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Evaluation of Generic Issue 125.II.7, Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During a Line Break

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ABSTRACT

This report presents the evaluation of the potential safety concerns identified in Generic Issue 125.II.7, related to the automatic auxiliary feedwater (AFW) isolation from a steam generator during a main steam or feed line break. For this review, existing probabilistic risk assessments (PRAs) were evaluated to identify specific event tree sequences where the AFW system had failed. These sequences were used to calculate the contribution of AFW isolation system to the accident sequence frequency. By using this methodology, the change in risk (based on a change in core damage frequency) could be calculated for a plant with an automatic AFW isolation system compared with the same plant with the automatic AFW isolation system removed. The review evaluated one Westinghouse plant, one Combustion Engineering plant, and two Babcock and Wilcox design versions of a plant. Sections 3 and 4 of this report describe the methodology used to evaluate the various designs and provides the technical findings. Section 5 presents the cost benefit analysis performed to evaluate the various alternatives that were considered to resolve this issue.

FIN No. A6881-8 – Generic Issue 125.II.7, Automatic Isolation of Feedwater From Steam Generator During a Line Break

EXECUTIVE SUMMARY

The issue designated as GI 125.II.7 addresses the specific concerns related to the automatic isolation of auxiliary feedwater (AFW) to a steam generator with a broken steam or feed line. Some pressurized water reactor (PWR) designs use a system to automatically isolate AFW to a depressurizing steam generator. Following the Davis-Besse Loss of Feedwater Event in June 1985, the benefits of this automatic isolation system versus its disadvantages were questioned.

The NRC auxiliary feedwater requirements related to this issue are documented in NUREG 0800, *Standard Review Plan,*[1] Section 10.4.9.I.13. This reference states that:

> *"13. The system design possesses the capability to automatically terminate auxiliary feedwater flow to a depressurized steam generator, and to automatically provide feedwater to the intact steam generator. Or, as an alternative, if it is shown that the intact steam generator will receive the minimum required flow without isolation of the depressurized steam generator and containment design pressure is not exceeded, then operator action may be relied upon to isolate the depressurized steam generator."*

The purpose of GI 125.II.7 is to reevaluate these requirements and to determine if the disadvantages of automatic AFW isolation may outweigh the benefits. The benefits of automatic AFW isolation are as follows:

1. In the event of a steam or feed line break, the steam generator blowdown inventory is minimized. While isolating AFW does not prevent the initial secondary side inventory blowdown, it does prevent continued blowdown after the initial inventory is expended, and thus minimizes the containment pressure rise.

2. Excessive primary system cooldown is minimized. As the primary system cools down due to the heat transfer to the depressurizing steam generator, a reactor restart could occur, especially if the reactor fuel is approaching end-of-life. This would introduce thermal energy to the transient. Shutting off AFW to the faulted steam generator will reduce this effect.

3. Excessive containment pressure is minimized. The containment is designed to accommodate the pressure increase caused by a primary system loss-of-coolant-accident (LOCA). A steam or feedwater line break within containment might cause the containment design pressure to be exceeded if the automatic AFW isolation were not operable. If a reactor restart does not occur, continued AFW flow to the steam generator would contribute to excessive primary system cooldown and pressurized thermal shock conditions.

4. In some plants, the AFW isolation is required to divert AFW from the affected steam generator for orderly and safe plant shutdown and to meet the single failure criterion in supplying feedwater to the intact steam generator(s).

The disadvantages of automatic AFW isolation are related to concerns that the automatic isolation system may reduce the reliability of the AFW function. Also, with operator error, the long term success of AFW for main feedwater transients, steam generator tube ruptures, and small-break LOCAs could be compromised. Failures that cause inadvertent actuation of the AFW isolation system could cause loss of all AFW system flow during accidents or transients. Additionally, during a controlled cooldown, the thresholds for automatic AFW isolation (such as low steam generator pressure or high steam generator to steam generator differential) may be crossed, which would require that the operator lock out the isolation logic as the steam generator parameters approach the isolation setpoint. During an accident scenario (not requiring isolation), the accompanying distractions could result in a failure to lock out the automatic isolation, thus AFW would not be available as predicted for the applicable accident analyses.

The safety significance of this issue is whether the positive aspects of automatic AFW isolation, resulting in decreased containment pressure and diversion of AFW to the functioning steam generator(s), are outweighed by the negative aspects of lower AFW system availability due to inadvertent or spurious actuation of its automatic isolation system.

To evaluate this issue, a methodology was developed wherein four typical PWR designs were selected for analyses. The PWR selections were based on, (a) PWRs utilizing an automatic isolation system, (b) Nuclear Steam Supply System (NSSS) supplier (Combustion Engineering, Babcock and Wilcox [two different designs], and Westinghouse), and (c) availability of an existing PRA.

One plant design each from Westinghouse (W) and Combustion Engineering (CE) and two different Babcock and Wilcox (B&W) plant designs were selected for this study. It was thought that the greatest risk associated with this issue would be for plants with marginal or no feed-and-bleed capabilities because with the loss of all secondary heat removal methods with primary system pressure greater than Residual Heat Removal (RHR) operating pressure there is no way of removing decay heat. This study included one such plant. Another significant consideration was that some plants without flow restrictors in their AFW pump discharge lines utilize the automatic AFW isolation system to prevent AFW pump runout conditions. The evaluation performed for this study also included three of these plants.

After the collection and review of the data, the contributions of the automatic AFW isolation systems to the AFW failure probabilities were determined based on failure rates estimated for the automatic isolation system design, probability of operator error, and possible common cause failure resulting in failure of all AFW system functions. Automatic AFW isolation system failure rates had to be estimated because all accident sequences containing automatic AFW isolation system failures had Core Damage Frequencies (CDF) that were less than the accident sequence CDFs considered in the individual PRAs. Also, the study estimated changes to the CDFs that would occur if the automatic AFW isolation systems were removed. These modified AFW system failure rates were then substituted into the PRA sequences that included the AFW system. By using this methodology, the change in CDF could be calculated for removing the automatic isolation system. The change in CDF was then utilized to calculate a change in risk. More details on the methodology used are given in Section 3.0. The evaluation indicated that the contribution of the AFW isolation system to CDF depends on the plant design, but the total contribution is small. Also, the evaluation showed that removing the AFW isolation system and relying on the operator to isolate a ruptured steam generator caused CDF to change in either direction (i.e., increased or decreased) depending primarily on the presence of AFW flow restrictors and the reliability of backup feed-and-bleed operations.

After calculating this delta risk, the cost associated with removing the automatic isolation systems was estimated. Cost benefit ratios were then calculated and summarized. The change in risk due to removing or disabling the AFW isolation system ranged from a decrease in plant risk of 44 man-rem for one plant (B&W without emergency feedwater initiation and control) to an increase of 12 man-rem over the plant life for Westinghouse design.

Cost estimates for the disabling or removing the AFW isolation system ranged from $351K for a plant with existing flow restrictors to $768K for a plant requiring installation of flow restrictors.

The cost benefit ratios calculated to determine the cost per man-rem reduction for the proposed plant modifications showed a best case (maximum man-rem reduction, minimum cost) of $7.9K per man-rem. Therefore, it was concluded that no plant modifications would be warranted for the resolution of this issue.

FOREWORD

This report provides an evaluation of Generic Issue 125.II.7, Reevaluate Provisions to Automatically Isolate Feedwater From a Steam Generator During a Line Break. This report was prepared for the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Division of Reactor and Plant Systems, by EG&G Idaho, Inc., NRC Technical Assistance Group.

The NRC Task Manager for this project is D.L. Basdekas.

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CONTENTS

FIGURES

TABLES

EVALUATION OF GENERIC ISSUE 125.II.7, REEVALUATE PROVISION TO AUTOMATICALLY ISOLATE FEEDWATER FROM STEAM GENERATOR DURING A LINE BREAK

1. INTRODUCTION

The issue (designated in 1985 as GI 125.II.7 by the Nuclear Regulatory Commission) of automatic feedwater isolation for a PWR steam generator during a steam or feed line break was identified from the findings of the Davis-Besse Incident Investigation Team as reported in NUREG-1154, *Loss of Main and Auxiliary Feedwater Event at the Davis-Besse Plant on June 9, 1985*.[2] The investigation identified the concern that the benefits of automatic auxiliary feedwater isolation could possibly be outweighed by the negative aspects of this feature. Periodic reference to this Davis-Besse event is made throughout this report.

The automatic isolation of AFW from a steam generator is provided to mitigate the consequences of a steam or feedwater line break. The isolation logic, for most plants that utilize the design, closes all main steam isolation valves and also isolates AFW from the broken depressurizing steam generator while continuing to feed the unaffected steam generator(s).

The benefits of AFW isolation are as follows:

1. The break blowdown inventory is minimized. While isolating AFW doesn't prevent the initial secondary side inventory from blowdown, it does preclude additional blowdown from continuing decay heat removal after the initial inventory is expended.

2. Excessive primary system cooldown is minimized. As the primary system cools down due to the heat transfer to the depressurizing steam generator, a reactor restart could occur, especially if the reactor fuel is approaching end-of-life. This would introduce thermal energy to the transient. Shutting off AFW to the faulted steam generator will reduce this effect.

3. Excessive containment pressure is minimized. The containment is designed to accommodate the pressure increase caused by a primary system loss-of-coolant-accident (LOCA). A steam or feedwater line break within containment might cause the containment design pressure to be exceeded if the automatic AFW isolation were not operable.

4. In some plants, AFW automatic isolation is required to divert AFW from the affected steam generator for orderly and safe plant shutdown and to meet the single failure criterion in supplying feedwater to the intact steam generator(s).

The disadvantages of automatic AFW isolation are related to concerns that the automatic isolation system may reduce the reliability of the AFW system and with operator error, the long term success of AFW for main feedwater transients, steam generator tube ruptures, and small-break LOCAs could be compromised. Failures that cause inadvertent actuation of the AFW isolation system could cause loss of all AFW system flow during accidents or transients. Additionally, during a controlled cooldown, the thresholds for automatic AFW isolation may be crossed, which would require that the operator lock out the isolation logic as the steam generator parameters approach the isolation setpoint. During an accident scenario, the accompanying distractions could result in a failure to lock out the automatic isolation, thus AFW would not be available as predicted for the applicable accident analyses.

The safety significance of this issue is whether the positive aspects of automatic AFW isolation resulting in decreased containment pressure and diversion of AFW to the functioning steam generator(s) are outweighed by the negative aspects of lower AFW system availability due to inadvertent or spurious actuation of its automatic isolation system.

The automatic isolation of AFW from a steam generator is provided to mitigate the consequences of a steam or feedwater line break. The isolation logic, usually triggered by a steam generator low pressure signal, closes all main steam isolation valves and also isolates AFW from the broken depressurizing steam generator while continuing to feed the unaffected steam generator(s).

The purpose of GI 125.II.7 is to evaluate typical PWR designs to determine if automatic AFW isolation results in an increase in the risk to the public. To accomplish this, a detailed technical review was performed in which four PWR designs were subjected to a detailed automatic AFW isolation analysis. Five generic accident sequences were identified in which automatic AFW isolation could affect plant safety. This evaluation calculated a change in risk based on the change to CDF caused by removing the automatic AFW isolation system.

2. SCOPE OF STUDY

This study evaluated, for representative PWR plant designs, the consequences of automatic isolation of AFW to the steam generators during a feed line or steam line break as well as other applicable accident sequences that rely on the AFW system for success. A plant representing each PWR vendor was analyzed to determine the impact of GI 125.II.7.

The study also (a) identified plant designs where such AFW isolation could impact plant safety by increasing the probability that decay heat removal capabilities may be lost, (b) evaluated plant modifications that could be implemented to mitigate the consequences of automatic isolation, and (c) performed a cost benefit analysis to assess the value-impact of removing the automatic isolation feature.

It should be noted that this study only evaluated plants that utilize an automatic AFW isolation system and did not address whether this type of system should be required. This scoping decision was made based on the definition of this issue as stated in NUREG-0933, *A Prioritization of Generic Safety Issues,*[3,4] because there are existing NRC criteria concerning these AFW system requirements (NUREG-0800[1], Section 10.4.9). Other Generic Issues, such as GI 124, Auxiliary Feedwater Reliability, and GI 125.II.1, Need for Additional Actions on AFW Systems, should also address this question of requiring PWR automatic AFW isolation systems.

3. EVALUATION METHODOLOGY

3.1 Evaluation Technique

A plant from each of the PWR vendors was selected for the evaluation, and available PRAs (References 6,10,11,13,14) for the selected plants were employed for this analysis. AFW systems were studied to determine how they functioned and how the AFW isolation system functioned. Accident sequences were evaluated to determine the contribution to the CDF of inadvertent or spurious AFW isolation system actuation. The following sections describe the technique employed in this analysis and the postulated sequences evaluated.

The AFW isolation system's contribution to the total CDF was calculated by reviewing completed PRAs for the selected plants. The AFW systems and the AFW isolation systems were studied to determine which of the basic accident sequences described in Section 3.2 were applicable to the particular plant. The accident sequences described in the respective plant PRAs, referenced above, were reviewed to identify the accident sequences that included failure of the AFW system. The AFW failure events in those sequences were then evaluated to determine the contribution of the automatic AFW isolation system to the AFW system failure probability.

Throughout this report, where probabilities or failure rates are utilized, they have been reviewed to determine if they were considered appropriate for the application. This determination was based on PRA experience and engineering judgement. In general, plant specific data from the plant PRA was reviewed and used as a first choice. In cases where the plant specific data was deemed inappropriate or the data was not available, previously used and justified data from the NUREG-0933[3] analysis related to this issue was reviewed and used as judged appropriate. In cases where neither the plant PRA data or NUREG-0933[3] data was available or judged appropriate for the specific application, data from other recognized and widely used data bases was used. The sources of the data used is indicated in the text of this report.

In general, the referenced plant PRAs identify several sets of accident sequences that lead to core damage. The calculated rate of occurrence for the accident sequence per year is called the core damage frequency. Usually, only the accident sequences with a CDF contribution greater than 1.0E-08 are discussed, because CDFs less than this are insignificant when added to the accident sequences with the higher contributions. These latter accident sequences are called the dominant sequences. The accident sequences are identified by the initiating event (e.g. loss of offsite power or loss of feed flow) that starts the accident sequence, and a series of letter designations that identify the systems or components that are failed or fail as the sequence progresses. For this study, the accident sequences of interest have initiating events that are perturbations in the power conversion system that result in a loss of main feed. In general, the other accident sequences do not rely on the AFW system for successful mitigation because the power conversion system is available to remove energy from the primary system. The referenced PRA labels the initiator as a transient, e.g. T1 through T4, Tdc., etc. The number following the T indicates the type of transient (e.g. loss of feed flow, loss of off site power, etc.). The letter "L" in the accident designator signifies failure of the AFW. Where a particular accident sequence is discussed in the following sections, the accident sequence designation is shown along with a very brief description of the accident. The CDF for the accident sequence is determined by multiplying the frequency of the initiating event times the probabilities of failure or unavailabilities of the failed systems or components in the sequence.

The probability of failure or the unavailabilities of failed systems in the accident sequences are determined as follows. Fault trees are developed for the system that identify all the combinations of component failures that will cause the system to fail to perform its intended function. Component failure rates or unavailabilities are then determined for the failed or unavailable components. Using Boolean algebra techniques the failure rates and unavailabilities of the various component failure combinations that lead to system failure are evaluated to determine the probability of system failure. There are many sets of component failure combinations that lead to system failure. These are called cutsets. A minimal cutset is a cutset that is made up of the smallest combination of component failures which, if they all occur, will cause the system to fail. The fault trees are usually so large that they must be evaluated by computer and even then they must be selectively reduced to allow computation in a reasonable time and at a reasonable cost. Generally, cutsets leading to system failure that have a probability below a value determined to be insignificant by the analyst are truncated to reduce the computation time and cost. See NUREG-0492, *Fault Tree handbook*,[5] for a more detailed discussion of fault tree analysis.

