DESIGN ALTERNATIVES
FOR THE
SYSTEM 80+ NUCLEAR POWER PLANT

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COMBUSTION ENGINEERING NUCLEAR, INC.
WINDSOR, CONNECTICUT

ABB
ASEA BROWN BOVERI

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ACRONYMS

ADV	Atmospheric Dump Valves
ALWR	Advanced Light Water Reactor
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
cal	Calories
CCW	Component Cooling Water
CET	Containment Event Tree
CRAC	Calculations of Reactor Accident Consequences
DC	Direct Current
DCH	Direct Containment Heating
EFWS	Emergency Feedwater System
H2	Hydrogen
HPSI	High Pressure Safety Injection
Hrs	Hours
HVAC	Heating, Ventilation, and Air Conditioning
IRWST	In-Containment Refueling Water Storage Tank
KAG	Key Assumptions and Groundrules
LOCA	Loss of Coolant Accident
M	Meters
MAAP	Modular Accident Analysis Program
MORV	Motor Operated Relief Valve
NRC	Nuclear Regulatory Commission
PDS	Plant Damage State
PRA	Probabilistic Risk Assessment
RC	Release Class
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
SCS	Shutdown Cooling System
sec	Second
SGTR	Steam Generator Tube Rupture
SIT	Safety Injection Tanks
y	year

1.0 INTRODUCTION

The U.S. Nuclear Regulatory Commission's policy related to severe accidents requires, in part, that an application for a design approval comply with the requirements of 10CFR50.34(f). Item (f)(1)(i) requires "performance of a plant site specific [PRA] the aim of which is to seek improvements in the reliability of core and containment heat removal systems as significant and practical and do not impact excessively on the plant." Reference 1 provides the base PRA of the System 80+ plant.

The NRC also requested the ALWR participants to evaluate design alternatives[3] that help mitigate the consequences of severe accidents. To address these requirements and requests, a review of potential modifications to the System 80+ design, beyond those included in the Probabilistic Risk Assessment (PRA), was conducted to evaluate whether potential severe accident mitigation design features could be justified on the basis of cost per person-rem averted.

This report summarizes the results of C-E's review and evaluation of eleven Design Alternatives that were considered in the System 80+ design. Improvements have been reviewed against conservative estimates of risk reductions based on the PRA and minimum order of magnitude costs, to determine what modifications are potentially attractive.

2.0 SUMMARY AND CONCLUSION

The System 80+ design is an Evolutionary Advanced Light Water Reactor design with improved design features to reduce the risk of core damage and mitigate the consequences if core damage should occur. The design process was integrated with the PRA to ensure that the risk was very low and distributed over all of the safety related systems (i.e., no single system carries a disproportional responsibility for plant safety). The design insured that no single accident sequence dominated the plant risk and the lessons learned from previous PRAs were addressed.

Eleven design alternatives were evaluated. These were selected based on the Design Alternatives evaluated for the Limerick plant[2] and the results from the System 80+ PRA performed by C-E. The Design Alternative analysis used a bounding technique. It was assumed that each Design Alternative worked perfectly and completely eliminated the accident sequences that the Design Alternative was to address. This approach maximizes the benefits associated with each Design Alternative. The benefits were the reduction in risk in terms of whole body person-rems per year received by the total population around the ALWR site. Using $1,000 per person- rem, and a levelized capital cost rate of 17.9%, this risk reduction was converted to a maximum capital benefit that was compared with capital costs.

Table 2-1 summarizes the results of the Design Alternative analysis. The first column, is the percent of the total person-rem/year reduction for each design alternative. The next column, labeled capital benefit, is an equivalent present worth of the annual dose reduction. It is also the maximum amount that could be spent in capital to be cost beneficial. The third column is a rough capital cost estimate for the design alternatives. The net benefit (capital benefit - capital cost) is given in the last column.

The System 80+ plant was designed to meet the stringent design goals in the EPRI Advanced Light Water Reactor (ALWR) Utility Requirements Document. The System 80+ design has a core damage frequency approximately two orders of magnitude lower than existing plants. Therefore, the benefits of improving the existing design are significantly lower than predicted for the Limerick Plant[2]. The analysis presented in this report overestimated the benefits of the Design Alternatives by assuming that they would work perfectly to eliminate the type of accident they are designed to address. Because of the small initial risk associated with the System 80+ design, none of the Design Alternatives are cost beneficial.

TABLE 2-1

SUMMARY OF THE RISK REDUCTIONS OF THE DESIGN ALTERNATIVES

	Design Alternative	PERSON-REM REDUCTION	CAPITAL BENEFIT*	CAPITAL COST	NET CAPITAL BENEFIT
1	CONTAINMENT SPRAY	90%	$27,600	$1,500,000	-$1,472,400
2	FILTERED VENT	86%	$26,300	$10,000,000	-$9,973,700
3	DC BATTERIES & EFWS	69%	$21,100	$2,000,000	-$1,978,900
4	RCP SEAL COOLING	16.5%	$6,034	$100,000	-$93,966
5	PRESSURIZER AUXILIARY SPRAY	6.7%	$2,050	$5,000,000	-$4,997,950
6	ATWS VALVES	4.2%	$1,290	$1,000,000	-$998,710
7	CONCRETE COMPOSITION	2.5%	$765	$5,000,000	-$4,999,244
8	REACTOR VESSEL EXTERIOR COOLING	2.5%	$765	$5,500,000	-$5,499,244
9	H2 IGNITORS	0.1%	$31	$1,000,000	-$999,969
10	HIGH PRESSURE SAFETY INJECTION	0.004%	0	$20,000,000	-$20,000,000
11	RCS DEPRESSURIZATION	0.002%	0	$500,000	-$500,000

* THE MAXIMUM CAPITAL COST ASSUMES NO MAINTENANCE OR TESTING COSTS FOR THE ADDITIONAL EQUIPMENT

3.0 METHODOLOGY

The Design Alternative evaluation followed the format and procedure used by the NRC in evaluating Design Alternatives for Limerick[2]. The Design Alternatives were evaluated in terms of cost benefit where the cost of the additional equipment is compared with the savings in terms of a reduced exposure risk to the general population. The savings, in person-rems per year, were converted to dollars using $1,000 per person-rem. The risk of the base System 80+ design is described in the System 80+ Standard Design PRA[1].

3.1 RISK REDUCTION

Risk (person-rem/year) in this analysis is the product of the frequency of core damage for each type of accident (events/y) times the consequence of the accident (person-rem/event). The total risk is the sum of the risks from all the types of accidents. For each Design Alternative, the reduction in total risk is the difference between the risk of the base System 80+ design and the risk with the Design Alternative added.

Risk is defined as the product of frequency and consequence. The frequency of core damage for various accident sequences are calculated. These sequences are then grouped ("binned") into releases classes depending on the timing of the accident and the conditions of the core, vessel, containment, and release characteristics for the sequence. Each Design Alternative is evaluated in terms of how it might affect each release class. For this analysis it is assumed that each Design Alternative is perfect: that is, if installed it completely eliminates all failures associated with the systems for which it is designed to be an alternative or addition. This implies that each Design Alternative is also tied to perfect support systems. This is a conservative upper limit approach since it overestimates the benefits associated with any design addition. If a Design Alternative is cost beneficial using this screening approach, then a more detailed analysis could be performed.

The Design Alternatives can be divided into two groups. One group prevents core damage and the other group protects the containment or reduces the releases. For the Design Alternatives that prevent core damage, the frequency of affected release classes was decreased to zero and the risk reduction was calculated. For example, an ideal pressurizer auxiliary spray Design Alternative is assumed to eliminate all core melt risk of a Steam Generator Tube Rupture (SGTR) by always getting the plant depressurized and into shutdown cooling. Therefore the frequency of release classes associated with SGTR was reduced to zero and a risk reduction was calculated.

Some Design Alternatives protect the containment or reduce the amount of radioactive material that is released in an accident. These Alternatives reduce the consequence of the accident and therefore reduce the risk (risk = frequency x consequence). Using the CRAC2 Code[5], the consequence in terms of dose to the general population is calculated for the ALWR site. This site conservatively represents the characteristics of most potential sites[6]. For Design Alternatives that prevent containment failure, the releases were changed from that of a failed containment to that of an intact containment and the risk was reevaluated. For Alternatives that filter the releases, an appropriate consequence and risk reduction was calculated.

3.2 COST ESTIMATES

In order to evaluate the effectiveness of the design alternatives, the benefits were compared to the costs of the Alternatives. The rough order of magnitude costs were estimated for each modification. These costs represent the incremental costs that would be incurred in incorporating the alternative in a new plant. The cost estimate for each of the modifications is given in Section 4 where the modification is discussed.

The cost estimates were intentionally biased on the low side, but all known or reasonably expected costs were accounted for in order that a reasonable assessment of the minimum cost would be obtained. Actual plant costs are expected to be higher than indicated in this evaluation. All costs are referenced to 1992 U.S. dollars.

3.3 COST BENEFIT COMPARISON

As described in Section 3.1, the benefit of a design alternative is risk reduction which was evaluated in terms of reduced exposure of the general population (in units of person-rem/y). The cost of additional equipment is in dollars, a one-time initial capital cost. To compare these two numbers, a common measure must be used. In this analysis, the risk reduction was converted to a single capital benefit which can be directly compared with the capital cost.

The benefits of a particular modification were defined as the risk reduction to the general public. Offsite factors evaluated were limited to health effects to the general public based on total exposure (in person-rem). Consistent with the standard used by the NRC to evaluate radiological impacts, health effects costs were evaluated based on a value of $1,000 per offsite person-rem averted due to the design modification. This factor converts person-rem/y to $/y.

The annual benefit in $/y is converted to an single capital benefit using a levelized capital cost rate. Using the method and values described in Ref. 7, and the economic assumptions given in Table 3-1, a levelized capital cost rate of 17.9% is estimated. The Design Alternative results are not very sensitive to the detailed economic assumptions used in calculating a levelized capital cost rate.

The offsite costs for other items, such as relocation of local residents, elimination of land use and decontamination of contaminated land are not considered. Economic losses, replacement power costs and direct accident costs incurred by the plant owner also are not considered in this evaluation. Again, this gives a conservative estimate of the net benefit.

The analysis presented here conservatively neglects any annual costs associated with the operation of the Design Alternatives. These Alternatives would have to be tested and maintained at regular intervals. Regular training would also be required. In a more detailed analysis, such costs would be converted to an annual cost and be used to reduce the annual benefits.

