ATTACHMENT A

DESIGN ALTERNATIVES

FOR THE

SYSTEM 80+ NUCLEAR POWER PLANT

(REV. 2)

SEPTEMBER 23, 1993

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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
CDF	Core Damage Frequency
CET	Containment Event Tree
CHRS	Containment Hydrogen Recombiner System
CSS	Containment Spray System
DA	Design Alternative
DC	Direct Current
DCH	Direct Containment Heating
DG	Diesel Generator
DHR	Decay Heat Removal
EFWS	Emergency Feedwater System
EPRI	Electric Power Research Institute
H2	Hydrogen
HMS	Hydrogen Mitigation System
HPSI	High Pressure Safety Injection
Hrs	Hours
HVAC	Heating, Ventilation, and Air Conditioning
IRWST	In-Containment Refueling Water Storage Tank
KAC	Key Assumptions and Groundrules
LOCA	Loss of Coolant Accident
LOFW	Loss Of FeedWater
M	Millions
MACCS	MELCOR Accident Consequence Code System
MORV	Motor Operated Relief Valve
MSSV	Main Steam Safety Valve
NRC	Nuclear Regulatory Commission
PARS	Passive Autocatalytic Recombiners
PDS	Plant Damage State
PPS	Plant Protection System
PRA	Probabilistic Risk Assessment
RC	Release Class
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
SAMDA	Severe Accident Mitigation Design Alternative
SCS	Shutdown Cooling System
sec	Second
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SIT	Safety Injection Tanks
URD	Utility Requirements Document
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." Section 15 to Chapter 19 provides the base PRA of the System 80+ plant.

The NRC also requested the ALWR participants to evaluate design alternatives[1] 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 Design Alternatives that were considered in the System 80+ design. Improvements have been reviewed against conservatively high estimates of risk reductions based on the PRA and conservatively low estimates of costs, to determine whether potential modifications are cost beneficial.

2.0 SUMMARY AND CONCLUSION

The System 80+ design is an evolutionary Advanced Light Water Reactor (ALWR) 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 ensured that no single accident sequence dominated the plant risk and the lessons learned from previous PRAs were addressed.

Sixty-three design alternatives were considered and the expected risk reduction from twenty-seven of those alternatives were quantified. These were selected based on the Design Alternatives (DAs) evaluated for the Limerick plant[2] , Comanche Peak[15], NUREG/CR-4920[16], GSI-163[17], and the results from the System 80+ PRA performed by C-E. The DAs were selected to address the sequences that either have the largest risk to the public or sequences that have high CDF. The analysis used a bounding technique. It was assumed that each DA worked perfectly and completely eliminated the accident sequences that the DA was to address. This approach maximizes the benefits associated with each DA. 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. Consistent with the standard used by NRC to evaluate offsite impacts, health and economic effect costs were evaluated based on a value of $1000 per offsite person-rem averted. Using this $1,000 per person-rem, and a levelized capital cost rate of 16.6%, 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 annual risk reduction to the general population using $1000 per 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 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 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 conservatively estimated the benefits of the DAs by assuming that they would work perfectly to eliminate the type of accident they are designed to address and would require no maintenance or testing. Because of the small initial risk associated with the System 80+ design, none of the DAs are cost beneficial.

TABLE 2-1
(Sheet 1 of 2)

SUMMARY OF THE RISK REDUCTIONS OF THE DESIGN ALTERNATIVES

	DESIGN ALTERNATIVE	ANNUAL RISK REDUCTION $/Y	CAPITAL BENEFIT*	CAPITAL COST	NET CAPITAL BENEFIT
5.1	ALT. CONTAINMENT SPRAY	$7.27	$44	$1,500,000	($1,499,956)
5.2	FILTERED VENT (CONTAINMENT)	$0.53	$3	$10,000,000	($9,999,997)
5.3	ALT. DC BATTERY AND EFWS	$1.87	$11	$2,000,000	($1,999,989)
5.5	ALT. PRESSURIZER AUX SPRAY	$90.44	$545	$5,000,000	($4,999,455)
5.6	ALT. ATWS RELIEF VALVES	$1.02	$6	$1,000,000	($999,994)
5.7	ALT. CONCRETE COMPOSITION	$4.87	$29	$5,000,000	($4,999,971)
5.8	RV EXTERIOR COOLING	$32.64	$197	$2,500,000	($2,499,803)
5.9	ALT. H2 IGNITORS	$0.75	$5	$1,000,000	($999,995)
5.10	ALT. HPSI	$83.38	$502	$2,200,000	($2,299,498)
5.11	ALT. RCS DEPRESSURIZATION	$15.14	$91	$500,000	($499,909)
5.12	100% SG INSPECTION	$100.38	$605	$1,500,000	($1,499,395)
5.13	MSSV AND ADV SCRUBBING	$97.30	$586	$9,500,000	($9,499,414)
5.14	THIRD DIESEL GENERATOR	$0.45	$3	$25,000,000	($9,999,997)
5.15	ATWS INJECTION SYSTEM	$1.02	$6	$300,000	($299,994)
5.16	DIVERSE PPS SYSTEM	$1.02	$6	$3,000,000	($2,999,994)
5.17	ALT. CONTAINMENT MONITORING SYSTEM	$1.67	$10	$1,000,000	($999,991)

TABLE 2-1
(Sheet 2 of 2)

SUMMARY OF THE RISK REDUCTIONS OF THE DESIGN ALTERNATIVES

DESIGN ALTERNATIVE	ANNUAL RISK REDUCTION $/Y	CAPITAL BENEFIT*	CAPITAL COST	NET CAPITAL BENEFIT
5.18 CAVITY COOLING	$32.64	$197	$50,000	($49,818)
5.19 12-HOUR BATTERIES	$0.71	$4	$300,000	($299,996)
5.20 TORNADO-PROTECTION FOR COMBUSTION TURBINE	$1.60	$10	$3,000,000	($2,999,991)
5.21 DIESEL SI PUMPS (2)	$83.79	$505	$2,000,000	($1,999,532)
5.23 EXTENDED RWST SOURCE	$32.80	$198	$1,000,000	($999,802)
5.28 SECONDARY SIDE GUARD PIPES	$0.73	$4	$820,000	($819,996)
5.29 PASSIVE AUTOCATALYTIC RECOMBINERS (PARS)	$0.75	$5	$760,000	($759,995)
5.31 FUEL CELLS	$1.87	$11	$2,000,000	($1,999,989)
5.32 HOOKUP FOR PORTABLE GENERATOR	$1.87	$11	$10,000	($9,989)
5.33 WATER COOLED RUBBLE BED	$4.87	$29	$18,800,000	($18,799,971)
5.34 REFRACTORY LINED CRUCIBLE	$4.87	$29	$108,000,000	($107,999,971)

* THE CAPITAL BENEFIT IS THE PRICE OF A PIECE OF EQUIPMENT THAT HAS A LEVELIZED (ANNUAL) COST EQUAL TO THE ANNUAL BENEFIT IN RISK REDUCTION AND ASSUMES NO MAINTENANCE OR TESTING OF ADDITIONAL EQUIPMENT

3.0 METHODOLOGY

The Design Alternative (DA) evaluation followed the format and procedure used by the NRC in evaluating DAs for Limerick[2]. The DAs 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 Section 15 of Chapter 19 of this CESSAR-DC, Amendment R.

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. The risk reduction was converted to an annual benefit ($/y) using $1000 per person/rem. This was then converted to an equivalent capital cost which could be compared to the estimated price for the DA.

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 DA 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 DA 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 DA 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 based on the sequences that were binned and the risk reduction was calculated. For example, an Alternative pressurizer auxiliary spray (DA5) 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 core damage for the Plant Damage States (PDS) with failure to

aggressively cooldown 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 S80SOR, a modified version of the ZISOR Code[3], the consequence in terms of dose to the general population is calculated for the ALWR site. The ALWR site was described in the May 1989 version of the KAG and was to represent 80% of the potential sites. The site was an existing site in South Carolina with the population increased to represent most potential sites[4]. For DAs that prevent containment failure, the releases were assumed to be reduced to zero and the risk was reevaluated.

3.2 COST ESTIMATES

In order to evaluate the effectiveness of the DAs, the benefits were compared to the costs of the Alternatives. Conservatively low cost estimates were made for each potential 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 for the DAs are in 1993 U.S. dollars.

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.

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 DA were defined as the risk reduction to the general public. Offsite factors evaluated were limited to whole body dose to the general public. Consistent with the standard used by the NRC to evaluate radiological impacts, health and economic 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 and represents both health effects and offsite economic losses[12].

The annual benefit in $/y is converted to a single capital benefit using a levelized capital cost rate. Using the method described in Ref. 5, a component and plant economic life of 60 years, and other assumptions given in Table 3-1, a levelized capital cost rate of 16.6% was used. The DA results are not very sensitive to the detailed economic assumptions used in calculating a levelized capital cost rate. If the calculations are performed using zero inflation and a reduced cost of capital (capital costs reflect expected inflation plus an expected real return), similar results are obtained.

The offsite costs for other items, such as relocation of local residents, elimination of land use and decontamination of contaminated land are considered as part of the $1000/person-rem. Economic losses, replacement power costs and direct accident costs incurred by the plant owner also are not considered in this evaluation.

TABLE 3-1

ECONOMIC ASSUMPTIONS FOR LEVELIZED CAPITAL COST RATE

ASSUMPTIONS	VALUE
BOND (DEBT) INTEREST RATE, %	10.48
DEBT 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
PLANT AND COMPONENT ECONOMIC LIFE, YRS.	60.0
RESULTING LEVELIZED CAPITAL COST RATE, %	16.6

4.0 PRA RELEASE CLASSES

In assessing the risk reduction of each Design Alternative (DA), the potential for each DA 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 Section 12 of Chapter 19 of the CESSAR-DC, Amendment R, 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.

S80SOR 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.

Table 4-1 presents a brief description for each release class with a frequency greater than or equal to 1.0E-10. This table is used to identify the effect of mitigation equipment (more details of each RC is given in Section 12.3 of Chapter 19, Amendment R). Table 4-2 gives the mapping of each PDS into each release class.

Also given in this table are the mapping of the CDF sequences into the PDSs. In addition, the description of each sequence and the sequence CDF is also presented. This table is used to reduce each RC frequency (column 2 of Table 4-2) for preventative DAs.

The sequence CDF (last column of Table 4-2) was used to calculate the risk reduction associated with DAs that prevent core damage. It was assumed that any prevention DA would completely eliminate the sequence that the DA would address. For example, a Safety Injection DA would reduce the RC1.1E by 55%. SIS failure appears in five of the sequences with a total sequence frequency of 7.15E-7. The sum of all the sequences contributing to RC1.1E is 12.89E-7 and therefore the DA is assumed to reduce this RC by 55% (7.15E-7 / 12.89E-7). Each release class is evaluated in this manner for each prevention DA.

Table 4-3 gives the ranking of the release classes in terms of risk to the general population (mr/y). It also gives the base frequency, and population dose for each RC that is used in the risk reduction analysis. The first three sequences are associated with steam generator tube ruptures. This table (used with the previous two tables) was used in selecting the DAs because it highlights the importance of the failure modes. Table 4.4 gives the ranking of the Level I sequences in terms of CDF. This table is useful for selecting DAs for preventing core damage.

Each release class was evaluated for total person-rem exposure using MACCS. Table 4-3 gave the initiating frequency, and total person-rem dose for the twenty three release classes with initiating frequencies greater than 1.0E-10. The lifetime doses were calculated for the people within 50 miles of the site and assumes the evacuation strategy used in NUREG 1150. 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 0.135 person-rem/y. These results are for the ALWR site which is representative of most of the current U.S. sites[4].

Table 4-1 summarizes the accident characteristics for each release class. These are the dominant sequences of the binned accidents. For each DA, the release class was reviewed assuming that the DA worked perfectly (failure rate = 0.0). This means that each DA had perfect support systems, power supplies and heat sinks. In addition, for each DA, no other failure modes were considered when the DA 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 DA is cost beneficial in this analysis then a

more detailed analysis addressing the actual failure rate of the Design Alternative can be undertaken.

Table 4-1
SUMMARY DESCRIPTION OF SYSTEM 80+ RELEASE CLASSES

Release Class	Release Class Definition
RC1.1E	Early core melt, Intact containment, Annulus filtering, Core damage less than 8 hrs
RC1.1M	Mid core melt, Intact containment, Annulus filtering, Core damage 8 to 24 hrs
RC2.1E	Early core melt, Late Containment failure, in-vessel scrubbing, no vaporization or revaporization releases, Core damage less than 8 hrs, H2 burn
RC2.2E	Early core melt, In-vessel scrubbing, no vaporization releases, revaporization releases, releases scrubbed, Late Containment failure, H2 burn
RC2.4E	Early core melt, Late Containment failure, in-vessel scrubbing, vaporization releases, no revaporization releases, releases scrubbed, Basemat melt through
RC2.5E	Early core melt, Late Containment failure, in-vessel scrubbing, vaporization releases, no revaporization releases, releases not scrubbed, Basemat melt thru
RC2.6E	Early core melt, Late Containment failure, in-vessel scrubbing, vaporization releases and revaporization releases, releases scrubbed, Basemat melt through
RC2.7E	Early core melt, Late Containment failure, in-vessel scrubbing, vaporization releases and revaporization releases, revaporization releases scrubbed, vaporization releases not scrubbed, Basemat melt
RC2.2M	Mid core melt, Late Containment failure, in-vessel scrubbing, no vaporization releases, revaporization releases, releases scrubbed, CSS fls, Steam failure
RC2.5M	Mid core melt, Late Containment failure, in-vessel scrubbing, vaporization releases, no revaporization releases, releases not scrubbed, Basemat melt thru
RC2.6M	Mid core melt, Late Containment failure, in-vessel scrubbing, vaporization releases and revaporization releases, releases scrubbed, Basemat melt through
RC2.7M	Mid core melt, Late Containment failure, in-vessel scrubbing, vaporization releases and revaporization releases, revaporization releases scrubbed, vaporization releases not scrubbed, Basemat melt

Table 4-1 SUMMARY DESCRIPTION OF SYSTEM 80+ RELEASE CLASSES	
Release Class	Release Class Definition
RC3.1E	Early core melt, early containment failure, in-vessel scrubbing, no vaporization or revaporization releases, Steam explosion
RC3.2E	Early core melt, early contnmnt failure, in-vessel scrubbing, no vaporization release, revaporization releases, releases scrubbed, Steam explosion
RC3.4E	Early core melt, early containment failure, in-vessel scrubbing, vaporization release, no revap. releases, releases scrubbed, steam explosion
RC3.6E	Early core melt, early containment failure, in-vessel scrubbing, vaporization release, revap. releases, releases scrubbed, steam explosion
RC3.2M	Mid core melt, early containment failure, in-vessel scrubbing, no vaporization release, revap. releases, releases scrubbed, CSS failed, steam explosion
RC3.6M	Mid core melt, early containment failure, in-vessel scrubbing, vaporization release, revap. releases, releases scrubbed, CSS failed, Steam explosion
RC4.4E	Early core melt(SGTR), in-vessel scrubbing, vaporization releases, no revaporization releases, releases scrubbed, isolation failure
RC4.8E	Early core melt, isolation failure, in-vessel scrubbing, vaporization releases, revaporization releases, vaporization releases scrubbed, revaporization releases not scrubbed
RC4.12E	Early core melt(SGTR), isolation failure, in-vessel scrubbing, vaporization releases, revaporization releases, releases scrubbed
RC4.18L	Late core melt(SGTR), isolation failure, in-vessel scrubbing, vaporization releases, revaporization releases, releases not scrubbed
RC5.1E	Early core melt, containment bypass, vaporization releases, releases scrubbed/attenuated in auxiliary building

TABLE 4-2
(Sheet 1 of 4)

MAPPING SEQUENCES INTO RELEASE CLASSES

RC	RC FREQ	PDS	SEQUENCE	DESCRIPTION	SEQ. FREQ*.
RC1.1E	1.36E-6	PDS235	LSSB-9A	(LSSB)(Safety Injection OK)(Failure to Deliver Feedwater)(Safety Depressurization for Bleed Fails)	2.2E-09
			LOFW-9A	(LOFW)(Emergency Feedwater Fails)(Safety Depressurization for Bleed Fails)	4.6E-07
			TOTH-9A	(Other Transients)(Feedwater Fails)(Safety Depressurization Fails)	2.7E-09
			LOOP-9A	(LOOP)(Failure to Deliver Emergency Feedwater)(Safety Depressurization for Bleed Fails)	3.8E-09
		PDS184	SGTR-16A	(SGTR)(Safety Injection Fails)(Aggressive Cooldown OK)(RHR Injection Fails)	1.5E-08
			SGTR-17A	(SGTR)(Injection Fails)(Aggressive Secondary Cooldown Fails)	2.7E-07
		PDS3	LL-3A	(LLOCA)(SITs Inject OK)(Safety Injection Fails)	1.1E-07
			LL-4A	(LLOCA)(SITs Fail to Inject)	4.7E-09
			VR-A	Vessel Rupture	1.0E-07
		PDS85	ML2-3A	(Medium LOCA 2)(Safety Injection Fails)	1.6E-07
		PDS201	SL-11A	(SLOCA)(Safety Injection Fails)(Aggressive Cooldown Fails)	1.6E-07
RC1.1M	3.81E-7	PDS148	LOFW-4E	(LOFW)(Emergency Feedwater OK)(Long-term Decay Heat Removal Fails)(Bleed OK)(Safety Injection for Feed Fails)	3.6E-08
			TOTH-4E	(Other Transients)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(Safety Injection for Feed Fails)	6.9E-08
			TRND-4E	Tornado, PSV reseats, EFSW OK, LTDHR Fails, Bleed OK, Feed Fails	2.5E-07
		PDS136	LOFW-4A	(LOFW)(Emergency Feedwater OK)(Long-term Decay Heat Removal Fails)(Bleed OK)(Safety Injection for Feed Fails)	3.6E-08
			TOTH-4A	(Other Transients)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(Safety Injection for Feed Fails)	6.9E-08
			TRND-4A	Tornado, PSV reseats, EFW OK, LTDHR Fails, Bleed OK, Feed Fails	2.5E-07
RC2.1E	3.46E-9	PDS3	LL-3A	(LLOCA)(SITs Inject OK)(Safety Injection Fails)	1.1E-07
			LL-4A	(LLOCA)(SITs Fail to Inject)	4.7E-09
			VR-A	Vessel Rupture	1.0E-07
		PDS235	LSSB-9A	(LSSB)(Safety Injection OK)(Failure to Deliver Feedwater)(Safety Depressurization for Bleed Fails)	2.2E-09
			LOFW-9A	(LOFW)(Emergency Feedwater Fails)(Safety Depressurization for Bleed Fails)	4.6E-07
			TOTH-9A	(Other Transients)(Feedwater Fails)(Safety Depressurization Fails)	2.7E-09
			LOOP-9A	(LOOP)(Failure to Deliver Emergency Feedwater)(Safety Depressurization for Bleed Fails)	3.8E-09
RC2.2E	2.04E-9	PDS201	SL-11A	(SLOCA)(Safety Injection Fails)(Aggressive Cooldown Fails)	1.6E-07
		PDS235	LSSB-9A	(LSSB)(Safety Injection OK)(Failure to Deliver Feedwater)(Safety Depressurization for Bleed Fails)	2.2E-09
			LOFW-9A	(LOFW)(Emergency Feedwater Fails)(Safety Depressurization for Bleed Fails)	4.6E-07
			TOTH-9A	(Other Transients)(Feedwater Fails)(Safety Depressurization Fails)	2.7E-09
			LOOP-9A	(LOOP)(Failure to Deliver Emergency Feedwater)(Safety Depressurization for Bleed Fails)	3.8E-09
RC2.4E	3.64E-8	PDS233	LOFW-9B	(LOFW)(Emergency Feedwater Fails)(Safety Depressurization for Bleed Fails)	4.6E-07
		PDS83	ML2-3B	(Medium LOCA 2)(Safety Injection Fails)	1.6E-07
		PDS18	ML1-3B	(Medium LOCA 1)(Safety Injection Fails)	1.4E-07
		PDS3	LL-3A	(LLOCA)(SITs Inject OK)(Safety Injection Fails)	1.1E-07
			LL-4A	(LLOCA)(SITs Fail to Inject)	4.7E-09
			VR-A	Vessel Rupture	1.0E-07
		PDS1	LL-3B	(LLOCA)(SITs Inject OK)(Safety Injection Fails)	1.1E-07
RC2.5E	2.84E-8	PDS241	LOFW-9F	(LOFW)(Emergency Feedwater Fails)(Safety Depressurization for Bleed Fails)	4.6E-07