For this evaluation, the AFW system minimal cutsets were examined to determine the effects of the AFW isolation system failure. In all cases, the probability of AFW failure due to a minimal cutset involving the AFW isolation system was well below the probability of failure for many other minimal cutsets. Thus, the AFW isolation system was not a significant contributor to AFW system failure. To estimate the contribution of the AFW isolation system to the CDF, minimal cutsets involving the AFW isolation system were developed and added separately. These probabilities for AFW failure involving the AFW isolation system were then substituted into the accident sequence for the failure of the AFW system to determine the contribution to the CDF for the AFW isolation system. The AFW isolation system contribution for each of the selected plants was determined by the above method.

In addition to the sequence evaluation described above, the effects on the CDF of removing the AFW isolation system was determined by estimating the frequency of a break in a steam or feed line, the probability of the operator failing to isolate the affected steam generator, and the probability of feed-and-bleed failure. These values were then combined, as appropriate for each plant.

By using this methodology, the change in CDF resulting from removal or disabling the automatic AFW isolation system was calculated. The change in CDF was then used to calculate a change in plant risk.

3.2 Postulated Accident Sequences Affected by the AFW Isolation System

The purpose of this evaluation is to investigate the positive and negative aspects of the automatic AFW isolation system. The positive aspects are that the isolation system will ensure that AFW is supplied to functional steam generators only, thus ensuring that the containment pressure and steam content are minimized and that AFW is not diverted to a ruptured steam generator, resulting in a loss of heat sink for decay heat removal. The negative aspects are the contribution of the automatic AFW isolation system to the inadvertent failure of the AFW system and the potential of causing a loss of feedwater transient. The first three accident sequences discussed below present an evaluation of the negative aspects of the automatic AFW isolation system, i.e. they present an estimation of the automatic AFW isolation system's contribution to the CDF. The last two accident sequences present an evaluation of the positive aspects of the automatic AFW isolation system by providing an estimate of the change to CDF if the automatic isolation system were removed. The "net worth" of the automatic AFW isolation system is determined by comparing the increase in CDF caused by removing the automatic AFW isolation system to the CDF of accident sequences involving failure of the AFW system caused by the automatic AFW isolation system. If the isolation system's failure rate contribution is higher than the change in CDF due to removing the system, then the automatic AFW isolation system is contributing more risk than it is mitigating.

Inadvertent or spurious actuation of the AFW Isolation System could be the cause of a transient. For example, a spurious signal could cause the main steam isolation valves and AFW isolation valves to close. Closure of the main steam isolation valves would trip the main feed pumps on most plants because the pumps are turbine-driven. Thus, the plant would be in a total loss of feedwater transient. If the operators cannot recover feedwater flow or initiate and maintain feed-and-bleed in the limited time available, usually about 30 minutes, the transient will lead to core damage due to a loss of heat removal capability.

Another area of concern is any accident sequence that relies on AFW operation. Spurious or inadvertent actuation of the AFW isolation system could be a significant contributor to the unavailability of the AFW system. Recovery actions may not be simple operations; at the June 1985 Davis-Besse event, the operators had to manually initiate the opening of the isolation valves because the valve motor torque limit had been improperly set, and the pumps had to be manually restarted.

Actuation of the isolation system during long term cooldown using the AFW system for heat removal is also an area of concern. During long term cooldown, secondary system conditions that cause actuation of AFW isolation system will eventually be reached, i.e. low steam generator pressure. If the operator has not locked out or bypassed the isolation system, the heat removal method will be lost and some type of recovery action will be required. The added stress (caused by this additional event) may cause the operators to make other errors complicating recovery from an accident sequence, and eventually leading to core damage.

The removal of the AFW isolation system is expected to have a negative impact (i.e., an increase in the consequences) on the two accident sequences described below. Both of these sequences address the concerns related to the original purpose of the AFW isolation system, i.e. to mitigate the effects of a main steam or feed line break.

Isolation of a depressurizing steam generator caused by a feedwater line break, is required to prevent either the diversion of flow from unaffected steam generators or the failure of all the AFW system due to pump runout or cavitation caused by higher than normal flow rates. Upon removal of the AFW isolation system, isolation of the affected steam generator will have to be performed manually. During a postulated accident sequence consisting of a feedwater line break, followed by the failure to isolate the affected steam generator, and the failure of feed-and-bleed (where this technique can be performed), the core damage contribution is expected to increase because timely operator action under stressful accident conditions is required to isolate the affected steam generator.

Another purpose of the AFW isolation system is to minimize steam blowdown to the containment for accident sequences involving steam line rupture while the AFW system still supplies flow to the affected steam generator. The concern for this accident scenario is containment failure due to overpressure. Although this scenario does not involve core damage, it does involve the potential release of radioactive material to the environment due to the release of primary coolant through potential steam generator tube leaks or low levels of contamination that may be present inside the containment. Hence, some safety consequence is expected due to this type of accident scenario. As described above, the consequences associated with this scenario are expected to increase since operator action will be require to isolate the affected steam generator.

4. PLANT ANALYSIS

This section presents the system and PRA sequence analyses performed for the four plant designs evaluated in this study. A description of the AFW system, details of the sequence analyses, and the results of the analyses are presented for each plant.

4.1 Plant A

This evaluation was based on the system descriptions and PRA evaluations contained in the IREP study of a Combustion Engineering plant in NUREG/CR-3511, *Interim Reliability Evaluation Program: Analysis of the Plant A Nuclear Power Plant*[6]

4.1.1 System Description. Plant A is a Combustion Engineering (CE) designed reactor system that has two U-tube steam generators. The AFW system (Figure 1) has one turbine-driven pump and one motor-driven pump. Each pump supplies both steam generators through a separate header, i.e., there are flow control and isolation valves for the turbine-driven pump on a header separate from the flow control and isolation valves for the motor-

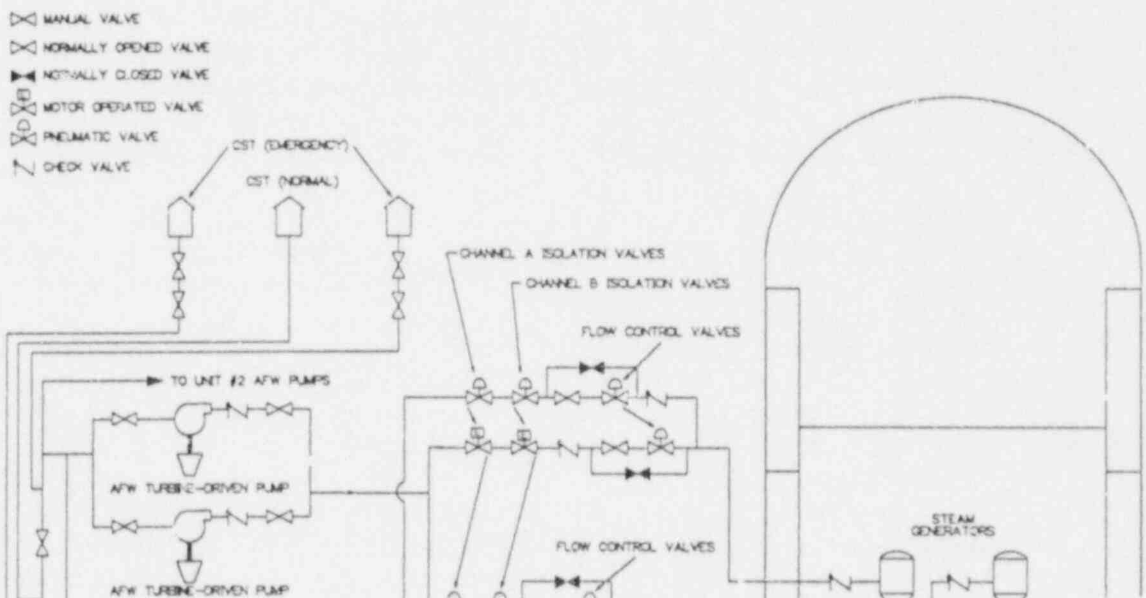

Figure 1. Plant A (CE) auxiliary feedwater system flow schematic.

driven pump to steam generator 1. There is another turbine-driven pump, but it must be manually lined up and started by the operator. The motor-driven pump from Unit 2 can be cross connected to Unit 1. The extra turbine-driven pump and cross-connecting of the motor-driven pump from Unit 2 were considered recovery actions for the applicable accident sequences by the IREP analysis.[6]

Each steam generator has its own automatic AFW isolation system, actuated by two independent channels (A and B). The AFW system isolation is accomplished by two valves in series in each header supplying each steam generator for a total of eight isolation valves. The valves have no other purpose in the system and are normally open. Isolation initiation circuit A closes one valve in each header on the affected steam generator and circuit B closes the other valve. Only one valve in each affected header (motor-driven and turbine-driven pump headers) must be closed to isolate AFW flow to the desired steam generator.

During a steam or feedwater line rupture event, the main steam isolation valves on both steam generators will close when the pressure of either steam generator is less than 500 psig. The AFW isolation valves on the affected steam generator will close on coincident low steam generator water level (less than 50 in.) and high steam generator differential pressure (greater than 100 psid). If the isolation signal is generated while the AFW system is in operation, half of the isolation signal will already be present because the AFW is initiated by the low steam generator water level signal. The actuation signals for each circuit are based on two of four coincidence from four independent transducers on each steam generator.

Each header has a throttle valve set to limit flow to 200 gpm; thus, failure to isolate AFW from a ruptured steam generator will not cause the loss of AFW to the other steam generator due to flow being diverted to the steam generator with the lower pressure. The operator can manually control the throttle valve setting if desired. The PRA[6] used to evaluate this plant indicates that the AFW system is assumed to fail if less than 400 gpm is delivered to one or both steam generators.

The AFW system must be used to maintain steam generator levels during startup and shutdown when the reactor power level is low (less than 5%).

4.1.2 Plant A (CE) Sequence Analysis. At Plant A (CE), two automatic AFW isolation system failure sequences are of interest; (a) failures of the isolation system that result in inadvertent isolation of both steam generators, and (b) the inadvertent isolation of one steam generator when one of the AFW pump trains is inoperable. If both steam generators are isolated, all AFW flow will be lost. If one steam generator is isolated while one of the pump trains is inoperable, the AFW flow will drop to 200 gpm. The operator must then take manual control of the appropriate throttle valve to increase AFW flow to greater than 400 gpm, or he must open one of the isolation valves to prevent core damage. Feed-and-bleed was not considered effective at Plant A because of the relatively low discharge head (1750 psi) of the high pressure injection pumps and the uncertainty as to whether or not the pressure could be reduced enough to initiate High Pressure Safety Injection (HPSI).[11,12]

AFW system failure caused by spurious actuation of the AFW isolation system would consist of; (a) a spurious signal to isolate one steam generator, (b) a common mode failure of the logic module to isolate the other steam generator, and (c) the operator failing to recover flow to greater than 400 gpm. The Plant A (CE) IREP study[6] indicates a spurious isolation has a probability of 7.2E-05. This is based on a 3.0E-06/hr failure rate for solid state components and a 24 hour exposure time. If a common mode failure probability (0.05) similar to that used in the NUREG-09333 evaluation of this event and a failure of recovery probability (0.04) similar to that used in the IREP study[6] are assumed, the resulting cutset will have a probability of 7.2E-05 * 0.05 * 0.04 = 1.44E-07.

Because there are two actuation channels, two cutsets will contribute to the change in CDF due to the aforementioned factors. For some accident sequences, both actuation channels will be active; thus, the AFW system failure probability will be 2.88E-07. For other accident sequences, one of the actuation channels is assumed failed due to maintenance or other independent failure, so the AFW failure probability will be as stated above, 1.44E-07.

The 3.0E-06/hr failure rate of solid state components used by the IREP[6] study for estimating the spurious block probability agrees well with solid state component "failure to function" failure rates found in other PRAs (References 11,13-15,17,18,19) and in reference material such as

WASH-1400, *Reactor Safety Study*,[9] and NUREG/CR-2728, *IREP Procedures Guide*.[6] Because a spurious output of a solid state device is only part of a solid state component's "failure to function," the value used is a conservative estimate.

The 0.05 probability for common mode failure was taken directly from the NUREG-0933[3] evaluation of this generic issue because it was judged to be a reasonable value based on the system design.

The 0.04 probability of failure of the operator to recover was estimated from the values used for failure to recover in the IREP study.[6] Several recovery actions of various complexities were considered in the study. The 0.04 value is representative of the relatively simple actions required and the multiple paths available to recover the AFW and the approximately 80 minutes of time available for recovery.

Failure events involving isolation of both steam generators caused by independent spurious signals to both steam generators have a probability of occurrence of about 5.0E-09 and were judged to be insignificant; therefore, they were not included in this evaluation.

Failure events that could lead to core damage caused by spurious isolation of one steam generator when the turbine-driven or motor-driven pump trains are not operating for some reason (such as during maintenance or valve failure) consist of (a) a spurious signal to isolate one steam generator and (b) the operator fails to recover AFW to greater than 400 gpm. Again, feed-and-bleed is not effective at Plant A (CE) because of the low discharge head (1750 psi) of the HPSI pumps. Using the same probabilities as those used above, the probability of a spurious signal actuating the AFW isolation system to isolate one steam generator and the operator failing to recover is 7.2E-05*0.04 = 2.88E-06.

Again, two cutsets will contribute to an accident sequence because there are two isolation actuation channels. These cutsets are only applicable when one of the pumping systems is down for reasons not related to the spurious isolation. Like the previous evaluation, only one cutset will apply to some accident sequences and both will apply to others.

In the preceding discussions, only the AFW is isolated by the spurious signals. The main steam isolation valves are on a different logic module and will require a different signal to cause them to close.

4.1.2.1 Spurious AFW Isolation Caused Transients. A spurious actuation of the AFW isolation system at Plant A (CE) will not cause a total loss of feedwater transient because the signal to close the main steam isolation valves (which will cause the main feed pumps to trip) comes from a different logic module than the signal that closes the AFW system isolation valves. Thus, two spurious actuation signals are required to initiate a transient and fail AFW. This accident sequence is covered by sequence T2L, which is the loss of the Power Conversion System (PCS) followed by loss of the AFW system. The PCS comprises the main feedwater and condensate system, the steam generators, and the main steam system which includes the turbines, the turbine bypass, the atmospheric dump valves, and the safety relief valves.

4.1.2.2 Spurious AFW Isolation During Transients Requiring AFW. The following six accident sequences in the dominant accident sequences involve failure of AFW:

TdcL Loss of DC power fails the AFW actuation system but starts the turbine-driven pump.

AFW isolation does not play a part in this accident sequence because the loss of DC power fails the AFW actuation system; thus, an inadvertent isolation signal cannot be generated.

T2L Loss of power conversion and secondary steam relief system followed by loss of AFW.

This accident sequence is dominated by cutsets that require two independent failures of AFW. Replacing the appropriate events with spurious isolation of one steam generator and with common mode isolation of the other results in a spurious AFW isolation contribution to the CDF of 4.67E-07 (see Appendix A for calculations).

T4ML Any transient other than those initiated by the loss of offsite power, loss of PCS, or transients requiring primary system pressure relief, followed by loss of power conversion and secondary steam relief system and AFW.

This accident sequence is dominated by failure of vital AC inverter #11 which fails the power conversion and secondary steam relief system and the motor-driven pump of the AFW system requiring another AFW failure to fail AFW. Direct failure of the power conversion and secondary steam relief system by local fault and failure of the AFW is also included. Replacing the appropriate AFW failure rate with the spurious isolation probability of occurrence results in 2.6E-08 contribution to accident sequence T4ML from spurious isolation of the AFW (see Appendix A for calculations).

T1L Loss of offsite power followed by failure of AFW.

This accident sequence consists of two single event cutsets and many double event cutsets in the AFW system. Replacing the appropriate AFW events with spurious isolation events results in 4.1E-08 contribution for accident sequence T1L to the CDF for spurious AFW isolation (see Appendix A for calculations).

T3ML Transient requiring primary system relief followed by loss of power conversion and secondary steam relief system and AFW.