TABLE 3-1

ECONOMIC ASSUMPTIONS FOR LEVELIZED CAPITAL COST RATE

ASSUMPTIONS	VALUE
BOND (DEPT) INTEREST RATE, %	10.48
DEPT FRACTION	0.55
RETURN ON EQUITY, %	12.48
INCOME TAX RATE, %	50.0
RATE OF INFLATION, %	4.0
ANNUAL PROPERTY TAX + INSURANCE, %	2.0
TAX DEPRECIATION LIFE, YRS.	20.0
COMPONENT ECONOMIC LIFE, YRS.	24.0
RESULTING LEVELIZED CAPITAL COST RATE, %	17.9

4.0 PRA RELEASE CLASSES

In assessing the risk reduction of each Design Alternative, the potential for each Design Alternative to reduce the frequency of occurrence or the consequence of each release class (RC) is assessed. To do this, an understanding of each RC is required.

In the System 80+ PRA[1], the containment event analysis describes the possible accident pathways in a containment event tree (CET). This CET was developed so that each end point of an accident sequence uniquely specified the mode of containment failure and the status of the various phenomena which have the potential to affect the source term characteristics. Therefore, each of the accident end points is a distinct release class. A release class (RC) can be fully characterized by the following parameters:

A) its frequency of occurrence,
B) the isotopic content and magnitude of the release,
C) the energy of the release,
D) the time of the release,
E) the duration of the release, and
F) the location of the release.

The RC frequency is determined directly from the cumulative frequency for its respective containment event tree end point. The location of the release was assigned as follows:

1) For overpressure containment failure RCs, the release was assumed to occur at the top of the containment building. This is at an elevation of 52.8 meters above grade.

2) For containment bypass RCs initiated by an interfacing systems LOCA and for containment melt-through RCs, the release from containment occurs in the region of the auxiliary building located below the containment sphere. The actual release to the environment occurs at grade level.

3) For all other RCs, the releases are assumed to occur at grade level.

MAAP[4] analyses were used to determine the isotopic content and magnitude of the source term and the time of the release. In general, releases were calculated for a period of 24 hours from the time of containment failure or from the time of vessel failure for containment bypass and containment isolation failure RCs. The MAAP analyses are based on specific accident sequences. To select the appropriate accident sequence for a specific RC, the following process was used:

1) The Plant Damage State (PDS) with the largest contribution to the RC's total frequency was determined.
2) The dominant plant accident sequence was used for the RC. This defined the initiating event and the status of the various plant systems.

3) The Containment Event Tree (CET) and the plant accident sequence definition were reviewed to determine if any special phenomenological conditions had to be specified.
4) A containment failure pressure, failure time or failure condition was specified based on the RC definition.

The following sections present a brief description for each release class with a frequency greater than or equal to 1.0E-10. The numbering system and order of the release classes is developed in the PRA and represents the approximate order they appear in the CET. The data for each release class is summarized in Table 4-1.

4.1 RELEASE CLASS RC7.1

Release class RC7.1 covers the releases from an intact containment. These releases are attributable to design basis leakage from containment. For this analysis, a design basis leakage of 0.34 volume percent per day was used which is consistent with that used in the Chapter 15 safety analysis of CESAR-CD. The containment leakage was adjusted to account for containment pressure variations throughout the accident. The cumulative frequency for this release class is 1.69E-6 per year.

This release class is characterized by a station blackout with battery depletion followed by late recovery of AC power and containment heat removal. The accident is initiated by a loss of offsite power with failure of the diesel generators and the alternate AC power source. The batteries supply control power for 8 hours and emergency feedwater is supplied by the turbine-driven emergency feedwater pumps. With battery depletion at 8 hours, emergency feedwater is assumed to be lost. Cavity flood is initiated just prior to battery depletion. Vessel failure is assumed to occur at about 15.5 hours. It was assumed that the hydrogen ignitors function to control hydrogen concentrations. At 48 hours, power was recovered and containment heat removal was reestablished, thus effectively terminating the accident sequence. The releases were assumed to occur at ground level. The characteristics of this release class are summarized in Table 4-1.

4.2 RELEASE CLASS RC6.2

Release class RC6.2 covers releases associated with a containment melt-through into the nuclear annex building with vaporization releases and no radioisotope scrubbing in the containment atmosphere prior to release. The cumulative release frequency for this release class is 1.38E-8 per year.

This release class is characterized by a large cold leg LOCA with failure of safety injection to initiate. Containment spray and containment heat removal are not available, and the cavity is not flooded. The hydrogen ignitors are assumed to be available for hydrogen control. Vessel failure is assumed to occur at 1.83 hours and the corium drops into a dry cavity. By 330 hours, the corium is assumed to have ablated through the lower subsphere region of the reactor building and into the shadow area of the nuclear annex building. The releases pressure. No deposition of radioactive material in the nuclear annex building is credited. The final release from the nuclear annex building is assumed to occur at ground level. The characteristics of this release class are summarizedin Table 4-1.

4.3 RELEASE CLASS RC6.4

Release class RC6.4 covers releases associated with a melt-through of the subsphere of the reactor building into the nuclear annex building with vaporization and revaporization releases and no radioisotope scrubbing in the containment atmosphere prior to release. The cumulative release frequency for this release class is 7.28E-9 per year.

This release class is characterized by a station blackout with battery depletion. The accident is initiated by a loss of offsite power with failure of the diesel generators and the alternate AC power source. The batteries supply control power for 8 hours and emergency feedwater is supplied by the turbine-driven emergency feedwater pumps. With battery depletion at 8 hours, emergency feedwater is assumed to be lost. The cavity is not flooded prior to battery depletion. The hydrogen ignitors function to control the hydrogen concentration. Vessel failure is assumed to occur at about 19.0 hours. Just prior to vessel failure, a hot leg failure occurs so that the corium is ejected at low pressure into a dry cavity. By 180 hours, the corium ablates through the lower subsphere of the reactor building and into the shadow area of the nuclear annex building located below the reactor building. The releases are discharged through the nuclear annex building, initially at a relatively high pressure. No deposition of radioactive material in the nuclear annex building is credited. The final release from the nuclear annex building occurs at ground level. The characteristics of this release class are summarized in Table 4-1.

4.4 RELEASE CLASS RC5.1

Release class RC5.1 covers releases associated with a late containment overpressure failure with no vaporization or revaporization releases and no radioisotope scrubbing in the containment atmosphere prior to release. The cumulative release frequency for this release class is 3.63E-8 per year.

This release class is characterized by a station blackout with battery depletion. The accident is initiated by a loss of offsite power with failure of the diesel generators and the alternate AC power source. The batteries supply control power for 8 hours and emergency feedwater is supplied by the turbine-driven emergency feedwater pumps. With battery depletion at 8 hours, emergency feedwater is assumed to be lost. Cavity flood is initiated just prior to battery depletion. The hydrogen ignitors function to control the hydrogen concentration. Vessel failure is assumed to occur at about 15.8 hours. Neither power nor containment heat removal are recovered. The containment fails due to overpressure at 128 hours. The containment failure, and hence, the releases are assumed to occur at the top of the containment at an elevation of 52.8 meters. The characteristics of this release class are summarized in Table 4-1.

4.5 RELEASE CLASS RC5.2

Release class RC5.2 covers releases associated with a late containment overpressure failure with revaporization releases, but no vaporization releases and no radioisotope scrubbing in the containment atmosphere prior to release. The cumulative release frequency for this release class is 6.59E-8 per year.This release class is characterized by a station blackout with battery depletion. The accident is initiated by a loss of offsite power with failure of the diesel generators and the alternative AC power source. The batteries supply control power for 8 hours and emergency feedwater is supplied by the turbine-driven emergency feedwater pumps. With battery depletion at 8 hours, emergency feedwater is assumed to be lost. Cavity flood is initiated just prior to battery depletion. The hydrogen ignitors function to control the hydrogen concentration. Vessel failure is assumed to occur at about 18.6 hours. Just prior to vessel failure, a hot leg failure occurs so that the corium is ejected at low pressure. Neither power nor containment heat removal are recovered. The containment fails due to overpressure at 94 hours. The containment failure, and hence, the releases are assumed to occur at the top of the containment at an elevation of 52.8 meters. The characteristics of this release class are summarized in Table 4-1.

4.6 RELEASE CLASS RC4.1

Release class RC4.1 covers releases associated with an early containment steam overpressure failure with no vaporization or revaporization releases but with radioisotope scrubbing from the containment atmosphere prior to release. The cumulative release frequency for this release class is 2.12E-10 per year.

This release class was characterized by a large cold leg LOCA with failure of safety injection from time zero. Containment spray and containment heat removal are available and the cavity is flooded. The hydrogen ignitors are assumed to be available for hydrogen control. Vessel failure occurred at 1.8 hours and the corium dropped into a wet cavity. The containment is forced to fail at a pressure of about 25 psi at 5.6 hours. (Note: the highest containment pressure during this event was about 40 psia immediately after the initiation of the LOCA. The containment was not failed at this point because there was no core damage and the vessel was still intact.) The containment failure, and hence, the releases are assumed to occur at the top of the containment at an elevation of 52.8 meters. The characteristics of this release class are summarized in Table 4-1.

4.7 RELEASE CLASS RC4.2

Release class RC4.2 covers releases associated with an early containment overpressure failure due to direct containment heating (DCH) with no radioisotope scrubbing in the containment atmosphere prior to release. The cumulative release frequency for this release class is 2.17E-10 per year.

This release class is characterized by a station blackout with battery depletion. The accident is initiated by a loss of offsite power with failure of the diesel generators and the alternative AC power source. The batteries supply control power for 8 hours and emergency feedwater is supplied by the turbine-driven emergency feedwater pumps. With battery depletion at 8 hours, emergency feedwater is assumed to be lost. The cavity is flooded just prior to battery depletion. The hydrogen ignitors are not functioning to control the hydrogen concentration. Vessel failure is assumed to occur at about 15.8 hours and the corium is ejected into the cavity at high pressure. A simultaneous hydrogen burn occurs and containment was assumed to fail at the peak pressure thus produced. Although MAAP calculates a peak pressure of about 31 psia at 16.5 hours which is below the containment design pressure, the pressures potentially attainable during a DCH event could be in the 100 psia range. The containment failure, and hence, the releases are assumed to occur at the top of the containment at an elevation of 52.8 meters. The characteristics of this release class are summarized in Table 4-1.