TABLE 4-2
(Sheet 2 of 4)

MAPPING SEQUENCES INTO RELEASE CLASSES

RC	RC FREQ	PDS	SEQUENCE	DESCRIPTION	SEQ. FREQ*.
RC2.6E	3.45E-8	PDS181	SGTR-17B	(SGTR)(Injection Fails)(Aggressive Secondary Cooldown Fails)	2.7E-07
		PDS199	SL-11B	(SLOCA)(Safety Injection Fails)(Aggressive Cooldown Fails)	1.6E-07
		PDS233	LOFW-9B	(LOFW)(Emergency Feedwater Fails)(Safety Depressurization for Bleed Fails)	4.6E-07
RC2.7E	1.62E-8	PDS241	LOFW-9F	(LOFW)(Emergency Feedwater Fails)(Safety Depressurization for Bleed Fails)	4.6E-07
RC2.2M	4.05E-9	PDS148	LOFW-4E	(LOFW)(Emergency Feedwater OK)(Long-term Decay Heat Removal Fails)(Bleed OK)(Safety Injection for Feed Fails)	3.6E-08
			TOTH-4E	(Other Transients)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(Safety Injection for Feed Fails)	6.9E-08
			TRND-4E	Tornado, PSV reseats, EFW OK, LTDHR Fails, Bleed OK, Feed Fails	2.5E-07
		PDS136	LOFW-4A	(LOFW)(Emergency Feedwater OK)(Long-term Decay Heat Removal Fails)(Bleed OK)(Safety Injection for Feed Fails)	3.6E-08
			TOTH-4A	(Other Transients)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(Safety Injection for Feed Fails)	6.9E-08
			TRND-4A	Tornado, PSV reseats, EFW OK, LTDHR Fails, Bleed OK, Feed Fails	2.5E-07
RC2.5M	3.95E-9	PDS242	SBOBD-F	Station Blackout with Battery Depletion	2.1E-08
			TRND-SBF	Tornado, Station Blackout with Battery Depletion	1.69-08
RC2.6M	9.08E-9	PDS134	LOFW-4B	(LOFW)(Emergency Feedwater OK)(Long-term Decay Heat Removal Fails)(Bleed OK)(Safety Injection for Feed Fails)	3.6E-08
			TOTH-4B	(Other Transients)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(Safety Injection for Feed Fails)	6.9E-08
		PDS148	LOFW-4E	(LOFW)(Emergency Feedwater OK)(Long-term Decay Heat Removal Fails)(Bleed OK)(Safety Injection for Feed Fails)	3.6E-08
			TOTH-4E	(Other Transients)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(Safety Injection for Feed Fails)	6.9E-08
			TRND-4E	Tornado, PSV reseats, EFW OK, LTDHR Fails, Bleed OK, Feed Fails	2.5E-07
RC2.7M	1.22E-8	PDS145	TRND-4F	Tornado, PSV reseats, EFW OK, LTDHR Fails, Bleed OK, Feed Fails	2.5E-07
		PDS242	SBOBD-F	Station Blackout with Battery Depletion	2.1E-08
			TRND-SBF	Tornado, Station Blackout with Battery Depletion	1.69-08
RC3.1E	6.58E-9	PDS235	LSSB-9A	(LSSB)(Safety Injection OK)(Failure to Deliver Feedwater)(Safety Depressurization for Bleed Fails)	2.2E-09
			LOFW-9A	(LOFW)(Emergency Feedwater Fails)(Safety Depressurization for Bleed Fails)	4.6E-07
			TOTH-9A	(Other Transients)(Feedwater Fails)(Safety Depressurization Fails)	2.7E-09
			LOOP-9A	(LOOP)(Failure to Deliver Emergency Feedwater)(Safety Depressurization for Bleed Fails)	3.8E-09
		PDS85	ML2-3A	(Medium LOCA 2)(Safety Injection Fails)	1.6E-07
		PDS3	LL-3A	(LLOCA)(SITs Inject OK)(Safety Injection Fails)	1.1E-07
			LL-4A	(LLOCA)(SITs Fail to Inject)	4.7E-09
			VR-A	Vessel Rupture	1.0E-07
RC3.2E	3.08E-9	PDS184	SGTR-16A	(SGTR)(Safety Injection Fails)(Aggressive Cooldown OK)(RHR Injection Fails)	1.5E-08
			SGTR-17A	(SGTR)(Injection Fails)(Aggressive Secondary Cooldown Fails)	2.7E-07
		PDS235	LSSB-9A	(LSSB)(Safety Injection OK)(Failure to Deliver Feedwater)(Safety Depressurization for Bleed Fails)	2.2E-09
			LOFW-9A	(LOFW)(Emergency Feedwater Fails)(Safety Depressurization for Bleed Fails)	4.6E-07
			TOTH-9A	(Other Transients)(Feedwater Fails)(Safety Depressurization Fails)	2.7E-09
			LOOP-9A	(LOOP)(Failure to Deliver Emergency Feedwater)(Safety Depressurization for Bleed Fails)	3.8E-09

TABLE 4-2
(Sheet 3 of 4)

MAPPING SEQUENCES INTO RELEASE CLASSES

RC	RC FREQ	PDS	SEQUENCE	DESCRIPTION	SEQ. FREQ*.
RC3.4E	6.73E-9	PDS235	LSSB-9A	(LSSB)(Safety Injection OK)(Failure to Deliver Feedwater)(Safety Depressurization for Bleed Fails)	2.2E-09
			LOFW-9A	(LOFW)(Emergency Feedwater Fails)(Safety Depressurization for Bleed Fails)	4.6E-07
			TOTH-9A	(Other Transients)(Feedwater Fails)(Safety Depressurization Fails)	2.7E-09
			LOOP-9A	(LOOP)(Failure to Deliver Emergency Feedwater)(Safety Depressurization for Bleed Fails)	3.8E-09
		PDS20	ML1-3A	(Medium LOCA 1)(Safety Injection Fails)	1.4E-07
		PDS3	LL-3A	(LLOCA)(SITs Inject OK)(Safety Injection Fails)	1.1E-07
			LL-4A	(LLOCA)(SITs Fail to Inject)	4.7E-09
			VR-A	Vessel Rupture	1.0E-07
		PDS85	ML2-3A	(Medium LOCA 2)(Safety Injection Fails)	1.6E-07
RC3.6E	3.12E-9	PDS184	SGTR-16A	(SGTR)(Safety Injection Fails)(Aggressive Cooldown OK)(RHR Injection Fails)	1.5E-08
			SGTR-17A	(SGTR)(Injection Fails)(Aggressive Secondary Cooldown Fails)	2.7E-07
		PDS235	LSSB-9A	(LSSB)(Safety Injection OK)(Failure to Deliver Feedwater)(Safety Depressurization for Bleed Fails)	2.2E-09
			LOFW-9A	(LOFW)(Emergency Feedwater Fails)(Safety Depressurization for Bleed Fails)	4.6E-07
			TOTH-9A	(Other Transients)(Feedwater Fails)(Safety Depressurization Fails)	2.7E-09
			LOOP-9A	(LOOP)(Failure to Deliver Emergency Feedwater)(Safety Depressurization for Bleed Fails)	3.8E-09
RC3.2M	1.80E-9	PDS148	LOFW-4E	(LOFW)(Emergency Feedwater OK)(Long-term Decay Heat Removal Fails)(Bleed OK)(Safety Injection for Feed Fails)	3.6E-08
			TOTH-4E	(Other Transients)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(Safety Injection for Feed Fails)	6.9E-08
			TRND-4E	Tornado, PSV reseats, EFW OK, LTDHR Fails, Bleed OK, Feed Fails	2.5E-07
		PDS136	LOFW-4A	(LOFW)(Emergency Feedwater OK)(Long-term Decay Heat Removal Fails)(Bleed OK)(Safety Injection for Feed Fails)	3.6E-08
			TOTH-4A	(Other Transients)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(Safety Injection for Feed Fails)	6.9E-08
			TRND-4A	Tornado, PSV reseats, EFW OK, LTDHR Fails, Bleed OK, Feed Fails	2.5E-07
RC3.6M	1.81E-9	PDS148	LOFW-4E	(LOFW)(Emergency Feedwater OK)(Long-term Decay Heat Removal Fails)(Bleed OK)(Safety Injection for Feed Fails)	3.6E-08
			TOTH-4E	(Other Transients)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(Safety Injection for Feed Fails)	6.9E-08
			TRND-4E	Tornado, PSV reseats, EFW OK, LTDHR Fails, Bleed OK, Feed Fails	2.5E-07
		PDS136	LOFW-4A	(LOFW)(Emergency Feedwater OK)(Long-term Decay Heat Removal Fails)(Bleed OK)(Safety Injection for Feed Fails)	3.6E-08
			TOTH-4A	(Other Transients)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(Safety Injection for Feed Fails)	6.9E-08
			TRND-4A	Tornado, PSV reseats, EFW OK, LTDHR Fails, Bleed OK, Feed Fails	2.5E-07
RC4.4E	5.98E-9	PDS184	SGTR-16A	(SGTR)(Safety Injection Fails)(Aggressive Cooldown OK)(RHR Injection Fails)	1.5E-08
			SGTR-17A	(SGTR)(Injection Fails)(Aggressive Secondary Cooldown Fails)	2.7E-07

TABLE 4-2
(Sheet 4 of 4)

MAPPING SEQUENCES INTO RELEASE CLASSES

RC	RC FREQ	PDS	SEQUENCE	DESCRIPTION	SEQ. FREQ*.
RC4.8E	1.12E-9	PDS235	LSSB-9A	(LSSB)(Safety Injection OK)(Failure to Deliver Feedwater) (Safety Depressurization for Bleed Fails)	2.2E-09
			LOFW-9A	(LOFW)(Emergency Feedwater Fails)(Safety Depressurization for Bleed Fails)	4.6E-07
			TOTH-9A	(Other Transients)(Feedwater Fails)(Safety Depressurization Fails)	2.7E-09
			LOOP-9A	(LOOP)(Failure to Deliver Emergency Feedwater)(Safety Depressurization for Bleed Fails)	3.8E-09
		PDS20	ML1-3A	(Medium LOCA 1)(Safety Injection Fails)	1.4E-07
		PDS3	LL-3A	(LLOCA)(SITs Inject OK)(Safety Injection Fails)	1.1E-07
			LL-4A	(LLOCA)(SITs Fail to Inject)	4.7E-09
			VR-A	Vessel Rupture	1.0E-07
		PDS85	ML2-3A	(Medium LOCA 2)(Safety Injection Fails)	1.6E-07
RC4.12E	6.54E-9	PDS184	SGTR-16A	(SGTR)(Safety Injection Fails)(Aggressive Cooldown OK) (RHR Injection Fails)	1.5E-08
			SGTR-17A	(SGTR)(Injection Fails)(Aggressive Secondary Cooldown Fails)	2.7E-07
		PDS181	SGTR-17B	(SGTR)(Injection Fails)(Aggressive Secondary Cooldown Fails)	2.7E-07
RC4.18L	5.56E-9	PDS194	SGTR-9F	(SGTR)(Safety Injection OK)(Deliver Feedwater OK)(RCS Pressure Control Fails)(SG not Isolated)(Failure to Refill IRWST)	4.4E-09
			SGTR-15F	(SGTR)(Safety Injection Fails)(Aggressive Cooldown OK)(SCS Injection OK)(Unisolable Leak in Ruptured SG)(Failure to Re-fill IRWST)	1.2E-09
RC5.1E	5.10E-10		ISLOCA	FAILURE OF CHECK AND ISOLATION VALVES IN ONE SCS line	5.1E-10

* FREQUENCY FOR CORE DAMAGE (LEVEL I)

TABLE 4-3

RANKING OF RELEASE CLASSES BY OFFSITE RISK

RANK	Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y
1	RC4.12E	6.54E-09	5.07E+06	3.32E-02
2	RC4.18L	5.56E-09	5.90E+06	3.28E-02
3	RC4.4E	5.98E-09	5.24E+06	3.13E-02
4	RC3.4E	6.73E-09	1.20E+06	8.08E-03
5	RC3.1E	6.58E-09	1.02E+06	6.71E-03
6	RC3.2E	3.08E-09	1.32E+06	4.07E-03
7	RC3.6E	3.12E-09	1.27E+06	3.96E-03
8	RC3.6M	1.81E-09	1.97E+06	3.57E-03
9	RC3.2M	1.80E-09	1.81E+06	3.26E-03
10	RC2.7M	1.22E-08	1.38E+05	1.68E-03
11	RC5.1E	5.10E-10	2.87E+ 6	1.46E-03
12	RC2.4E	3.64E-08	2.38E J4	8.66E-04
13	RC2.6E	3.45E-08	2.35E+04	8.11E-04
14	RC2.5E	2.84E-08	2.35E+04	6.67E-04
15	RC2.2M	4.05E-09	1.31E+05	5.31E-04
16	RC2.1E	3.46E-09	1.37E+05	4.74E-04
17	RC2.7E	1.62E-08	2.35E+04	3.81E-04
18	RC2.2E	2.04E-09	1.37E+05	2.79E-04
19	RC2.6M	9.08E-09	3.02E+04	2.74E-04
20	RC4.8E	1.12E-09	1.86E+05	2.08E-04
21	RC2.5M	3.95E-09	4.73E+04	1.87E-04
22	RC1.1E	1.36E-06	1.19E+02	1.62E-04
23	RC1.1M	3.81E-07	1.09E+02	4.15E-05
	SUM =	1.93E-06		1.35E-01

Table 4-4
(Sheet 1 of 2)

RANKING OF SEQUENCES BY CDF

SEQUENCE CODE	SEQUENCE	CDF EV/YEAR
LOFW-9	(LOFW)(Emergency Feedwater Fails)(SDS for Bleed Fails)	4.6E-7
SGTR-17	(SGTR)(Injection Fails)(Aggressive Secondary Cooldown Fails)	2.7E-7
SL-11	(SLOCA)(Safety Injection Fails)(Aggressive Cooldown Fails)	1.6E-7
ML2-3	(Medium LOCA 2)(Safety Injection Fails)	1.6E-7
ML1-3	(Medium LOCA 1)(Safety Injection Fails)	1.4E-7
LL-3	(LLOCA)(SITs Inject OK)(Safety Injection Fails)	1.1E-7
VR	Vessel Rupture	1.0E-7
TOTH-4	(Other Transients)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(SIS for Feed Fails)	6.9E-8
ATWS-29	(ATWS)(Adverse MTC)	4.7E-8
LOFW-4	(LOFW)(Emergency Feedwater OK)(Long-term DHR Fails)(Bleed OK)(SIS for Feed Fails)	3.6E-8
SBO	Station Blackout with Battery Depletion	2.1E-8
LOFW-8	(LOFW)(Emergency Feedwater Fails)(Bleed OK)(Safety Injection for Feed Fails)	2.1E-8
SGTR-16	(SGTR)(Safety Injection Fails)(Aggressive Cooldown OK)(RHR Injection Fails)	1.5E-8
LOOP-12	(LOOP)(PSV Fails to Reseat)(SIS Injection Fails)	1.3E-8
SL-10	(SLOCA)(Safety Injection Fails)(Aggressive Cooldown (RHR Injection Fails)	9.0E-9
SL-4	(SLOCA)(Safety Injection OK)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(SDS Fails)	8.9E-9

Table 4-4
(Sheet 2 of 2)

RANKING OF SEQUENCES BY CDF

SEQUENCE CODE	SEQUENCE	CDF EV/YEAR
TOTH-5	(Other Transients)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(SDS Fails)	6.9E-9
SGTR-12	(SGTR)(Safety Injection OK)(Feedwater Fails) (SDS - Bleed Fails)	6.3E-9
LOFW-5	(LOFW)(Emergency Feedwater OK)(Long-term DHR Fails)(SDS for Bleed Fails)	5.6E-9
LL-4	(LLOCA)(SITs Fail to Inject)	4.7E-9
SGTR-9	(SGTR)(Safety Injection OK)(EFW OK)(RCS Pressure Control Fails)(SG not Isolated)(Failure to Refill IRWST)	4.4E-9
LOOP-9	(LOOP)(Failure to Deliver Emergency Feedwater) (SDS for Bleed Fails)	3.8E-9
LHV-5	(LHVAC)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(SDS for Bleed Fail)	3.6E-9
TOTH-9	(Other Transients)(Feedwater Fails)(Safety Depressurization Fails)	2.7E-9
LSSB-9	(LSSB)(Safety Injection OK)(EFW Failure (Safety Depressurization for Bleed Fails)	2.2E-9
ATWS-9	(ATWS)(PSVs Open and Re-close OK)(No Consequential SGTR)(Deliver Feedwater OK)(Failure to Borate by Charging Pumps)(Safety Depressurization Fails)	2.1E-9
SGTR-15	(SGTR)(Safety Injection Fails)(Aggressive Cooldown OK)(SCS Injection OK)(Unisolable Leak in Ruptured SG)(Failure to Re-fill IRWST)	1.2E-9

5.0 DESIGN ALTERNATIVES

Potential modifications to the System 80+ design were derived from a survey of the dominant failure modes are shown in Table 4-1 through Table 4-4. Others were suggested by the PRA or design engineering staff. Some of the DAs were suggested by a foreign utility. Table 5-1 gives the DAs considered and how they were treated.

The risk reduction values of twenty-seven DAs were quantified. These were selected based on the SAMDAs for the Limerick plant[2] Comanche Peak SAMDA[15], NUREG/CR-4920[16], GSI-163[17], and a review of the dominant failure modes for the System 80+ plant. In addition, suggestions from C-E personnel with technical expertise in containment response were employed. Design Alternatives from earlier plant studies were also considered.

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 DA that prevent core damage, the frequency of affected release classes are reduced by the fraction that the sequence contributes to the RC 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 alternative RCP seal cooling system.

At the beginning of the design process, it was recognized that the steam generator integrity was important to safety and plant economics. The risk of SRTR in the System 80+ design is two orders of magnitude below current plants but SGTR represents over half the off site risk. SGTR represents System 80+ is designed to prevent MSSV actuation following SGTR as described below and also includes new or enhanced features for the prevention of SGTRs.