This accident, similar to T2ML, is dominated by loss of vital AC inverter #11 which fails the power conversion and secondary steam relief system and the motor-driven pump of the AFW, requiring another failure in the AFW system to fail the total system. Direct failure of the power conversion and secondary steam relief system and the AFW is also covered. Replacing the AFW failure rates with the appropriate spurious isolation of AFW events results in 7.1E-09 core damage contribution for accident sequence T3ML by spurious AFW isolation (see Appendix A for calculations).

T1LCC' Loss of offsite power followed by loss of AFW, CSSI (containment spray system injection), and CARCS (containment air recirculation and cooling system).

This accident is the same as T1L except for the additional failures of the CSSI and CARCS. Replacing the appropriate AFW events in the cutsets with spurious isolation of AFW events results in 8.6E-09 contribution to the CDF for accident sequence T1LCC' by spurious AFW isolation (see Appendix A for calculations).

4.1.2.3 Operator Failure To Lock Out Isolation System During Cooldown. This will not cause a problem at Plant A (CE) because the AFW isolation signal is generated from a high steam generator differential pressure. During a long term cooldown the steam generators will remain at approximately the same pressure; thus, the isolation signal will not be actuated.

4.1.2.4 Feedwater Line Break Initiated Transient. This accident sequence considers the impact of the AFW isolation system following a feedwater line break. Generally, the affected steam generator is isolated to prevent pumping water out of the break. Failure to isolate the affected steam generator could lead to the failure of the remaining trains of AFW due to the diversion of a sufficient amount of flow from the break, which would fail the AFW function. Because Plant A (CE) has flow limiting valves in the system headers, this sequence is not affected by the removal of the AFW isolation system. No operator action is required to prevent the diversion of AFW flow out of the break.

4.1.2.5 Main Steam Line Break Initiated Transient. This section evaluates the impact of removing the AFW isolation system during a steam line break accident. This accident involves a transient initiated by a steam line break. Steam line break accident sequences that lead to core damage have such a low CDF that they are not included in this analysis. The primary concern due to this postulated accident sequence is containment failure due to overpressurization. The frequency of occurrence was calculated on a generic basis in NUREG-0933[3] as 1.0E-06. The NUREG-0933[3] analysis was used because the pipe rupture frequency and operator error and containment failure rates are consistent with similar events found in the PRAs[6,11,13-15] used in this evaluation and other documents[5,7,9,12,18,19] that contain generic failure rates. The NUREG-0933[3] evaluation is:

1.0E-03 * 0.1 * 0.01 = 1.0E-06

where the frequency of steam and feedwater line breaks is estimated as 1.0E-03, with 10% (0.1) of these assumed to be steam line breaks. Failure of the operator to manually isolate the affected steam generator has been estimated as 0.01. NUREG-0933[3] also assumes that, given the occurrence of this sequence of events, the probability of containment failure is 0.03. It should be noted that NUREG-0933[3] considers this a "highly conservative assumption." Using this value, the estimated frequency of containment failure due to a steam line break is 3.0E-08. This value will be used later in determining the impact this issue will have on consequences (total man-rem).

4.1.3 Total CDF Contribution for Plant A (CE). At plant A (CE), removing the automatic AFW isolation system will decrease the CDF by 5.5E-07. This is determined by adding the contributors from each accident sequence previously discussed as shown below:

T2L	4.7E-07
T4ML	2.6E-08
T1L	4.1E-08
T3ML	7.1E-09
T1LCC'	8.3E-09
Total	5.5E-07

For this plant there are no new accident sequences leading to core damage caused by removing the automatic AFW isolation system.

Although removal of the AFW isolation system is expected to have a negative impact on consequences through the steam line break accident sequence, because of the increased probability of

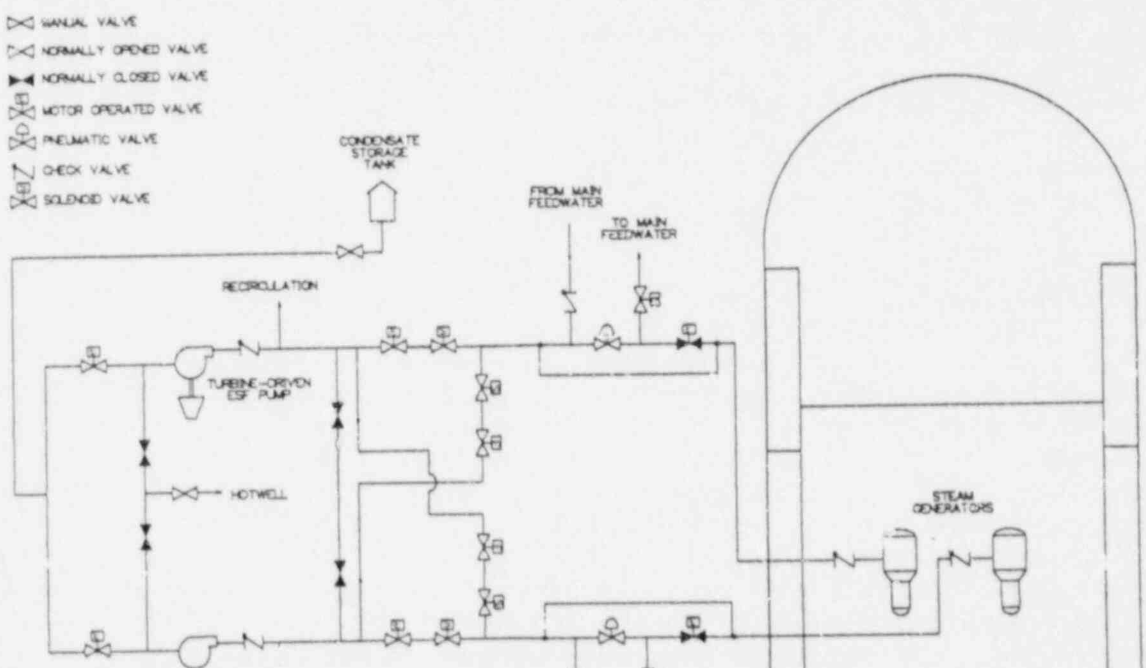

Figure 2. Plant B (B&W) emergency system flow schematic.

containment failure, no increase in the core damage frequency due to feedwater line breaks is expected because the plant has flow restrictors in the AFW supply headers which will prevent flow being diverted to the depressurized steam generator or pump failure due to cavitation or pump run out. Therefore, the net impact of this issue (i.e., removal of the AFW isolation system) is a CDF reduction of 5.97E-07.

4.2 Plant B

The Babcock and Wilcox AFW description and evaluation are based on NUREG/CR-2515, *Plant B Safety Study*[11] along with the description and effects of the emergency feedwater initiation and control system (EFIC) which was installed after the Safety Study was issued.

4.2.1 System Description. Plant B is a PWR designed by Babcock and Wilcox (B&W) with two once-through steam generators. At typical B&W plants, the main feedwater system can be used for supplying feedwater during startups and shutdowns; thus, an AFW system is not required for this purpose. They do have, however, a comparable system that can maintain steam generator water levels should the main feed system be out of service. This system is called the Emergency Feedwater System (EFS). The EFS (Figure 2) has one steam turbine-driven pump train and one motor-driven pump train. The two trains are cross-connected so that each pump supplies both steam generators through each generator's single feed header.

The Emergency Feedwater Initiation and Control (EFIC) system controls the operation of the EFS. The system has four actuation and monitoring channels and uses a logic technique called 1 out of 2 taken twice. This means that actuation signals from channels A and C or A and D or B and C or B and D will initiate the EFS. The actual initiation circuitry is located in the channel A and B cabinets. Signals from channels C and D combined in the A and B cabinets. Channel A actuates the motor-driven pump train and channel B actuates the turbine-driven pump train. The EFS will be actuated by (a) loss of main feed pumps with reactor power greater than 20%, (b) less than six inches of water level in either steam generator, (c) loss of reactor coolant pumps, (d) less than 600 psig in either steam generator, and (e) high pressure injection actuation on channel A and B Engineered Safeguards Actuation System (ESAS). After initiation, the EFIC system controls the steam generator water level change as a function of steam generator pressure.

The EFIC system initiation logic module contains logic that will isolate the main steam line and the main feedwater system on a steam generator pressure less than 600 psig in either steam generator. Only the main steam line on the steam generator with the low pressure is isolated. Two actuation signals are required to cause the isolation. A vector logic module in the EFIC system will isolate the EFS from a steam generator indicating rupture conditions. There is one header from the motor-driven pump to each steam generator and one header from the turbine-driven pump to each steam generator with two isolation valves in series in each header. Channel A will hold open or close one set of valves in the motor-driven pump headers, and channel D will do the same for the other set of valves. Channels B and C will do the same for the turbine-driven pump headers. If one steam generator is below 600 psig, the vector logic modules in the respective EFIC channels will close the isolation valves to that steam generator. If both steam generators are below 600 psig and there is less than 125 psi difference between the generators, all EFS isolation valves will be held open or allowed to be controlled by the EFS water level signals. If both generators are below 600 psig and there is a greater than 125 psi pressure difference, the EFS will be isolated from the steam generator with the lower pressure. Only one signal is required to close the appropriate valves.

The EFS does not provide flow restrictors or control valves that limit flow to a ruptured steam generator. Based on the referenced PRA, the EFS is assumed successful if at least one pump is supplying water to at least one steam generator.

4.2.2 Plant B (B&W) Sequence Analysis. Like plant A, only two spurious EFS isolation sequences are of interest for this plant design. The first sequence is: (a) a fault in the EFIC system causes isolation of one steam generator, (b) a common mode fault causes the other steam generator to isolate, and (c) the operator fails to open at least one valve in either header. Approximately 20 minutes are available for the operator to open the valve. If event probabilities similar to plant A are assumed, this sequence will have a probability of 7.2E-05 * 0.05 * 0.04 = 1.44E-07.

Similar event probabilities can be assumed because the actuation systems at plants A and B are similar. Also, although the recovery time is shorter, there would be immediate notification of the isolation and the recovery actions are simple; therefore, the recovery failure probability of 0.04 is used as a reasonable failure rate probability.

The other event sequence is a fault in the EFIC system that causes isolation of one steam generator while EFS to the other steam generator is isolated for some other reason, such as valve failure or valve maintenance, and the operator fails to recover flow to one of the steam generators. Using event probabilities similar to those above, the probability of this event sequence is 7.2E-05 * 0.04 = 2.88E-06.

Other spurious actuations of the EFS isolation system, such as independent spurious signals to both steam generators, have a very low probability of occurrence, on the order of 5.0E-09, and were not considered in this evaluation.

4.2.2.1 Spurious Isolation-Caused Transient. A spurious actuation of the EFS isolation system at plant B will not cause a total loss of feedwater transient because the signal to isolate the main steam line and the main feed line comes from a different logic module than the signal that closes the EFS isolation valves. Thus, two spurious actuation signals are required to initiate a transient and fail the EFS at the same time. This sequence has a CDF contribution of about 5.0E-10, which is about two orders of magnitude less than the CDF contribution for other accident sequences with EFS isolation

4.2.2.2 Loss of EFS During Another Transient Caused by Spurious Isolation. The Plant B (B&W) PRA[11] calculates loss of EFS due to all causes for two conditions: with offsite power available and without offsite power. The probabilities for these two events are:

With offsite power	3.4E-04
Without offsite power	1.8E-03

Spurious EFS isolation contributes 5.8E-07 to each of these failure rates. This was determined by multiplying the failure probability determined in Section 4.2.2 for spurious isolation of both steam generators by four, since the event could happen four ways, and by substituting the previously determined single steam generator isolation probability for single steam generator failures in the EFS fault trees and then adding the results. This was necessary because the spurious isolation cutsets are all below the dominant cutsets used in the PRA[11] calculation of the EFS failure probability.

Five accident sequences contributing to CDF include failure of the EFS. The EFS isolation system contribution to these accident sequences was determined by dividing the PRA[11] calculated accident sequence core damage frequency by the appropriate EFS failure probability used in the PRA[11] and multiplying by the isolation system failure probability determined above. The results of these calculations are shown below:

T1-T1A (MLU) — Transient that initially has the heat sink available with subsequent failure of the heat sink, the EFS, and primary makeup (feed and bleed).

The spurious isolation events contribution to the CDF is 7.0E-09 (see Appendix B for calculation).

T2A (MLU) — Loss of offsite power which fails the secondary system followed by failure of the EFS and primary makeup.

The spurious isolation events contribution to the CDF is 4.5E-09 (see Appendix B for calculation).

T2A (MLUO) — Same as the T2A(MLU) accident sequence with the addition of failure of the containment pressure reduction system.

The spurious isolation events contribution to the CDF is 1.7E-08 (see Appendix B for calculation).

T2A (MLU) Same as the T2A(MLUO) above accident sequence except the containment spray system fails instead of the containment pressure reduction system.

The spurious isolation events contribution to the CDF is 8.0E-10 (see Appendix B for calculation).

T2-T2A (MLU) This is a loss of secondary system transient with subsequent failure of the EFS and primary makeup.

The contribution of spurious isolation events to the CDF is 1.5E-08 (see Appendix B for calculation).

4.2.2.3 Operator Failure To Lock Out Isolation System During Cooldown. The EFIC system will not initiate automatic isolation of the EFS during a long term cooldown as long as the steam generator pressures remain within 125 psi of one another. The EFIC vector logic module contains logic that will maintain the EFS isolation valves open and allow steam generator level control as long as both steam generators are below 600 psig and there is less than a 125 psi pressure difference between them. No operator action is required to activate this logic.

4.2.2.4 Feedwater Line Break Initiated Transient. The impact of removing the AFW isolation system on sequences initiated by a feedwater line break is analyzed in this section. Plant B (B&W) does not have flow limiting devices in the EFS headers; thus, the operator must manually isolate a ruptured steam generator to prevent failure of the EFS due to all flow being diverted to the ruptured steam generator or failure of the pumps due to cavitation or pump runout. The EFS system will fail if the operator does not isolate a ruptured steam generator, because the EFS pumps are cross connected, which would allow all of the EFS flow to be diverted to the steam generator with the low pressure. Using failure rate data similar to that used in the NUREG-0933[3] analysis, the increase to the CDF due to removal of the EFS isolation system is estimated to be 1.0E-03*0.01*0.014 = 1.4E-07.

The 1.0E-03 is the frequency for rupture of a large pipe, taken from NUREG-0933.[3] This number agrees well with values generated from WASH-1400[9] and IEEE-STD-500.[12] The 0.01 term is the probability that the operator fails to isolate a ruptured steam generator, and 0.014 term is the probability of failure of feed-and-bleed taken from the Plant B (B&W) PRA.[11] The probability for the operator failing to isolate the affected steam generator was taken from NUREG-0933[3] because it was judged to be a reasonable value for the Plant B (B&W) system.

4.2.2.5 Steam Line Break Initiated Transient. This section describes the impact of removing the EFS isolation system for steam line break initiated transient at Plant B (B&W). This accident is identical to that described in Section 4.1.2.5 of this report for Plant A (CE). The primary concern due to this postulated accident sequence is containment failure due to overpressurization. The frequency of occurrence was calculated on a generic basis in NUREG-0933[3] as 1.0E-06. The NUREG-0933[3] analysis was used because the pipe rupture frequency and operator error and containment failure rates are consistent with similar events found in the PRAs (References 6, 11, 13-15) used in this evaluation and other documents (References 5,10,12,18,19) that contain generic failure rates. The NUREG-0933[3] evaluation is 1.0E-03 * 0.1 * 0.01 = 1.0E-06, where the frequency of steam and feedwater line breaks is estimated as 1.0E-03, with 10% (0.1) of these assumed to be steam line breaks. Failure of the operator to manually isolate the affected steam generator has been estimated as 0.01. NUREG-0933[3] also assumes that, given the occurrence of this sequence of events, the probability of containment failure is 0.03. It should be noted that NUREG-0933[3] considers this a "highly conservative assumption." Using this value, the estimated frequency of containment failure due to a steam line break is 3.0E-08. This value will not be combined with those previously presented since it does not involve core damage. However, this value will be used later in determining the impact this issue will have on consequences (total man-rem).