4.8 RELEASE CLASS RC3.1

Release class RC3.1 covers releases associated with a containment overpressure failure prior to core damage with no vaporization releases and no radioisotope scrubbing in the containment atmosphere prior to release. The cumulative release frequency for this release class is 4.82E-8 per year.

This release class is characterized by a transient involving loss of main feedwater with a subsequent failure to deliver emergency feedwater. Feed and bleed cooling is successfully established with the bleed valves being opened shortly after primary safety valve lift. However, containment heat removal is not available so the containment temperature and pressure increase until containment fails at about 36 hours. Safety injection fails just after containment failure. Cavity flooding is initiated just after containment failure. Vessel failure is assumed to occur at 51.2 hours. The containment failure, and hence, the releases are assumed to occur at the top of the containment at an elevation of 52.8 meters. The characteristics of this release class are summarized in Table 4-1.

4.9 RELEASE CLASS RC2.2

Release class RC2.2 covers releases associated with a containment isolation failure with no vaporization or revaporization releases prior to release. The cumulative release frequency for this release class is 1.12E-9 per year.

This release class is characterized by a small (0.02 ft^2) hot leg LOCA with failure of safety injection and emergency feedwater. The safety depressurization valves fail to open. Vessel failure is assumed to occur at 3 hours, causing the corium to be initially ejected into the cavity under high pressure. Containment spray is available for containment heat removal and fission product scrubbing. Because the containment sprays are functioning, the cavity is also flooded prior to vessel failure. At time zero, a containment isolation failure equivalent to a 6 inch diameter hole is assumed to occur. The containment isolation failure, and hence, the releases are assumed to occur at ground level. The characteristics of this release class are summarized in Table 4-1.

4.10 RELEASE CLASS RC2.4

Release class RC2.4 covers releases associated with a containment isolation failure with no vaporization or revaporization releases but with no radioisotope scrubbing in the containment atmosphere prior to release. The cumulative release frequency for this release class is 2.44E-9 per year.

This release class is characterized by a station blackout with battery depletion.The accident is initiated by a loss of offsite power with failure of the diesel generators and the alternate AC power source. The batteries supply control power for 8 hours and emergency feedwater is supplied by the turbine-driven emergency feedwater pumps. With battery depletion at 8 hours, emergency feedwater is assumed to be lost. Cavity flooding is initiated just prior to battery depletion. The hydrogen ignitors function to control the hydrogen concentration. Vessel failure is assumed to occur at about 15.5 hours causing the corium to be ejected into the cavity at high pressure. Neither power nor containment heat removal are recovered. At time zero a containment isolation failure equivalent to a 6 inch diameter hole is assumed to occur. The containment isolation failure, and hence, the releases are assumed to occur at ground level. The characteristics of this release class are summarized in Table 4-1.

4.11 RELEASE CLASS RC1.4

Release class RC1.4 covers releases associated with a containment bypass failure with no vaporization releases and no radioisotope scrubbing in the containment atmosphere prior to release. The cumulative release frequency for this release class is 7.02E-9 per year.

This release class is characterized by a steam generator tube rupture with a stuck open Atmospheric Dump Valve (ADV) on the affected generator. The accident is initiated by a tube rupture in one of the steam generators. Safety injection is successfully initiated and emergency feedwater is delivered to the intact generator. However, the ADV on the affected generator is assumed to fail open at about 30 minutes. As a result of this, primary inventory and IRWST inventory continued to flow through the ruptured tube and is exhausted to the atmosphere. Actions to stabilize the plant and terminate the accident are unsuccessful.

Cavity flood is manually initiated while there is still sufficient inventory in the IRWST to successfully flood the cavity. At 10.6 hours, the IRWST inventory is depleted, and safety injection flow is lost. (At this point emergency feedwater is also assumed to be lost.) At 29.9 hours, core uncovery is assumed to occur and the major releases begins. At 32.6 hours the vessel is assumed to fail, and the corium is ejected into the cavity. The releases through the ADV are assumed to occur at ground level. The characteristics of this release class are summarized in Table 4-1.

4.12 RELEASE CLASS RC1.7

Release class RC1.7 covers releases associated with a containment bypass failure with vaporization releases and no radioisotope scrubbing in the containment atmosphere prior to release but with the source term attenuated due to deposition in the nuclear annex building. The cumulative release frequency for this release class is 3.00E-9 per year.

This release class is characterized by a failure of the check and isolation valves in one Shutdown Cooling System (SCS) line resulting in a catastrophic failure of this line outside containment (Interfacing System LOCA). Safety injection is successful. However, the primary system inventory and the IRWST inventory is being discharged outside of containment. At 3.5 hours, the IRWST inventory is depleted and safety injection flow is lost. Vessel failure occurredat 7.1 hours, and the corium is dropped into a dry cavity. The release path is through the broken SCS line into the lower levels of the nuclear annex building in the shadow region. The radioactive material then passes upward through the nuclear annex building until it eventually finds a release point to the environment at ground level. The characteristics of this release class are summarized in Table 4-1.

Table 4-1

Release Parameter Data for System 80+ Release Classes

RELEASE CLASS	RELEASE START	RELEASE DURATION	RELEASE HEIGHT	RELEASE ENERGY
RC7.1	15.5 Hrs	10.0 Hrs	0 M	1.44E+4 cal/sec
RC6.2	330.0 Hrs	1.0 Hrs	0 M	3.08E+8 cal/sec
RC6.4	180.0 Hrs	1.0 Hrs	0 M	2.21E+8 cal/sec
RC5.1	128.0 Hrs	1.0 Hrs	52.8 M	[illegible].64E+8 cal/sec
RC5.2	94.0 Hrs	1.0 Hrs	52.8 M	3.83E+8 cal/sec
RC4.1	5.6 Hrs	1.0 Hrs	52.8 M	1.74E+7 cal/sec
RC4.2	16.5 Hrs	1.0 Hrs	52.8 M	1.40E+7 cal/sec
RC3.1	36.0 Hrs	10.0 Hrs	52.8 M	1.58E+6 cal/sec
RC2.2	3.0 Hrs	10.0 Hrs	0 M	5.68E+4 cal/sec
RC2.4	15.5 Hrs	10.0 Hrs	0 M	9.66E+5 cal/sec
RC1.4	29.9 Hrs	10.0 Hrs	0 M	2.87E+3 cal/sec
RC1.7	7.1 Hrs	1.0 Hrs	0 M	1.52E+7 cal/sec

5.0 DESIGN ALTERNATIVES

Eleven design alternatives were evaluated. These were selected based on the SAMDAs for the Limerick plant[2] and a review of the System 80+ PRA[1]. In addition, suggestions from C-E personnel with technical expertise in containment response were employed. Design alternatives from earlier plant studies were also considered.

Each release class which is described in Section 4 was evaluated for total person-rem exposure using CRAC2. Table 5-1 gives the initiating frequency, dose at 0.5 miles, and total person-rem dose for the twelve release classes with initiating frequencies greater than 1.0E-10. The risk for each release class is the product of frequency (events/year) times the total person-rem exposure per event. This product gives person rem per year and is a measure of the risk. The total risk of the dominant release classes is 5.48 person-rem/y. These results are for the ALWR site which is representative of most of the current U.S. sites[6].

Table 5-2 summarizes the accident characteristics for each release class. These are the dominant sequences of the binned accidents. For each Design Alternative, the release class was reviewed assuming that the Design Alternative worked perfectly (failure rate = 0.0). This means that each Design Alternative had perfect support systems, power supplies and heat sinks. In addition, for each Design Alternative, no other failure modes were considered when the Design Alternative was employed. For example, when the pressurizer auxiliary spray Design Alternative is employed to ensure that the primary coolant pressure can be decreased to enter SCS operation, the SCS system is assumed to always work. This represents an upper limit scoping analysis and maximizes the benefit of each Design Alternative. If a Design Alternative is cost beneficial in this analysis then a more detailed analysis addressing the actual failure rate of the Design Alternative can be undertaken.

The Design Alternatives can be divided into two groups. One group prevents core damage and the other group protects the containment or reduces the releases. For the Design Alternatives that prevent core damage, the frequency of affected release classes are put equal to zero and the total risk reduction is calculated. This group includes the high capacity HPSI systems, improved DC Battery and EFWS, ATWS pressure relief valves, improved pressurizer auxiliary spray, improved primary depressurization system, and advanced reactor coolant pump seal injector.

For the Design Alternatives that protect the containment, the releases are changed from that of a failed containment to that of an intact containment and then the risk is reevaluated. These Design Alternatives include the advanced containment sprays, filtered vent, concrete composition, reactor vessel exterior cooling, and H2 ignitors. The person-rem exposure for an intact containment (RC7.1) is substituted for the higher dosed when mitigation type Design Alternatives are evaluated. For the filtered vent Design Alternative, the doses represent immediate release of the noble gasses after scrubbing (RC4.1).

The following sections discuss each Design Alternative.

5.1 CONTAINMENT SPRAY

A perfect containment spray system prevents the high pressure containment failures caused by slow steam pressurization (H2 burns are expected to still be possible). This system is assumed to have a perfect power supply and heat sink and work in all release classes where the containment is challenged regardless of the sequence of events or equipment failures that led t core damage and containment challenge. These assumptions overestimate the benefits of this design alternative. It also reduces the releases in all the release classes where no scrubbing of fission products was initially predicted. This Design Alternative reduces the dose of five of the release classes to the dose associated with an intact containment (RC7.1) but does not affect the frequencies of the events. The risk (frequency x dose) of these five RCs therefore has been reduced. The total risk is reduced by 90% (see Table 5-3). Using a risk conversion factor of $1,000 per person-rem, this Design Alternative would have an annual value of $4,900/y. The annual benefit of the Design Alternative could be converted to a capital benefit using the levelized capital cost rate of 17.9% developed in Section 2. The ideal containment spray system would be cost beneficial if it could be installed for less than $27,600 and have no maintenance and testing costs. Any annual operating costs would have to be subtracted from the annual risk reduction benefits.

The above analysis assumes that the system has a failure rate of 0.0 in terminating the accident by protecting the containment. The capital benefit is inversely proportional to the reliability of the system. For example, if the design had a conditional reliability of 0.5 in these accident sequences, then the Design Alternative would have to cost less than $13,800 to be cost effective.

