in the bay. The pump would have to be connected to existing 24" lines for service water. The construction would be seismic after the discharge isolation valve, even for the non-seismic option. In this plant the existing intake structure serves both the service water and the circulating water systems. In case of problems with the service water system, the water can be delivered from the circulating water intake to the service water intake. In case of plant B, we took their non-seismic number as being more representative since the swing pump would not necessarily have to be seismically qualified.

Our third estimate comes from using the Energy Economic Data Base (see Section 6) and NUREG/CR-4627. The EEDB considers a closed cycle SW system which should give us a low estimate. However, at least one GI-130 plant also has a closed cycle system. The breakdown of green-field direct cost from those sources is as follows:

Ultimate Heat Sink Structure:	\$6.3M
One Pump:	\$1.0M
10' of Piping:	\$0.4M
Instrumentation, Control, Power:	\$0.2M
Total Direct Cost:	\$7.9M

Applying the labor productivity factors as above, we arrive at the total cost of this backfit of \$14M. Keeping in mind the sharing of this option by the two reactors, the average direct cost from the three estimates is \$29M, with a low of \$7M and a high of \$38M.

<u>Option 4</u>: Updating Technical Specifications, Procedures and Crossties Testing. For changing tech specs, plant B gave us an estimate of \$100K which would be divided by the two reactors. NUREG/CR-4627 gives an estimate of \$18K per reactor. Therefore, the average cost for changing tech specs is \$34K per reactor, with a minimum of \$18K and a maximum of \$50K.

For changing procedures, NUREG/CR-4627 gives an estimate of \$1K for a simple procedure change. NUREG/CR-5102 (interfacing systems LOCAs in PWRs) gives an estimate of \$20K for writing a new procedure. Therefore, for a procedure change, the average estimate is \$5K per reactor (both reactors will have the same procedure), with a minimum of \$0.5K and a maximum of \$10K.

For crosstie valve testing, plant B gave an estimate of \$10K per test. Discounted and integrated cost then comes out to \$160K or \$80K per reactor. This would involve just the stroke testing of the valve. NUREG/CR-5102 quotes a generic cost estimate of \$1K per test for leak testing of isolation valves, for an integrated and discounted cost of \$16K or \$8K per reactor. This gives an average cost of \$44K per reactor with a low of \$8K and a high of \$80K.

The total average cost for option 4 is therefore 34 + 5 + 44 = \$83K per reactor, with a minimum of \$27K and a maximum of \$140K.

The NRC cost for this option (column 5 of Table 6.16) would be the summation of $11K per reactor as per NUREG/CR-4627 (as explained in Section 6) for the technical specification change, and $11K per reactor for the procedure change (procedure change assumed to take the same amount of NRC time as tech spec change). Hence, the total NRC cost for this option would be $22K, while for all the other options it is $61K as pointed out in Section 6.

Option 5: A High Pressure AC-Independent Pump for RCP Seal Charging. Plant A gave a direct cost of $2M for this option. This would be a diesel driven pump. Plant B gave an estimate of $15M for a non-safety pump. This pump would be powered by a dedicated diesel generator. The pump would be in a low elevation in the auxiliary building and there would be problems with access and congestion. Control and instrumentation cables would have to be put in. If a long run of the pipe is used there will be more room for the pump, otherwise the pump will have space limitations. Cooling to the diesel generator had to be provided. Radiation exposure would be experienced at the tie-in to the charging system, which also had to be seismically qualified.

NUREG-1109 estimates the total cost of this option as $1.5M. This includes direct, indirect and O&M costs. Based on our multipliers for indirect and O&M costs, the direct cost should be $0.8M. This is for a steam turbine-driven or a steam-turbine generator powered system. For a diesel-generator powered system, the cost is 2.5 times lower, or $0.3M. We used the higher value ($0.8M) since it didn't make much of a difference in the final analysis. Therefore, the average cost of this option is $5.9M per reactor (each reactor will have this system), with a low of $0.8M and a high of $15M.

Option 5a: Firewater Cooling for Thermal Barriers. This option has actually been installed at plant B and there the direct cost was $200K. This is the same as the average value quoted in the NRC references in Section 6. These references give a low of $127K for plants that do not need a 500 foot section of 4 inch return pipe and a high of $237K for plants that do need this pipe. Therefore, the best estimate for this option is $200K with a minimum of $127K and a maximum of $273K.