4.2.3 Total Contribution To CDF for Plant B (B&W). The total estimated contribution to the CDF for accident sequences involving spurious actuation of the steam generator isolation system is shown below:

Loss of EFS sequences:

T1-T1A(MLU)	7.0E-09
T2A(MLU)	4.5E-09
T2A(MLUO)	1.7E-08
T2A(MLUO')	8.0E-10
T2-T2A(MLU)	1.5E-08
Total	4.4E-08

It is estimated that there is an increase in the CDF of 1.4E-07 due to the feedwater line break sequence, if the EFS isolation system is removed (Section 4.2.2.5).

For Plant B (B&W) the net change to the CDF for deleting the EFS isolation system is an increase of 9.6E-08. This consists of a decrease in CDF of 4.4E-08 caused by deleting the isolation system and an increase in CDF of 1.4E-07 due to the increased frequency of accident sequences that the EFS was protecting against.

4.3 Plant BB (B&W)

The Plant BB (B&W) emergency feedwater system (EFS) description and evaluation are based on NUREG/CR-2515, *Plant BB Safety Study*.[11] This plant is the same as the Plant B (B&W) design except that it does not have the EFIC system.

4.3.1 System Description. The EFS (Figure 3) is actuated by low steam generator water level in both steam generators or loss of both main feedwater pumps. A steam line rupture matrix will activate to close the main steam line isolation valves, the main feedwater valves, and the EFS valves on a steam generator indicating a steam line rupture. Only the affected steam generator is isolated; the main steam isolation valves on the unaffected steam generator remain open. The steam line rupture matrix is actuated by two pressure switches, one set at 725 psig and one set at 600 psig. There are two independent actuation channels (A and B) for each steam generator.

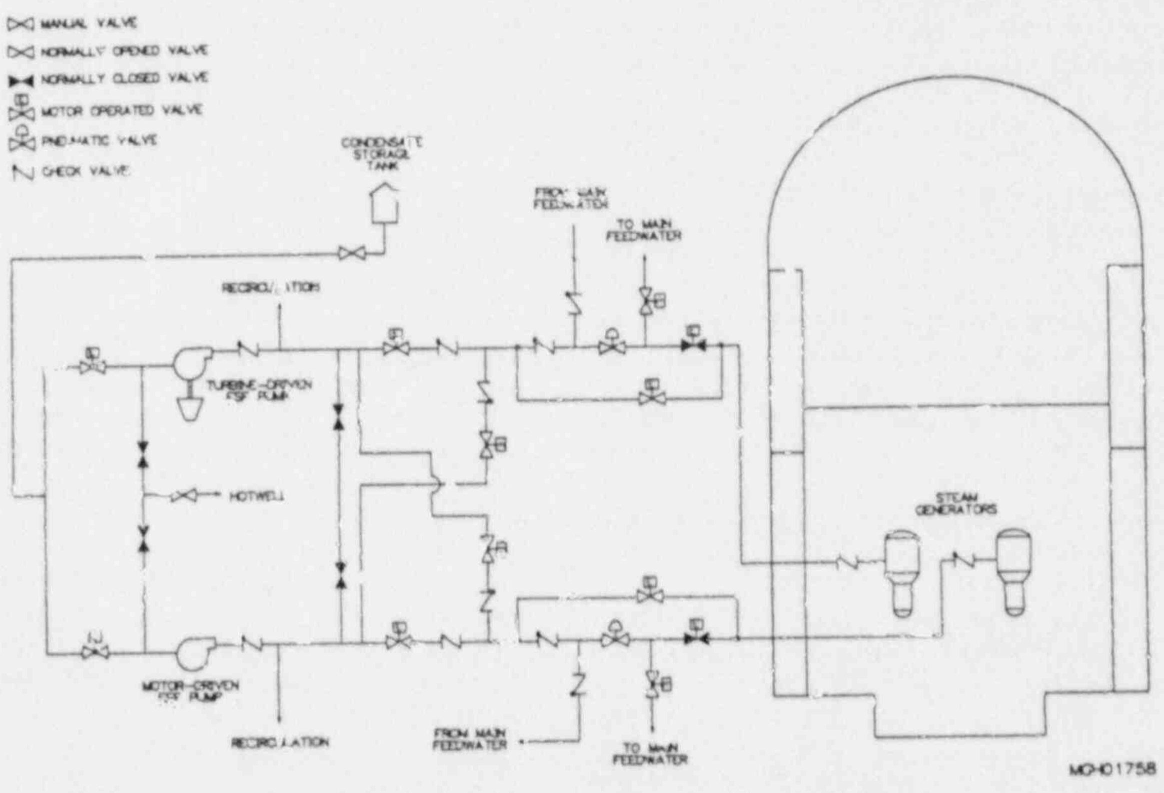

Figure 3. Plant BB (B&W) emergency feedwater system flow schematic.

The EFS discharge lines do not have flow restrictors or control valves that will automatically limit flow to a ruptured steam generator. The EFS is assumed successful if at least one pump is supplying water to at least one steam generator.

4.3.2 Plant BB (B&W) Sequence Analysis. Like Plant B (B&W), only two spurious EFS isolation events are of interest for this study. The first spurious isolation event is (a) a fault in the steam line rupture matrix relay causes isolation of one steam generator, (b) a common mode fault causes the other steam generator to isolate, and (c) the operator fails to open one valve in either header. Approximately 20 minutes are available to perform the recovery. If event probabilities similar to Plant B (B&W) are assumed, this event sequence will have the probability of 7.2E-05*0.05*0.04 = 1.44E-07.

The other event sequence is a fault in the steam line rupture matrix relay that causes isolation of one steam generator while EFS to the other steam generator is isolated for some other reason (such as valve failure or valve maintenance) and the operator fails to recover flow to one of the steam generators. Using the same event probabilities as those used for the Plant B (B&W) evaluation, the probability of this event would be 7.2E-05*0.04 = 2.88E-06. Because there are two actuation channels, two cutsets will contribute to the core melt frequency. As discussed for Plant A (CE) both actuation channels will be active for some accident sequences and only one channel will be active for others.

Other spurious actuations of the EFS isolation system, such as spurious signals to both steam generators, have a very low probability of occurrence, on the order of 5.0E-9, and were not considered in this evaluation.

4.3.2.1 Spurious Isolation Caused Transient. At Plant BB (B&W) a spurious steam generator isolation signal can initiate a transient that could lead to core damage if the operator failed to recover flow to at least one of the steam generators and he failed to correctly initiate feed and bleed. The Safety Study[8] used to evaluate this plant indicates that the probability of feed and bleed failure is 0.014. The spurious isolation of both steam generators coupled with operator failure to recover EFS flow, and failure of feed and bleed would have a CDF of:

1.0E-07*8760*0.75*0.05*0.04*0.014 = 1.8E-08

Where:

1.0E-07/hr is the spurious failure rate of a relay taken from IEEE-STD-500-1984.[12]

8760 is the number of hours in a year and 0.75 is the approximate annual reactor operating factor,

0.05 is the probability of a common mode failure of the other steam generator isolation relay,

0.04 is the probability of operator failure to recover EFS flow, and

0.014 is the probability of feed-and-bleed failure as reported by the Safety Study.[10]

The spurious isolation of one steam generator when EFS to the other steam generator is isolated, coupled with the failure of the operator to recover EFS flow and establish feed-and-bleed, was not considered due to its relatively low frequency when compared to the above CDF.

4.3.2.2 Spurious Isolation Caused Loss of EFS During Another Transient. The Plant BB (B&W) PRA calculates loss of EFS due to all causes for two conditions: with offsite power available and without offsite power. The probabilities for these two events are:

With offsite power	3.4E-04
Without offsite power	1.8E-03

Spurious EFS isolation contributes 5.8E-07 to each of these failure rates. This was determined by multiplying the failure probability determined in Section 4.2.2 for spurious isolation of both steam generators by four, since there are four ways the event could happen, and by substituting the previously determined single steam generator isolation probability for single steam generator failures in the EFS fault trees and then summing the results. This was necessary because the spurious isolation cutsets are all below the dominant cutsets used in the PRA calculation of the EFS failure probability. Five accident sequences contributing to CDF include failure of the EFS. The EFS isolation system contribution to these accident sequences was determined by dividing the PRA calculated accident sequence CDF by the appropriate EFS failure probability used in the PRA and multiplying by the isolation system failure probability determined above. The results of these calculations are shown below:

T1-T1A (MLU) — Transient that initially has the heat sink available with subsequent failure of the heat sink, the EFS, and primary makeup (feed-and-bleed).

The spurious isolation events contribution to the CDF is 7.0E-09 (see Appendix B for calculation).

T2A (MLU) — Loss of offsite power which fails the heat sink followed by failure of the EFS and primary makeup.

The spurious isolation events contribution to the CDF is 4.5E-09 (see Appendix B for calculation).

T2A (MLUO) — Same as the T2A(MLU) accident sequence with the addition of failure of the containment pressure reduction system.

The spurious isolation events contribution to the CDF is 1.7E-08 (see Appendix B for calculation).

T2A (M[illegible]) — Same as the T2A(MLUO) accident sequence except thecontainment spray system fails instead of the containment pressure reduction system.

The spurious isolation events contribution to the CDF is 8.0E-10 (see Appendix B for calculation).

T2-T2A (MLU) — This is a loss of heat sink transient with subsequent failure of the EFS and primary makeup.

The spurious isolation events contribution to the CDF is 1.5E-08 (see Appendix B for calculation).

4.3.2.3 Steam Generator Pressure Transient Caused Isolation. Another accident sequence that must be considered for Plant BB (B&W) is a low pressure transient in the steam generators that causes the steam generators to be isolated even though there is no rupture in the system. Because of the relatively small water to steam ratio in the once-through steam generators, they are more sensitive to large pressure transients than the other steam generator designs. This is because of the smaller volume of water near saturation conditions that could flash to steam to keep the pressure up, and because of the design of the isolation actuation signal. Plant BB (B&W) actuates the EFS isolation system on low steam generator pressure, whereas the other plants evaluated actuated the isolation system on a high steam pressure differential between steam generators.

The postulated accident sequence is some type of event that causes a pressure transient in the steam generators, such as a sudden opening of the steam relief valves, that causes the steam generator pressure to drop below the isolation actuation setpoint. To cause total EFS isolation, both steam generators must experience the same pressure transient. Once the steam generators are isolated, core damage will occur if the operator fails to recover feedwater to the steam generators (either main feed or EFS) and feed-and-bleed of the primary system fails.

A method similar to the one used by NUREG-0933[3] for the evaluation of this accident sequence was used, except some of the failure rates were made more representative of the specific plant circumstances. From the NUREG-0933[3] analysis, sudden opening of the safety relief valves has a frequency of 0.04 occurrences per year and the probability of the operator failing to recover EFS flow is 0.01. The 0.1 probability of the sudden relief valve opening resulting in a pressure decrease to below the isolation system actuation setpoint used by NUREG-0933[3] is considered too high for this situation. The isolation system actuation setpoints are 725 psig and 600 psig and typical relief valve setpoints are 1150 psig; thus, the steam generator pressure must fall almost 50 percent to cause isolation. Therefore, a probability of 0.01 seems more reasonable for a pressure decrease of this amount. The PRA[11] used for evaluating this plant provides a feed and bleed failure probability of 0.014, which is utilized in this evaluation.

The contribution to the CDF for this accident sequence is 0.04*0.01*0.01*0.014 = 5.6E-08.

4.3.2.4 Operator Failure To Lock Out Isolation System During Cooldown. Because the EFS isolation signal is derived from steam generator pressure, the actuation point will be passed during the cooldown; thus, the operator will have to lock out the isolation system to avoid an inadvertent loss of EFS.

The Plant BB (B&W) Safety Study[11] does not evaluate this event and provides minimal information for evaluating this accident sequence. An estimate of this sequence was made in the NUREG-0933[3] evaluation of this issue. The analysis presented here duplicates that estimate with the exception of the value employed in NUREG-0933[3] for the failure of feed-and-bleed (0.014/demand). The frequency of nonrecoverable loss of main feedwater (0.64), the probability of operator failure to lock out the isolation system (0.01), and the probability of failing to recover the EFS were extracted from the NUREG-0933[3] analysis. Upon review of both NUREG-0933[3] and the Plant BB (B&W) PRA,[11] it was judged that the PRA[11] value for feed and bleed is more appropriate in estimating this event at Plant BB (B&W) because the PRA[11] value is based on a specific analysis of the Plant BB (B&W) facility whereas the NUREG-0933[3] value is a very conservative generic value. It should also be noted that the NUREG-0933 value for the transient initiating event was evaluated to be more appropriate than that of 1.78 as reported in the PRA.[11] Based upon these values, the estimate of the contribution to CDF from this sequence is estimated as 0.64 * 0.01 * 0.01 * 0.014 = 8.96E-07.

4.3.2.5 Feedwater Line Break Initiated Transient. The impact of removing the AFW isolation system on sequences initiated by a feedwater line break is analyzed in this section. Plant BB (B&W) does not have flow limiting devices in the EFS headers; thus, the operator must manually isolate a ruptured steam generator. The EFS system will fail if the operator does not isolate a ruptured steam generator because the EFS pumps are cross-connected, which would allow all of the EFS flow to be diverted to the steam generator with the low pressure. Using failure rate data similar to that used in the NUREG-0933[3] analysis, the increase to the CDF due to removal of the EFS isolation system is estimated to be 1.0E-03*0.01*0.014 = 1.4E-7.

The 1.0E-03 is the frequency for rupture of a large pipe, taken from NUREG-0933.[3] This number agrees well with values generated from WASH-1400[9] and IEEE-STD 500.[12] The 0.01 is the probability that the operator fails to isolate a ruptured steam generator, and 0.014 is the probability of failure of feed and bleed extracted from the PRA[11] used to evaluate Plant BB (B&W). The value for the operator failing to isolate the affected steam generator was taken from NUREG-0933[3] because it is representative of the complexity for recovery actions and agrees with similar events in the reference PRA.[11]

4.3.2.6 Steam Line Break Initiated Transient. This section describes the impact of removing the EFS isolation system at Plant BB (B&W) for steam line break initiated transients. This accident is identical to that described in Section 4.1.2.5 of this report for Plant A (CE). The primary concern due to this postulated accident sequence is containment failure due to overpressurization. The frequency of occurrence was calculated on a generic basis in NUREG-0933[3] as 1.0E-06. The NUREG-0933[3] analysis was used because the pipe rupture frequency and operator error and containment failure rates are consistent with similar events found in the PRAs (References 6,11,13-15) used in this evaluation and other documents (References 5,10,12,18,19) that contain generic failure rates. The NUREG-0933[3] evaluation is 1.0E-03 * 0.1 * 0.01 = 1.0E-06, where the frequency of steam and feedwater line breaks is estimated as 1.0E-03, with 10% (0.1) of these assumed to be steam line breaks. Failure of the operator to manually isolate the affected steam generator has been estimated as 0.01. NUREG-0933[3] also assumes that, given the occurrence of this sequence of events, the probability of containment failure is 0.03. It should be noted that NUREG-0933[3] considers this a "highly conservative assumption." Using this value, the estimated frequency of containment failure due to a steam line break is 3.0E-08. This value will not be combined with those previously presented since it does not involve core damage. However, this value will be used later in determining the impact this issue will have on consequences (total man-rem).

4.3.3 Total Contribution To Core Melt Frequency for Plant BB (B&W). The total estimated contribution to the CDF for accident sequences involving spurious actuation of the steam generator isolation system is shown below:

Initiates transient	1.8E-8
Loss of EFS sequences:	
T1-T1A(MLU)	7.0E-09
T2A(MLU)	4.5E-09
T2A(MLUO)	1.7E-08
T2A(MLUO')	8.0E-10
T2-T2A(MLU)	1.5E-08
TOTAL (loss of EFS)	4.4E-08
Pressure transient in steam generator	5.6E-08
Operator does not lock out	8.96E-07
Total EFS contribution to CDF	1.04E-06

The EFS contribution to CDF is dominated by the accident sequence involving the operator failing to lock out the steam generator isolation system during a long term cooldown. It should be noted that the value of 0.01 for failure to lock out the EFS isolation system employed in the evaluation of this sequence is conservative, based upon the consideration that this action will be required "late" in the sequence and that this action should be familiar to the operator since it is similar to actions required during normal plant shutdown. However, due to uncertainties in the actual conditions existing during such a sequence, and lacking a detailed plant specific analysis of this operator action, it was decided to employ the NUREG-0933[3] value for conservatism.