Estimating the cost to design and build a perfect containment spray system is not realistically possible. However, one option would be to provide piping from the containment spray header to the exterior of the Nuclear Annex for a temporary hook-up of a fire truck should all containment spray and shutdown cooling pumps be unavailable. The cost of the additional Class 2 piping, pipe supports, valves, on-site fire truck with the required pumping capacity and pump head and building to store the fire truck is estimated to exceed $1.5 million.

5.2 FILTERED VENT

The filtered vent Design Alternative prevents all slow high pressure containment failures and therefore reduces the doses in three release classes (see Table 5-4). Since the noble gases are not held up, the doses associated with the filtered vent were approximated with the doses associated with a scrubbed release without any fission product vaporization (RC4.1). As with all mitigation Design Alternatives, this design alternative does not affect the frequencies of the RCs. This Design Alternative reduces the risk by 86%. Using a value of $1,000 per person-rem avoided, this Design Alternative has a benefit of $4,700/y. Using a levelized capital cost rate of 17.9%, a system with a capital cost of $26,300 would just be cost effective.

The Swiss recently purchased a filtered vent system for one of their BWRs for approximately \$3 million[8] for equipment along. With the building structure to house the equipment the total cost is estimated to exceed \$10 million.

5.3 DC BATTERIES AND EFWS

This Design Alternative addresses the release classes where emergency feedwater is lost after battery depletion during a station blackout. The System 80+ design already has an improved battery system that will carry the DC loads for 8 hours. There are still accident sequences where the batteries are depleted and emergency feedwater is lost leading to core damage. The improved DC batteries and EFWS Design Alternative is assumed to have the capability to remove decay heat using batteries and the turbine feedwater pump for whatever time period that is required (without any failures). This Design Alternative prevents core damage and therefore removes six of the release classes. The risk is reduced by 69% (see Table 5-5). Using a \$1,000 per averted person/rem and a levelized cost rate or 17.9%, such a system would be cost beneficial if it cost less than \$21,100.

Design of a battery system with unlimited capacity is not possible. However, to increase the existing battery capacity for the EFW pumps from the current System 80+ design capacity of 8 hours to 72 hours will require 9 times the number of current battery cells and thus approximately 9 times the space for building storage. The increased building space will also increase the HVAC requirements. The cost for the extra battery cells, building volume and increased HVAC requirements is estimated to exceed \$2 million.

5.4 RCP SEAL COOLING

The System 80+ employs a type of Reactor Coolant Pumps (RCP) seal which can withstand a loss of cooling and not produce a LOCA. This type of seal design has been employed in the operating C-E plants and experience has shown that the seals do not fail when seal cooling is lost[9]. Therefore seal failure was not modeled in the System 80+ PRA. To estimate a risk reduction associated with an improved seal cooling system, a different approach is used than used for the other Design Alternatives. It is assumed that for all station blackouts that last four hours, the seal fails and leads to core damage (frequency = 3.64E-8/y). It is also assumed that total loss of the component cooling water (CCW) also leads to seal failure and core damage. Appendix A contains the fault tree for the analysis of total loss of CCW. Total loss of CCW requires loss of four CCW pumps or loss of the four service water pumps and would probably require a common cause failure. The frequency for this event is 2.25E-5/y. The containment is assumed to remain intact (conditional containment failure probability = 0.099) and the expected dose is 4.8E+4 (RC7.1). Therefore, the risk of seal loss leading to core damage is 1.08 person-rem/y (2.25E-5 events/y x 4.8E+4 person-rem/event). The RCP seal cooling Design Alternative is assumed to completely eliminate this risk and therefore reduces the total risk of the plant by 16.5%. Using a value of $1,000 per person-rem avoided, this Design Alternative has a benefit of $1,080/y. Using a levelized capital cost rate of 17.9%, a system with a capital cost of $6,034 would just be cost effective.

The reliability of the reactor coolant pump seal cooling could be improved by adding a small dedicated positive displacement pump for diverse seal injection. This pump would be powered from the Alternate AC Source (Combustion Turbine). It would take suction from the boric acid storage tank and would connect to the normal supply line for seal injection inside the containment. This design addition will provide additional diversity for RCP seal cooling and provide a seal cooling system that is not dependent on CCW. The cost of the additional pump, piping, valves, containment penetration, instrumentation, electrical cable and building space is estimated to exceed $100,000.

5.5 PRESSURIZER AUXILIARY SPRAY

This Design Alternative was introduced to specifically address steam generator tube rupture (SGTR) and eliminates the one RC initiated by SGTR (RC1.4). The analysis assumes that during a SGTR, the auxiliary spray will always depressurize the primary system to the SCS operation mode with sufficient speed and the SCS system will always remove decay heat. The elimination of the risk of SGTR in the System 80+ design has a 6.7% risk reduction (see Table 5-6). Using a $1,000 per averted person-rem and a levelized cost rate or 17.9%, such a system would be cost beneficial if it cost less than $2,050.

Designing a perfect pressurizer auxiliary spray system is not possible. However, increased reliability and diversity can be obtained by increasing the redundancy and diversity of the pressurizer spray valves and providing a diverse positive displacement charging pump that is powered from a diverse power source. The reliability of the shutdown cooling system can be improved by providing a diverse shutdown cooling pump with a diverse power source and providing a diverse heat sink. The cost for the additional components, piping, power supplies, instrumentation and building volume is estimated to exceed $5 million.

5.6 ATWS PRESSURE RELIEF VALVES

This Design Alternative consists of a system of relief valves that can prevent any equipment damage from a primary coolant pressure spike in an ATWS accident sequence. This Design Alternative is assumed to eliminate all the ATWS core damage sequences. The initiating frequency for each release class consists of one or more accident sequence. The descriptions of each RC given in Section 3 and Table 4-2 are typical accident sequences that contribute to the release class frequency. ATWS events represent 55% of the initiating frequency of RC7.1 and 50% of the initiating frequency of RC4.2. The elimination of the ATWS events therefore reduces the frequency of RC7.1 by 55% and RC1.4 by 50%. This leads to a total risk reduction of 4.2% (see Table 5-7). Using a $1,000 per averted person-rem and a levelized cost rate or 17.9%, such a system would be cost beneficial if it cost less than $1,290.

To implement this design alternative, the safety relief valve sizes and discharge piping size would need to be increased. It may also require additional safety relief valves and thus additional safety relief valve discharge piping and supports. In addition, the size and possible the number of safety valve nozzles on top of the pressurizer would need to be increased. The cost of this design alternative is estimated to exceed $1 million.

5.7 CONCRETE COMPOSITION

The containment building for System 80+ uses a spherical containment with an area below it that can be considered part of the nuclear annex building. It is assumed that in accident sequences where corium/concrete interaction are not stopped, containment failure would lead to releases through the nuclear annex building. This Design Alternative assumes that an ideal concrete composition could be developed that prevents basemat melt-through. This would reduce two RCs (RC6.2 and RC6.4) where basemat melt-through is modeled and would produce a 2.5% risk reduction. Table 5-8 summarizes the risk reduction. Using a $1,000 per averted person-rem and a levelized cost rate or 17.9%, such a system would be cost beneficial if it cost less than $765.

An advanced concrete composition to prevent corium/concrete interaction is not currently available. However, additional concrete could be added to increase the time before containment failure would occur. In order to increase the thickness of the concrete at the bottom of the reactor cavity the containment diameter would need to be increased. An increase in containment diameter also requires an increase in containment plate thickness. The cost of increasing the concrete thickness by two feet is estimated to exceed $5 million.

5.8 REACTOR VESSEL EXTERIOR COOLING

A reactor vessel exterior cooling system is assumed to prevent vessel melt-through and subsequent basemat attack. It also prevents Direct Containment Heating (DCH) sequences by preventing vessel failure because it promotes cooling the corium in the vessel. This Design Alternative reduces the consequences of four RCs by reducing the doses to that of an intact containment (RC7.1). It has a dose risk reduction of 2.5% (see Table 5-9). Using a $1,000 per averted person-rem and a levelized cost rate or 17.9%, such a system would be cost beneficial if it cost less than $765.

The current arrangement for the IRWST will allows wetting the bottom of the reactor vessel through the cavity flood system. However, the water level of the IRWST is not high enough to flood the entire cavity for reactor vessel exterior cooling. Reactor vessel exterior cooling could be accomplished by raising the elevation of the IRWST such that the entire reactor cavity can be flooded up the reactor flange. Raising the elevation of the IRWST would require an increase in containment diameter. In addition, the reactor vessel would have to be qualified to demonstrate that wetting the exterior will not cause a thermal shock which would cause vessel failure. To implement this design alternative is estimated to exceed $5.5 million.

5.9 H2 IGNITORS

Ideal hydrogen (H_2) ignitors would prevent release classes associated with containment failures from hydrogen burns or explosions. The System 80+ design already has H_2
ignitors and therefore only one release class (RC4.2) has containment failure from hydrogen burning. This Design Alternative reduces the doses to that of an intact containment and therefore reduces the risk by 0.1% (see Table 5-10). Such a system would have to have a negligible cost to be cost beneficial.

Providing perfect hydrogen ignitors which have no probability of failure is not possible. However, the reliability of the hydrogen ignitors could be improved by either providing dedicated batteries for the existing design (glow plug ignitors) or by providing catalytic hydrogen ignitors which do not require a power source. Since catalytic hydrogen ignitors are not fully developed, possible failure modes, including common cause failure modes, are not known. Therefore, they are not being selected for the System 80+ design at this time. The addition of dedicated batteries for the hydrogen ignitors along with the additional equipment such as battery chargers and invertor and the additional building space to store this equipment is estimated to exceed $1 million.

5.10 HIGH PRESSURE SAFETY INJECTION

The System 80+ design has a very reliable four train HPSI system to begin with. The high pressure safety injection Design Alternative assumes that most sequences with HPSI failures can be eliminated (see Table 5-11). This Design Alternative eliminates RC4.1 and 2.2. For these two RCs, the core damage frequency goes to zero. It is assumed that the HPSI Design Alternative does not prevent core damage for RC3.1. In this release class, an early containment failure produces HPSI pump cavitation because the rapid depressurization caused the primary coolant to flash. This Design Alternative produces a 0.004% risk reduction and therefore would have to have a negligible cost to be cost beneficial.