Features to prevent SGTRs include:

- Steam generator tubes made of thermally treated Inconel 690, which has favorable corrosion resistance properties including superior resistance to primary and secondary stress corrosion cracking

- A deaerator in the condensate/feedwater system for removal of oxygen

- Condensate system with full flow condensate polisher to remove dissolved and suspended impurities

- Main condenser with provisions for early detection of tube leaks, and segmented design permitting repair of leaks while

operating at reduced power

- Steam, feedwater and condensate generator blowdown system and SG secondary side recirculation system for chemistry control during wet layup

The response to Unresolved Safety Issue A-4 in CESSAR-DC, Appendix A further describes design features to assure SG tube integrity. New or enhanced System 80+ features which help to mitigate SGTRs include:

- Larger steam generator secondary volume

- Larger pressurizer

- Four train safety injection system

- Four train emergency feedwater system

- Electrical system upgrades including alternate AC gas turbine and 8-hour batteries

- Safety depressurization and vent system

- Component cooling water system upgrade to four 100% capacity pumps and heat exchangers

- Highly reliable turbine bypass system, discharging all steam to condenser, not partially to atmosphere as in earlier designs

- Radiation monitors on the steam lines

- N-16 monitors for the steam generators

The System 80+ design meets the EPRI ALWR requirement of preventing main steam safety valve actuation following a SGTR. A reactor trip on high SG water level, actuation of the turbine bypass system and controlled depressurization of the RCS using the safety depressurization and vent system (SDVS) limit secondary side pressure below the MSSV setpoint. The turbine bypass valves discharge steam to the main condenser, which minimizes the radioactive release to the environment. The intent of the ALWR URD was to meet the above requirement on a best-estimate basis (i.e., credit for operator action and use of control-grade equipment is acceptable) to provide an effective and economical design.

The consequences of a worst-case steam generator tube rupture (SGTR) with loss of offsite power (LOOP) where the containment is bypassed due to malfunction of a main steam system valve has been analyzed. The analysis presented in CESSAR-DC Section 15.6.3.3, SGTR with LOOP and Single Failure, calculated the worst-case

releases for an SGTR event with LOOP and a stuck open ADV on the affected steam generator.

The analysis simulated a double-ended break of one SG tube. The analysis contained conservative assumptions regarding atmospheric dispersion factors, initial RCS and SG activity levels, and iodine spiking. Mitigating operator actions based on the approved CE emergency procedure guidelines (EPGs), CEN-152, were simulated. The analysis showed that no fuel failures were expected for this event.

The ADV on the affected SG was assumed to stick open when the operator tried to reseat the ADV to isolate the affected SG. After 30 minutes of steaming through the stuck-open ADV, the operator isolated this path by closing the ADV block valve. However, the leak of RCS liquid through the tube break continues for the duration of the analysis (8 hours) due to the conservative nature of the analysis models. In order to avoid overfilling the SG, the operator periodically steams from the affected SG per the EPGs. This additional steaming increased the total radiation dose. The total releases are well within regulatory limits.

It was recognized that the SGTR event represented a significant fraction of the offsite risk in this SAMDA analysis and DAs were selected specifically to address these sequences. These DAs include the Alternative Pressurizer Auxiliary Spray (DA5.5), Ideal 100% SG inspection (DA5.12), MSSV and ADV Scrubbing (DA5.13), Alternative SIS (DA5.11), and Diesel SIS Pump (DA5.19). The last two DAs address failure to inject for RC4.12E. DA5.23 specifically addresses refilling the RWST during a SGTR. Secondary side guard pipes (DA5.28) are also evaluated.

For the DAs that protect the containment, the releases are put to zero and then the risk is reevaluated. These DAs include the improved containment sprays, filtered vent, concrete composition, reactor vessel exterior cooling, and H2 ignitors.

The following sections discuss each Design Alternative.

5.1 ALTERNATIVE CONTAINMENT SPRAY

An alternative containment spray system is assumed to prevents the high pressure containment failures caused by slow steam pressurization (RC2.2M) and eliminate the sequences where scrubbing does not occur. 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 to 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 DA reduces the risk of six of the release classes (see Table 5-3). Using a risk conversion factor of $1,000 per person-rem, this DA would have an annual value of $7.27/y. The annual benefit of the Design Alternative could be converted to a capital benefit using the levelized capital cost rate of 16.6% developed in Section 2. The ideal containment spray system would be cost beneficial if it could be installed for less than $44 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 DA would have to cost less than $22 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. This design modification has been included in the design.

5.2 FILTERED VENT (CONTAINMENT)

The filtered vent Design Alternative prevents all slow high pressure containment failures and therefore reduces the doses in RC2.2M (see Table 5-4). Using a value of $1,000 per person-rem avoided, this Design Alternative has a benefit of $0.53/y. Using a levelized capital cost rate of 16.6%, a system with a capital cost of $3 would just be cost effective.

The cost estimates for a filtered vent system range from $2.8 Million to $25 Million. IDCOR Technical Report 19.1, July 1983 estimated a cost of $25M for larger system than our design and sized to handle ATWS. In the ABWR SAMDA, a cost of $3M was quoted. This is probably a smaller design taking credit for scrubbing in the BWR suppression pool. The Comanche Peak SAMDA[15] estimated the cost from $15M to $22.3M and the Limerick SAMDA[2] gives a range from $2.8M to $11.3M. The System 80+ estimate of $10M is for a non-ATWS sized, fully Category I facility and is bounded by the other estimates.

5.3 ALTERNATIVE 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 DA 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 two of the release classes. Using a $1,000 per averted person/rem and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than $11.

Design of a battery system with unlimited capacity is not possible. However, to increase the existing battery capacity for the EFWS 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. In the Comanche Peak SAMDA[15] additional batteries were estimated to cost between $1.3M and $3M.

5.4 RCP SEAL COOLING

The System 80+ employs a type of Reactor Coolant Pump (RCP) seal which can withstand a loss of cooling and not result in 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[8]. 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 design addition will provide additional diversity for RCP seal cooling and provide a seal cooling system that is not dependent on CCW. Such a RCP seal cooling pump has been added to the System 80+ plant as a result on NRC's questions on testing of the RCP seals and therefore a cost benefit analysis is not needed.

5.5 ALTERNATIVE PRESSURIZER AUXILIARY SPRAY

This Design Alternative was introduced to specifically address steam generator tube rupture (SGTR) which is the initiating event for the largest three RCs. 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. This reduces the risk of SGTR in the System 80+ design has for six RCs (see Table 5-6). Using a $1,000 per averted person-rem and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than $545.

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 SCS 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 ALTERNATIVE ATWS PRESSURE RELIEF VALVES

This Design Alternative was selected because the System 80+ design uses an advanced digital plant protection system that has raised much interest. It 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 DA is assumed to eliminate all the ATWS core damage sequences. ATWS does not show up as a dominant PDS but represents 3% of the CDF (see Figure 15.2.1 of the PRA). Therefore the risk of all release classes from transients was reduced by 3% (see Table 5-7). Using a $1,000 per averted person-rem and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than $6.

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 ALTERNATIVE 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 eliminate seven RCs where basemat melt-through is modeled (see Table 5-8) Using a $1,000 per averted person-rem

and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than $29.

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. Currently additional concrete can not be added to the reactor cavity, since there would be an interference with the incore instrumentation tubes which exit the bottom of the reactor vessel. In order to add an additional two feet of concrete the NSSS would have to be raised by two feet to avoid interference with the incore instrumentation tubes. Raising the NSSS would also require the crane wall height to be increased by two feet in order to have adequate clearance to lift the reactor head and service other NSSS components. In order to increase the crane wall height the containment diameter would have to be increased by approximately two feet in order to avoid an interference between the crane wall and containment vessel and to allow adequate space for spray coverage. An increase in containment diameter may also require an increase in containment plate thickness. An increase in containment plate thickness will require post-weld heat treatment for the construction of the containment vessel since the current thickness is at the limit allowed by the ASME Code before post-weld heat treatment is required. An increase in containment diameter will also require an increase in the diameter of the concrete shield building. The added cost for an additional two feet of concrete in the reactor cavity floor is small. However, the added cost of additional steel for the increased containment diameter and thickness, post-weld heat treatment required for the increased containment plate thickness, additional concrete and rebar for the increase in crane wall height and shield building diameter is estimated to exceed $5 million.

Because the dominant risks are associated with containment bypass events, the risk reduction associated with the additional thickness of the containment was not quantified. In events where no decay heat removal is available, the containment failure would still be postulated.

5.8 REACTOR VESSEL EXTERIOR COOLING

A reactor vessel exterior cooling system is assumed to prevent vessel melt-through and subsequent basemat attack or steam explosions. This Design Alternative reduces the consequences of eleven RCs (see Table 5-9). Using a $1,000 per averted person-rem and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than $197.

The current arrangement for the IRWST will not allow wetting of the reactor vessel. The elevation of the IRWST was selected to ensure that wetting of the vessel would not occur should the holdup volume

and cavity flood valves inadvertently open during power operation. This will prevent thermal shock of the vessel. However, water can be induced into the reactor cavity for exterior vessel cooling from external sources such as the Boric Acid Tank which provides a makeup source to the IRWST or by inducing water through the temporary hookup on the containment spray line discussed in Design Alternative 5.1 above and cost $1.5M. However, to utilize this option it must first be demonstrated that the reactor vessel will not breach do to thermal shock of the vessel from the cold water. The analysis to demonstrate this is estimated to cost $1M. This is based on the FERC prudence hearings for Yankee Atomic Electric Co. where it was reported that demonstration of vessel integrity would be a "multi-million dollar cost"[14]. The total cost would be $2.5 M.

Given such a design modification was licensable, an inadvertent wetting of the reactor vessel during power, and no actual failures occurred, the event would require extensive testing and inspection before the plant would be permitted to startup. Such costs and additional economic risks have not been quantified but it is believed that these risks would outweigh any advantage of vessel flooding.

5.9 ALTERNATIVE H2 IGNITORS

Ideal hydrogen (H2) ignitors would prevent release classes associated with containment failures from hydrogen burns or explosions. The System 80+ design has two different hydrogen control systems as described in Section 6.2.5 of the CESSAR-DC. The Containment Hydrogen Recombiner System (CHRS) is designed to control the H2 concentrations in the containment following a LOCA. The CHRS prevents the concentration of hydrogen from reaching the lower flammability limit of 4% by volume in air or steam-air mixtures. During a degraded core accident, hydrogen will be produced at a greater rate than that of a design basis LOCA. The Hydrogen Mitigation System (HMS)is designed to accommodate the hydrogen produced from 100% fuel clad metal-water reaction and limit the average hydrogen concentration in the containment to a 10% for a degraded core accident. The HMS consists of 80 Glow Plug Ignitors distributed through out the containment. Their placement is based on a detailed assessment of the flow paths to fully cover all of the containment. Section 19.11.4.1.3 of the CESSAR-DC discussed hydrogen in severe accidents. System 80+ already has a degraded core H2 control system and only two release classes (RC2.1E and 2.2E) have containment failure from hydrogen burning. This Design Alternative reduces the risk of these RCs (see Table 5-10). Such a system would have to cost $5 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 recombiners which do not require a power source. Since catalytic hydrogen recombiners 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. In the Comanche Peak SAMDA[15] additional batteries were estimated to cost between $1.3M and $3M and a Ignition system was estimated to cost $5.8M to $8M.

5.10 ALTERNATIVE 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 all sequences with HPSI failures can be eliminated (see Table 5-11). This Design Alternative would have to cost $502 to be cost beneficial.

As shown in Table 19.6.3.6-5 of the PRA, the dominant failure mode (80% of the total for small break LOCA) is common cause failure of the four check valves or four motor operated isolation valves. The Alternative SIS would have eight additional valves, each one with piping to parallel the existing valves. The estimated cost for this modification is $2.2M. It is assumed that these valves are not subject to common cause failures. Testing and maintenance has been neglected.

5.11 ALTERNATIVE 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 pumps are effective in getting coolant into the core and removing decay heat. This DA eliminates all sequences in Table 4-2 where the SDS fails. The risk reduction, shown in Table 5.12 is worth $91 in capital to be cost beneficial.

Designing a perfect safety depressurization system is not possible. However, increased reliability and diversity of the system can be obtained by increasing the redundancy of the safety depressurization valves and/or providing valves that are diverse. Providing the additional valves, piping and instrumentation is estimated to exceed $500,000. In the Comanche Peak SAMDA[15] an

Alternate depressurization system were estimated to cost between \$1.9M and \$3.7M.

5.12 100% SG INSPECTION

Inspection of 100% of the tubes in a steam generator is not really a design alternative but is a maintenance practice. It was selected because it is has reasonable costs, and can be executed with a management decision. This DA was introduced to specifically address steam generator tube rupture (SGTR) which is the initiating event for the largest three RCs. The analysis assumes that all SGTR are eliminated. This reduces the risk of SGTR in the System 80+ design has for six RCs (see Table 5-6). Using a \$1,000 per averted person-rem and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than \$605.

The increased cost of performing eddie current testing on 100% of the steam generator tubes compared to a 20% random inspection of the steam generator tubes is \$1.5 million per refueling outage. Assuming an eighteen month refueling, this would cost \$1.0M/y or be equivalent to a capital cost of \$5.59M.

5.13 MSSV AND ADV SCRUBBING

The discharges of the main steam safety valves (MSSVs) and atmospheric dump valves (ADVs) could be scrubbed by routing the discharges through a structure with a water spray condense the steam and remove most of the fission products. This DA was introduced to specifically address steam generator tube rupture (SGTR) where isolation fails (the largest three RCs). Table 5-14 gives the risk reduction of this DA. The risk reduction is worth \$544 dollars in capital to be cost beneficial.

This modification would require building structure over the valve discharges and installing a header system to distribute water. In addition, a pump, piping, water supply and instrumentation and drain system would be needed. Conceptually, this system is similar to a containment spray system for which a cost estimate of \$9.5M was give in the Commanche Peak SAMDA analysis[15] and that cost estimate will be used in this analysis.

5.14 THIRD DIESEL GENERATOR

The System 80+ plant is designed to have two diesel generators (DGs), a combustion turbine and two independent switchyards. Many plants are using a third DG as a swing unit or during a refueling when one DG is out for maintenance. This DA was selected to address the risk reduction of installing an additional unit. It

was assumed that the unit was effected by common cause failure and had a conditional failure rate[10] (γ) of 0.76/d given that the other DGs had failed. This reduced the risk of the two RCs for station blackout by 24% (see Table 5-15). Using a $1,000 per averted person-rem and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than $3.

Addition of a third diesel generator to lower the probability of station blackout would require the addition of a 6.4 MW diesel generator, its associated support systems, additional component cooling water piping to and from the diesel generator cooling water heat exchanger, an addition of a swing bus, additional cabling for connecting the diesel generator to the electrical distribution system, an additional diesel generator building to house the diesel, an additional fuel oil storage tank and storage tank structure, and additional HVAC systems for the diesel generator building and fuel oil storage tank structure. A study conducted for Duke Power Company's McGuire Nuclear Station estimates the cost of adding a similar swing diesel to be in excess of $25 Million. This McGuire study investigated the cost that other utilities incurred in installing additional diesel generators. Pennsylvania Power and Light installed a swing diesel at their Susquehanna plant. This job was originally bid at $30 Million; however, final installation ended up costing $130 Million. Northern States Power added additional diesel generators at the Prairie Island site. The initial bid for the project was $60 Million; however the final price was around $78 Million. The cost estimates for an additional diesel was $18.4M to $19M in the Comanche Peak SAMDA[15]. For this analysis the additional diesel will be estimated to cost $25 million.

5.15 ATWS INJECTION SYSTEM

An "ink" injection system was proposed for the heavy water New Production Reactor as a shutdown system diverse from the mechanical rods. also a foreign utility also showed some interest in this concept. Therefore, this DA was selected for evaluation. In terms of risk reduction benefits, this DA has the same advantage as the ATWS pressure relief valves (see Table 5-7) and would have an equivalent capital value of $6.

For estimating the cost of this DA, it was assumed that the RCP seal cooling pump could be used with existing sources of boron and existing piping and valves. The cost of this DA is $300,000 and is associated with the instrumentation and control system to activate the pump and align it.

5.16 DIVERSE PPS

This Design Alternative was selected because the System 80+ design uses an advanced digital plant protection system that has raised much interest. A foreign utility also inquired about this DA. In this analysis, it was assumed that the redundant PPS eliminated all ATWS. The System 80+ has an alternate protection system (APS) in order to meet the ATWS rule. The APS contains an alternate scram system and a diverse emergency feedwater actuation system (DEFAS). The DA considered here is a third, diverse PPS to resolve I&C diversity concerns. This DA has the same risk reduction as the ATWS pressure relief valves (see Table 5-7) and using a $1,000 per averted person-rem and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than $6. The cost of a diverse PPS was estimated to be $3,000,000.

5.17 ALTERNATIVE CONTAINMENT MONITORING SYSTEM

The alternative containment monitoring system was selected to address the RCs where containment bypass is predicted. It does not address steam generator tube rupture where failure to isolate the SG is predicted. This DA is assumed to eliminate the containment bypass RC4.8E and the interfacing LOCA RC5.1E (see Table 5-16). Using a $1,000 per averted person-rem and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than $10.

This modification would require the addition of a redundant and diverse limit switch to each containment isolation valve, and the addition of control and fiber optic cabling to the plant computer. The cost of this modification would be in excess of $1 million. In the Comanche Peak SAMDA[15] an alternative bypass instrumentation system was estimated to cost $2.7M.

5.18 CAVITY COOLING

The Cavity Cooling DA uses the existing SCS heat exchangers in the IRWST to cool the reactor vessel cavity under natural circulation. It uses existing piping and equipment but only increases the size of the pipes to ensure that natural circulation is effective. In the upper limit as modeled here, this DA assumes the existing SCS equipment always works and it is assumed to eliminate vessel failure, steam explosions and concrete interactions. It has the same advantages and risk reduction worth as the reactor vessel cooling system (see Section 5.8 and Table 5-9) but has a lower capital cost because it uses existing equipment. This modification would be cost beneficial if the cost was less than $197.

This modification would require increasing the size of the existing cavity flood lines and performing analysis that there is adequate mixing between the reactor cavity and IRWST. The cost of this modification would be in excess of $50,000. In the Comanche Peak SAMDA[15] an alternative cavity flooding system was estimated to cost between $1.2M and $2.3M.

5.19 <u>12-HOUR BATTERIES</u>

The DA described in Section 5.3 is for an ideal battery system. This DA is for a specific and technically realistic design alternative of using a battery system that would maintain load for twelve hours. Such an improvement would decrease the failure to restore offsite power from 0.081 to 0.031[10], a 38% improvement. In terms of risk reduction benefits, this DA reduces the risk of two RCs (see Table 5-17) and would have an equivalent capital value of $4.

Increasing the current battery size to accommodate a 12-hour duty cycle for station blackout loads rather than a 8-hour duty cycle would require more plates per cell (minimum of 25% increase). Preliminary estimates show that the existing 8-hour duty cycle requires a large number of plates per cell (assuming 60 cell battery). Therefore, a 25% increase in plates per cell may exceed the number of plates that can be placed in a typical cell and may not be possible. However, if cells are available in sufficient size, they would be larger per cell and would require an additional mounting rack, which would require at a minimum 1.5 times existing battery building space. The more likely scenario would require another 60 cell battery or two 58 cell batteries connected in parallel. Thus, the required space would be 2 times existing space. The cost of this modification would be in excess of $300,000.