It is estimated that there is an increase in the CDF of 1.4E-7 due to the feedwater line break sequence. For Plant BB (B&W), the net change to the CDF for deleting the EFS isolation system is a decrease of 9.0E-07. This consists a decrease in CDF of 1.04E-06 caused by deleting the isolation system and an increase in CDF of 1.4E-07 due to the increased CDF of accident sequences that the automatic EFS isolation system was designed to prevent.

4.4 Plant C

The Plant C (W) AFW system description and evaluation is based on NUREG/CR-4550 Vol. 5, *Analysis of Core Damage Frequency From Internal Events: Plant C*[13] and EPRI NP-3382, *Plant C Nuclear Power Plant Availability and Safety Assessment.*[14]

4.4.1 System Description. Plant C is a Westinghouse (W) designed reactor system with four U-tube steam generators. The AFW system (Figure 4) has three pumps, one turbine-driven and two motor-driven. The turbine-driven pump supplies all four steam generators and each motor-driven pump supplies two steam generators. Each steam generator has two headers supplying AFW, one from the motor-driven pump and one from the turbine-driven pump. Each header has its own level control valve.

Plant C (W) does not have a dedicated AFW isolation system to isolate a ruptured steam generator, but does have components that will provide the same function. All main steam isolation valves will close on high containment pressure or high steam flow (steam line rupture indication). The containment pressure requires two of three coincidence signals to actuate and the high steam flow requires two of four coincidence signals. Each AFW header has a pressure switch that will close the level control valve in that header on a low downstream pressure signal that indicates a steam or feedwater line rupture.

The AFW is assumed to fail if less than two steam generators are supplied with feedwater for all accident sequences evaluated by the PRA[13] except anticipated transients without scram (ATWS). ATWS accident sequences require AFW flow to at least three steam generators to provide adequate cooling.

4.4.2 Plant C (W) Sequence Analysis. The only spurious AFW isolation event at Plant C (W) that will cause a significant contribution to CDF is receipt of a spurious signal to close one of the level control valves on the operating motor-driven pump, when the turbine-driven pump system and one of the motor-driven pump systems are out of service. With all AFW pumps operating, six level control

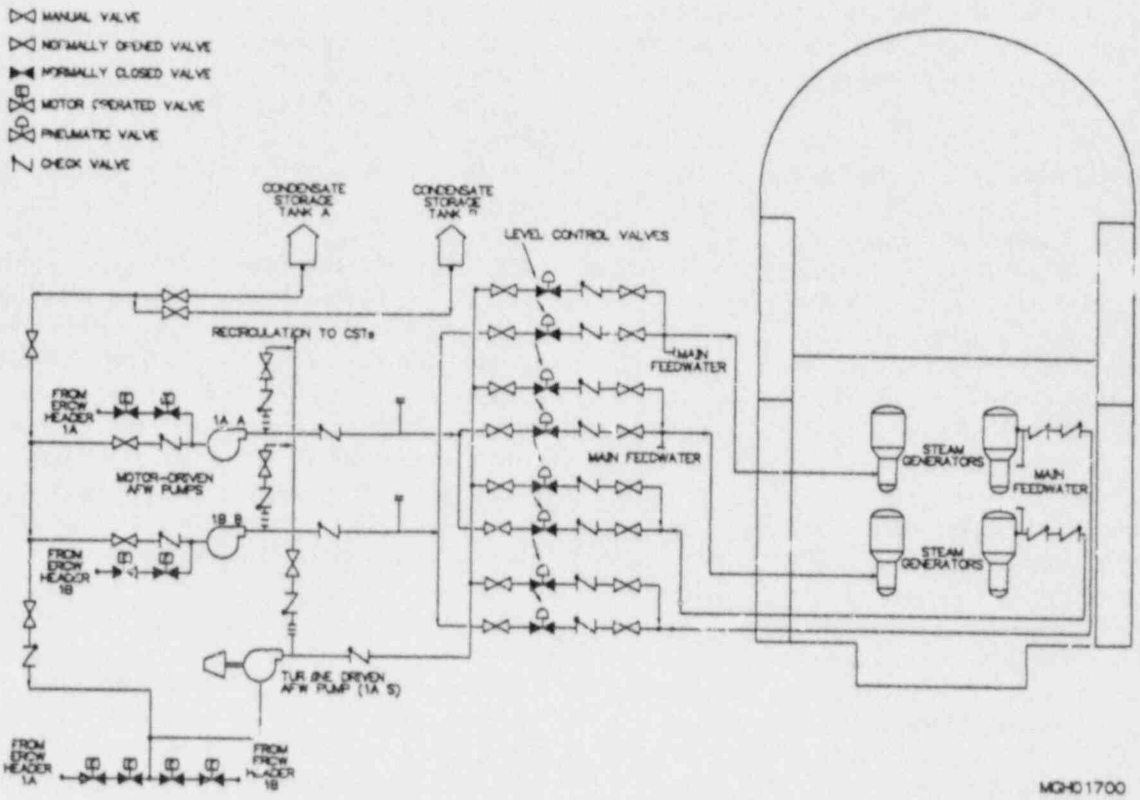

Figure 4. Plant C (W) auxiliary feedwaster system flow schematic.

valves must close to fail the AFW. With the turbine-driven pump or both motor-driven pumps out of service, three level control valves (LCV) must close to fail the AFW. Both of the later events have a very low probability of occurrence.

Using a failure probability of 1.0E-03 for an inadvertent closure of a motor operated valve and because there are two valves that could close to cause loss of AFW function, the probability of loss of AFW for one motor-driven pump is 2.0E-03. The valve failure rate used above is based on actual plant data from plant A (CE).[6]

Because the recovery actions must occur during a very stressful time,i.e., during a significant transient, and two out of three AFW trains are already failed, the recovery failure probability should be about twice as high as the value used in previous evaluations, or about 0.10. Therefore, the probability of spurious isolation of the AFW and operator failure to recover is 2.0E-04.

4.4.2.1 Spurious Isolation Caused Transient. A spurious AFW isolation signal will not cause a total loss of feedwater transient at plant C (W) because the signals that isolate the AFW (low pressure on at least six pressure switches) are not the same signals that isolate the main steam line (which trips the main feed pumps). The probability of spurious signals closing six level control valves combined with the probability of spurious signals for two high steam flow or two high containment pressure channels closing the main steam isolation valves is extremely remote.

4.4.2.2 Spurious AFW Isolation During Transients That Requires AFW. The contribution to the CDF from spurious isolation of the AFW is dominated by two accident sequences, Tdc1L1P1 and Tdc2L1P1. These sequences are essentially the same: both fail one of the motor-driven pumps. The contribution to CDF due to spurious closure of one of the level control valves on the operating motor-driven pump is determined by substituting the spurious isolation probability determined in Section 4.3.2 for other motor-driven pump system failures in the accident sequence cutsets with turbine-driven pump failures. For accident sequence Tdc1L1P1 this results in a CDF contribution of 2.0E-08.

Because accident sequence Tdc2L1P1 is the same except the operating and failed motor-driven pumps are switched, the total contribution will be twice the value calculated above or 4.0E-08.

4.4.2.3 Operator Failure To Lock Out AFW Isolation System.255D As previously mentioned in Section 4.4.1, Plant C (W) does not have a dedicated AFW isolation system, but it does have pressure switches that will close the respective level control valve on each of the eight headers supplying AFW to the steam generators. The operator will have to lock out the pressure switches during a long term cooldown; however, it will be a very low probability event to fail to lock out the isolation system because the AFW system is the normal feed system for startups and shutdowns, there is ample time for any recovery actions. Thus, lockout of the isolation system is considered to be a routine event for the operators. Therefore, the contribution to the CDF is negligible.

4.4.2.4 Feedwater Line Break Initiated Transient. This section describes the impact of removing the AFW isolation system on a main feedwater line break. If the pressure switches controlling the level control valves are deactivated, isolation of the AFW due to a main feedwater line break would be defeated because the level control valves would remain open and coolant would continue to flow to the depressurizing steam generator. The AFW system would not fail immediately if the operator did not act to isolate a ruptured steam generator, because the flow from the turbine-driven pump and one of the motor-driven pumps would be diverted to the ruptured steam generator, but the other motor-driven pump would still provide flow to two operable steam generators. As previously noted, the failure criteria for AFW is AFW flow to less than two steam generators for most accident sequences and flow to less than three steam generators for ATWS events. ATWS events did not contribute significantly to CDF at Plant C (W) (less than 1.0E-08); thus, they are not a significant contributor to the CDF for the AFW isolation system. Therefore, the non-ATWS events, with failed AFW, will overshadow the ATWS events. The increase to the CDF for defeating the AFW header isolation pressure switches is estimated to be 1.0E-03*0.01*4.4E-02 = 4.4E-07.

The 4.4E-02 value is the probability of the failure in the AFW system and the failure of feed-and-bleed calculated from the Plant C (W) dominant accident sequences. The other two values are the same as those used in the Plant B (B&W) evaluation above, which were extracted from NUREG-0933.[3]

4.4.2.5 Steam Line Break Initiated Transient. This section describes the impact of removing the AFW isolation system from Plant C (W). This accident sequence is identical to that described in Section 4.1.2.5 of this report for Plant A (CE). The primary concern due to this postulated accident sequence is containment failure due to overpressurization. The frequency of occurrence was calculated on a generic basis in NUREG-0933[3] as 1.0E-06. The NUREG-0933[3] analysis was used because the pipe frequency, operator error, and containment failure rates are consistent with similar events found in the PRAs (References 6,11,13-15) used in this evaluation and other documents (References 5,10,12,18,19) that contain generic failure rates. NUREG-0933[3] also assumes that, given the occurrence of this sequence of events, the probability of containment failure is 0.03. Although NUREG-0933[3] considers this a "highly conservative assumption," this value was evaluated as not applicable to the Plant C (W) plant, since Plant C (W) has an ice condenser containment. This type of containment has a significantly smaller free volume than the other containment types. For Plant C (W), failure of containment will be conservatively assumed to be 1.0, given the occurrence of the previously described accident sequence. Using this value, the estimated frequency of containment failure due to a steam line break is 1.0E-06. Section 4.1.2.5 of this report contains details of this estimate. This value will not be combined with those previously presented since it does not involve core damage. However, this value will be used later in determining the impact this issue will have on consequences (total man-rem).

4.4.3 Total Contribution to CDF for Plant C (W). The total estimated contribution to CDF for the Plant C (W) automatic AFW isolation system is estimated as 4.0E-08. This comes from sequences in which the turbine-driven pump and one motor-driven pump trains are inoperable and a spurious isolation signal isolates the remaining AFW train from one of the two steam generators that it is supplying. Removing the automatic AFW isolation system features would decrease the CDF by 4.0E-08; however, there is an estimated increase in the CDF of 4.4E-07 from sequences involving feedwater line breaks. For Plant C (W) the net change in the CDF for removing the automatic AFW isolation system is estimated as an increase of approximately 4.0E-07.

4.5 Operator Inadvertently Initiates AFW Isolation System

During the June 1985 Davis-Besse incident, the operator initiated the AFW isolation system accidently. AFW flow had initiated automatically as designed, but the operator, attempting to back it up by manually initiating AFW flow, pushed the wrong set of buttons, which isolated AFW to both steam generators. Although there was immediate notification of the error, it took several minutes to restore AFW flow because of other unrelated events. For plants like Plant A (CE) that do not have feed-and-bleed capability this event could be quite serious because there is no other means of removing decay heat.

Evaluation of the AFW control systems for plants with and without automatic AFW isolation systems shows that the operator of either type of plant has about the same chance of making an error similar to the error made at Davis-Besse. Plants with automatic AFW isolation have controls that allow the operator to manually isolate any or all of the steam generators and they have controls to manually initiate or shutoff the AFW. Plants without the automatic isolation system do not have the specific controls to manually isolate the steam generators, but they do have valve controllers that can isolate any or all of the steam generators and they have controls to manually initiate or shutoff the AFW. Specific plant control panel design and operator training will determine the actual probability of the operator making the error. However, for the same plant with the same level of operator training and human factors engineering, the probability of the operator inadvertently isolating AFW was determined to be approximately equal for the same plant with or without an automatic AFW isolation system. Therefore, operator error of commission has a negligible effect on CDF and is not included in the calculations.

It should be noted that the issue of the operator inadvertently isolating the AFW has been evaluated by Generic Issue 124, AFW System Reliability. Generic Issue 124 combined issues 68, 122.1.a, b, and c, and 125.II.1.b. Issue 122.1.c., Interruption Of AFW Flow, included the event of the operator inadvertently isolating the AFW. Issue 124 concluded that the AFW system was acceptable if the plants had a high AFW system reliability (between 1.0E-04 and 1.0E-05). Plants that did not meet this criteria were identified and were or are being evaluated to propose modifications that will bring them up to the required reliability.

4.6 Technical Findings of CDF Analysis

Four PWRs, one each CE and W designs and two B&W designs, were evaluated to determine the AFW isolation system's contribution to CDF. Three of the plants selected did not have flow restrictors to limit flow to a ruptured steam generator, one of them could not be cooled successfully by feed-and-bleed, and one had a very diverse AFW isolation system.

The evaluation indicates that the effects of the AFW isolation system are strongly dependent on the particular plant's design. The estimated contribution to CDF due to AFW isolation system were reasonably low, but the difference between the highest and the lowest value was an order of magnitude.

At Plant A (CE) removing the isolation system will not cause a failure of the AFW system because the plant has flow restrictors in the AFW headers that limit flow to a ruptured steam generator and maintain flow to the intact steam generators. Removing the isolation system at this plant would decrease the CDF by 5.5E-07. At Plant B (B&W) removing the automatic AFW isolation system would cause AFW system failure without operator action because the plant does not have flow restrictors in the AFW headers and the pump trains are cross connected. Thus, all AFW flow would be directed to the ruptured steam generator and the pumps could be damaged due to low net positive suction head caused by the high flow rate. Removing the automatic isolation system would cause a CDF increase of 9.6E-08. At Plant BB (B&W) removing the automatic isolation system would also cause AFW failure without operator action because the plant does not have flow restrictors in the AFW headers and the pumps are cross-connected. Removing the automatic AFW isolation system would cause a CDF decrease of 9.0E-07. At Plant C (W), only part of the AFW system would fail if the automatic isolation features were removed and the operator took no action to isolate a ruptured steam generator. A ruptured steam generator would cause the flow from one of the motor-driven pumps and the turbine-driven pump to be diverted to the break, but the other motor-driven pump would still supply two intact steam generators, which is the AFW success criterion for most accidents. Removing the automatic isolation system would cause a CDF increase of 4.0E-07. Table 1 summarizes the changes to the CDF caused by removing the automatic AFW isolation system from the four plants evaluated.

5. COST BENEFIT ANALYSIS

5.1 Cost Benefit Analysis Methodology

The consideration of possible plant modifications (section 5.2) is based on the value of the modification in terms of the safety benefit derived, that is, the risk reduction achieved and the cost of implementing the modification (Section 5.3). The modifications focus on increasing the probability of maintaining steam generator water inventory and thereby eliminating the loss of steam generator decay heat removal capability. Best estimates for equipment failure probabilities were used whenever possible in the analyses for core damage. The risk reduction resulting from the proposed modifications is represented by the difference between the base case before any plant modifications and the adjusted case that results from implementing the modifications. Plant specific estimates of the change in the CDFs were combined with containment failure probabilities and generic off-site dose release to calculate the estimated change in risk.