Providing a perfect high pressure safety injection system is not possible; however, the reliability of the system could be improved slightly by adding two more diesel generators. The cost of adding two more diesel generators, with the associated support systems and the building space is estimated to exceed $20 million.

5.11 RCS DEPRESSURIZATION

The System 80+ design has motor operated relief valves (MORVs) that permit residual heat removal using the valves and HPSI pumps in a "feed and bleed" mode of operation. This Design Alternative models a perfect MORV system that permits the primary coolant system to be quickly depressurized so that the Safety Injection Tanks (SITs) and Safety Injection pumps are effective in getting coolant into the core and removing decay heat. This Design Alternative eliminates the one RC where the HPSI fails or feed and bleed fails because of failure of the four train HPSI system. It assumes that the SITs, and RHR systems also have zero failures for these sequences and therefore the core damage frequency goes to zero for RC2.2. It is assumed that the RCS Depressurization system does not prevent core damage for RC3.1. In this release class, an early containment failure produces HPSI pump cavitation because the rapid depressurization caused the primary coolant to flash. This is the least effective design alternative considered in this study with only a 0.002% dose risk reduction (see Table 5-12). Such a system would have to have negligible costs to be cost beneficial.

Designing a perfect safety depressurization system is not possible. However, increased reliability and diversity of the syste[illegible] be obtained by increasing the redundancy of the safety depressurization val[illegible]/or providing valves that are diverse. Providing the additional valves, p[illegible] and instrumentation is estimated to exceed $500,000.

TABLE 5-1

FREQUENCIES AND PERSON-REM EXPOSURES FOR RELEASE CLASSES

RC#	INITIATING FREQUENCY	PERSON-REMS AT 0.5MI	TOTAL PERSON-REM	RISK PERSON-REM/Y
7.1	1.69E-6	25.9	4.80E+4	8.11E-2
6.2	1.38E-8	28.2	1.24E+6	1.71E-2
6.4	7.28E-9	566.0	1.66E+7	1.21E-1
5.1	3.63E-8	630.0	1.24E+7	4.50E-1
5.2	6.59E-8	2510.0	4.72E+7	3.11E-0
4.1	2.12E-10	76.3	5.39E+5	1.14E-4
4.2	2.17E-10	3830.0	2.14E+7	4.64E-3
3.1	4.82E-8	1540.0	2.60E+7	1.25E-0
2.2	1.12E-9	36.7	9.01E+4	1.01E-4
2.4	2.44E-9	4280.0	8.76E+6	2.14E-2
1.4	7.02E-9	71900.0	5.22E+7	3.66E-1
1.7	3.00E-9	6140.0	1.94E+7	5.82E-2

TABLE 5-2

SUMMARY DESCRIPTION OF RELEASE CLASSES

RC#	INIT FREQ	TOTAL PERSON-REM	DESCRIPTION
7.1	1.69E-6	4.80E+4	BLACKOUT, DC DEPLETION, RESTORED CONT HEAT REMOVAL, CONTAINMENT INTACT
6.2	1.38E-8	1.24E+6	LARGE LOCA, NO SI, NO CAVITY FLOOD, CONT MELT-THRU, VAPORIZATION RELEASE, NO SCRUBBING
6.4	7.28E-9	1.66E+7	BLACKOUT, DC DEPLETION, RESTORE CONT HR, NO CAVITY FLOOD, CONT MELT-THRU, VAPORIZATION RELEASE, RE-VAPORIZATION, NO SCRUBBING
5.1	3.63E-8	1.24E+7	BLACKOUT, DC DEPLETION, CAVITY FLOOD, NO AC, NO CONT HR, LATE OVERPRESSURE F, NO VAP., NO REVAP., NO SCRUBBING
5.2	6.59E-8	4.72E+7	BLACKOUT, DC DEPLETION, CAVITY FLOOD, NO AC, NO CONT HR, LATE OVERPRESSURE F, NO VAP., REVAP., NO SCRUBBING
4.1	2.12E-10	5.39E+5	LARGE LOCA, NO SI, CONT SPRAY & HR, CORIUM DROP INTO WET CAVITY, EARLY STEAM OVERPRESSURE F, NO VAPORIZATION, NO REVAP., SCRUBBING
4.2	2.17E-10	2.14E+7	BLACKOUT, DC DEPLETION, CAVITY FLOOD, H2 IGNITORS F, H2 BURN + DCH, EARLY OVERPRESSURE F, NO VAP., NO REVAP., NO SCRUBBING
3.1	4.82E-8	2.60E+7	LOSS MFW+EFW, FEED+BLEED, NO CONT HR, CONT F PRIOR TO CM, SI FAILS, CAVITY FLOOD, NO VAP., NO REVAP., NO SCRUBBING
2.2	1.12E-9	9.01E+4	SMALL LOCA, SI F, EFW F, DEP. VLVS F OPEN, CONT ISOLATION F, NO VAP., NO REVAP., SCRUBBING
2.4	2.44[illegible]	8.76E+6	BLACKOUT, DC DEPLETION, CAVITY FLOOD, H2 IGNITORS OK, NO AC, NO CONT HR, CONT ISOLATION F, NO VAP., NO REVAP., NO SCRUBBING
1.4	7.02E-9	5.22E+7	STEAM GEN TUBE RUPT, SI OK, EFW OK, ADV STUCK OPEN, PRIMARY INVENTORY LOST, CAVITY FLOODED, CONT BYPASS, NO VAP., NO REVAP., NO SCRUBBING
1.7	3.0[illegible]	1.94E+7	CHECK AND ISOL VLV F IN RHR (INTERFACE LOCA), SI OK, PRIM INVENTORY DEPLETED, DRY CAVITY, CONT BYPASS, VAP., NO SCRUBBING, DEPOSITION IN AUX BLDG

TABLE 5-3

RISK REDUCTION EVALUATION FOR CONTAINMENT SPRAY

RC#	INIT FREQ	TOTAL PERSON-REM	BASE PERSON-REM/YEAR	REVISED PERSON-REM/YEAR	COMMENTS
7.1	1.69E-6	4.80E+4	8.11E-2	8.11E-2	
6.2	1.38E-8	1.24E+6	1.71E-2	6.62E-4	IMPROVED
6.4	7.28E-9	1.66E+7	1.21E-1	3.49E-4	IMPROVED
5.1	3.63E-8	1.24E+7	4.50E-1	1.74E-3	IMPROVED
5.2	6.59E-8	4.72E+7	3.11E-0	3.16E-3	IMPROVED
4.1	2.12E-10	5.39E+5	1.14E-4	1.14E-4	
4.2	2.17E-10	2.14E+7	4.64E-3	4.64E-3	H2 BURN
3.1	4.82E-8	2.60E+7	1.25E-0	2.31E-3	IMPROVED
2.2	1.12E-9	9.01E+4	1.01E-4	1.01E-4	
2.4	2.44E-9	8.76E+6	2.14E-2	2.14E-2	CONT. OPEN
1.4	7.02E-9	5.22E+7	3.66E-1	3.66E-1	
1.7	3.00E-9	1.94E+7	5.82E-2	5.82E-2	
			5.48E-0	0.54E-0	(90% RISK REDUCTION)

TABLE 5-4

RISK REDUCTION EVALUATION FOR FILTERED VENT

RC#	INIT FREQ	TOTAL PERSON-REM	BASE PERSON-REM/YEAR	REVISED PERSON-REM/YEAR	COMMENTS
7.1	1.69E-6	4.80E+4	8.11E-2	8.11E-2	
6.2	1.38E-8	1.24E+6	1.71E-2	1.71E-2	
6.4	7.28E-9	1.66E+7	1.21E-1	1.21E-1	
5.1	3.63E-8	1.24E+7	4.50E-1	1.96E-2	IMPROVED
5.2	6.59E-8	4.72E+7	3.11E-0	3.55E-2	IMPROVED
4.1	2.12E-10	5.39E+5	1.14E-4	1.14E-4	
4.2	2.17E-10	2.14E+7	4.64E-3	4.64E-3	
3.1	4.82E-8	2.60E+7	1.25E-0	2.60E-2	IMPROVED
2.2	1.12E-9	9.01E+4	1.01E-4	1.01E-4	
2.4	2.44E-9	8.76E+6	2.14E-2	2.14E-2	
1.4	7.02E-9	5.22E+7	3.66E-1	3.66E-1	
1.7	3.00E-9	1.94E+7	5.82E-2	5.82E-2	
			5.48E-0	0.75E-0	(86% RISK REDUCTION)

TABLE 5-5

RISK REDUCTION EVALUATION FOR DC BATTERIES & EFWS

RC#	INIT FREQ	TOTAL PERSON-REM	BASE PERSON-REM/YEAR	REVISED PERSON-REM/YEAR	COMMENTS
7.1	1.69E-6	4.80E+4	8.11E-2	0.0	IMPROVED
6.2	1.38E-8	1.24E+6	1.71E-2	1.71E-2	
6.4	7.28E-9	1.66E+7	1.21E-1	0.0	IMPROVED
5.1	3.63E-8	1.24E+7	4.50E-1	0.0	IMPROVED
5.2	6.59E-8	4.72E+7	3.11E-0	0.0	IMPROVED
4.1	2.12E-10	5.39E+5	1.14E-4	1.14E-4	
4.2	2.17E-10	2.14E+7	4.64E-3	0.0	IMPROVED
3.1	4.82E-8	2.60E+7	1.25E-0	1.25E-0	
2.2	1.12E-9	9.01E+4	1.01E-4	1.01E-4	
2.4	2.44E-9	8.76E+6	2.14E-2	0.0	IMPROVED
1.4	7.02E-9	5.22E+7	3.66E-1	3.66E-1	
1.7	3.00E-9	1.94E+7	5.82E-2	5.82E-2	
			5.48E-0	1.69E-0	(69% RISK REDUCTION)