5.20 <u>TORNADO-PROTECTION FOR COMBUSTION TURBINE</u>

The PDSs in Table 4-2 with the designator "TRND" are for tornados and it was assumed that offsite power was lost and the combustion turbine was not available. For these three sequences, it was assumed that the DA completely protected the turbine and it was available to supply AC with a failure rate[10] of 0.025/d. This reduced the risk of two RCs (see Table 5-18) and would be cost beneficial if it could be installed for less than $10.

The cost of this DA was estimated at over $3M ind includes protection of the turbine, fuel tank, and tunneling for cooling line. The cost could be as high as $4M depending on tunneling distances.

5.21 DIESEL SI Pumps (2)

The System 80+ design has a very reliable four train HPSI system to begin with. The high pressure safety injection Design Alternative (Section 5.10) assumes that all sequences with HPSI failures can be eliminated (see Table 5-11). This Design Alternative is more specific. It assumes that two of the electric SIS pumps are replaced with diesel pumps. This reduces common cause failures of all four pumps and also reduces the risk of station blackout. Using the failure rates and common cause dependencies in Reference 10, the reliability or the SIS would be increased by factor of 60. Station blackout was assumed to be eliminated. Table 5-19 shows that nineteen RCs are reduced with a risk worth of $83.79/y. This DA would be cost beneficial in terms of offsite risk reduction if it could be installed for less than $505 in capital.

This modification would require replacing the electric motors on two of the safety injection pumps with diesel engines. The diesel engines will also require addition support systems and additional building volume to house the diesel drives and support systems compared to electric motor drives. The cost of this modification would be in excess of $2 million.

5.22 ALTERNATIVE STARTUP FEEDWATER SYSTEM

The startup feedwater system introduces the feedwater upstream of the main feedwater control valves and is assumed to be unavailable for transients such as loss of feedwater. The alternative startup feedwater system would be available as a backup to the EFWS. It is assumed to eliminate the sequences in Table 4-2 where the EFWS fails. This reduces the risk of thirteen RCs (see Table 5-20) and would be cost beneficial if it could be installed for a cost under $198.

The System 80+ startup feedwater system has been modified such that it can be utilized as a back up to the Emergency Feedwater System. The System 80+ startup feedwater pumps are powered from the Combustion Turbine such that they are available on a loss of offsite power event. The condensate storage tank provides the water source for the startup feedwater pumps. Since, the startup feedwater system is non-safety the water from the startup feedwater pump is supplied upstream of the main feedwater isolation valves. Should the transient cause the main feedwater isolation valves to close on a Main Steam Isolation Signal, the signal can be bypassed and the valves reopened. The instrument air compressors are also powered from the Combustion Turbine. Therefore, they will be available to provide the air source for reopening the main feedwater isolation valves. Since this DA is included in the System 80+ design, no cost benefit analysis is necessary.

5.23 EXTENDED RWST SOURCE

In the important SGTR sequences to public risk (RC4.18L), the RWST source expires as a makeup source. This DA consists of a ground level tank of borated water and a pump and piping to pump the water to the elevated RWST. It is assumed that the supply of water is sufficient to permit corrective actions before it also is exhausted. This DA is assumed to eliminate RC4.18L (see Table 5-21). Using a $1,000 per averted person-rem and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than $198.

A detailed design for the extended RWST source has not been performed but it would require a ground level tank of borated water and a pump and piping to pump the water to the elevated RWST, and instrumentation and control system. It is estimated to cost in excess of $1 Million.

5.24 N-16 MONITORS

The N-16 monitors have been added to the System 80+ design. Its purpose is to assist the operators in identifying SGTR events. This DA was not quantified since it has been included in the design.

5.25 INCREASE SECONDARY SIDE PRESSURE

Upgrading the design pressure of the secondary system including the MSSVs to 1500 psia from the current 1200 psia was considered early in the System 80+ design process. It was determined that an increased design pressure would not significantly reduce the probability of containment bypass and release to the environment during a SGTR event.

During a SGTR with loss of offsite power, the condenser is not available for plant cooldown. The decay heat of the core and the stored energy in components are released to the atmosphere via the MSSVs, then via the SG ADVs. The steaming will continue until reaching shutdown cooling system entry conditions. The total heat to be removed (or the total steam release) is only slightly reduced by increasing the secondary design pressure and MSSV setpoints. Hence, using conservative safety analysis assumptions and methods, the overall radiation release would be essentially unchanged.

During a SGTR with offsite power available, the operator will act to mitigate this event according to the Emergency Procedures Guidelines, using both control grade and safety grade equipment if required. Therefore, for a "real-world" scenario, an increased

design pressure would not significantly decrease the likelihood of lifting the MSSVs.

There are several technical disadvantages of increasing the secondary system design pressure to 1500 psia:

1. Steam generator would increase by up to 100 tons each. The added weight would increase containment heat sinks, and increase thermal stresses on the steam generator shell and main steam piping. These factors would likely impact the volume and arrangement of the containment. The additional weight would also increase the handling difficulties during fabrication.

2. The RCS support system would need to be redesigned and/or reevaluated to accommodate the increased loads. Any contribution to containment sizing must also be assessed.

3. For decreased heat removal events, RCS temperature and pressure would rise to a much higher value than in current plants. Pressurizer safety valve actuation would be more likely.

4. Unless the entire steam system and turbine are upgraded to 1500 psia, a second set of secondary side relief valves would be required downstream of the MSIVs to protect the low pressure portion of the steam system.

5. Feedwater systems would have to be compatible with the higher design pressure. Increasing secondary design pressure would require a major redesign effort and increase design complexity,which are not consistent with the evolutionary ALWR goals.

In summary, the issue of including an upgrade to the secondary side design pressure was considered from design considerations. Based on this review, this DA poses serious design drawbacks with limited benefits. A cost benefit analysis was not performed for this DA because very limited benefits were expected for extensive costs.

5.26 PASSIVE SECONDARY SIDE COOLERS

Secondary heat rejection for System 80+ has been considered at the conceptual level. Passive secondary heat rejection was included in the conceptual design for SIR, a much smaller plant.

The passive secondary heat rejection concept that is often promoted consists of an elevated condenser designed to full secondary side pressures. The heat sink for this condenser can be either water or air. If it is water, then in addition to the elevated condenser,

there is an elevated water tank that gravity feeds into the condenser and is allowed to boil to atmosphere. Use of air in natural circulation results in a large increase of the surface area of the condenser but it has the potential of continuous long term operation without support. The water tank concept requires a periodic refill.

The base system is relatively simple. However, several supporting functions are required to initiate the system. Isolation of the affected steam generator will be required, otherwise one must assume the entire cooling loop will go water solid with pressures equal to RCS pressure. An alternative is to have a continuous drain system that maintains a suitable free surface in the steam generator. This requires coordination with the RCS makeup system. If the design basis is isolation, will that require redundant systems on each steam generator. Control of cooldown rates is expected to be required, adding additional complexity. Heat rejection capacity sufficient to avoid early releases is expected to result in excessive cooldown rates later.

While simple in concept, the implementation of secondary closed loop cooling is expected to require major changes in the plant structures. A workable system will be more complex than the conceptual presentations being offered. Because of the redundancies in the current System 80+ design, and the potential high cost of this DA, this DA will not be further studied as a SAMDA.

5.27 VENTING THE MSSV IN CONTAINMENT

ABB does not plan to divert MSSV steam releases back to the containment. While such a system would reduce radiological releases to the environment for selected accident scenarios, such a system does not significantly reduce public risk and does carry several disadvantages. It should be noted that this feature does not eliminate releases to the environment.

The technical disadvantages of the MSSV-containment steam return system are summarized below for two hypothetical systems. In the first system, the steam is simply returned to the containment atmosphere. In the second system, the steam is discharged into the IRWST where it would be condensed.

Direct discharge of MSSV into containment has several serious disadvantages.

1. The secondary system return will place an additional loading burden on the containment and restrict plant operators in responding to accidents when containment sprays are unavailable. This could lead to the addition of a containment vent to address those concerns which in itself introduces

another means of inadvertent containment bypass.

2. Any condensed steam discharge will drain to the IRWST, diluting the boron concentration. A minimum IRWST boron concentration for safety injection is necessary for mitigating LOCA and non-LOCA events.

3. The release to containment atmosphere has the potential to cause personal injury.

An MSSV return system directed to the IRWST has similar drawbacks to Items 1 and 2 described above and poses the additional complication that discharge of steam flows typical of the MSSVs may produce excessive loadings within the IRWST.

Either return path would require a major redesign effort and increase design complexity, which are not consistent with evolutionary ALWR goals. Also, this provision will not eliminate radiological releases to the environment from a SGTR.

In summary, the issue of including an MSSV discharge return to the containment was considered from design considerations. Based on this review, this DA poses serious design drawbacks. ABB-CE does not believe that the secondary steam should be piped and vented inside containment. These events are characterized as AOO events and filling the containment with steam during these events would be both damaging to the equipment and dangerous to operators. A cost benefit analysis was not performed for this DA because as it would require an assessment of equipment degradation, injuries, and loss of plant availability after secondary side venting into containment.

5.28 SECONDARY SIDE GUARD PIPES

The secondary side guard pipe was proposed to address a Main Steam Line Break (MSLB) outside containment. This event is postulated to trigger multiple steam generator tube failures which could then result in a core melt because of depletion of coolant inventory. This sequence also bypasses the containment. The guard pipe would extend from the containment to the MSIVs and would be designed to prevent depressurization, given a MSLB in the specific section of pipe. MSLB represented 0.5% of the CDF for System 80+ and consequential SGTR was not modeled. It was assumed that this DA would halve the risk associated with intersystem LOCAs (RC5.1E) and halve the risk associated with all steam line break sequences because it is assumed that half of the lengths of main steam lines are guarded. Table 5-22 quantifies the risk reduction value of this DA. Using a $1000 per averted person-rem and a levelized capital cost of 16.6%, such a modification would be cost beneficial if it cost less than $4.40.

The cost for the guard pipes was taken from GSI-163[17] and adjusted for the different number and size. The original estimate of $1.1M was for a four loop plant. This estimate was first halved for a two loop plant and then increased by 50% to account for the larger size. The final cost of $820,000 was used in this analysis. This cost neglects the increased inspection and maintenance cost of the main steam lines because they are no longer accessible.

5.29 PASSIVE AUTOCATALYTIC RECOMBINERS (PARS)

Passive Autocatalytic Recombiners (PARS) are arrays of a palladium catalyst that will combine molecular hydrogen and oxygen gases into water. These units are currently in the development stage and have not been used in existing U.S. plants. They have a low conversion efficiency and therefore would have to be used in combination with existing H2 ignitors. The advantage of the PARS is that they require no electrical power and therefore would operate during a station blackout. The success of the PARS to prevent a H2 burn would depend on the speed of the production and release of the H2. For this analysis, it was conservatively assumed that the PARS worked perfectly and therefore would prevent release classes associated with containment failures from hydrogen burns or explosions. The System 80+ design already has H2 ignitors with redundant power backup via either DGs , batteries, or CT. Therefore, only two release classes (RC2.1E and 2.2E) have containment failure from hydrogen burning. This Design Alternative reduces the risk of these RCs (see Table 5-10, Alternative H2 Ignitors). Such a system would have to cost $5 to be cost beneficial.

EPRI[13] has been developing PARS technology and estimates that 40 units would be needed for large dry containments. EPRI estimates the units would cost $19,000 each, and the cost for the PARS in System 80+ would be $760,000. This costs neglects any annual costs of cleaning, inspection and testing. Also, both NRC and ACRS have expressed concern about the expected relative slow response time of the PARS.

5.30 HYDROGEN PURGE LINE

An existing System 80+ design feature that could be utilized in venting the containment is the hydrogen purge vent. System 80+ is equipped with two 3 inch diameter hydrogen purge vents which can be used for purposes of containment venting. This design feature is shown in CESSAR-DC Figure 6.2.5-1. The vents are intended for use in post LOCA condition for diverting hydrogen to the secondary containment (annulus) should the hydrogen recombiners be inoperative. The annulus ventilation system then collects and

filters the secondary containment atmosphere before release.

An analysis of the potential application of the venting capabilities of the hydrogen purge piping was performed using the MAAP computer code. This analysis conservatively simulated hydrogen purge as a 0.049 ft^2 equivalent area opening in the containment. A hypothetical accident management strategy was considered, whereby the hydrogen purge system is used to vent at the time the containment reaches 80 psia will enable the containment to maintain its pressure well below the containment failure threshold.

Since there are 4 AC electric motor operated valves in series on each division that must be opened to purge the containment and the annulus ventilation system requires AC power for operation, this feature can not be credited for mitigating severe accidents resulting from a complete loss of AC power.

This DA has already been included in the System 80+ design and no cost benefit analysis is necessary.

5.31 FUEL CELLS

In addition alternative battery types to the traditional lead battery were investigated. Alternative battery types such as lithium or zinc are not commercially available in the necessary sizes to provide the capacity required by System 80+. Fuel cells are available in the size required for System 80+; however, they are not proven technologies in nuclear station applications and are not available as Class 1E equipment. In addition, the use of fuel cells presents the problem of heat generation since a typical fuel cell will operate at a temperature of 300 to 1000 °C. HVAC systems would have to be capable of removing the heat. Also, a safety related fuel delivery and exhaust system would be required for each battery. Design, development and installation of this type of fuel cell system would cost well over $2 million more than a conventional lead acid battery arrangement.

This Design Alternative addresses the release classes where emergency feedwater is lost after battery depletion during a station blackout. This DA is assumed to have the capability to remove decay heat using the turbine feedwater pump for whatever time period that is required (without any failures). This Design Alternative prevents core damage and therefore removes two of the release classes (same as Alternative DC Batteries and EFWS, see Table 5-5). Using a $1,000 per averted person-rem and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than $11.

5.32 HOOKUP FOR PORTABLE GENERATORS

Instead of increasing the battery capacity for the turbine driven EFW pump train, portable generators could be brought in and hooked up for continued operation of the turbine driven EFW pump train after the batteries are depleted. This would require temporary hookup connections so that the portable generators could be connected in a timely manner. These temporary hook up connections would need to be located in an area that was easily accessible for installing the portable generators and would have to be located in an appropriate environment for running the generators during station blackout conditions. The cost of adding these temporary hookup connections, including the cabling to an appropriate location for hookup would be in excess of $10,000.

The diesel driven fire pump was investigated as an alternate feedwater source. This pump is only capable of producing 100 psia pressure. Therefore it does not have adequate head to feed the steam generators which would be in excess of 1000 psia pressure."

This Design Alternative addresses the release classes where emergency feedwater is lost after battery depletion during a station blackout. This Design Alternative prevents core damage and therefore removes two of the release classes (same as Alternative DC Batteries and EFWS, see Table 5-5) and would be cost beneficial if it cost less than $11.

5.33 WATER COOLED RUBBLE BED

The purpose of the water cooled rubble bed is to achieve a coolable debris bed below the vessel and remove decay heat. This DA consist of a floodable rubble bed in the bottom of the vessel cavity. The rubble bed would be kept dry until the corium had penetrated into it, thus minimizing the potential for steam explosion. This DA would have the same risk reduction potential as the ideal, Alternative Concrete Composition (DA5.7). This DA would eliminate seven RCs where basemat melt-through is modeled (see Table 5-8) Using a $1,000 per averted person-rem and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than $29.

The cost of the water cooled rubble bed is estimated[2] to be between $35.5 and $38.5 Million. Another source[18] estimated the cost to be $18.8 Million. Neither source included the cost of actually developing the system. Periodic testing and maintenance of the device which could be significant. For this analysis, the lower cost of $18.8M will be used.

5.34 REFRACTORY LINED CRUCIBLE

The purpose of the refractory lined crucible is to achieve a coolable debris bed below the vessel and remove decay heat. This DA consist of a ceramic lined crucible with cooling located in the vessel cavity. This DA would have the same risk reduction potential as the ideal, Alternative Concrete Composition (DA5.7). This DA would eliminate seven RCs where basemat melt-through is modeled (see Table 5-8) Using a $1,000 per averted person-rem and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than $29.

The cost of the water cooled rubble bed is estimated[2] to be between $108 and $119 Million. Neither source included the cost of actually developing the system. Periodic testing and maintenance of the device which could be significant. For this analysis, the lower cost of $108M will be used.

5.35 AUTOMATIC OVERPRESSURE PROTECTION

ABB-CE conducted an extensive evaluation of the System 80+ standard design to respond to interfacing system LOCA challenges, to address Staff concerns raised in SECY-90-016 and SECY-93-087. ABB-CE and the Staff worked closely in the development of an acceptance criteria and performance of a system-by-system evaluation of ISLOCA challenges. The evaluation was documented in an ABB-CE special report which has been incorporated in CESSAR-DC as Appendix 5E. Table 2-1 of Appendix 5E summarizes the design changes made to achieve ISLOCA responses acceptable to the Staff. Section 4 of Appendix 5E presents the evaluation of design alternatives and rationale for the selected design approach for each potential ISLOCA pathway. Since this issue has been designated by the Staff as technically resolved, no further evaluation or reporting will be provided.

5.36 VACUUM BUILDING

ABB-CE developed a conceptual design for a vacuum building which was designed to reduce emissions from severe accidents and described is in Reference 11. The cost was estimated as $30 M in 1983. A separate IDCOR sponsored study (IDCOR Technical Report 19.1, July, 1983) also estimated the cost to be $30M (approximately $42M in 1993 dollars). Because of the high costs, and because most of the significant releases are bypass events for which the vacuum building would not help, this DA was not quantified.

5.37 RIBBED CONTAINMENT

A ribbed containment was proposed to address failure of the containment from buckling during a seismic event coupled with an inadvertent actuation of the containment spray. This combination of events might lead to a vacuum in containment and some potential buckling. The ribs would not increase the maximum containment overpressure strength because the containment is assumed to fail at a weak point in the containment located between the ribs. Therefore since none of the RCs have containment failure due to a vacuum, no benefits were quantified for this DA. The cost of this DA is in the $10s of millions because the ribs complicate manufacturing and construction and would require field heat treating. Given that this DA has a high cost and no quantifiable benefit, it will not be further quantified.

5.38 DIGITAL LBLOCA PROTECTION

The likelihood of Plant Protection System (PPS) or Engineered Safety Feature (ESF) component system failure has been made extremely low through redundancy, hardware qualification, and a rigorous quality assurance program which has been reviewed by the NRC (see CESSAR-DC Section 7.2.1.1.2.5). Large Break LOCAs represent only 6.6% of the CDF and steam line breaks represent 0.5% of the CDF. These events are not major contributors to offsite risk because they tend to be in containment. Therefore only minor benefits in terms of public risk would be expected.The Large Break LOCA (LBLOCA) and steam line break within containment events can be assured through operator action in response to symptoms of precursor leakage (Leak Before Break, LBB). The instrumentation available to detect the leakage includes:

- Acoustic leak monitoring system alarm and trending
- Containment Temperature Level
- Containment Radiation
- Containment Humidity

The capacity of Nuplex 80+ makes possible tracking of leakage within containment and correlation of multiple symptoms. In addition to increased costs and complexity of additional trios and ESF actuation paths, the additional trips could decrease plant availability and increase the potential for equipment challenge (false actuation leading to transients) for a negligible improvement in plant safety. Because of the small public risk associated with the LBBLOCA and the sophistication of the current protection system, This DA will not be further considered.