In evaluating the associated change in risk, the containment failure probabilities and the release categories for a specific accident sequence were extracted from the PRAs (References 6, 11, 13-15). The release categories are those defined in WASH-1400.[9] In addition to this analysis, estimated changes in risk were also calculated using the containment failure probabilities and release categories described in the NUREG-0933[3]. These values were utilized to provide a conservative assessment of the change in risk which is assumed to be representative of the change in risk on a generic plant basis.

Estimated public dose in terms of man-rem was assigned to the WASH-1400[9] release categories in accordance with the data presented in NUREG-0933.[3] The data presented in NUREG-0933[3] was calculated based on a typical mid-west site adjusted to reflect the mean of the population density within a 50-mile radius of U. S. nuclear power plants. Other assumptions used in the NUREG-0933[3] calculations and also used in this study due to their generic applicability are:

Table 1. Change to CDF caused by removing the automatic AFW isolation system

Plant	Decrease In CDF Caused By Deactivating AFW Isolation System	Increase In Main Feed Line Break CDF Caused By Deactivating AFW Isolation System	Total Change To CDF
A (CE)	5.50E-07	0	-5.5E-07
B (B&W)	4.40E-08	1.4E-07	+9.6E-08
BB (B&W)	1.04E-06	1.4E-07	-9.0E-07
C (W)	4.00E-08	4.4E-07	+4.0E-07

1. Dose consequences represent whole body population dose commitment (man-rem) received within 50 miles of the site.

2. An exclusion area of one half mile radius was assumed, with a uniform population density of 340 persons per square mile beyond the one half mile radius (this is the projected average 50-mile-radius population density around U. S. LWRs for the year 2000).

3. Evacuation was not considered.

4. Meteorological data was taken from the U. S. Weather Service station at Moline, Illinois.

5. The core inventory at the time of the accident was assumed to be represented by a 3412 MWt (1120 MWe) plant.

6. All exposure pathways were considered, including selected ingestion pathways of which farm land parameters for the State of Illinois were used.

5.2 Description of Modifications

The modification proposed for resolution of this issue consists of electrically disabling the automatic AFW isolation system by disconnecting the automatic enable circuits. This will provide the AFW system with manual control once the system has been activated and will necessitate additional operator training and revised plant operating procedures. A further concern has been identified for plants that use the automatic AFW isolation system to prevent AFW pump runout. If the automatic AFW isolation system were disabled on these plants, further plant modification would be required to prevent pump runout.

A survey performed on all operating PWRs indicated that 27 plants would be affected by this issue. Further, 19 of these 27 plants would be affected by AFW pump runout considerations. These plants, if modified, would require further modification, i.e., installation of pump discharge flow restrictors or throttle valves, to prevent pump runout.

These changes would also require detailed reanalysis of steam and feed line break accidents for *Final Safety Analysis Report (FSAR)* revision and amendments. Technical Specification changes would be required to reflect the modified design and to provide for periodic testing of the modified AFW system.

5.3 Risk Evaluation

To evaluate the proposed modifications on a risk change versus cost basis, the risk change associated with the scenarios of concern was calculated. Utilizing the reduction in CDF calculations presented in Section 4 of this report and the methodology identified in Section 5.1 to determine the containment failure rate and the offsite dose releases, total risk change was estimated using the following relationship:

Change in CDF (events/yr) x Containment Failure Probability x Offsite Radiation Dose (man-rem) = Risk change (man-rem/year)

To calculate the total change to the potential population exposure or risk per plant life due to this issue, the above relationship was extended over the plant life, taking into account plant down time. The total change in population exposure over the remaining plant lifetime is calculated as follows:

Change in Risk (man-rem/year) x Remaining Plant Life (years) x Plant Utilization Factor = Total Change in Population Risk (man-rem)

The potential change in risk due to the proposed AFW modification, for the selected plants, was evaluated using the plant specific containment failure and release category information delineated in the respective PRAs.[6,11,15] To extrapolate the estimated man-rem/year risk to total change in plant risk, the plant life was estimated utilizing the expected remaining lifetime of 23 years, with an associated utilization factor of 75%. These values were taken from the NUREG-0933[3] analysis.

To estimate the change in risk caused by a steam line rupture without a subsequent core damage event for plant C (W), a very conservative analysis was made using a WASH-1400[9] release category. This was required because the consequence analysis used in the Plant C (W) PRA[15] established consequence categories that did not allow a straightforward determination of the fraction of total plant risk due to steam line breaks without core damage. Section 4.4.2.5 calculated the frequency of containment failure from steam line breaks as 1.0E-06. Using a PWR 8 release category from WASH-1400[9], which is a release category that does not does not involve core damage, the release to the environment would be 7.5E-04 man-rem/year. This value is very low, but it is conservative because it represents a release of the radioactive elements contained in the primary coolant whereas the accident being evaluated involves a release of radioactive elements that may be present in the secondary system (a much lower amount). Combining the containment failure frequency and the release rate results in a change in risk of 0.075 man-rem/year or 1.3 man-rem over the estimated plant lifetime. Adding this value to the change in risk calculated for the accident sequences involving automatic AFW isolation system that lead to core damage results in a total plant lifetime risk of 13.3 man-rem/year.

To provide additional information with regard to the potential impact from implementing the proposed modification, simple sensitivity analyses were performed. These analyses consisted of utilizing the NUREG-0933[3] containment failure probabilities to calculate a lower bound for the change in risk. Table 2 presents the various values utilized for the containment failure modes and the consequences associated with a specific PWR release category.

Table 3 shows the estimated change in risk (man-rem) for the plants evaluated (23 years at 75%).

From Table 3, it can be seen that the change in risk values calculated using the NUREG-0933[3] data are much lower than those determined by using the plant specific PRA information. This difference is due to the value used in determining the containment failure probabilities. The containment failure probabilities used in the reference PRAs are based upon WASH-1400[9] containment response analyses, except for Plant C (W), which is based upon the plant specific consequence analysis contained in NUREG/CR-4551, *Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Plant C.*[15] The NUREG-0933[3] values reflect additional information gained over the past several years due to the significant amount of research performed on the response of the containment under accident conditions. Based upon this, the NUREG-0933[3] values are judged to be more reflective of the best-estimate results with regard to containment failure considerations. Hence, the risk change values presented in Table 3 under the Plant PRA Data column are more conservative (excluding Plant C [W]).

5.4 Proposed Modifications Cost Analysis

NUREG/CR-4568, *A Handbook for Quick Cost Estimates,*[16] provides guidance for preparing estimates. Using this guidebook, the costs of implementing the proposed modifications were analyzed. A cost analysis for disabling the automatic feedwater isolation system was also conducted by the NRC staff, as documented in a memorandum from A. J. Dipalo to G. R. Mazetis, dated February 5, 1988. The results of these two cost analyses were in close agreement. Table 4 presents the results of the NRC analysis with the exception that replacement power costs were added to the cost estimate for the case of flow restrictor installation.

In order to determine the cost effectiveness of the proposed modification for each of the plants, a cost benefit analysis was performed. The cost benefit analysis was performed according to the following equation:

Estimated cost of Modification ($) x Change in Risk (man-rem) = Cost Benefit ($/man-rem)

The values employed in this analysis were the largest decrease in risk from Table 3 and the smallest cost from Table 4. This approach was taken to add conservatism to the analysis.

The results of this analysis are shown in Table 5. The cost benefit analysis was compared against the 1,000 per man-rem screening value to evaluate the cost effectiveness of the proposed modification.

TABLE 2. CONTAINMENT FAILURE MODES AND CONSEQUENCE INFORMATION

	PWR Release Category								
	1	2	3	4	5	6	7	8	9
Consequences of Release Category (man-rem)	5.4 E06	4.8 E06	5.4 E06	2.7 E06	1.0 E06	1.5 E05	2.3 E03	7.5 E04	120
Plant A Sequences Containment Failure Probabilities									
T_2L	α = 1 E-04	--	γ + δ = 0.7	--	β = 7 E-03	--	ε = 0.3	--	--
T_4ML	α = 1 E-04	--	γ + δ = 0.7	--	β = 7 E-03	-	ε = 0.3	--	--
T_1L	α = 1 E-04	--	γ + δ = 0.7	--	β = 7 E-03	--	ε = 0.3	--	--
T_3ML	α = 1 E-04	--	γ + δ = 0.7	--	β = 7 E-03	--	ε = 0.3	--	--
T_1LCC'	α = 1 E-04	δ = 0.8	δ' = 0.2	β = 7 E-03	--	--	--	--	--
Steam line break[a]	--	--	--	--	--	--	--	CF = 3 E-02	--
Plant B & BB Sequences Containment Failure Probabilities									
Spurious AFW isolation signal initiates transient[b]	α = 1 E-04	--	γ = 0.5	--	β = 7 E-03	--	ε = 0.5	--	--
T_1-$T_{1A}MLC$	α = 1 E-04	--	γ = 0.5	--	β = 7 E-03	--	ε = 0.5	--	--
$T_{2A}MLU$	α = 1 E-04	--	γ = 0.5	--	β = 7 E-03	--	ε = 0.5	--	--
$T_{2A}MLUD$	α = 1 E-04	γ = 0.5	--	β = 7 E-03	--	ε = 0.5	--	--	--
$T_{2A}MLUD'$	α = 1 E-04	δ = 0.5	--	β = 7 E-03	--	ε = 0.5	--	--	--
T_2MLU	α = 1 E-04	--	γ = 0.5	--	β = 7 E-03	--	ε = 0.5	--	--
Long term cooldown[b]	α = 1 E-04	--	γ = 0.5	--	β = 7 E-03	--	ε = 0.5	--	--
Feedwater line break[a]	α = 1 E-04	--	γ = 0.5	--	β = 7 E-03	--	ε = 0.5	--	--
Steam line break[a]	--	--	--	--	--	--	--	CF = 3 E-02	--
Plant C Sequences	AFW system accident sequences grouped in plant damage state TIYYY which contributes 1.29% to the total population dose of 200 man-rem/year. TIYYY has a CDF of 1.6 E-06/yr.								
NUREG-0933 Containment Failure Probabilities									
All sequences including feedwater line break[a]	--	γ = 3 E-02	--	--	β = 5 E-03	--	ε = 0.965	--	--
Steamline break[a]	--	--	--	--	--	--	--	CF = 3 E-02	--
	--	--	--	--	--	--	--	= 1.0 for	--
	-	--	--	--	--	--	--	Plant C only	--

a. These sequences represent the potential increase in risk, if the AFW isolation system is removed.

b. Reactor BB only.

NOTES: Containment Failure Modes

α = Vessel steam explosion; β = Containment leakage; γ = Hydrogen burning

δ = Overpressure; δ' = Delayed overpressure; ε = Basemat meltthrough

Table 3. Risk change due to proposed AFW system modification

Plant	Plant PRA Data Data (man-rem)	NUREG-0933[3] Data (man-rem)
A (CE)	36.2 Decrease	1.5 Decrease
B (B&W)	4.54 Increase	0.25 Increase
BB (B&W)	44.4 Decrease	3.0 Decrease
C (W)	13.3 Increase	

Table 4. Cost estimate for proposed plant modification

Cost Category	Cost to Disable Automatic AFW Isolation Systems Without Flow Restrictors Installed ($1000)	Cost to Disable Automatic AFW Isolation Systems with Flow Restrictors Installed ($1000)
Design, Hardware, and Installation	Not Applicable	$75[a]
Utility Licensing[b]	$250c	550c
Operator Training	43	43
NRC Review	58	100
Total if modifications are performed during a scheduled outage	$351	$768
Replacement Power Cost	Not Applicable	$6,000[d]
Total if modifications are performed during a nonscheduled outage	$351	$6,768

a. Estimate includes design, installation, calibration, and testing.
b. Estimate includes Technical Specification, FSAR, and procedure changes and amendments.
c. Estimate based on reanalyses required of selected DBAs
d. Estimate based on the powe replacement costs of $300K/day associated with a 2- day nonscheduled outage.

Table 5. Summary of cost benefits in dollars per man-rem reduction[a]

For Plants Not Requiring Hardware Modifications	For Plants Requiring Hardware Modifications	Do Proposed Modifications Show A Viable Cost Benefit?
\$351K/44.4 man-rem = \$7905/man-rem	\$768K/44.4 man-rem = \$17,290/man-rem	No[b]

a. Based on the most conservative values from Table 3.

b. Based on a screening value of \$1000 per man-rem of reduction.

Table 6. Uncertainties associated with the various tasks

Event Description	Error Factor	Source/Comments
Spurious signal results in AFW isolation	10	IREP/NREP This is applicable to either those events resulting in a transient or where the spurious signal occurs following some other initiator.
Failure to recover AFW	10	Engineering judgement based on review of NUREG/CR-4772.[17]
Failure of Feed and Bleed	10	Engineering judgement based on review of the various values employed for this event and NUREG/CR-4772.[17] This error factor was evaluated to be an upper bound for the various values used in this analysis.
Failure of operator to bypass AFW isolation logic during long term cooldown	10	Engineering judgement based on review of NUREG/CR-4772.[17]

6. UNCERTAINTIES

The individual tasks performed during the evaluation of GI 125.II.7 are subject to some level of uncertainty. The purpose of this section is to identify the major uncertainties associated with the various tasks and to evaluate the sensitivity of the recommendations for the resolution of GI 125.II.7.

6.1 Consequence Uncertainties

The study performed for GI 125.II.7 consisted of the following tasks: evaluation of the contribution to various sequences due to the automatic AFW isolation system, assignment of containment failure probabilities, and evaluation of the offsite dose factor which are presented below.

In this study, the major uncertainties associated with the evaluation of the core damage contribution due to the automatic AFW isolation system are in the assessment of the values for the events of interest. The specific events and their associated error factors are shown in Table 6. As can be seen from this table, all the events were assessed to have an error factor of ten.

One method which could be employed to determine the uncertainty in the estimated offsite consequences would be to employ a Monte-Carlo analysis and propagate the distributions through the models. However, based on statistical methodology for the log-normal distributions, the combined error factor can be approximated to or less than the largest individual error factor of the events used in the estimation of the contribution to CDF. Therefore, an upper bound on the combined error factor is assumed to be equal to the largest individual error factor.

Uncertainties associated with the probability of containment failure will not be specifically addressed due to the complexity of the analysis that would be required to properly treat this issue. However, containment failure will be evaluated using the plant specific containment failure probabilities from the plants' PRA as well as the generic containment failure probabilities from NUREG-0933.[3] This calculation is performed to demonstrate the sensitivity of the change in offsite consequences calculations to changes in containment failure probabilities.The offsite dose release factors (R) used in the GI 125.II.7 study were those presented in NUREG-0933,[3] with the exception of Plant C. The NUREG-0933[3] factors represent the offsite dose calculated for a typical plant. Certain plant specific characteristics such as assumed source terms and population density surrounding a specific plant introduces some uncertainty in the calculated offsite consequences. However, the NUREG-0933[3] values are considered representative in lieu of a detailed plant specific evaluation of the offsite consequences. The NUREG-0933[3] information was not used to evaluate the offsite consequences for Plant C since recent detailed offsite consequence information was available.

6.2 Cost Estimate Uncertainties

The cost estimate used to calculate the cost benefit ratios are also subject to some uncertainty. These costs were estimated using NUREG/CR-4568[16] as guidance, and were therefore assumed to be relatively accurate. One area of uncertainty is whether the proposed modification can be completed during a scheduled outage. Table 4 shows the costs associated with the bounding cases (i.e., estimated cost when the modification requires an outage--the upper bound on estimated cost, and the estimated cost when the modification is performed during a scheduled plant outage--the lower bound on estimated cost).

6.3 Sensitivity of Cost Benefit Summary

Based on the previous discussion of the estimated uncertainties, the use of an error factor of ten was assumed to be representative of the total uncertainties of the factors used to calculate the cost benefit ratio. This approach is acceptable because of the nature of this analysis and the application of the results. The analytical results are only needed to (a) provide an approximate evaluation of the sensitivity of the recommendations to the uncertainty of the factors used in the analysis, and (b) provide an aid to engineering judgement.