TABLE 5-6

RISK REDUCTION EVALUATION FOR PRESSURIZER AUXILIARY SPRAY

RC#	INIT FREQ	TOTAL PERSON-REM	BASE PERSON-REM/YEAR	REVISED PERSON-REM/YEAR	COMMENTS
7.1	1.69E-6	4.80E+4	8.11E-2	8.11E-2	
6.2	1.38E-8	1.24E+6	1.71E-2	1.71E-2	
6.4	7.28E-9	1.66E+7	1.21E-1	1.21E-1	
5.1	3.63E-8	1.24E+7	4.50E-1	4.50E-1	
5.2	6.59E-8	4.72E+7	3.11E-0	3.11E-0	
4.1	2.12E-10	5.39E+5	1.14E-4	1.14E-4	
4.2	2.17E-10	2.14E+7	4.64E-3	4.64E-3	
3.1	4.82E-8	2.60E+7	1.25E-0	1.25E-0	
2.2	1.12E-9	9.01E+4	1.01E-4	1.01E-4	
2.4	2.44E-9	8.76E+6	2.14E-2	2.14E-2	
1.4	7.02E-9	5.22E+7	3.66E-1	0.0	IMPROVED
1.7	3.00E-9	1.94E+7	5.82E-2	5.82E-2	
			5.48E-0	5.11E-0	(6.7% RISK REDUCTION)

TABLE 5-7

RISK REDUCTION EVALUATION FOR ATWS VALVES

RC#	INIT FREQ	TOTAL PERSON-REM	BASE PERSON-REM/YEAR	REVISED PERSON-REM/YEAR	COMMENTS
7.1	1.69E-6	4.80E+4	8.11E-2	3.65E-2	55% IMPROVED
6.2	1.38E-8	1.24E+6	1.71E-2	1.71E-2	
6.4	7.28E-9	1.66E+7	1.21E-1	1.21E-1	
5.1	3.63E-8	1.24E+7	4.50E-1	4.50E-1	
5.2	6.59E-8	4.72E+7	3.11E-0	3.11E-0	
4.1	2.12E-10	5.39E+5	1.14E-4	1.14E-4	
4.2	2.17E-10	2.14E+7	4.64E-3	4.64E-3	
3.1	4.82E-8	2.60E+7	1.25E-0	1.25E-0	
2.2	1.12E-9	9.01E+4	1.01E-4	1.01E-4	
2.4	2.44E-9	8.76E+6	2.14E-2	2.14E-2	
1.4	7.02E-9	5.22E+7	3.66E-1	1.83E-1	50% IMPROVED
1.7	3.00E-9	1.94E+7	5.82E-2	5.82E-2	
			5.48E-0	5.25E-0	(4.2% RISK REDUCTION)

TABLE 5-8

RISK REDUCTION EVALUATION FOR CONCRETE COMPOSITION

RC#	INIT FREQ	TOTAL PERSON-REM	BASE PERSON-REM/YEAR	REVISED PERSON-REM/YEAR	COMMENTS
7.1	1.69E-6	4.80E+4	8.11E-2	8.11E-2	
6.2	1.38E-8	1.24E+6	1.71E-2	6.62E-4	IMPROVED
6.4	7.28E-9	1.66E+7	1.21E-1	3.49E-4	IMPROVED
5.1	3.63E-8	1.24E+7	4.50E-1	4.50E-1	
5.2	6.59E-8	4.72E+7	3.11E-0	3.11E-0	
4.1	2.12E-10	5.39E+5	1.14E-4	1.14E-4	
4.2	2.17E-10	2.14E+7	4.64E-3	4.64E-3	
3.1	4.82E-8	2.60E+7	1.25E-0	1.25E-0	
2.2	1.12E-9	9.01E+4	1.01E-4	1.01E-4	
2.4	2.44E-9	8.76E+6	2.14E-2	2.14E-2	
1.4	7.02E-9	5.22E+7	3.66E-1	3.66E-1	
1.7	3.00E-9	1.94E+7	5.82E-2	5.82E-2	
			5.48E-0	5.34E-0	(2.5% RISK REDUCTION)

TABLE 5-9

RISK REDUCTION EVALUATION FOR VESSEL EXTERIOR COOLING

RC#	INIT FREQ	TOTAL PERSON-REM	BASE PERSON-REM/YEAR	REVISED PERSON-REM/YEAR	COMMENTS
7.1	1.69E-6	4.80E+4	8.11E-2	8.11E-2	
6.2	1.38E-8	1.24E+6	1.71E-2	6.62E-4	IMPROVED
6.4	7.28E-9	1.66E+7	1.21E-1	3.49E-4	IMPROVED
5.1	3.63E-8	1.24E+7	4.50E-1	4.50E-1	
5.2	6.59E-8	4.72E+7	3.11E-0	3.11E-0	
4.1	2.12E-10	5.39E+5	1.14E-4	1.02E-5	IMPROVED
4.2	2.17E-10	2.14E+7	4.64E-3	1.04E-5	IMPROVED
3.1	4.82E-8	2.60E+7	1.25E-0	1.25E-0	
2.2	1.12E-9	9.01E+4	1.01E-4	1.01E-4	
2.4	2.44E-9	8.76E+6	2.14E-2	2.14E-2	
1.4	7.02E-9	5.22E+7	3.66E-1	3.66E-1	
1.7	3.00E-9	1.94E+7	5.82E-2	5.82E-2	
			5.48E-0	5.34E-0	(2.5% RISK REDUCTION)

TABLE 5-10

RISK REDUCTION EVALUATION FOR H2 IGNITORS

RC#	INIT FREQ	TOTAL PERSON-REM	BASE PERSON-REM/YEAR	REVISED PERSON-REM/YEAR	COMMENTS
7.1	1.69E-6	4.80E+4	8.11E-2	8.11E-2	
6.2	1.38E-8	1.24E+6	1.71E-2	1.71E-2	
6.4	7.28E-9	1.66E+7	1.21E-1	1.21E-1	
5.1	3.63E-8	1.24E+7	4.50E-1	4.50E-1	
5.2	6.59E-8	4.72E+7	3.11E-0	3.11E-0	
4.1	2.12E-10	5.39E+5	1.14E-4	1.14E-4	
4.2	2.17E-10	2.14E+7	4.64E-3	1.04E-5	IMPROVED
3.1	4.82E-8	2.60E+7	1.25E-0	1.25E-0	
2.2	1.12E-9	9.01E+4	1.01E-4	1.01E-4	
2.4	2.44E-9	8.76E+6	2.14E-2	2.14E-2	
1.4	7.02E-9	5.22E+7	3.66E-1	3.66E-1	
1.7	3.00E-9	1.94E+7	5.82E-2	5.82E-2	
			5.48E-0	5.43E-0	(0.1% RISK REDUCTION)

TABLE 5-11

RISK REDUCTION EVALUATION FOR HPSI

RC#	INIT FREQ	TOTAL PERSON-REM	BASE PERSON-REM/YEAR	REVISED PERSON-REM/YEAR	COMMENTS
7.1	1.69E-6	4.80E+4	8.11E-2	8.11E-2	
6.2	1.38E-8	1.24E+6	1.71E-2	1.71E-2	
6.4	7.28E-9	1.66E+7	1.21E-1	1.21E-1	
5.1	3.63E-8	1.24E+7	4.50E-1	4.50E-1	
5.2	6.59E-8	4.72E+7	3.11E-0	3.11E-0	
4.1	2.12E-10	5.39E+5	1.14E-4	0.0	IMPROVED
4.2	2.17E-10	2.14E+7	4.64E-3	4.64E-3	
3.1	4.82E-8	2.60E+7	1.25E-0	1.25E-0	
2.2	1.12E-9	9.01E+4	1.01E-4	0.0	IMPROVED
2.4	2.44E-9	8.76E+6	2.14E-2	2.14E-2	
1.4	7.02E-9	5.22E+7	3.66E-1	3.66E-1	
1.7	3.00E-9	1.94E+7	5.82E-2	5.82E-2	
			5.48E-0	5.48E-0	(0.004 RISK REDUCTION)

TABLE 5-12

RISK REDUCTION EVALUATION FOR RCS DEPRESSURIZATION

RC#	INIT FREQ	TOTAL PERSON-REM	BASE PERSON-REM/YEAR	REVISED PERSON-REM/YEAR	COMMENTS
7.1	1.69E-6	4.80E+4	8.11E-2	8.11E-2	
6.2	1.38E-8	1.24E+6	1.71E-2	1.71E-2	
6.4	7.28E-9	1.66E+7	1.21E-1	1.21E-1	
5.1	3.63E-8	1.24E+7	4.50E-1	4.50E-1	
5.2	6.59E-8	4.72E+7	3.11E-0	3.11E-0	
4.1	2.12E-10	5.39E+5	1.14E-4	1.14E-4	
4.2	2.17E-10	2.14E+7	4.64E-3	4.64E-3	
3.1	4.82E-8	2.60E+7	1.25E-0	1.25E-0	
2.2	1.12E-9	9.01E+4	1.01E-4	0.0	IMPROVED
2.4	2.44E-9	8.76E+6	2.14E-2	2.14E-2	
1.4	7.02E-9	5.22E+7	3.66E-1	3.66E-1	
1.7	3.00E-9	1.94E+7	5.82E-2	5.82E-2	
			5.48E-0	5.48E-0	(0.002% RISK REDUCTION)

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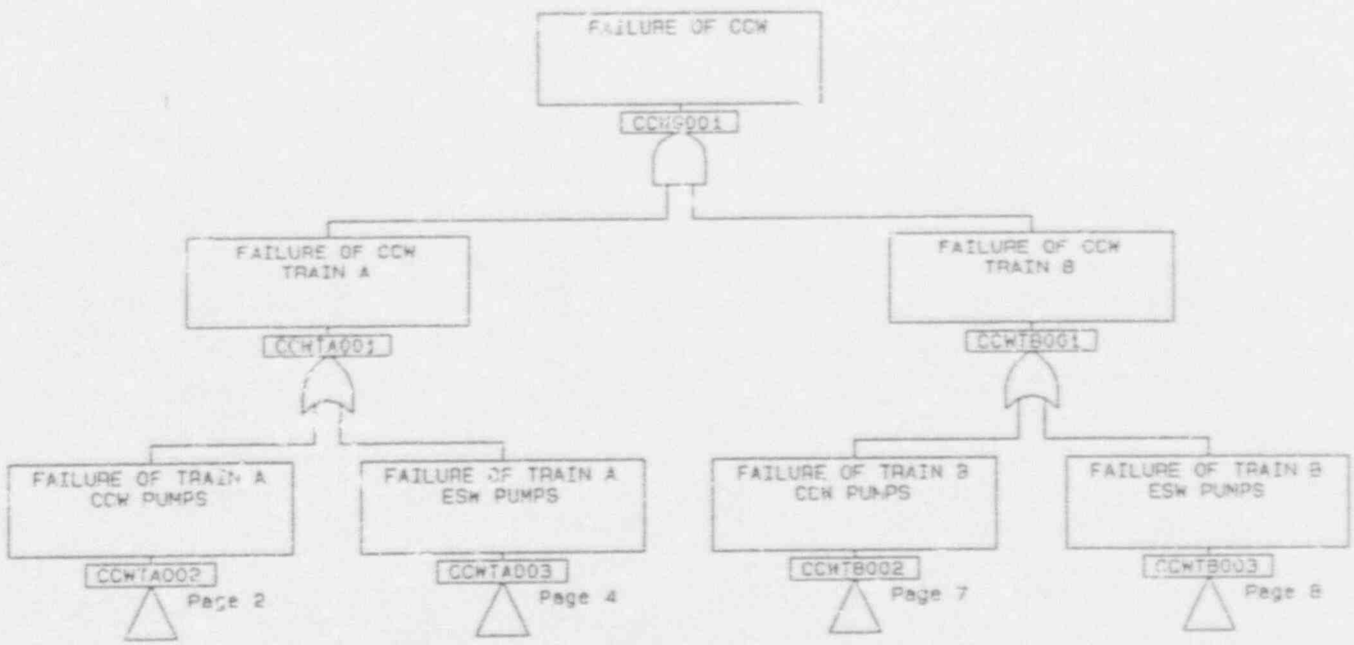
APPENDIX A