5.39 SEISMIC CAPABILITY

The System 80+ Plant is designed for a Safe Shutdown Earthquake (SSE) of 0.3g acceleration. The Seismic margins analysis (Section 19.7.5 of CESSAR-DC) addresses the margins associated with the seismic design and demonstrates that the plant High Confidence of Low Probability of Failure (HCLPF) value is 0.6g acceleration. Therefore, there is a 95% confidence that existing equipment has less than a 5% probability of failure at twice the SSE level. To meet this stringent design goal, the containment design and SG support design may be modified. Recent Commission policy decisions state that ALWRs need to only demonstrate a HCLPF of 0.5g. The seismic capability is considered adequate for the System 80+ design and no additional changes are considered.

5.40 FIRE AND FLOOD CAPABILITY

The System 80+ Plant is designed with four quadrants, two in each of two divisions with permanent barriers between the divisions. Also sources of flooding were reduced in the annex building and drains were specifically designed to reduce flooding potential. These design features are described in Sections 9.5 (Fire Protection) and 3.4 (Flood Design) of CESSAR-DC. This capability is considered adequate for the System 80+ design and no additional changes are considered for fire and flood.

TABLE 5-1
(SHEET 1 OF 2)

DESIGN ALTERNATIVES CONSIDERED

	DESIGN ALTERNATIVE	CATEGORY*
1.	LARGER PRESSURIZER	1
2.	LARGER STEAM GENERATORS	1
3.	HIGH-PRESSURE SHUTDOWN COOLING SYSTEM (SCS)	1
4.	FUNCTIONALLY INTERCHANGEABLE SCS AND CONTAINMENT SPRAY SYSTEM (CSS) PUMPS	1
5.	MULTIPLE INDEPENDENT CONNECTIONS TO THE GRID	1
6.	TURBINE-GENERATOR RUNBACK CAPABILITY	1
7.	DEDICATED STARTUP FEEDWATER SYSTEM	1
8.	IMPROVED CONTROL ROOM DESIGN	1
9.	IMPROVED NORMALLY OPERATING COMPONENT COOLING WATER SYSTEM (CCWS)/STATION SERVICE WATER SYSTEM (SSWS)	1
10.	FOUR TRAIN SAFETY INJECTION SYSTEM (SIS) WITH DIRECT VESSEL INJECTION	1
11.	SAFETY DEPRESSURIZATION SYSTEM (SDS)	1
12.	FOUR TRAIN EMERGENCY FEEDWATER SYSTEM	1
13.	TWO EMERGENCY DIESEL GENERATORS AND A STANDBY ALTERNATE AC SOURCE (COMBUSTION TURBINE)	1
14.	SIX VITAL BATTERIES	1
15.	IN-CONTAINMENT REFUELING WATER STORAGE TANK (IRWST)	1
16.	CROSS-CONNECTED CSS AND SCS TRAINS	1
17.	IMPROVED CONTROL ROOM DESIGN	1
18.	LARGE SPHERICAL CONTAINMENT	1
19.	REACTOR CAVITY DESIGNED FOR CORIUM DISENTRAINMENT	1
20.	REACTOR CAVITY DESIGNED FOR DEBRIS COOLABILITY	1
21.	IRWST AND SDS INTERCONNECTED	1
22.	HYDROGEN MITIGATION SYSTEM	1
23.	ALTERNATIVE CONTAINMENT SPRAY	2
24.	FILTERED VENT	2
25.	ALTERNATIVE DC BATTERIES AND EFWS	2
26.	RCP SEAL COOLING	1
27.	ALTERNATIVE PRESSURIZER AUXILIARY SPRAY	2
28.	ALTERNATIVE ATWS PRESSURE RELIEF VALVES	2
29.	ALTERNATIVE CONCRETE COMPOSITION	2
30.	REACTOR VESSEL EXTERIOR COOLING	2
31.	ALTERNATIVE H2 IGNITORS	2
32.	ALTERNATIVE HIGH PRESSURE SAFETY INJECTION	2
33.	ALTERNATIVE RCS DEPRESSURIZATION	2
34.	100% SG INSPECTION	2
35.	MSSV SCRUBBING	2
36.	THIRD DIESEL GENERATOR	2
		2

TABLE 5-1
(SHEET 2 OF 2)

DESIGN ALTERNATIVES CONSIDERED

	DESIGN ALTERNATIVE	CATEGORY*
38.	BORON INJECTION SYSTEM (ATWS)	2
39.	DIVERSE PPS	2
40.	ALTERNATIVE CONTAINMENT MONITORING SYSTEM VALVES	2
41.	ALTERNATIVE CAVITY COOLING	2
42.	12 HOUR BATTERIES	2
43.	TORNADO PROTECTION FOR COMBUSTION TURBINE	2
44.	DIESEL SI PUMPS (2)	2
45.	ALTERNATIVE STARTUP FEEDWATER SYSTEM	2
46.	VACUUM BUILDING	3
47.	RIBBED CONTAINMENT LINER	3
48.	EXTENDED RWST SOURCE	2
49.	N-16 MONITOR	1
50.	INCREASE SECONDARY SIDE PRESSURE	3
51.	PASSIVE SECONDARY SIDE COOLERS	3
52	VENTING MSSV TO CONTAINMENT	3
53.	SECONDARY SIDE GUARD PIPES	2
54.	PASSIVE AUTOCATALYTIC RECOMBINERS (PARS)	2
55.	HYDROGEN PURGE LINE	1
56.	FUEL CELLS	2
57.	HOOKUP FOR PORTABLE GENERATOR	2
58.	WATER COOLED RUBBLE BED	2
59.	REFRACTORY LINED CRUCIBLE	2
60.	AUTOMATIC OVERPRESSURE PROTECTION	3
61.	DIGITAL LBLOCA PROTECTION	3
61.	SEISMIC CAPABILITY	3
63.	FIRE AND FLOOD CAPABILITY	3

* Category: 1 Modification is applicable to the System 80+ and already incorporated in the design. No further evaluation is needed.

2 Modification was quantified in this report and not included in the System 80+.

3 Modification was not quantified because of high costs or small benefits.

Table 5-2

DESIGN ALTERNATIVES EVALUATED

NUMBER	DESIGN ALTERNATIVE
DA5.1	ALTERNATIVE CONTAINMENT SPRAY
DA5.2	FILTERED VENT (CONTAINMENT)
DA5.3	ALTERNATIVE DC BATTERIES AND EFWS
DA5.4	RCP SEAL COOLING
DA5.5	ALTERNATIVE PRESSURIZER AUXILIARY SPRAY
DA5.6	ALTERNATIVE ATWS PRESSURE RELIEF VALVES
DA5.7	ALTERNATIVE CONCRETE COMPOSITION
DA5.8	REACTOR VESSEL EXTERIOR COOLING
DA5.9	ALTERNATIVE H2 IGNITORS
DA5.10	ALTERNATIVE HIGH PRESSURE SAFETY INJECTION
DA5.11	ALTERNATIVE RCS DEPRESSURIZATION
DA5.12	100% SG INSPECTION
DA5.13	MSSV AND ADV SCRUBBING
DA5.14	THIRD DIESEL GENERATOR
DA5.15	ATWS INJECTION SYSTEM
DA5.16	DIVERSE PPS
DA5.17	ALTERNATIVE CONTAINMENT MONITORING SYSTEM
DA5.18	CAVITY COOLING
DA5.19	12 HOUR BATTERIES
DA5.20	TORNADO PROTECTION FOR COMBUSTION TURBINE
DA5.21	DIESEL SI PUMPS (2)
DA5.22	ALTERNATIVE STARTUP FEEDWATER SYSTEM
DA5.23	EXTENDED RWST SOURCE
DA5.24	N-16 MONITOR
DA5.25	INCREASE SECONDARY SIDE PRESSURE
DA5.26	PASSIVE SECONDARY SIDE COOLERS
DA5.27	VENTING MSSV TO CONTAINMENT
DA5.28	SECONDARY SIDE GUARD PIPES
DA5.29	PASSIVE AUTOCATALYTIC RECOMBINERS (PARS)
DA5.30	HYDROGEN PURGE LINE
DA5.31	FUEL CELLS
DA5.32	HOOKUP FOR PORTABLE GENERATOR
DA5.33	WATER COOLED RUBBLE BED
DA5.34	REFRACTORY LINED CRUCIBLE
DA5.35	AUTOMATIC OPERPRESSURE PROTECTION
DA5.36	VACUUM BUILDING
DA5.37	RIBBED CONTAINMENT LINER
DA5.38	DIGITAL LBLOCA PROTECTION
DA5.39	SEISMIC CAPABILITY
DA5.40	FIRE AND FLOOD CAPABILITY

Table 5-3

RISK REDUCTION EVALUATION FOR
ALTERNATIVE CONTAINMENT SPRAY

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.05	$0.03
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	1.00	$0.53
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.36	$0.07
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.78	$1.31
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.78	$2.54
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.78	$2.78
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
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SUM	1.93E-06		1.35E-01		$7.27

Table 5-4

RISK REDUCTION EVALUATION FOR
FILTERED VENT (CONTAINMENT)

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	1.00	$0.53
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
SUM	1.93E-06		1.35E-01		$0.53

Table 5-5

RISK REDUCTION EVALUATION FOR
ALTERNATIVE DC BATTERIES AND EFWS

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00
RC2.5M	3.95E-09	4.73E+04	1.87E-04	1.00	$0.19
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00
RC2.7M	1.22E-08	1.38E+05	1.68E-03	1.00	$1.68
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
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SUM	1.93E-06		1.35E-01		$1.87

Table 5-6

RISK REDUCTION EVALUATION FOR
ALTERNATIVE PRESSURIZER AUXILIARY SPRAY

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.21	$0.03
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.36	$1.46
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.36	$1.43
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.95	$29.77
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.97	$32.16
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.78	$25.59
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
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SUM	1.93E-06		1.35E-01		$90.44

Table 5-7

RISK REDUCTION EVALUATION FOR ALTERNATIVE ATWS PRESSURE RELIEF VALVES

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.03	$0.00
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.03	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.03	$0.01
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.03	$0.01
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.03	$0.03
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.03	$0.02
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.03	$0.02
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.03	$0.02
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.03	$0.01
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.03	$0.20
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.03	$0.12
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.03	$0.24
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.03	$0.12
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.03	$0.10
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.03	$0.11
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.03	$0.01
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
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SUM	1.93E-06		1.35E-01		$1.02

Table 5-8

RISK REDUCTION EVALUATION FOR
ALTERNATIVE CONCRETE COMPOSITION

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00
RC2.4E	3.64E-08	2.38E+04	8.66E-04	1.00	$0.87
RC2.5E	2.84E-08	2.35E+04	6.67E-04	1.00	$0.67
RC2.6E	3.45E-08	2.35E+04	8.11E-04	1.00	$0.81
RC2.7E	1.62E-08	2.35E+04	3.81E-04	1.00	$0.38
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00
RC2.5M	3.95E-09	4.73E+04	1.87E-04	1.00	$0.19
RC2.6M	9.08E-09	3.02E+04	2.74E-04	1.00	$0.27
RC2.7M	1.22E-08	1.38E+05	1.68E-03	1.00	$1.68
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
SUM	1.93E-06		1.35E-01		$4.87

Table 5-9

RISK REDUCTION EVALUATION FOR
REACTOR VESSEL EXTERIOR COOLING

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00
RC2.4E	3.64E-08	2.38E+04	8.66E-04	1.00	$0.87
RC2.5E	2.84E-08	2.35E+04	6.67E-04	1.00	$0.67
RC2.6E	3.45E-08	2.35E+04	8.11E-04	1.00	$0.81
RC2.7E	1.62E-08	2.35E+04	3.81E-04	1.00	$0.38
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00
RC2.6M	9.08E-09	3.02E+04	2.74E-04	1.00	$0.27
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00
RC3.1E	6.58E-09	1.02E+06	6.71E-03	1.00	$6.71
RC3.2E	3.08E-09	1.32E+06	4.07E-03	1.00	$4.07
RC3.4E	6.73E-09	1.20E+06	8.08E-03	1.00	$8.08
RC3.6E	3.12E-09	1.27E+06	3.96E-03	1.00	$3.96
RC3.2M	1.80E-09	1.81E+06	3.26E-03	1.00	$3.26
RC3.6M	1.81E-09	1.97E+06	3.57E-03	1.00	$3.57
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
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SUM	1.93E-06		1.35E-01		$32.64

Table 5-10

RISK REDUCTION EVALUATION FOR
ALTERNATIVE H2 IGNITERS

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	1.00	$0.47
RC2.2E	2.04E-09	1.37E+05	2.79E-04	1.00	$0.28
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
SUM	1.93E-06		1.35E-01		$0.75

Table 5-11

RISK REDUCTION EVALUATION FOR
ALTERNATIVE HIGH PRESSURE SAFETY INJECTION

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.55	$0.09
RC1.1M	3.81E-07	1.09E+02	4.15E-05	1.00	$0.04
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.16	$0.08
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.38	$0.11
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.48	$0.42
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.48	$0.39
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	1.00	$0.53
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00
RC2.6M	9.08E-09	3.02E+04	2.74E-04	1.00	$0.27
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.87	$1.46
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.32	$2.15
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.38	$1.54
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.42	$3.39
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.38	$1.51
RC3.2M	1.80E-09	1.81E+06	3.26E-03	1.00	$3.26
RC3.6M	1.81E-09	1.97E+06	3.57E-03	1.00	$3.57
RC4.4E	5.98E-09	5.24E+06	3.13E-02	1.00	$31.34
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.42	$0.09
RC4.12E	6.54E-09	5.07E+06	3.32E-02	1.00	$33.16
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
SUM	1.93E-06		1.35E-01		$83.38

Table 5-12

RISK REDUCTION EVALUATION FOR
ALTERNATIVE RCS DEPRESSURIZATION

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.36	$0.06
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.69	$0.33
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.75	$0.21
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.42	$0.36
RC2.5E	2.84E-08	2.35E+04	6.67E-04	1.00	$0.67
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.52	$0.42
RC2.7E	1.62E-08	2.35E+04	3.81E-04	1.00	$0.38
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.56	$3.76
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.62	$2.52
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.48	$3.88
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.62	$2.46
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.48	$0.10
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
SUM	1.93E-06		1.35E-01		$15.14

Table 5-13

RISK REDUCTION EVALUATION FOR
100% SG INSPECTION

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.21	$0.03
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.38	$1.54
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.38	$1.51
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00
RC4.4E	5.98E-09	5.24E+06	3.13E-02	1.00	$31.34
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00
RC4.12E	6.54E-09	5.07E+06	3.32E-02	1.00	$33.16
RC4.18L	5.56E-09	5.90E+06	3.28E-02	1.00	$32.80
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
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SUM	1.93E-06		1.35E-01		$100.38

Table 5-14

RISK REDUCTION EVALUATION FOR
MSSV AND ADV SCRUBBING

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00
RC4.4E	5.98E-09	5.24E+06	3.13E-02	1.00	$31.34
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00
RC4.12E	6.54E-09	5.07E+06	3.32E-02	1.00	$33.16
RC4.18L	5.56E-09	5.90E+06	3.28E-02	1.00	$32.80
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
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SUM	1.93E-06		1.35E-01		$97.30

Table 5-15

RISK REDUCTION EVALUATION FOR
THIRD DIESEL GENERATOR

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.24	$0.04
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.24	$0.40
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
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SUM	1.93E-06		1.35E-01		$0.45

Table 5-16

RISK REDUCTION EVALUATION FOR
ALTERNATIVE CONTAINMENT MONITORING SYSTEM

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00
RC4.8E	1.12E-09	1.86E+05	2.08E-04	1.00	$0.21
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00
RC5.1E	5.10E-10	2.87E+06	1.46E-03	1.00	$1.46
SUM	1.93E-06		1.35E-01		$1.67

Table 5-17

RISK REDUCTION EVALUATION FOR
12-HOUR BATTERIES

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.38	$0.07
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.38	$0.64
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
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SUM	**1.93E-06**		**1.35E-01**		**$0.71**

Table 5-18

RISK REDUCTION EVALUATION FOR
TORNADO-PROTECTION FOR COMBUSTION TURBINE

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.43	$0.08
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.90	$1.52
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
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SUM	1.93E-06		1.35E-01		$1.60

Table 5-19

RISK REDUCTION EVALUATION FOR
DIESEL SI PUMPS (2)

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.55	$0.09
RC1.1M	3.81E-07	1.09E+02	4.15E-05	1.00	$0.04
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.16	$0.08
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.38	$0.11
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.48	$0.42
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.48	$0.39
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	1.00	$0.53
RC2.5M	3.95E-09	4.73E+04	1.87E-04	1.00	$0.19
RC2.6M	9.08E-09	3.02E+04	2.74E-04	1.00	$0.27
RC2.7M	1.22E-08	1.38E+05	1.68E-03	1.00	$1.68
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.32	$2.15
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.38	$1.54
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.42	$3.39
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.38	$1.51
RC3.2M	1.80E-09	1.81E+06	3.26E-03	1.00	$3.26
RC3.6M	1.81E-09	1.97E+06	3.57E-03	1.00	$3.57
RC4.4E	5.98E-09	5.24E+06	3.13E-02	1.00	$31.34
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.42	$0.09
RC4.12E	6.54E-09	5.07E+06	3.32E-02	1.00	$33.16
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
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SUM	1.93E-06		1.35E-01		$83.79

Table 5-20

RISK REDUCTION EVALUATION FOR
ALTERNATIVE STARTUP FEEDWATER SYSTEM

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.36	$0.06
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.69	$0.33
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.75	$0.21
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.42	$0.36
RC2.5E	2.84E-08	2.35E+04	6.67E-04	1.00	$0.67
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.52	$0.42
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.56	$0.94
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.62	$4.16
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.48	$1.95
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.62	$5.01
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.75	$2.97
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.75	$2.44
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.48	$15.04
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
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SUM	1.93E-06		1.35E-01		$34.57

Table 5-21

RISK REDUCTION EVALUATION FOR
EXTENDED RWST SOURCE

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00
RC4.18L	5.56E-09	5.90E+06	3.28E-02	1.00	$32.8
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00
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SUM	1.93E-06		1.35E-01		$32.80

Table 5-22

RISK REDUCTION EVALUATION FOR SECONDARY SIDE GUARD PIPES

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	Savings $/y
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.001	$0.00
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.002	$0.00
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.002	$0.00
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.001	$0.00
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.001	$0.00
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.001	$0.00
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.001	$0.00
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.50	$0.73
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SUM	1.93E-06		1.35E-01		$0.73

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12 Memorandum from W. J. Dircks to the Commission, "Bases for Quantification of Offsite Costs," October 23, 1985.

13 "Quantification of Passive Autocatalytic Recombiners for Combustible Gas Control in ALWR Containments," EPTR ALWR Program, April 8, 1993.

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18. "Survey of the Art in Mitigation Systems," NUREG/CR-3908, December, 1985.