Table 7 presents the base information utilized in performing the sensitivity analysis. This table is a compilation of data previously presented. Table 8 presents the results of the sensitivity analysis. The sensitivity of the results presented in this table as to the uncertainties in the cost benefit ratios were calculated using an error factor of ten as described above. Cost benefit ratios were not calculated for those plants (Plants B and C) for which a net increase in the CDF due to implementing the proposed modification was estimated.

As can be seen from Table 8, all the estimated upper cost benefit ratios are below $1000/man-rem with the exception of Plant A using the PRA containment response information. The upper cost benefit ratio of $970/man-rem was evaluated as overly conservative and does not justify implementation of the proposed modification. This latter evaluation is based on engineering judgement in consideration of the following:

1. The cost benefit ratios of both the PRA and NUREG-0933[3] columns are based on offsite consequences (man-rem per accident) estimates for a generic plant as developed in NUREG-0933.[3] These estimates were based upon the conservative assumption that no evacuation would occur. This leads to a conservative estimation of the cost benefit ratio.

2. Comparison of the upper cost benefit ratio estimated by using the NUREG-0933[3] containment failure information.

3. The best-estimate cost benefit ratio using the PRA containment failure information.

4. The fact that for log-normal distributions, the combined error factor will be equal to or less than the largest individual error factor.

Table 7. Base data employed in the sensitivity analysis

Plant	Total Change In CDF (per Rx-year)	Offsite Consequences (Total man-rem) PRA[6,11,15]	NUREG-0933[3]	Cost ($1000)	Cost Benefit Ratio ($1000/man-rem)
A (CE)	5.5E-07 (decrease)	36.2	1.5	351	8.9
B (B&W)	9.6E-08 (increase)	4.5	0.25	768	*
BB (B&W)	9.0E-07 (decrease)	44.4	3.0	768	17
C (W)	4.0E-07 (increase)	13.3	**	351	*

* Cost benefit ratios were not calculated for plants where the implementation of this issue would result in an increase in the estimated risk.

** Consequences using the NUREG-0933[3] information were not estimated for this plant since the resulting value would not be comparable to the plant specific value. The values are not comparable due to the different assumptions and techniques employed in the two analyses to determine offsite consequences.

Table 8. Sensitivity analysis results

Plant		Cost Benefit Ratio ($/man-rem) PRA[6,11,15] Containment Failure Information	NUREG-0933[3] Containment Failure Information
A (CE)			
	Upper Bound	970	23,400
	Best Estimate	9700	234,000
	Lower Bound	97,000	2,340,000
BB (B&W)			
	Upper Bound	1700	25,600
	Best Estimate	17,000	256,000
	Lower Bound	172,000	2,560,000

7. CONCLUSIONS

Four PWRs, one from each reactor vendor (two B&W AFW designs), were evaluated to determine the AFW isolation system's contribution to the CDF. It was thought that the greatest risk associated with this issue would be for plants with marginal or no feed-and-bleed capabilities. This study included one such plant. Another significant consideration was that some plants utilize the automatic AFW isolation system to prevent AFW pump runout conditions with resultant possible pump damage and AFW system failure when supplying water to a depressurized (steam or feed line break condition) steam generator. The evaluation performed for this study also included three of these plants.

The evaluation indicates that the effects of the AFW isolation system are dependent on the particular plant and its design. The estimated reduction in CDF due to AFW isolation system's contribution were reasonably low, but the difference between the highest and the lowest value was an order of magnitude. At the CE design plant, deleting the AFW isolation system would not cause a failure of the AFW system if the operator did not take action to isolate a ruptured steam generator because flow restrictors are provided in the system design. For the B&W design plants, the AFW could be assumed to fail due to EFS flow diverted to the ruptured steam generator if no operator action was taken. At the W design plant, because of the diverse motor-driven AFW pumps, only part of the AFW would fail if no operator action was taken. If the AFW isolation systems were deleted or disabled, the net change in CDF for the CE plant would decrease by 5.5E-07, one B&W design plant would decrease its CDF by 9.0E-7, the other B&W design plant would have its CDF increase by 9.6E-08, and the W design plant would have a CDF increase of 4.0E-07.

Because these changes in CDF are low and the calculated cost for removing the automatic AFW isolation system were relatively high ($351K for lowest cost), the cost benefit ratios indicate that no significant benefit would be realized by removing or disabling existing automatic AFW isolation systems.

Additionally some plants, as documented in IE Bulletin 80-04, may show unacceptable containment pressure analyses results if the existing automatic isolation systems were removed.

Even though only four plant designs were evaluated in this study, engineering judgement indicates that the results of this study can be extrapolated to address this issue on a generic basis. This judgement is based on the following factors:

1. The four plants evaluated in this study include plants with different automatic AFW isolation system designs. One of these designs would most likely represent system designs used at other plants. Cost analysis and cost benefit ratio calculations indicate that no significant benefit can be realized by removing the automatic AFW isolation system on the plants included in this study. Therefore, it can be assumed that plants with an AFW isolation system similar to one of the designs evaluated in this study would show a similar cost benefit ratio.

2. If automatic AFW isolation designs are used at some plants which are significantly different than those evaluated in this study, the findings of this study related to differences in CDF compared to system design can be extrapolated. This study indicated that the differences in isolation system design had little bearing on the change in CDF. The major factor affecting the CDF calculations was the presence, or absence, of flow restrictors in the AFW system. All PWRs will either have AFW flow restrictors or will not. This study showed the worst case (most risk reduction) was for plants that do not have flow restrictors. Even these plants showed no significant cost benefit.

3. The cost benefit ratios calculated for this study were performed very conservatively. As noted in Item 2 above, the greatest risk reduction associated with removal of the automatic AFW isolation system was for plants that do not use separate AFW flow restrictors. If the existing isolation system were removed, these plants would incur the highest cost because a plant hardware modification would be required. However, the most favorable cost benefit ratio (approximately $8K/man-rem) calculated during this study used the highest

risk reduction value calculated for a plant without AFW flow restrictors and used the least expensive cost (for plants not requiring a hardware modification). This method was used to add conservatism to the analysis to account for analytical uncertainty and also provide some assurance that differences not addressed in Items 1 and 2 above would not change the conclusions of this study.

8. REFERENCES

1. U.S. Nuclear Regulatory Commission, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, LWR Edition, NUREG-0800 (formerly issued as NUREG-75/087), June 1987.

2. U.S. Nuclear Regulatory Commission, *Loss of Main and Auxiliary Feedwater Event at the Davis-Besse Plant on June 9, 1985*, NUREG-1154, July 1985.

3. U.S. Nuclear Regulatory Commission Division of Safety Technology, *A Prioritization of Generic Safety Issues*, NUREG-0933, Rev. 4, December 1983.

4. Memorandum from H. R. Denton to T. Speis, September 10, 1986, on *Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During Line Break.*

5. U.S. Nuclear Regulatory Commission, *Fault Tree Handbook*, NUREG-0492, January 1981.

6. Arthur C. Payne, Jr., Principal Investigator, *Interim Reliability Evaluation Program: Analysis of the Plant A Nuclear Power Plant, 1*, NUREG/CR-3511/1 of 2, SAND83-2086/1 of 2, March 1984.

7. U.S. Nuclear Regulatory Commission Memorandum to Karl Kniel, Generic Issues Branch, DST, from Brian W. Sheron, Reactor Systems Branch, DSI, on *The Status of Feed-and-Bleed for Emergency Decay Removal*, March 31, 1981.

8. Memorandum from B.W. Sheron, NRC, to T. P. Speis, NRC, Subject: *Feed-and-Bleed Capability in CE Plants Both With and Without PORVs*, February 11, 1982.

9. U.S. Nuclear Regulatory Commission, "Quantitative Results of Accident Sequences," *Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, Appendix V*, NUREG-75/014, WASH-1400, October 1975.

10. *Interim Reliability Evaluation Program Procedures Guide*, Sandia National Laboratories, NUREG/CR-2728, SAND82-1100, January 1983.

11. A. A. Garcia, et al., *Plants B and BB Safety Study Main Report, 1*, NUREG/CR-2515, SAND81-7229/1, December 1981.

12. IEEE-STD 500-1984, *IEEE Guide to the Collection and Presentation of Electrical, Electronic, Sensing Component, and Mechanical Equipment Reliability Data for Nuclear-Power Generating Stations*, New York, NY, by The Institute of Electrical and Electronics Engineers, Inc., 1983.

13. U.S. Nuclear Regulatory Commission, *Analysis of Core Damage Frequency From Internal Events: Plant C*, NUREG/CR-4550, *5*, February 1987.

14. R. J. Mulvihill, et al., *Plant C Nuclear Power Plant Availability and Safety Assessment*, EPRI NP-3382 Research Project 1842-4 Interim Report, December 1983.

15. A. S. Benjamin, et al., *Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Plant C, 2*, NUREG/CR-4551, SAND86-1309, February 1987.

16. J. R. Ball, "A Method for Developing Quick Approximate Estimates of Costs for Generic Actions for Nuclear Power Plants," *A Handbook for Quick Cost Estimates*, NUREG/CR-4568, ANL/EES-TM-297, April 1986.

17. Alan D. Swain, *Accident Sequence Evaluation Program Human Reliability Analysis Procedure,* NUREG/CR-4772, SAND86-1996, February 1987.

18. U.S. Nuclear Regulatory Commission, *Shutdown Decay Heat Removal Analysis of a Westinghouse 3-Loop Pressurized Water Reactor,* NUREG/CR-4762, SAND86-2377, March 1987.

19. U.S. Nuclear Regulatory Commission, *Catalog of PRA Dominant Accident Sequence Information,* NUREG/CR-3301, EGG-2259, July 1985.

APPENDIX A

ACCIDENT SEQUENCE CALCULATIONS FOR PLANT A (CE)

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ACCIDENT SEQUENCE CALCULATIONS FOR PLANT A (CE)

INTRODUCTION

This appendix provides the cutsets used to develop the accident sequence frequency for the accidents involving the AFW isolation system at Plant A (CE). Also presented in this appendix are the basic event failure probabilities used in establishing the accident sequence frequencies.

This plant has two steam generators, one motor-driven AFW pump and one turbine-driven AFW pump. Each of four pump headers has a flow control valve set to limit flow to 220 GPM. The AFW is assumed to fail if less than 400 GPM is delivered to the steam generators. The AFW isolation system will cause failure of the AFW if both steam generators are isolated or if one steam generator is isolated when one of the AFW pump trains is down if the operator does not adjust the flow to the operating steam generator.

ACCIDENT SEQUENCE FREQUENCIES

Sequence T2L

Accident Sequence T2L is a loss of the PCS followed by loss of the AFW (L) system. The following lists the cutsets and frequencies of each cutset considered in the analysis.

Cutset	Frequency
T2*AFWISO-2SG	4.60E-07
T2*RA-3*AFWP11-PTD-LF*AFWISO-1SG	1.74E-09
T2*RA-3*AFWP13-PMD-LF*AFWISO-1SG	1.37E-09
T2*RA-3*CBP13-BOO-LF*AFWISO-1SG	1.10E-09
T2*RA-3*AFWP11-PTD-PRMN*AFWISO-1SG	1.37E-09
T2*RA-2*ELCOO11A-INV-LF*AFWISO-1SG	4.40E-10
T2*RA-3*AFWS903A-NOC-LF*AFWISO-1SG	3.74E-10
T2*RA-3*AFW3987A-NOC-LF*AFWISO-1SG	3.74E-10
T2*RA-3*ESFSONCA-LOG-LF*AFWISO-1SG	2.35E-10
Total	4.67E-07

Sequence T4ML

Sequence T4ML is any transient not considered elsewhere followed by the loss of the power conversion system (M) and loss of the AFW (L). The following lists the cutsets and frequencies of each cutset considered in the analysis.

Cutset	Frequency
T4*RA-1*PCS-LF*AFWISO-2SG	1.88E-08
T4*RA-2*ELCOO11A-INV-LF*AFWISO-1SG	3.68E-09
T4*RA-2*ELCOO12A-INV-LF*AFWISO-1SG	3.74E-09
T4*RA-2*ELCOO11A-CBL-LF*AFWISO-1SG	1.27E-10
Total	2.63E-08

Sequence T1L

Sequence T1L is a loss of offsite power followed by loss of AFW (L). The following lists the cutsets and frequencies of each cutset considered in the analysis.

Cutset	Frequency
T1*RA-LOSP*AFWISO-2SG	3.63E-08
T1*RA-LOSP*RA-3*AFWP11-PTD-LF*AFWISO-1SG	1.36E-10
T1*RA-LOSP*RA-3*AFWISO-1SG*ELCOO11A-GEN-LF	1.56E-10
T1*RA-LOSP*RA-3*AFWS903A-NOC-LF*AFWISO-1SG	3.00E-10
T1*RA-LOSP*RA-3*AFWP11-PTD-PRMN*AFWISO-1SG	1.08E-10
T1*RA-LOSP*RA-3*AFW3987A-NOC-LF*AFWISO-1SG	3.00E-11
T1*RA-3*AFWP11-PTD-LF*AFWISO-1SG	3.02E-10
T1*RA-3*AFWP13-PMD-LF*AFWISO-1SG	2.37E-10
T1*RA-3*AFWISO-1SG*CBP13-BOO-LF	1.92E-10
T1*RA-LOSP*RA-3*AFWISO-1SG*ELCOO11A-G-PRMN	1.92E-10
T1*RA-3*AFWP11-PTD-PRMN*AFWISO-1SG	2.40E-10
T1*RA-LOSP*RA-16*AFW4530-N-PRMN*AFWISO-1SG	1.44E-11
T1*RA-LOSP*RA-16*AFW4520-N-PRMN*AFWISO-1SG	1.44E-11
T1*RA-LOSP*RA-3*AFWISO-1SG*ELCOO11A-G-PRMN	1.92E-10
T1*RA-LOSP*RA-3*AFWP11-PTD-PRMN*AFWISO-1SG	1.08E-10
T1*RA-LOSP*RA-3*AFWISO-1SG*ELCOO11A-G-FRFT	1.44E-10
T1*RA-LOSP*RA-4*AFWP11-PTD-PRTS*AFWISO-1SG	1.04E-11
T1*RA-LOSP*RA-4*AFWISO-1SG*ELCOO11A-GEN-LF	3.96E-10
T1*RA-LOSP*RA-3*SDSSQNCA-LOG-LF*AFWISO-1SG	1.12E-10
T1*RA-LOSP*RA-3*SRW1587A-NCC-LF*AFWISO-1SGN-LF	8.82E-11
T1*RA-LOSP*RA-3*AFWISO-1SG*ELC1103A-BOO-LF	8.32E-11
T1*RA-LOSP*RA-3*AFWISO-1SG*DGVCT11A-BOO-LF	8.32E-11
T1*RA-LOSP*RA-3*AFWISO-1SG*DGVOT11A-DCC-LF	8.32E-11
T1*RA-LOSP*RA-3*AFWISO-1SG*DGVRC11A-DCO-LF	8.32E-11
T1*RA-LOSP*RA-3*AFWISO-1SG*DGV[illegible]11A-DCC-LF	8.32E-11
T1*RA-LOSP*RA-3*AFWISO-1SG*SWS1105-BOO-LF	8.32E-11
T1*RA-LOSP*RA-3*AFWISO-1SG*SRWA011A-BOO-LF	8.32E-11
T1*RA-2*AFWISO-1SG*ELCOO11A-INV-LF	7.84E-11
T1*RA-LOSP*RA-3*AFWISO-1SG*ELC1103A-BOO-CC	7.35E-11
T1*RA-LOSP*RA-3*AFWISO-1SG*SWS5210A-NTC-CC	7.35E-11
T1*RA-LOSP*RA-3*AFWISO-1SG*SWS5150A-NOC-CC	7.35E-11
T1*RA-LOSP*RA-3*AFWISO-1SG*AFW0103-X-FRFT	5.60E-12
T1*RA-LOSP*RA-3*AFWISO-1SG*AFWM911X-X-PRMN	4.80E-12

T1*RA-3*AFWP11-PTD-LF*AFWISO-1SG	6.40E-11
T1*RA-3*AFWP13-PMD-PRMN*AFWISO-1SG	6.40E-11
T1*RA-3*AFWS903A-NOC-LF*AFWISO-1SG	6.40E-11
T1*RA-3*AFW3987A-NOC-LF*AFWISO-1SG	6.40E-11
T1*RA-LOSP*RA-3*AFWISO-1SG*ELC0011A-G-PRTS	4.40E-11
T1*RA-3*CBP13-BOO-LF*AFWISO-1SG	1.96E-10
T1*RA-LOSP*RA-3*AFWISO-1SG*AFWS903A-NOC-LF	2.96E-11
T1*RA-LOSP*RA-3*AFWISO-1SG*AFW3987A-NOC-LF	2.96E-11
Total	4.07E-08

Sequence T3ML

Sequence T3ML is a transient that requires primary pressure relief followed by failure of the power conversion system (M) and the AFW system (L).