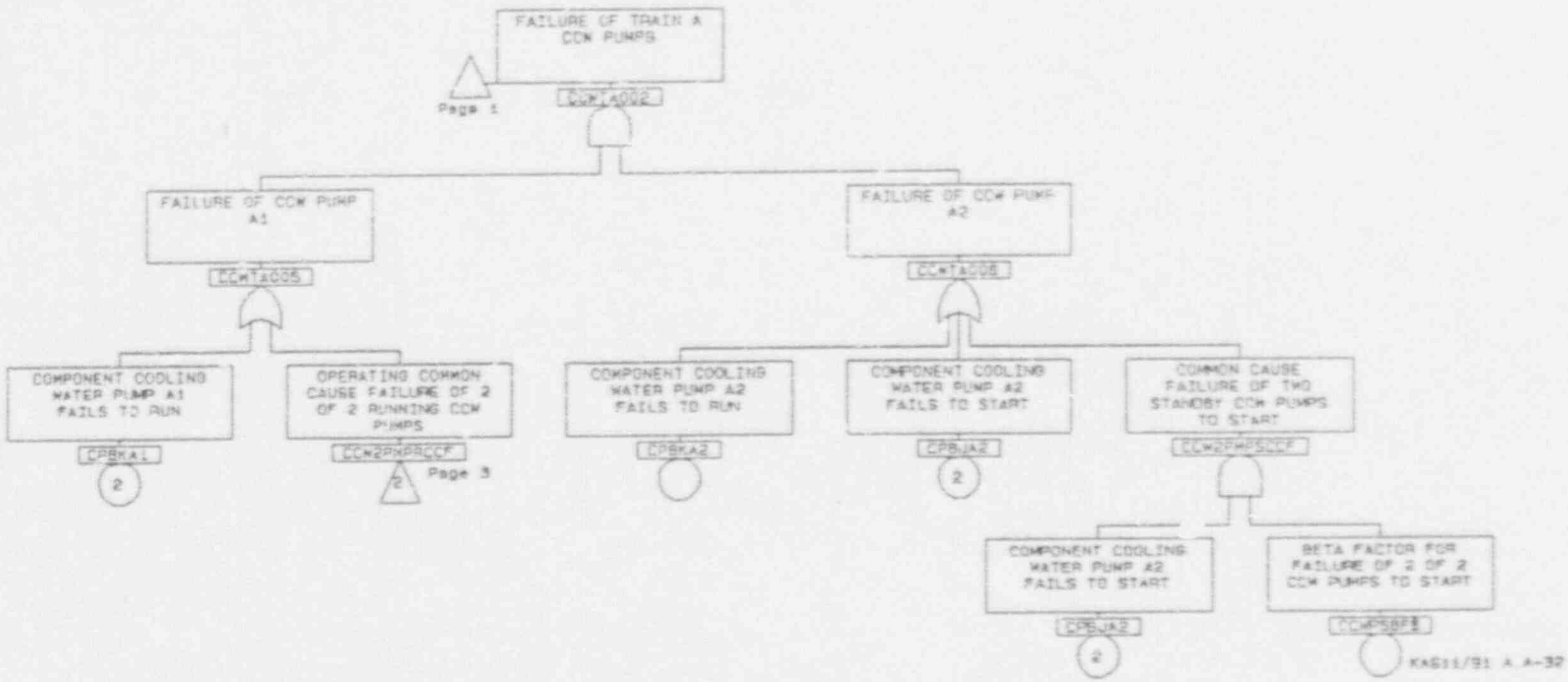
FAULT TREE ANALYSIS FOR TOTAL LOSS ⌐F

COMPONENT COOLING WATER

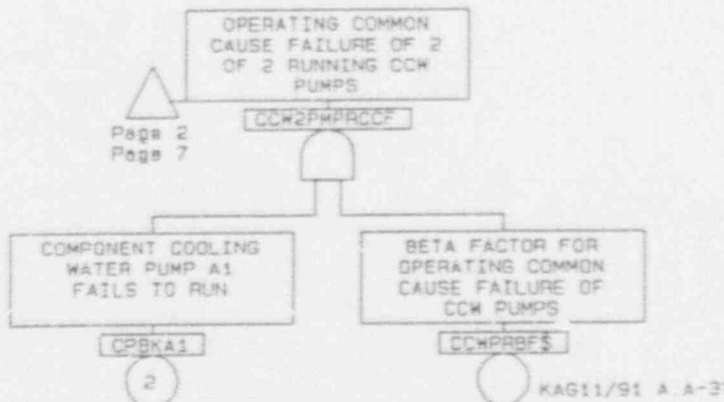
FAULT TREE ANALYSIS FOR TOTAL LOSS OF COMPONENT COOLING WATER

There are two trains of Component Cooling Water (CCW). Each train has one pump operating and one pump in standby. In addition, each CCW trains require service water. Each CCW train is supported by two service water pumps, one operating and one on standby. The following fault tree models total loss of CCW including common cause failures. This tree was evaluated using the CAFTA code and the resulting failure frequency was 2.25E-5/y.


FAILURE OF CCW
CCWS001
FAILURE OF CCW TRAIN A
CCWTA001
FAILURE OF CCW TRAIN B
CCWTB001
FAILURE OF TRAIN A CCW PUMPS
CCWTA002
Page 2
FAILURE OF TRAIN A ESW PUMPS
CCWTA003
Page 4
FAILURE OF TRAIN B CCW PUMPS
CCWTB002
Page 7
FAILURE OF TRAIN B ESW PUMPS
CCWTB003
Page 8