ATTACHMENT B

THE INCLUSION OF AVERTED ONSITE COSTS IN THE EVALUATION OF DESIGN ALTERNATIVES FOR THE SYSTEM 80+ NUCLEAR POWER PLANT

SEPTEMBER 23, 1993

PURPOSE

An evaluation of Design Alternatives (DAs) for the System 80+ design was issued[1] which was based on the reduction of health and economic risk to the offsite population. The purpose of this analysis is to evaluate the same design alternatives but include credit for Averted Onsite Costs (AOC).

SUMMARY

Section 4 of Reference 1 describes the Release Classes (RCs) and the accident sequences that were binned into each RC. To evaluate the risk reduction of each DA, the frequency of each RC was decreased proportionally to the contribution that each DA makes to the RC frequency. The AOC benefit was estimated as the product of the change in core damage frequency (CDF) times the total onsite cost for loosing the plant.

Table 1 summarizes the results of the Design Alternative quantification including AOC. The first column, is the annual risk reduction to the Combined License (CL) applicant for each DA for both AOC and dose risk to the general population using $1000 per person-rem/year reduction. The next column, labeled capital benefit, is an equivalent present worth of the annual risk reduction. It is also the maximum amount that could be spent in capital to be cost beneficial. The third column is a 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 ALWR Utility Requirements Document. The System 80+ design has a core damage frequency approximately two orders of magnitude lower than existing plants. The analysis presented in this report conservatively estimated the benefits of the DAs by assuming that they would work perfectly to eliminate the type of accident they are designed to address and would require no maintenance or testing. Because of the small initial risk associated with the System 80+ design, none of the DAs are cost beneficial.

ANALYSIS

For plant modifications that reduced the Core Damage Frequency (CDF), the annual benefit was increased by an amount proportioned to the present worth of the reduction in risk of Averted Onsite Costs (AOC) and dose reduction to the general public. Modifications that reduced the probability of containment failure, or reduced the amount of fission products leaving the site were

assumed to have no significant AOC reduction.

AOC included replacement power costs, direct accident costs (including cleanup), and the economic loss of the plant. Credit is given for property and replacement-power insurance. Evaluation of the AOC includes the following considerations:

1) The replacement power costs used ($386,000/day) is a replacement power cost for the Palo Verde Reactor (a System 80 plant) averaged for 1993 as predicted by ANL[4]. This cost is applied for a three year period because it is assumed that the utility will contract with an Independent Power Producer (IPP) during that period for power at a comparable cost as that incurred in the nuclear plant. Currently, IPPs can build new facilities in 12 months and IPP rates are very competitive. Therefore a three year replacement power period is a reasonable assumption. Replacement power costs are estimated at $423 Million (M) but will be partially offset by replacement power insurance of $365M[2].

2) Direct accident costs, including cleanup costs were assumed to be $2 Billion (B). This is partially offset by the primary and excess nuclear-property insurance of $1.625B[2,3]. Most new, large plants and publicly owned plants carry the maximum amount of coverage. The NRC requires the plant owners to carry over $1B.

3) The economic value of the facility at the time of the accident was calculated assuming that the initial plant invested cost was $1.4B based on DOE cost guidelines. It is also assumed that a straight line depreciation value is used over a twenty year period and the accident is equally probable during any year in the plants sixty year life. The economic value of the plant averages $233M and is assumed lost. The inclusion of both the value of the plant and its output (replacement power) is conservatively exaggerate the size of the AOC.

The total AOC is estimated at $666M. This figure neglects credit for premature decommissioning insurance or elimination of annual capital expenditures. Such credits would further reduce the AOC.

The maximum value for a capital expense for which AOC avoidance is cost beneficial can now be calculated. The core damage frequency (CDF) is approximately 1.93E-6/y. If an unspecified modification completely eliminated core damage, it would be worth (1.93E-6/y x $666M) or $1.29K/y in AOC avoidance. Using the economic assumptions given in Table 3-1 of Reference 1, a levelized capital cost rate of 16.6% is predicted. A capital expense of $7,770 would be justified for AOC avoidance if it completely eliminated any

chance of core damage and had no annual maintenance, testing, or training costs.

Section 5 of Reference 1 gives an analysis of the dose risk reduction for each DA. Tables 2 through 14 presents the annual risk reduction for thirteen of the DAs that reduce Core Damage Frequency (CDF) and have an AOC benefit. The ATWS Injection DA and the Diverse PPS DA were not evaluated because they have the same risk reduction benefits as the ATWS Pressure Relief Valves (Table 4). Also the fuel cell DA and the portable generator DA were not evaluated because they have the same risk reduction benefits as the alternative DC battery and AFWS (Table 2).

REFERENCES

1. "Design Alternatives for the System 80+ Nuclear Power Plant (Rev. 2), ABB Combustion Engineering, Inc., September, 1993.

2. "Nuclear Insurance Newsletter," Johnson & Higgins Inc., January, 1990 (90-1).

3. "Nuclear Insurance Newsletter," Johnson & Higgins Inc., July, 1990 (90-2).

4. Nucleonics Week, December 3, 1992, Page 13.

TABLE 1
(Sheet 1 of 2)

SUMMARY OF THE RISK REDUCTIONS (INCLUDING AOC) OF THE DESIGN ALTERNATIVES

	DESIGN ALTERNATIVE	ANNUAL RISK REDUCTION $/Y	CAPITAL BENEFIT*	CAPITAL COST	NET CAPITAL BENEFIT
5.1	ALT. CONTAINMENT SPRAY	$7.27**	$44	$1,500,000	($1,499,956)
5.2	FILTERED VENT (CONTAINMENT)	$0.53**	$3	$10,000,000	($9,999,997)
5.3	ALT. DC BATTERY AND EFWS	$12.63	$76	$2,000,000	($1,999,924)
5.5	ALT. PRESSURIZER AUX SPRAY	$293	$1765	$5,000,000	($4,998,235)
5.6	ALT. ATWS RELIEF VALVES	$38.64	$233	$1,000,000	($999,767)
5.7	ALT. CONCRETE COMPOSITION	$4.87**	$29	$5,000,000	($4,999,971)
5.8	RV EXTERIOR COOLING	$32.64**	$197	$2,500,000	($2,499,803)
5.9	ALT. H2 IGNITERS	$0.75**	$5	$1,000,000	($999,995)
5.10	ALT. HPSI	$890.57	$5365	$2,200,000	($2,294,635)
5.11	ALT. RCS DEPRESSURIZATION	$403.18	$2429	$500,000	($497,571)
5.12	100% SG INSPECTION	$304.20	$1833	$1,500,000	($1,498,167)
5.13	MSSV AND ADV SCRUBBING	$97.30**	$586	$9,500,000	($9,499,414)
5.14	THIRD DIESEL GENERATOR	$3.03	$18	$25,000,000	($9,999,982)
5.15	ATWS INJECTION SYSTEM	$38.64	$233	$300,000	($299,767)
5.16	DIVERSE PPS SYSTEM	$38.64	$233	$3,000,000	($2,999,767)
5.17	ALT. CONTAINMENT MONITORING SYSTEM	$2.76	$17	$1,000,000	($999,983)

TABLE 1
(Sheet 2 of 2)

SUMMARY OF THE RISK REDUCTIONS (INCLUDING AOC) OF THE DESIGN ALTERNATIVES

	DESIGN ALTERNATIVE	ANNUAL RISK REDUCTION $/Y	CAPITAL BENEFIT*	CAPITAL COST	NET CAPITAL BENEFIT
5.18	CAVITY COOLING	$32.64**	$197	$50,000	($49,803)
5.19	12-HOUR BATTERIES	$4.80	$27	$300,000	($299,973)
5.20	TORNADO-PROTECTION FOR COMBUSTION TURBINE	$10.04	$60	$3,000,000	($2,999,940)
5.21	DIESEL SI PUMPS (2)	$894.66	$5390	$2,000,000	($1,994,610)
5.23	EXTENDED RWST SOURCE	$36.50	$220	$1,000,000	($999,780)
5.28	SECONDARY SIDE GUARD PIPES	$1.81	$11	$820,000	($819,989)
5.29	PASSIVE AUTOCATALYTIC RECOMBINERS (PARS)	$0.75**	$5	$760,000	($759,995)
5.31	FUEL CELLS	$12.63	$76	$2,000,000	($1,999,924)
5.32	HOOKUP FOR PORTABLE GENERATR	$12.63	$76	$10,000	($9,924)
5.33	WATER COOLED RUBBLE BED	$4.87**	$29	$18,800,000	($18,799,971)
5.34	REFRACTORY LINED CRUCIBLE	$4.87**	$29	$108,000,000	($107,999,971)

* THE CAPITAL BENEFIT IS THE PRICE OF A PIECE OF EQUIPMENT THAT HAS A LEVELIZED (ANNUAL) COST EQUAL TO THE ANNUAL BENEFIT IN RISK REDUCTION AND ASSUMES NO MAINTENANCE OR TESTING OF ADDITIONAL EQUIPMENT.

** NO AOC WAS CREDITED TO DOSE MITIGATION DESIGN ALTERNATIVES THAT DOES NOT REDUCE CDF.

Table 2

RISK REDUCTION EVALUATION (INCLUDING AOC) FOR ALTERNATIVE DC BATTERIES AND EFWS

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	MR Risk Savings $/y	CDF Reduction
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00	0
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00	0
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00	0
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00	0
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00	0
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00	0
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00	0
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00	0
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00	0
RC2.5M	3.95E-09	4.73E+04	1.87E-04	1.00	$0.19	3.950E-09
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00	0
RC2.7M	1.22E-08	1.38E+05	1.68E-03	1.00	$1.68	1.220E-08
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00	0
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00	0
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00	0
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00	0
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00	0
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00	0
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00	0
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00	0
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00	0
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00	0
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00	0
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SUM	1.93E-06		1.35E-01		$1.87	1.615E-08

AOC ($)	6.66E+08
AOC RISK REDUCTION	$10.76
MR RISK REDUCTION	$1.87

TOTAL RISK REDUCTION	$12.63

Table 3

RISK REDUCTION EVALUATION (INCLUDING AOC) FOR ALTERNATIVE PRESSURIZER AUXILIARY SPRAY

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	MR Risk Savings $/y	CDF Reduction
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.21	$0.03	2.86E-07
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00	0
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00	0
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00	0
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00	0
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00	0
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00	0
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00	0
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00	0
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00	0
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00	0
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00	0
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00	0
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.36	$1.46	1.109E-09
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00	0
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.36	$1.43	1.123E-09
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00	0
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00	0
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.95	$29.77	5.681E-09
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00	0
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.97	$32.16	6.344E-09
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.78	$25.59	4.337E-09
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00	0
SUM	1.93E-06		1.35E-01		$90.44	3.04E-07

AOC ($)	6.66E+08
AOC RISK REDUCTION	$202.59
MR RISK REDUCTION	$90.44
TOTAL RISK REDUCTION	$293.04

Table 4

RISK REDUCTION EVALUATION (INCLUDING AOC) FOR ALTERNATIVE ATWS PRESSURE RELIEF VALVES

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	MR Risk Savings $/y	CDF Reduction
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.03	$0.00	4.080E-08
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.03	$0.00	1.143E-08
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.03	$0.01	1.038E-10
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.03	$0.01	6.120E-11
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.03	$0.03	1.092E-09
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.03	$0.02	8.520E-10
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.03	$0.02	1.035E-09
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00	0
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.03	$0.02	1.215E-10
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00	0
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.03	$0.01	2.724E-10
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00	0
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.03	$0.20	1.974E-10
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.03	$0.12	9.240E-11
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.03	$0.24	2.019E-10
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.03	$0.12	9.360E-11
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.03	$0.10	5.400E-11
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.03	$0.11	5.430E-11
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00	0
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.03	$0.01	3.360E-11
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00	0
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00	0
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00	0
SUM	1.93E-06		1.35E-01		$1.02	5.650E-08

AOC ($)	6.66E+08
AOC RISK REDUCTION	$37.63
MR RISK REDUCTION	$1.02
TOTAL RISK REDUCTION	$38.64

Table 5

RISK REDUCTION EVALUATION (INCLUDING AOC) FOR ALTERNATIVE HIGH PRESSURE SAFETY INJECTION

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	MR Risk Savings $/y	CDF Reduction
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.55	$0.09	7.48E-07
RC1.1M	3.81E-07	1.09E+02	4.15E-05	1.00	$0.04	3.81E-07
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.16	$0.08	5.536E-10
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.38	$0.11	7.752E-10
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.48	$0.42	1.747E-08
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00	0
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.48	$0.39	1.656E-08
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00	0
RC2.2M	4.05E-09	1.31E+05	5.31E-04	1.00	$0.53	4.050E-09
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00	0
RC2.6M	9.08E-09	3.02E+04	2.74E-04	1.00	$0.27	9.080E-09
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.87	$1.46	1.061E-08
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.32	$2.15	2.106E-09
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.38	$1.54	1.170E-09
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.42	$3.39	2.827E-09
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.38	$1.51	1.186E-09
RC3.2M	1.80E-09	1.81E+06	3.26E-03	1.00	$3.26	1.800E-09
RC3.6M	1.81E-09	1.97E+06	3.57E-03	1.00	$3.57	1.810E-09
RC4.4E	5.98E-09	5.24E+06	3.13E-02	1.00	$31.34	5.980E-09
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.42	$0.09	4.704E-10
RC4.12E	6.54E-09	5.07E+06	3.32E-02	1.00	$33.16	6.540E-09
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00	0
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00	0
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SUM	1.93E-06		1.35E-01		$83.38	1.21E-06

AOC ($)	6.66E+08
AOC RISK REDUCTION	$807.19
MR RISK REDUCTION	$83.38

TOTAL RISK REDUCTION	$890.57

Table 6

RISK REDUCTION EVALUATION (INCLUDING AOC) FOR
ALTERNATIVE RCS DESPRESSURIZATION

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	MR Risk Savings $/y	CDF Reduction
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.36	$0.06	4.90E-07
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00	0
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.69	$0.33	2.387E-09
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.75	$0.21	1.530E-09
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.42	$0.36	1.529E-08
RC2.5E	2.84E-08	2.35E+04	6.67E-04	1.00	$0.67	2.840E-08
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.52	$0.42	1.794E-08
RC2.7E	1.62E-08	2.35E+04	3.81E-04	1.00	$0.38	1.620E-08
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00	0
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00	0
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00	0
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00	0
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.56	$3.76	3.685E-09
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.62	$2.52	1.910E-09
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.48	$3.88	3.230E-09
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.62	$2.46	1.934E-09
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00	0
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00	0
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00	0
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.48	$0.10	5.376E-10
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00	0
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00	0
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00	0
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SUM	1.93E-06		1.35E-01		$15.14	5.83E-7

AOC ($)	6.66E+08
AOC RISK REDUCTION	$388.04
MR RISK REDUCTION	$15.14

TOTAL RISK REDUCTION	$403.18

Table 7

RISK REDUCTION EVALUATION (INCLUDING AOC) FOR
100% SG INSPECTION

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	MR Risk Savings $/y	CDF Reduction
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.21	$0.03	2.86E-07
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00	0
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00	0
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00	0
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00	0
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00	0
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00	0
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00	0
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00	0
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00	0
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00	0
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00	0
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00	0
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.38	$1.54	1.170E-09
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00	0
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.38	$1.51	1.186E-09
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00	0
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00	0
RC4.4E	5.98E-09	5.24E+06	3.13E-02	1.00	$31.34	5.980E-09
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00	0
RC4.12E	6.54E-09	5.07E+06	3.32E-02	1.00	$33.16	6.540E-09
RC4.18L	5.56E-09	5.90E+06	3.28E-02	1.00	$32.80	5.560E-09
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00	0
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SUM	1.93E-06		1.35E-01		$100.38	3.06E-07

	6.66E+08
AOC ($)	$203.82
AOC RISK REDUCTION	$100.38
MR RISK REDUCTION	----------
TOTAL RISK REDUCTION	$304.20

Table 8

RISK REDUCTION EVALUATION (INCLUDING AOC) FOR THIRD DIESEL GENERATOR

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	MR Risk Savings $/y	CDF Reduction
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00	0
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00	0
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00	0
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00	0
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00	0
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00	0
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00	0
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00	0
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00	0
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.24	$0.04	9.480E-10
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00	0
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.24	$0.40	2.928E-09
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00	0
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00	0
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00	0
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00	0
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00	0
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00	0
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00	0
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00	0
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00	0
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00	0
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00	0
SUM	1.93E-06		1.35E-01		$0.45	3.88E-09

AOC ($)	6.66E+08
AOC RISK REDUCTION	$2.58
MR RISK REDUCTION	$0.45
TOTAL RISK REDUCTION	$3.03

Table 9

RISK REDUCTION EVALUATION (INCLUDING AOC) FOR ALTERNATIVE CONTAINMENT MONITORING SYSTEM

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	MR Risk Savings $/y	CDF Reduction
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00	0
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00	0
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00	0
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00	0
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00	0
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00	0
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00	0
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00	0
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00	0
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00	0
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00	0
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00	0
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00	0
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00	0
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00	0
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00	0
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00	0
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00	0
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00	0
RC4.8E	1.12E-09	1.86E+05	2.08E-04	1.00	$0.21	1.120E-09
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00	0
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00	0
RC5.1E	5.10E-10	2.87E+06	1.46E-03	1.00	$1.46	5.100E-10
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SUM	1.93E-06		1.35E-01		$1.67	1.630E-09

AOC ($)	6.66E+08
AOC RISK REDUCTION	$1.09
MR RISK REDUCTION	$1.67

TOTAL RISK REDUCTION	$2.76

Table 10

RISK REDUCTION EVALUATION (INCLUDING AOC) FOR 12-HOUR BATTERIES

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	MR Risk Savings $/y	CDF Reduction
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00	0
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00	0
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00	0
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00	0
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00	0
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00	0
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00	0
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00	0
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00	0
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.38	$0.07	1.501E-09
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00	0
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.38	$0.64	4.636E-09
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00	0
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00	0
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00	0
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00	0
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00	0
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00	0
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00	0
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00	0
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00	0
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00	0
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00	0
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SUM	1.93E-06		1.35E-01		$0.71	6.137E-09

AOC ($)	6.66E+08
AOC RISK REDUCTION	$4.09
MR RISK REDUCTION	$0.71

TOTAL RISK REDUCTION	$4.80

Table 11

RISK REDUCTION EVALUATION (INCLUDING AOC) FOR TORNADO-PROTECTION FOR COMBUSTION TURBINE

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	MR Risk Savings $/y	CDF Reduction
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00	0
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00	0
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00	0
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00	0
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00	0
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00	0
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00	0
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00	0
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00	0
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.43	$0.08	1.699E-09
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00	0
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.90	$1.52	1.098E-08
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00	0
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00	0
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00	0
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00	0
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00	0
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00	0
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00	0
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00	0
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00	0
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00	0
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00	0
SUM	1.93E-06		1.35E-01		$1.60	1.268E-08

AOC ($)	6.66E+08
AOC RISK REDUCTION	$8.44
MR RISK REDUCTION	$1.60
TOTAL RISK REDUCTION	$10.04

Table 12

RISK REDUCTION EVALUATION (INCLUDING AOC) FOR DIESEL SI PUMPS (2)