This accident sequence has the same cutsets as sequence T4ML except for the initiator. The accident sequence can be determined by a simple ratio of initiator frequencies:

CDF FOR T4ML*T3/T4
2.63E-08 * 1.85/6.8 7.16E-09

Sequence T1LCC'

Sequence T1LCC'is a loss of offsite power followed by loss of the AFW (L), containment spray injection and the containment air recirculation systems.

Because the cutsets for this accident sequence are almost the same as those of accident sequence T1L, the CDF can be estimated by ratioing the accident sequence CDFs and multiplying that times the AFW isolation system contribution to T1L.

(CDF FOR T1LCC')/(CDF FOR T1L) * CDF FOR T1L AFW ISO = 1.0E-06/4.9E-06 * 4.07E-08
8.31E-09

BASIC EVENT FAILURE PROBABILITIES

Except where noted, the following values were extracted from the PRA used to evaluate this plant.

Event Code	Event Description	Probability
T1	Loss of offsite power	f = 0.14/yr
T2	Loss of power conversion system	f = 0.8/yr
T3	Transients requiring primary system pressure relief	f = 1.85/yr
T4	All other transients requiring reactor trip	f = 6.8/yr
AFWISO-2SG	Spurious isolation of one steam generator with common mode isolation of the other and the operator does not recover - 4 ways of occurring (developed for this study)	p = 5.76E-07

Event Code	Event Description	Probability
AFWISO-1SG	Spurious isolation of one steam generator and the operator does not recover - 4 ways of occurring (developed for this study)	p = 1.15E-05
RA-1	Operator fails to realign AFW suction to CST #11 and start locked-out AFW turbine-driven pump #12; all actions must be done locally	p = 0.1
RA-2	Operator fails to manually actuate AFW motor-driven pump #13 (given failure of auto star)t	p = 0.02
RA-3	Operator fails to manually start locked-out AFW turbine-driven pump #12	p = 0.04
RA-LOSP	Failure to recover offsite power within 1 hour	p = 0.45
AFWP11-PTD-LF	AFW turbine-driven pump #11 local fault	p = 4.7E-03
AFWP11-PTD-PRMN	AFW turbine-driven pump #11 maintenance	p = 3.7E-03
AFWP11-PTD-PRTS	AFWS turbine-driven pump #11 unavailable due to test	p = 1.4E-03
AFWO103-X-FRFT	AFW turbine-driven pump #11 discharge valve, fail to return from test	p = 2.0E-04
AFWM911X-X-PRMN	Maintenance of valve in turbine driven pump #11 steam admission line	p = 1.6E-04
AFWP13-PMD-LF	AFW motor-driven pump #13 local fault	p = 3.7E-03
AFWP13-PMD-PRMN	AFW motor-driven pump #13 maintenance	p = 3.7E-03
AFWS903A-NOC-LF	Local fault of steam admission valve to AFW turbine-driven pump #11	p = 1.0E-03
AFW3987A-NOC-LF	Local fault of steam admission valve To AFW turbine-driven pump #11	p = 1 [illegible]E-03
AFW4520-N-PRMN	Maintenance of valve in AFW turbine pumps feedwater lines fail delivery by both AFW pumps	p = 2.0E-04
AFW4530-N-PRMN	Maintenance of valve in AFW turbine pumps feedwater lines fails delivery by both AFW pumps	p = 2.0E-04
CBP13-BOO-LF	AFW motor-driven pump #13 circuit breaker	p = 3.0E-03

Event Code	Event Description	Probability
DGVCT11A-BOO-LF	Local fault of power breaker to diesel generator #11 room coolers fails DG # 11 which fails 1/2 of all ESF and the motor-driven AFW pump	p = 3.0E-03
DGVOT11A-DCC-LF	Damper fails to operate, fails DG #11 which fails 1/2 of all ESF and the motor-driven AFW pump	p = 3.0E-03
DGVRC11A-DCO-LF	Damper fails open, fails DG#11 which fails 1/2 of all ESF and the motor driven AFW pump	p = 3.0E-03
DGVIN11A-DCC-LF	Damper fails to operate, fails DG #11 which fails 1/2 of all ESF and the motor-driven AFW pump	p = 3.0E-03
ELC0011A-G-FRFT	Diesel generator #11 not returned from test, fails 1/2 of all ESF systems and motor-driven AFW pump	p = 5.0E-03
ELC0011A-G-PRMN	Maintenance of diesel generator #11 fails motor-driven pump # 13 and 1/2 of all ESF systems	p = 6.6E-03
ELC0011A-G-PRTS	Diesel generator #11 unavailable due to test, fails 1/2 of all ESF and the motor-driven AFW pump	p = 1.5E-03
ELC0011A-GEN-LF	Local fault in diesel generator #11 fails AFW motor-driven pump #13 and 1/2 of all ESF systems	p = 5.4E-02
ELC0011A-INV-LF	11A vital AC bus, fails AFW turbine driven steam admission valve 4071 due to no actuation signal and fails motor-driven AFW pump	p = 2.4E-03
ELC0011A-CBL-LF	Local fault of cable from vital AC inverter #11; same effect as inverter fault above	p = 7.5E-05
ELC0012A-INV-LF	Similar to ELC0011A-INV-LF above except steam admission valve 4070 fails closed	p = 2.4E-03
ELC1103A-BOO-CC	Control circuit fault of DG # 11 output breaker, fails 1/2 of all ESF and the motor-driven AFW pump	p = 2.5E-03
ELC1103A-BOO-LF	Local fault of diesel generator #11 breaker, fails 1/2 of all ESF and motor-driven AFW pump	p = 3.0E-03
ESFSONCA-LOG-LF	Faults in ESFAS sequencer fail AFAS auto actuation of AFW motor driven pump #13	p = 6.4E-04
PCS-LF	Local fault causes failure of PCS	p = 4.8E-03

Event Code	Event Description	Probability
SDSSQNCA-LOG-LF	Shutdown sequencer logic unit fails to sequence loads to DG #11, fails 1/2 of all ESF and motor-driven AFW pump	p = 3.8E-03
SRWA011A-BOO-LF	Local fault of SRW pump #11 power breaker, fails DG #11 which fails 1/2 of all ESF and the motor-driven AFW pump	p = 3.0E-03
SRW1587A-NCC-LF	Local fault of diesel generator #11 cooling outlet valve, fails diesel generator cooling and fails AC power to 1/2 of all ESF systems and motor-driven pump	p = 3.0E-03
SWS1105-BOO-LF	Local fault of SWS pump #11 power breaker, fails DG #11 cooling and AC power to 1/2 of all ESF and the motor driven AFW pump	p = 3.0E-03
SWS5150A-NOC-CC	Control circuit fault of inlet valve on SRW heat exchanger #11 fails DG # 11 which fails 1/2 of all ESF and the motor-driven AFW pump	p = 2.5E-03
SWS5210A-NTC-CC	Control circuit fault of outlet valve on SRW heat exchanger #11 fails DG # 11 which fails 1/2 of all ESF and the motor-driven AFW pump	p = 2.5E-03

APPENDIX B

ACCIDENT SEQUENCE CALCULATION FOR PLANT B AND BB (B&W)

APPENDIX B

ACCIDENT SEQUENCE CALCULATION FOR PLANT B AND BB (B&W)

This appendix provides the cutsets used to develop the accident sequence frequency for the accidents involving the AFW isolation system at Plant B and Plant BB (B&W):

This plant has two steam generators one motor-driven EFS pump and one turbine-driven EFS pump. The pumps are cross connected so that each pump supplies both steam generators. The plant does not have flow restrictors in the EFS headers. Thus, the EFS will fail if one steam generator ruptures and that generator is not isolated. The EFS is assumed to be successful if at least one pump is supplying feedwater to at least one steam generator. For this evaluation, the EFS will fail if the automatic isolation system isolates both steam generators or if it isolates one steam generator when the other is isolated and the operator does not recover flow to one of the steam generators.

The PRA used to evaluate these plants calculated the failure probability of the EFS for two conditions, with and without offsite power. With offsite power available, the EFS failure probability is EF1 = 3.4E-04.

With offsite power unavailable, the EFS failure probability is EF2 = 1.8E-03.

These events are made up of the following cutsets:

EF1 = E01 + E4 + EA*EB + EA*EM2 + EB*EM1 + E3*E02
EF2 = E01 + E4 + EA*EB + EA*EM2 + EB'*EM1 + E3*EO2

Where:

E01	Auto actuation locked out and operator fails to recover	p = 1.0E-04
E4	Coupled check valve faults (FWV-43 & 44)	p = 1.0E-05
EA	E1 + EX1	
E1	Train A hardware faults	p = 2.2E-02
EX1	DC power train B fails w/offsite pwr	p = neg
	wo/offsite pwr	p = 3.2E-03
EB	E2 + EX2	
E2	Train B hardware faults	p = 3.5E-03
EX2	AC train A fails and DC power train A	
	fails w/offsite pwr	p = neg
	wo/offsite pwr	p = 3.52E-02
EM2	Train B maintenance and test outages	p = 5.5E-03
	EM1	Train A maintenance and test outage
EB'	E5 + EX2	
E5	Train B hardware faults	p = 3.6E-03
E3	Valve plugged	p = 1.0E-04
E02	Operator fails to recover plugged valve	p = 0.1

Isolation system contribution:

AFWISO-2SG	Isolation of 2 steam generators due to spurious actuation of the isolation system to isolate 1 S/G and common mode isolation of the other S/G - 4 ways to occur	p = 5.76E-07
AFWISO-1SG	Isolation of 1 steam generator due to spurious actuation of the isolation system - 4 ways to occur	p = 1.15E-05

Only event E4 isolates the steam generators; all of the others fail one of the pump trains, which leaves the other pump pumping to both steam generators. Actuation of the automatic EFS isolation system to isolate one steam generator would not fail the EFS because the running pump would simply deliver its flow to the other steam generator because the plant does not have flow restrictors to limit flow.

Event E4 is a coupled failure of two check valves at the inlet to the steam generators. The failure probability is made up of (a) check valve failure, 1.0E-04 and (b) common mode failure of the other, 0.1.

The probability of one check valve failing closed and the automatic isolation system isolating the other steam generator is CHKVLV-LF * AFWISO-1SG = 1.0E-04 * 1.15E-05 = 1.15E-09.

The total contribution to the EFS system failure for the automatic EFS isolation system is 5.76E-07 + 1.15E-09 = 5.77E-07.

The contribution to the accident sequence frequency of the auto EFS isolation system can be determined by dividing the accident sequence frequency of all sequences with EFS failures found in the PRA by the appropriate EFS system failure rate and then multiplying the result by the automatic EFS isolation system failure rate contribution found above.

The contribution of the auto EFS isolation system to the accident sequence frequency is (note--L is the indication of EFS failure):

T1-T1A(MLU)	4.1E-06/3.4E-04*5.8E-07	f = 7.0E-09
T2A(MLU)	1.4E-05/1.8E-03*5.8E-07	f = 4.5E-09
T2A(MLUO')	2.5E-06/1.8E-03*5.8E-07	f = 8.0E-10
T2A(MLUO)	5.4E-05/1.8E-03*5.8E-07	f = 1.7E-08
T2-T2A(MLU)	8.6E-06/3.4E-04*5.8E-07	f = 1.5E-08
Total		f = 4.4E-08

APPENDIX C

ACCIDENT SEQUENCE CALCULATIONS FOR PLANT C (W)

APPENDIX C

ACCIDENT SEQUENCE CALCULATIONS FOR PLANT C (W)

INTRODUCTION

This appendix provides the cutsets used to develop the accident sequence frequency for the accidents involving the AFW isolation system at Plant C (W):

For all but ATWS accident sequences, the AFW is assumed to fail if less than two steam generators are supplied with feedwater. For ATWS, three steam generators must be supplied; however, ATWS events do not contribute to the dominant accident sequences for Plant C (W), so they will not be considered for this analysis. The turbine-driven pump supplies all four steam generators and each motor-driven pump supplies two steam generators. Therefore, if the turbine-driven pump is running, the success criteria will be met if less than three of the LCVs close. If one of the motor-driven pumps and the turbine-driven pump are out of service, only one LCV must close to fail the AFW. For Plant C (W) then, only accident sequences that fail one of the motor-driven pumps and the turbine-driven pump will be of interest.

ACCIDENT SEQUENCE FREQUENCIES

Sequence Tdc2L1P1

Sequence Tdc2L1P1 is a loss of the 125 V DC bus, failure of the AFW system (L1), and failure of the PORVs to successfully open (P1) for feed-and-bleed. Loss of the 125 V DC bus fails motor-driven pump 1B.

The following lists the cutsets and frequencies of each cutset considered in the analysis.

Cutset	Frequency
Tdc2L1P1*AFW-PSF-LF-PS415*AFWISO-1SG-#1	f = 9.3E-09
Tdc2L1P1*AFW-PSF-LF-PS415*AFWISO-1SG-#2	f = 9.3E-09
Tdc2L1P1*AFW-MOV-CC-151*AFWISO-1SG-#1	f = 6.9E-10
Tdc2L1P1*AFW-MOV-CC-151*AFWISO-1SG-#2	f = 6.9E-10
Total	f = 2.0E-08

Sequence Tdc1L1P1

Accident sequence Tdc1L1P1 is the same as the above except the other motor-driven pump is failed by the DC failure and the LCV on steam generator 3 and 4 would have to isolate to fail the AFW. Its frequency will be the same -- 2.0E-08.

Other Sequences

For the remaining accident sequences, the turbine-driven pump, one of the motor-driven pumps and one of the LCVs must fail, or one of the motor-driven pumps and three of the LCVs must fail. The accident frequencies of these events are on the order of 1.0E-14; thus, they will not be considered further.

The total contribution of the AFW isolation system to the CDF for Plant C (W) is 4.0E-08.

BASIC EVENT FAILURE PROBABILITIES

The following tabulates the basic event failure probabilities used in the above cutsets:

Event Code	Event Description	Probability
Tdc2L1P1	Loss of 125 VDC bus II	f = 9.0E-04
AFW-PSF-LF-PS415	Faults in turbine-driven pump pipe segment 415 fails turbine driven pump	p = 5.2E-02
AFW-MOV-CC-151	Failure of turbine throttle valve-fails turbine-driven pump	p = 3.8E-03
AFWISO-1SG-#1	LCV to steam generator #1 closes	p = 2.0E-04
AFWISO-1SG-#2	LCV to steam generator #2 closes	p = 2.0E-04

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This report presents the evaluation of the potential safety concerns identified in Generic Issue 125.II.7, related to the automatic auxiliary feedwater (AFW) isolation from a steam generator during a main steam or feedwater line break. For this review, existing probabilistic risk assessments (PRAs) were evaluated to identify specific event tree sequences where the AFW system had failed. These sequences were utilized to calculate the contribution of the AFW isolation system to the accident sequence frequency. By utilizing this methodology, the change in risk could be calculated for a plant with an automatic AFW isolation system compared to the same plant with the automatic AFW isolation system removed. The review evaluated one Westinghouse plant, one Combustion Engineering plant, and two Babcock and Wilcox design versions of a plant.

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ISOLATE FEEDWATER FROM STEAM GENERATOR DURING A LINE BREAK