FAILURE OF CCW	.\TREES\CCWSAMDA.CAF	4-15-92	Page 2



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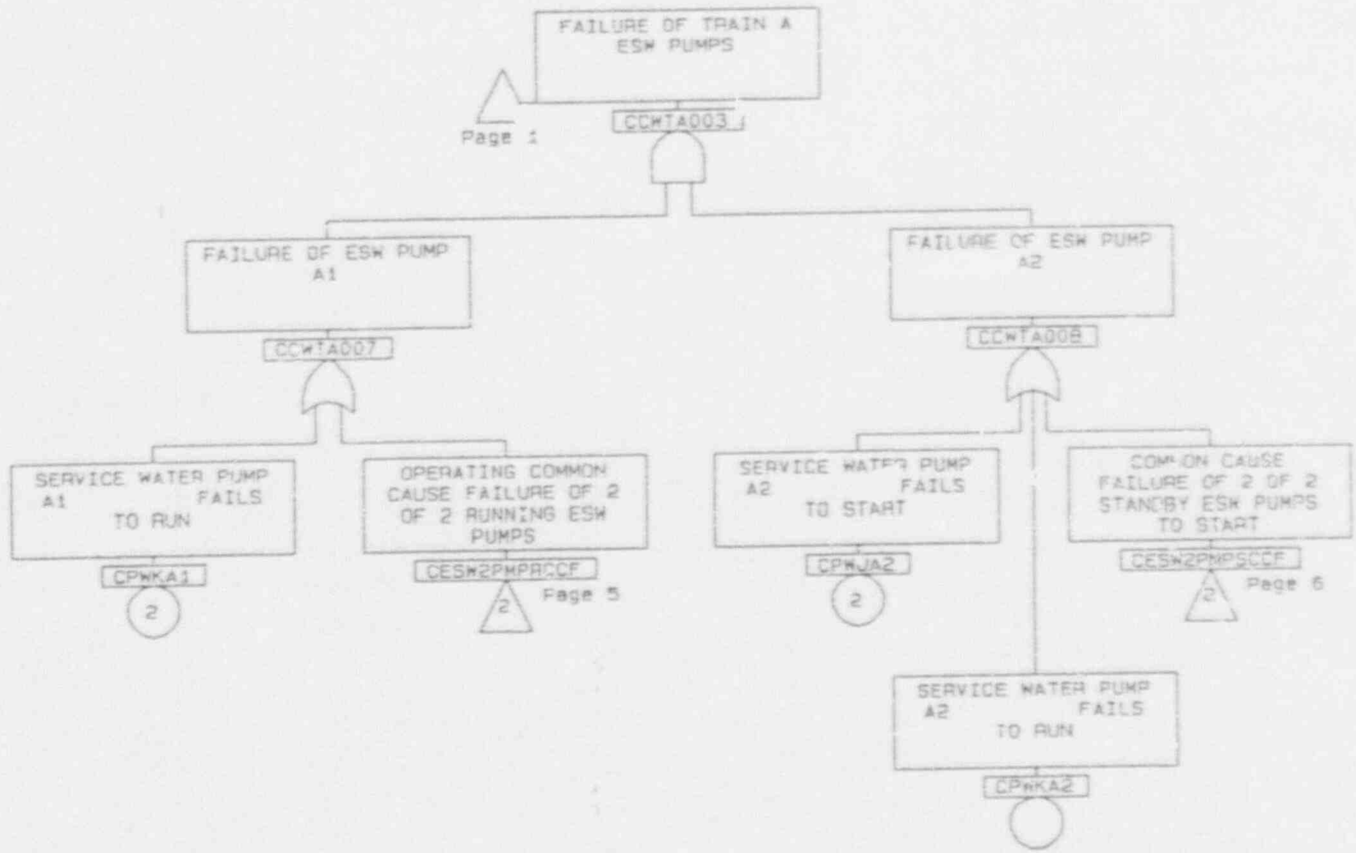
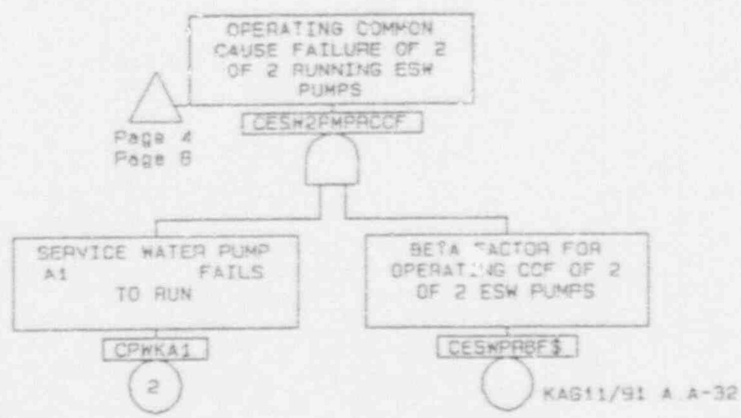
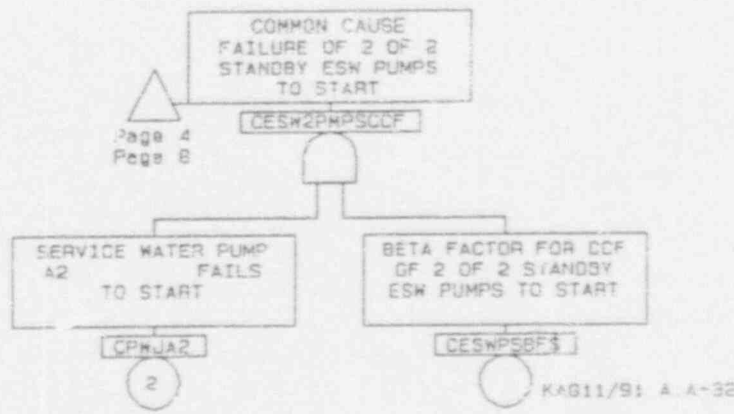
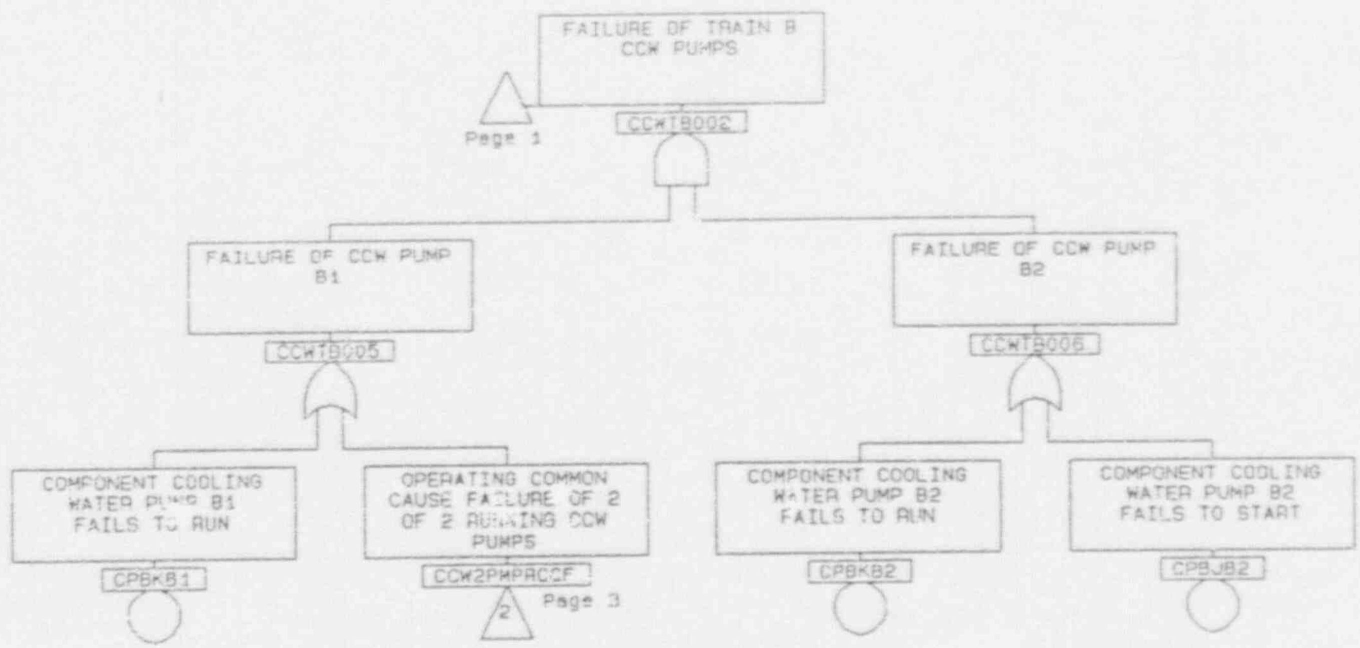


FAILURE OF TRAIN A ESW PUMPS
Page 1
CCWTA003
FAILURE OF ESW PUMP A1
CCWTA007
FAILURE OF ESW PUMP A2
CCWTA008
SERVICE WATER PUMP A1 FAILS TO RUN
CPWKA1
2
OPERATING COMMON CAUSE FAILURE OF 2 OF 2 RUNNING ESW PUMPS
CESW2PMPRCCF
2
Page 5
SERVICE WATER PUMP A2 FAILS TO START
CPWJA2
2
COMMON CAUSE FAILURE OF 2 OF 2 STANDBY ESW PUMPS TO START
CESW2PMPSCCF
2
Page 6
SERVICE WATER PUMP A2 FAILS TO RUN
CPWKA2

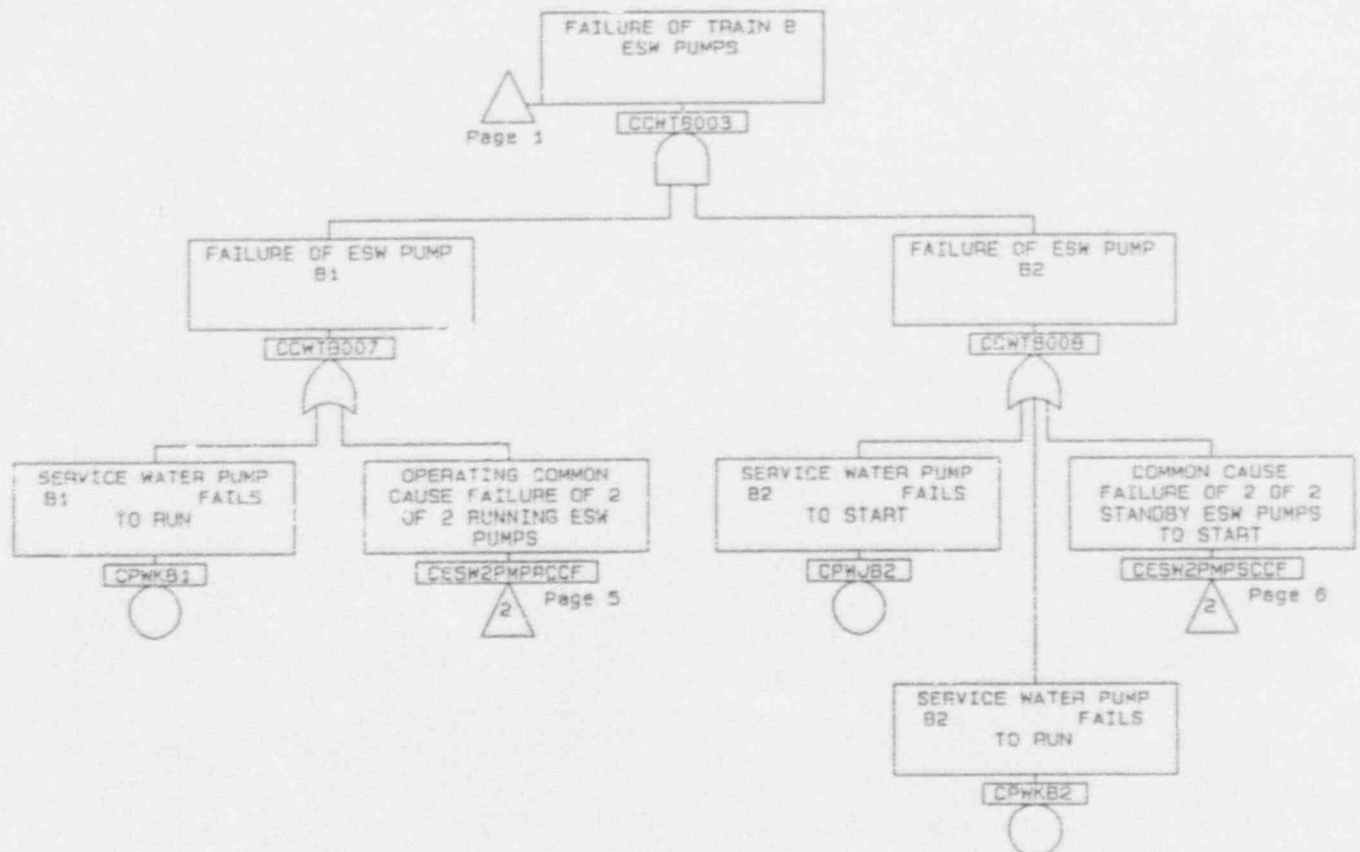


FAILURE OF CCW	.\TREES\CCWSAMDA.CAF	4-15-92	Page 5


COMMON CAUSE FAILURE OF 2 OF 2 STANDBY ESW PUMPS TO START
CESW2PMPSCCF
Page 4
Page 8
SERVICE WATER PUMP A2 FAILS TO START
CPWJA2
2
BETA FACTOR FOR CCF OF 2 OF 2 STANDBY ESW PUMPS TO START
CESWPSBFS
KAG11/91 A.A-32



FAILURE OF CCW	.\TREES\CCWSAMDA.CAF	4-15-92	Page 7



FAILURE OF CCW | .\TREES\CCWSAMDA.CAF | 4-15-92 | Page 8