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	MR Risk Savings $/y	CDF Reduction
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.55	$0.09	7.48E-07
RC1.1M	3.81E-07	1.09E+02	4.15E-05	1.00	$0.04	3.81E-07
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.16	$0.08	5.536E-10
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.38	$0.11	7.752E-10
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.48	$0.42	1.747E-08
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00	0
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.48	$0.39	1.656E-08
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00	0
RC2.2M	4.05E-09	1.31E+05	5.31E-04	1.00	$0.53	4.050E-09
RC2.5M	3.95E-09	4.73E+04	1.87E-04	1.00	$0.19	3.950E-09
RC2.6M	9.08E-09	3.02E+04	2.74E-04	1.00	$0.27	9.080E-09
RC2.7M	1.22E-08	1.38E+05	1.68E-03	1.00	$1.68	1.220E-08
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.32	$2.15	2.106E-09
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.38	$1.54	1.170E-09
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.42	$3.39	2.827E-09
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.38	$1.51	1.186E-09
RC3.2M	1.80E-09	1.81E+06	3.26E-03	1.00	$3.26	1.800E-09
RC3.6M	1.81E-09	1.97E+06	3.57E-03	1.00	$3.57	1.810E-09
RC4.4E	5.98E-09	5.24E+06	3.13E-02	1.00	$31.34	5.980E-09
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.42	$0.09	4.704E-10
RC4.12E	6.54E-09	5.07E+06	3.32E-02	1.00	$33.16	6.540E-09
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00	0
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00	0
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SUM	1.93E-06		1.35E-01		$83.79	1.22E-06

AOC ($)	6.66E+08
AOC RISK REDUCTION	$810.87
MR RISK REDUCTION	$83.79

TOTAL RISK REDUCTION	$894.66

Table 13

RISK REDUCTION EVALUATION (INCLUDING AOC) FOR
EXTENDED RWST SOURCE

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	MR Risk Savings $/y	CDF Reduction
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.00	$0.00	0
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00	0
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.00	$0.00	0
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.00	$0.00	0
RC2.4E	3.64E-08	2.38E+04	5.66E-04	0.00	$0.00	0
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00	0
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00	0
RC2.7E	1.62E-08	2.[illegible]5E+04	3.81E-04	0.00	$0.00	0
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00	0
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00	0
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00	0
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00	0
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.00	$0.00	0
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.00	$0.00	0
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.00	$0.00	0
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.00	$0.00	0
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00	0
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00	0
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00	0
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00	0
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00	0
RC4.18L	5.56E-09	5.90E+06	3.28E-02	1.00	$32.80	5.56E-09
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.00	$0.00	0
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SUM	1.93E-06		1.35E-01		$32.80	5.56E-09

AOC ($)	6.66E+08
AOC RISK REDUCTION	$ 3.70
MR RISK REDUCTION	$32.80

TOTAL RISK REDUCTION	$ 36.50

Table 14

RISK REDUCTION EVALUATION (INCLUDING AOC) FOR
SECONDARY SIDE GUARD PIPES

Release Class	Frequency Events/y	Mean Dose mr/event	Dose Risk mr/y	Benefit fract. reduct.	MR Risk Savings $/y	CDF Reduction
RC1.1E	1.36E-06	1.19E+02	1.62E-04	0.001	$0.00	1.36E-09
RC1.1M	3.81E-07	1.09E+02	4.15E-05	0.00	$0.00	0
RC2.1E	3.46E-09	1.37E+05	4.74E-04	0.002	$0.00	6.92E-12
RC2.2E	2.04E-09	1.37E+05	2.79E-04	0.002	$0.00	4.08E-12
RC2.4E	3.64E-08	2.38E+04	8.66E-04	0.00	$0.00	0
RC2.5E	2.84E-08	2.35E+04	6.67E-04	0.00	$0.00	0
RC2.6E	3.45E-08	2.35E+04	8.11E-04	0.00	$0.00	0
RC2.7E	1.62E-08	2.35E+04	3.81E-04	0.00	$0.00	0
RC2.2M	4.05E-09	1.31E+05	5.31E-04	0.00	$0.00	0
RC2.5M	3.95E-09	4.73E+04	1.87E-04	0.00	$0.00	0
RC2.6M	9.08E-09	3.02E+04	2.74E-04	0.00	$0.00	0
RC2.7M	1.22E-08	1.38E+05	1.68E-03	0.00	$0.00	0
RC3.1E	6.58E-09	1.02E+06	6.71E-03	0.001	$0.00	6.58E-12
RC3.2E	3.08E-09	1.32E+06	4.07E-03	0.001	$0.00	3.08E-12
RC3.4E	6.73E-09	1.20E+06	8.08E-03	0.001	$0.00	6.73E-12
RC3.6E	3.12E-09	1.27E+06	3.96E-03	0.001	$0.00	3.12E-12
RC3.2M	1.80E-09	1.81E+06	3.26E-03	0.00	$0.00	0
RC3.6M	1.81E-09	1.97E+06	3.57E-03	0.00	$0.00	0
RC4.4E	5.98E-09	5.24E+06	3.13E-02	0.00	$0.00	0
RC4.8E	1.12E-09	1.86E+05	2.08E-04	0.00	$0.00	0
RC4.12E	6.54E-09	5.07E+06	3.32E-02	0.00	$0.00	0
RC4.18L	5.56E-09	5.90E+06	3.28E-02	0.00	$0.00	0
RC5.1E	5.10E-10	2.87E+06	1.46E-03	0.50	$0.73	2.30E-10
SUM	1.93E-06		1.35E-01		$0.73	1.62E-09

AOC ($)	6.66E+08
AOC RISK REDUCTION	$ 1.08
MR RISK REDUCTION	$ 0.73
TOTAL RISK REDUCTION	$ 1.81

ATTACHMENT C

RESPONSE TO RAIS

ON

ABB-CE SAMDA ANALYSIS FOR SYSTEM 80+

(RECEIVED FROM PEST/SPSB, 9/1/93)

SEPTEMBER 23, 1993

1. Design alternatives are said to be based on a similar analysis performed for the Limerick plant. However, no mention was made of whether plant improvements considered as part of the NRC Containment Performance Improvement (CPI) program (e.g., NUREG/CR-5567, -5630, and -5662) or the Comanche Peak SAMDA analysis (NUREG-0775 supplement) were also considered. Please justify that the set of design alternatives considered for System 80+ include all relevant design improvements considered in these other two studies. ABB-CE should also evaluate NUREG/CR-4920 for additional alternatives? In addition, it is not clear whether the evaluation is for a 40-year or 60-year plant life.

Response:

ABB-CE has reviewed the Comanche Peak SAMDA and NUREG/CR-4920. The Design Alternatives (DAs) discussed in other references are contained in the above two references. The analysis was also redone using a 60 year plant life and Table 3-1 in the revised report has been revised.

2. Where available, provide a comparison of CE cost estimates to those for similar design alternatives considered in previous analyses, including the Comanche Peak and Limerick SAMDA analyses, the February 1987 draft of NUREG-1150 and NRC-sponsored work in support of the review of GESSAR (NUREG/CR-3908, -4025, -4242, -4243, -4244).

Response:

Where ever possible, the costs for the DAs are compared with other sources in the revised report.

3. The discussion of the offsite costs for other items such as economic losses, replacement power costs, etc. (SSAR page 19A-7) states they are not considered in this evaluation. ABB-CE should discuss in the SSAR why they are not evaluated here, since no rational was given. Other applicants have included a discussion of averted on-site costs in the SSAR. This approach seems reasonable and if ABB-CE did the same, it would help defend ABB-CE arguments against the modification.

Response:

It was agreed on September 10, 1993 that ABB-CE would continue to use the $1000/mr for offsite costs and to keep the averted onsite cost calculation as a separate document. The $1000/mr

represents both health and offsite economic costs as documented in the following reference: Memorandum from W. J. Dircks to the Commission, "Bases for Quantification of Offsite Costs," October 23, 1985.

4. Describe and justify the population and meteorological data used in the System 80+ analysis, especially in view of the fact that the EPRI requirements document for evolutionary plants has significantly changed in this area and no longer includes such data. Provide a copy of Reference 4.

Response:

The ALWR site was described in the May 1989 version of the KAG and was to represent 80% of the potential sites. The site was an existing site in South Carolina with the population increased. The attached Annex B from the 5/89 version of KAG is a summary of Reference 4 and further describes the site.

5. Describe the release path and point of release (including elevation) for over-pressure containment failures. Justify the use of a 52.8 meter elevation for this release given: (a)use of the hydrogen purge vent, and (b)failure of a penetration, which is the expected mode of failure.

Response:

A sensitivity analysis was performed to investigate the effect of release height on dose at the site boundary (Table 19.14.2-1 of CESSAR-DC). The effect of having all the releases occur at ground level would increase the probability of exceeding 25 Rem at the site boundary by 3%. The effect of having all the releases occur at the top of containment would decrease the probability of exceeding 25 Rem at the site boundary by 14%. These sensitivities are small compared to the difference between the costs and benefits and would not change any results.

6. It appears that the source terms used for the evaluation of design alternatives were based on the MAAP code, whereas the source terms used in the updated Level 2/3 PRA analysis are based on the S80SOR code. The criteria for selecting a representative sequence for each release class was also modified in the updated analysis. Please provide an assessment of the risk reduction for each design alternative using the source terms and sequence selection scheme from the updated Level 2/3 analysis.

Response:

The SAMDA analysis used the latest source terms from the revised PRA (Amendment R to CESSAR-DC) and the report will be changed to clearly reference it including the use of S80SOR.

7. Table 4-2 shows the same PDSs (e.g., PDS 184 and PDS 235) being mapped into more than one RC, without giving the fraction of the PDS frequency assigned to each RC. The sequence frequency information reported in the last column is also not very useful because it represents the total sequence frequency, rather than the frequency each sequence contributes to the particular RC. In this regard, provide a breakdown of RC frequency by PDS (e.g., the frequency contribution from each PDS). Also provide a breakdown of the fraction or frequency each sequence contributes to the PDS contribution.

Response:
Section 4.0 has been expanded to include a better discussion of the mapping of design improvements into release classes. The following revision will be substituted for the last paragraph on page 19a-10.

S80SOR 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.

Table 4-1 presents a brief description for each release class with a frequency greater than or equal to 1.0E-10. This table is used to identify the effect of mitigation equipment (more details of each RC is given in Section 12.3 of Chapter 19). Table 4-2 gives the mapping of each PDS into each release class. Also given in this table are the mapping of the CDF sequences into the PDSs. In addition, the description of each sequence and the sequence CDF is also presented. This table is used to reduce each RC frequency (column 2 of Table 4-2) for preventative DAs.

The sequence CDF (last column of Table 4-2) was used to calculate the risk reduction associated with DAs that prevent core damage. It was assumed that any prevention DA would completely eliminate the sequence that the DA would address. For example, a Safety Injection DA would reduce the RC1.1E by 55%. SIS failure appears in five of the sequences with a total sequence frequency of 7.15E-7. The sum of all the sequences contributing to RC1.1E is 12.89E-7 and therefore the

DA is assumed to reduce this RC by 55% (7.15E-7 / 12.89E-7). Each release class is evaluated in this manner for each prevention DA.

8. SGTR sequences account for over 70% of the total risk from System 80+. Contributors to these sequences include: RHR injection, aggressive secondary cooldown failure, failure to isolate the SG, and failure to refill the IRWST. It is not clear that a systematic search has been made for design alternatives that would serve to reduce these contributors, e.g., improved reliability/automation of IRWST refill, backup of injection using existing equipment such as startup pumps, diesel-driven firewater pumps, or fire trucks. Provide an assessment of additional design improvements specifically oriented towards reducing the observed risk from SGTR.

Response:

The following paragraphs are to be added to Section 5.0, after the third paragraph:

At the beginning of the design process, it was recognized that the steam generator integrity was important to safety and plant economics. The risk of SRTR in the System 80+ design is two orders of magnitude below current plants but SGTR represents over half the off site risk. System 80+ is designed to prevent MSSV actuation following SGTR as described below and also includes new or enhanced features for the prevention of SGTRs.

Features to prevent SGTRs include:

- Steam generator tubes made of thermally treated Inconel 690, which has favorable corrosion resistance properties including superior resistance to primary and secondary stress corrosion cracking

- A deaerator in the condensate/feedwater system for removal of oxygen

- Condensate system with full flow condensate polisher to remove dissolved and suspended impurities

- Main condenser with provisions for early detection of tube leaks, and segmented design permitting repair of leaks while operating at reduced power

- Steam, feedwater and condensate generator blowdown system and SG secondary side recirculation system for chemistry control during wet layup

The response to Unresolved Safety Issue A-4 in CESSAR-DC, Appendix A further describes design features to assure SG tube integrity. New or enhanced System 80+ features which help to mitigate SGTRs include:

- Larger steam generator secondary volume
- Larger pressurizer
- Four train safety injection system
- Four train emergency feedwater system
- Electrical system upgrades including alternate AC gas turbine and 8-hour batteries
- Safety depressurization and vent system
- Component cooling water system upgrade to four 100% capacity pumps and heat exchangers
- Highly reliable turbine bypass system, discharging all steam to condenser, not partially to atmosphere as in earlier designs
- Radiation monitors on the steam lines
- N-16 monitors for the steam generators

The System 80+ design meets the EPRI ALWR requirement of preventing main steam safety valve actuation following a SGTR. A reactor trip on high SG water level, actuation of the turbine bypass system and controlled depressurization of the RCS using the safety depressurization and vent system (SDVS) limit secondary side pressure below the MSSV setpoint. The turbine bypass valves discharge steam to the main condenser, which minimizes the radioactive release to the environment. The intent of the ALWR URD was to meet the above requirement on a best-estimate basis (i.e., credit for operator action and use of control-grade equipment is acceptable) to provide an effective and economical design.

The consequences of a worst-case steam generator tube rupture (SGTR) with loss of offsite power (LOOP) where the containment is bypassed due to malfunction of a main steam system valve has been analyzed. The analysis presented in CESSAR-DC Section 15.6.3.3, SGTR with LOOP and Single Failure, calculated the worst-case releases for an SGTR event with LOOP and a stuck open ADV on the affected steam generator.

The analysis simulated a double-ended break of one SG tube. The analysis contained conservative assumptions regarding

atmospheric dispersion factors, initial RCS and SG activity levels, and iodine spiking. Mitigating operator actions based on the approved CE emergency procedure guidelines (EPGs), CEN-152, were simulated. The analysis showed that no fuel failures were expected for this event.

The ADV on the affected SG was assumed to stick open when the operator tried to reseat the ADV to isolate the affected SG. After 30 minutes of steaming through the stuck-open ADV, the operator isolated this path by closing the ADV block valve. However, the leak of RCS liquid through the tube break continues for the duration of the analysis (8 hours) due to the conservative nature of the analysis models. In order to avoid overfilling the SG, the operator periodically steams from the affected SG per the EPGs. This additional steaming increased the total radiation dose. The total releases are well within regulatory limits.

It was recognized that the SGTR event represented a significant fraction of the offsite risk in this SAMDA analysis and DAs were selected specifically to address these sequences. These DAs include the Alternative Pressurizer Auxiliary Spray (DA5.5), Ideal 100% SG inspection (DA5.12), MSSV and ADV Scrubbing (DA5.13), Alternative SIS (DA5.11), and Diesel SIS Pump (DA5.19). The last two DAs address failure to inject for RC4.12E. DA5.23 specifically addresses refilling the RWST during a SGTR. Secondary side guard pipes (DA5.28) are also evaluated.

The following section will be added to Section 5:

5.23 Extended RWST Source

In the important SGTR sequences to public risk (RC4.18L), the RWST source expires as a makeup source. This DA consists of a ground level tank of borated water and a pump and piping to pump the water to the elevated RWST. It is assumed that the supply of water is sufficient to permit corrective actions before it also is exhausted. This DA is assumed to eliminate RC4.18L (see Table 5-21). Using a $1,000 per averted person-rem and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than $198.

A detailed design for the extended RWST source has not been performed but it would require a ground level tank of borated water and a pump and piping to pump the water to the elevated RWST, and instrumentation and control system. It is estimated to cost in excess of $1 Million.

9. Several SGTR containment bypass related improvements that have been or are now under consideration are not reflected in the

System 80+ design alternative evaluation document, such as addition of N-16 monitors, the use of a secondary system passive cooler, or increase in MSSV setpoints/SG shell pressure rating (Reference ABB-CE response to DSER Open Item 15.3.8-1). These design alternatives should also be included in the design alternative SSAR discussion. Please modify the document to include discussion and evaluation of all design alternatives evaluated and/or incorporated by CE.

Response:

The following DAs will be added to the SAMDA report:

5.24 N-16 Monitors

The N-16 monitors have been added to the System 80+ design. Its purpose is to assist the operators in identifying SGTR events. This DA was not quantified since it has been included in the design.

5.25 Increase Secondary Side Pressure

Upgrading the design pressure of the secondary system including the MSSVs to 1500 psia from the current 1200 psia was considered early in the System 80+ design process. It was determined that an increased design pressure would not significantly reduce the probability of containment bypass and release to the environment during a SGTR event.

During a SGTR with loss of offsite power, the condenser is not available for plant cooldown. The decay heat of the core and the stored energy in components are released to the atmosphere via the MSSVs, then via the SG ADVs. The steaming will continue until reaching shutdown cooling system entry conditions. The total heat to be removed (or the total steam release) is only slightly reduced by increasing the secondary design pressure and MSSV setpoints. Hence, using conservative safety analysis assumptions and methods, the overall radiation release would be essentially unchanged.

During a SGTR with offsite power available, the operator will act to mitigate this event according to the Emergency Procedures Guidelines, using both control grade and safety grade equipment if required. Therefore, for a "real-world" scenario, an increased design pressure would not significantly decrease the likelihood of lifting the MSSVs.

There are several technical disadvantages of increasing the secondary system design pressure to 1500 psia:

1. Steam generator would increase by up to 100 tons each. The added weight would increase containment heat sinks, and increase thermal stresses on the steam generator shell and main steam piping. These factors would likely impact the volume and arrangement of the containment. The additional weight would also increase the handling difficulties during fabrication.

2. The RCS support system would need to be redesigned and/or reevaluated to accommodate the increased loads. Any contribution to containment sizing must also be assessed.

3. For decreased heat removal events, RCS temperature and pressure would rise to a much higher value than in current plants. Pressurizer safety valve actuation would be more likely.

4. Unless the entire steam system and turbine are upgraded to 1500 psia, a second set of secondary side relief valves would be required downstream of the MSIVs to protect the low pressure portion of the steam system.

5. Feedwater systems would have to be compatible with the higher design pressure. Increasing secondary design pressure would require a major redesign effort and increase design complexity,which are not consistent with the evolutionary ALWR goals.

In summary, the issue of including an upgrade to the secondary side design pressure was considered from design considerations. Based on this review, this DA poses serious design drawbacks with limited benefits. A cost benefit analysis was not performed for this DA because very limited benefits were expected for extensive costs.

5.26 Passive Secondary Side Coolers

Secondary heat rejection for System 80+ has been considered at the conceptual level. Passive secondary heat rejection was included in the conceptual design for SIR, a much smaller plant.

The passive secondary heat rejection concept that is often promoted consists of an elevated condenser designed to full secondary side pressures. The heat sink for this condenser can be either water or air. If it is water, then in addition to the elevated condenser, there is an elevated water tank that gravity feeds into the condenser and is allowed to boil to atmosphere. Use of air in natural circulation results in a large increase of the surface area of the condenser but it has the potential of continuous long term operation without

support. The water tank concept requires a periodic refill.

The base system is relatively simple. However, several supporting functions are required to initiate the system. Isolation of the affected steam generator will be required, otherwise one must assume the entire cooling loop will go water solid with pressures equal to RCS pressure. An alternative is to have a continuous drain system that maintains a suitable free surface in the steam generator. This requires coordination with the RCS makeup system. If the design basis is isolation, will that require redundant systems on each steam generator. Control of cooldown rates is expected to be required, adding additional complexity. Heat rejection capacity sufficient to avoid early releases is expected to result in excessive cooldown rates later.

While simple in concept, the implementation of secondary closed loop cooling is expected to require major changes in the plant structures. A workable system will be more complex than the conceptual presentations being offered. Because of the redundancies in the current System 80+ design, and the potential high cost of this DA, this DA will not be further studied as a SAMDA.

10. Multiple tube SGTR events, ABB-CE was asked during the January 4, 1993 PRA meeting to address, within SAMDAs, the design alternatives discussed in GSI-163, "Multiple Steam Generator Tube Leakage". The staff acknowledged that USIs and GSIs emerging after 6 months prior to the application date would technically not have to be addressed; nevertheless, the issue appears to be safety-significant and should be dispositioned. Given the uncertainties in understanding SG tube degradation in the Palo Verde steam generators and the unspecified tube plugging criteria in the System 80+ technical specifications, ABB-CE should evaluate the design alternatives presented in the NRC GSI write-up (e.g., guard pipe, MSIV inside containment, etc.).

Response:

The SAMDA alternatives discussed in GSI-163 has been added:

5.27 Venting the MSSV in Containment

ABB does not plan to divert MSSV steam releases back to the containment. While such a system would reduce radiological releases to the environment for selected accident scenarios, such a system does not significantly reduce public risk and does carry several disadvantages. It should be noted that this feature does not eliminate releases to the environment.

The technical disadvantages of the MSSV-containment steam return system are summarized below for two hypothetical systems. In the first system, the steam is simply returned to the containment atmosphere. In the second system, the steam is discharged into the IRWST where it would be condensed.

Direct discharge of MSSV into containment has several serious disadvantages.

1. The secondary system return will place an additional loading burden on the containment and restrict plant operators in responding to accidents when containment sprays are unavailable. This could lead to the addition of a containment vent to address those concerns which in itself introduces another means of inadvertent containment bypass.

2. Any condensed steam discharge will drain to the IRWST, diluting the boron concentration. A minimum IRWST boron concentration for safety injection is necessary for mitigating LOCA and non-LOCA events.

3. The release to containment atmosphere has the potential to cause personal injury.

An MSSV return system directed to the IRWST has similar drawbacks to Items 1 and 2 described above and poses the additional complication that discharge of steam flows typical of the MSSVs may produce excessive loadings within the IRWST.

Either return path would require a major redesign effort and increase design complexity, which are not consistent with evolutionary ALWR goals. Also, this provision will not eliminate radiological releases to the environment from a SGTR.

In summary, the issue of including an MSSV discharge return to the containment was considered from design considerations. Based on this review, this DA poses serious design drawbacks. ABB-CE does not believe that the secondary steam should be piped and vented inside containment. These events are characterized as AOO events and filling the containment with steam during these events would be both damaging to the equipment and dangerous to operators. A cost benefit analysis was not performed for this DA because as it would require an assessment of equipment degradation, injuries, and loss of plant availability after secondary side venting into containment.

5.28 Secondary Side Guard Pipes

The secondary side guard pipe was proposed to address a Main Steam Line Break (MSLB) outside containment. This event is postulated to trigger multiple steam generator tube failures which could then result in a core melt because of depletion of coolant inventory. This sequence also bypasses the containment. The guard pipe would extend from the containment to the MSIVs and would be designed to prevent depressurization, given a MSLB in the specific section of pipe. MSLB represented 0.5% of the CDF for System 80+ and consequential SGTR was not modeled. It was assumed that this DA would halve the risk associated with intersystem LOCAs (RC5.1E) and halve the risk associated with all steam line break sequences because it is assumed that half of the lengths of main steam lines are guarded. Table 5-22 quantifies the risk reduction value of this DA. Using a $1000 per averted person-rem and a levelized capital cost of 16.6%, such a modification would be cost beneficial if it cost less than $4.40.

The cost for the guard pipes was taken from GSI-163[17] and adjusted for the different number and size. The original estimate of $1.1M was for a four loop plant. This estimate was first halved for a two loop plant and then increased by 50% to account for the larger size. The final cost of $820,000 was used in this analysis. This cost neglects the increased inspection and maintenance cost of the main steam lines because they are no longer accessible.

11. DA 5.9 - Hydrogen Igniters - This section should be rewritten to provide a high level discussion of the actual new design of the HMS, including the electrical arrangement for a minimum set of igniters powered from the station batteries and reference SSAR Section 6.2.5 and 19.11 for details. The discussion of passive autocatalytic recombiners (PARs) and their potential vulnerabilities should be expanded and clarified (e.g.slow removal rates, poor efficiency, etc...). These claims would appear to contradict the Electric Power Research Institute (EPRI) position on PAR efficacy for the System 80+ design. ABB-CE should clearly explain why a passive hydrogen control system is not cost beneficial.

Response:

The first paragraph to Section 5.9 has been rewritten as shown below:

Ideal hydrogen (H2) ignitors would prevent release classes associated with containment failures from hydrogen burns or

explosions. The System 80+ design has two different hydrogen control systems as described in Section 6.2.5 of the CESSAR-DC. The Containment Hydrogen Recombiner System (CHRS) is designed to control the H2 concentrations in the containment following a LOCA. The CHRS prevents the concentration of hydrogen from reaching the lower flammability limit of 4% by volume in air or steam-air mixtures. During a degraded core accident, hydrogen will be produced at a greater rate than that of a design basis LOCA. The Hydrogen Mitigation System (HMS)is designed to accommodate the hydrogen produced from 100% fuel clad metal-water reaction and limit the average hydrogen concentration in the containment to a 10% for a degraded core accident. The HMS consists of 80 Glow Plug Ignitors distributed through out the containment. Their placement is based on a detailed assessment of the flow paths to fully cover all of the containment. Section 19.11.4.1.3 of the CESSAR-DC discussed hydrogen in severe accidents. System 80+ already has a degraded core H2 control system and only two release classes (RC2.1E and 2.2E) have containment failure from hydrogen burning. This Design Alternative reduces the risk of these RCs (see Table 5-10). Such a system would have to cost $5 to be cost beneficial.

The following DA has been added to Section 5:

5.29 Passive Autocatalytic Recombiners (PARS)

Passive Autocatalytic Recombiners (PARS) are arrays of a palladium catalyst that will combine molecular hydrogen and oxygen gases into water. These units are currently in the development stage and have not been used in existing U.S. plants. They have a low conversion efficiency and therefore would have to be used in combination with existing H2 ignitors. The advantage of the PARS is that they require no electrical power and therefore would operate during a station blackout. The success of the PARS to prevent a H2 burn would depend on the speed of the production and release of the H2. For this analysis, it was conservatively assumed that the PARS worked perfectly and therefore would prevent release classes associated with containment failures from hydrogen burns or explosions. The System 80+ design already has H2 ignitors with redundant power backup via either DGs , batteries, or CT. Therefore, only two release classes (RC2.1E and 2.2E) have containment failure from hydrogen burning. This Design Alternative reduces the risk of these RCs (see Table 5-10, Alternative H2 Ignitors). Such a system would have to cost $5 to be cost beneficial.

EPRI[13] has been developing PARS technology and estimates that 40 units would be needed for large dry containments. EPRI estimates the units would cost $19,000 each, and the cost for

the PARS in System 80+ would be $760,000. This costs neglects any annual costs of cleaning, inspection and testing. Also, both NRC and ACRS have expressed concern about the expected relative slow response time of the PARS.

12. The filtered vent is estimated to cost in excess of $10 million. This estimate appears high relative to estimates developed elsewhere for foreign-installed systems. In this regard, provide a more complete accounting of associated costs. The significant increment above the $3 million cost cited for the Swiss vent would not appear to be related to the cost of the building to house the system, since the cost of other options (e.g., additional batteries) which also involve increasing building volumes are on the order of $300,000 to $2 million.

Response:

The cost estimates for a filtered vent system range from $2.8 Million to $25 Million. IDCOR Technical Report 19.1, July 1983 estimated a cost of $25M for larger system than our design and sized to handle ATWS. In the ABWR SAMDA, a cost of $3M was quoted. This is probably a smaller design taking credit for scrubbing in the BWR suppression pool. The Comanche Peak SAMDA estimated the cost from $15M to $22.3M and the Limerick SAMDA gives a range from $2.8M to $11.3M. The System 80+ estimate of $10M is for a non-ATWS sized, fully Category I facility and is bounded by the other estimates.

13. ABB-CE should discuss the feasibility of providing a filtration system on the "existing" hydrogen purge line. This 3-inch diameter purge line is presented in SSAR Chapter 19.11 as relieving containment pressure challenge at 80 psi through manual initiation.

Response:

A filtration system on the 3-inch hydrogen purge line already exists. This DA is to address late overpressure failures of containment. The 3 inch diameter hydrogen purge line is located in the secondary containment (annulus). The secondary containment is maintained at a negative pressure during accident conditions by the Annulus Ventilation System. Thus, all containment leakage into the annulus is collected by the Annulus Ventilation System and is filtered through HEPA and carbon filters before release through the Unit Vent. The 3 inch diameter hydrogen purge line was located inside the annulus such that the annulus ventilation filters would filter the releases of any post-accident purges.

14. An unfiltered vent using existing equipment should be considered as a design alternative given the late containment failure times and significant fission product removal that would occur over this time. As such, the hydrogen purge vent should be treated as an implemented design improvement in the same manner as the alternative containment spray and RCP seal cooling options.

<u>Response:</u>

The following will be added to Section 5:

<u>5.30 Hydrogen Purge Line</u>

An existing System 80+ design feature that could be utilized in venting the containment is the hydrogen purge vent. System 80+ is equipped with two 3 inch diameter hydrogen purge vents which can be used for purposes of containment venting. This design feature is shown in CESSAR-DC Figure 6.2.5-1. The vents are intended for use in post LOCA condition for diverting hydrogen to the secondary containment (annulus) should the hydrogen recombiners be inoperative. The annulus ventilation system then collects and filters the secondary containment atmosphere before release.

An analysis of the potential application of the venting capabilities of the hydrogen purge piping was performed using the MAAP computer code. This analysis conservatively simulated hydrogen purge as a 0.049 ft^2 equivalent area opening in the containment. A hypothetical accident management strategy was considered, whereby the hydrogen purge system is used to vent at the time the containment reaches 80 psia will enable the containment to maintain its pressure well below the containment failure threshold.

Since there are 4 AC electric motor operated valves in series on each division that must be opened to purge the containment and the annulus ventilation system requires AC power for operation, this feature can not be credited for mitigating severe accidents resulting from a complete loss of AC power.

This DA has already been included in the System 80+ design and no cost benefit analysis is necessary.

15. DA 5.3 - ABB-CE states that increasing the existing battery capacity for EFW pumps from the current System 80+ design capacity of 8 hours to 72 hours will require 9 times the number of cells. This is assuming lead-acid batteries. ABB-CE should discuss alternatives such as other types of

batteries (lithium, zinc secondary batteries) and fuel cells which may not impact the HVAC or structural design.

Response:

The following will be added to Section 5:

5.31 Fuel Cells

In addition alternative battery types to the traditional lead battery were investigated. Alternative battery types such as lithium or zinc are not commercially available in the necessary sizes to provide the capacity required by System 80+. Fuel cells are available in the size required for System 80+; however, they are not proven technologies in nuclear station applications and are not available as Class 1E equipment. In addition, the use of fuel cells presents the problem of heat generation since a typical fuel cell will operate at a temperature of 300 to 1000 °C. HVAC systems would have to be capable of removing the heat. Also, a safety related fuel delivery and exhaust system would be required for each battery. Design, development and installation of this type of fuel cell system would cost well over $2 million more than a conventional lead acid battery arrangement.

This Design Alternative addresses the release classes where emergency feedwater is lost after battery depletion during a station blackout. This DA is assumed to have the capability to remove decay heat using the turbine feedwater pump for whatever time period that is required (without any failures). This Design Alternative prevents core damage and therefore removes two of the release classes (same as Alternative DC Batteries and EFWS, see Table 5-5). Using a $1,000 per averted person-rem and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than $11.

16. Additional, lower cost design alternatives should be evaluated as an alternative to improved DC batteries and EFWS. These alternatives include: (a)use of portable generators already on site, and (b)use of existing diesel-driven pumps (such as the firewater pump) for feedwater injection.

Response:

The following will be added to Section 5:

5.32 Hookup for Portable Generators

Instead of increasing the battery capacity for the turbine driven EFW pump train, portable generators could be brought in

and hooked up for continued operation of the turbine driven EFW pump train after the batteries are depleted. This would require temporary hookup connections so that the portable generators could be connected in a timely manner. These temporary hook up connections would need to be located in an area that was easily accessible for installing the portable generators and would have to be located in an appropriate environment for running the generators during station blackout conditions. The cost of adding these temporary hookup connections, including the cabling to an appropriate location for hookup would be in excess of $10,000.

The diesel driven fire pump was investigated as an alternate feedwater source. This pump is only capable of producing 100 psia pressure. Therefore it does not have adequate head to feed the steam generators which would be in excess of 1000 psia pressure."

This Design Alternative addresses the release classes where emergency feedwater is lost after battery depletion during a station blackout. This Design Alternative prevents core damage and therefore removes two of the release classes (same as Alternative DC Batteries and EFWS, see Table 5-5) and would be cost beneficial if it cost less than $11.

17. DA 5.19 - 12 hour batteries. This DA discussion is unclear with respect to the increase in the number of cells. ABB-CE should clarify this remark to indicate that this increase would mean that for the current battery requirements and design, the design alternative would create additional batteries (and subsequent additional cells for the entire plant), since technically the design alternative would not increase the number of cells in each 125 vdc battery (e.g., 58 to 60 cells per battery). Also, the DA would probably not require a 1.5 times increase in the number of cells for the entire plant. A utility would procure an 8-h ampere-hour rated battery and load shed to get 12 hours worth just like System 80+ has 2-hour rated batteries with load-management for 8-hour capability. A lead-acid battery with 12-hour capability would still be approximately 58 to 60 cells. ABB-CE should evaluate the cost differential between a 2-hour rated battery and a battery with an increased rating.

Response:

The current ABB-CE design does not specify a 2-hour rated battery. Rather, a 2-hour duty cycle for design base accident loads and an 8-hour duty cycle for station blackout loads are defined. Batteries are sized according to the worst case duty cycle (8-hours for System 80+). The last paragraph of DA 5.19 will be revised as follows:

Increasing the current battery size to accommodate a 12-hour duty cycle for station blackout loads rather than a 8-hour duty cycle would require more plates per cell (minimum of 25% increase). Preliminary estimates show that the existing 8-hour duty cycle requires a large number of plates per cell (assuming 60 cell battery). Therefore, a 25% increase in plates per cell may exceed the number of plates that can be placed in a typical cell and may not be possible. However, if cells are available in sufficient size, they would be larger per cell and would require an additional mounting rack, which would require at a minimum 1.5 times existing battery building space. The more likely scenario would require another 60 cell battery or two 58 cell batteries connected in parallel. Thus, the required space would be 2 times existing space. The cost of this modification would be in excess of $300,000.

18. The evaluation of design alternatives to deal with core concrete interactions needs to be expanded to include consideration of the following: (a)the use of refractory materials in the floor of the reactor cavity, and (b)the use of a rubble bed or some other alternative to increase the likelihood of debris coolability.

<u>Response:</u>

The following sections will be added to Section 5 and the results will be added to the summary tables.

<u>5.33</u> <u>Water Cooled Rubble Bed</u>

The purpose of the water cooled rubble bed is to achieve a coolable debris bed below the vessel and remove decay heat. This DA consist of a floodable rubble bed in the bottom of the vessel cavity. The rubble bed would be kept dry until the corium had penetrated into it, thus minimizing the potential for steam explosion. This DA would have the same risk reduction potential as the ideal, Alternative Concrete Composition (DA5.7). This DA would eliminate seven RCs where basemat melt-through is modeled (see Table 5-8) Using a $1,000 per averted person-rem and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than $29.

The cost of the water cooled rubble bed is estimated[2] to be between $35.5 and $38.5 Million. Another source[18] estimated the cost to be $18.8 Million. Neither source included the cost of actually developing the system. Periodic testing and maintenance of the device which could be significant. For this analysis, the lower cost of $18.8M will be used.

5.34 Refractory Lined Crucible

The purpose of the refractory lined crucible is to achieve a coolable debris bed below the vessel and remove decay heat. This DA consist of a ceramic lined crucible with cooling located in the vessel cavity. This DA would have the same risk reduction potential as the ideal, Alternative Concrete Composition (DA5.7). This DA would eliminate seven RCs where basemat melt-through is modeled (see Table 5-8) Using a $1,000 per averted person-rem and a levelized cost rate of 16.6%, such a system would be cost beneficial if it cost less than $29.

The cost of the water cooled rubble bed is estimated[2] to be between $108 and $119 Million. Neither source included the cost of actually developing the system. Periodic testing and maintenance of the device which could be significant. For this analysis, the lower cost of $108M will be used.

19. Provide the following additional information regarding the design alternative related to increased concrete thickness: (a)justification that increasing the thickness of concrete would require an increase in containment diameter, (b)breakdown of the cost estimate for increasing the concrete and containment plate thickness, and (c)an assessment of the additional risk reduction that would be achieved by increasing the containment diameter and plate thickness (an increased containment volume and plate thickness would eliminate some early over-pressure failures and would delay the time of late over-pressure).

Response:

The last paragraph of DA 5.7 will be revised as follows:

"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. Currently additional concrete can not be added to the reactor cavity, since there would be an interference with the incore instrumentation tubes which exit the bottom of the reactor vessel. In order to add an additional two feet of concrete the NSSS would have to be raised by two feet to avoid interference with the incore instrumentation tubes. Raising the NSSS would also require the crane wall height to be increased by two feet in order to have adequate clearance to lift the reactor head and service other NSSS components. In order to increase the crane wall height

the containment diameter would have to be increased by approximately two feet in order to avoid an interference between the crane wall and containment vessel and to allow adequate space for spray coverage. An increase in containment diameter may also require an increase in containment plate thickness. An increase in containment plate thickness will require post-weld heat treatment for the construction of the containment vessel since the current thickness is at the limit allowed by the ASME Code before post-weld heat treatment is required. An increase in containment diameter will also require an increase in the diameter of the concrete shield building. The added cost for an additional two feet of concrete in the reactor cavity floor is small. However, the added cost of additional steel for the increased containment diameter and thickness, post-weld heat treatment required for the increased containment plate thickness, additional concrete and rebar for the increase in crane wall height and shield building diameter is estimated to exceed $5 million.

Because the dominant risks are associated with containment bypass events, the risk reduction associated with the additional thickness of the containment was not quantified. In events where no decay heat removal is available, the containment failure would still be postulated.

20. DA 5.8 - Reactor Vessel Exterior Cooling - The description of the Cavity Flood System (CFS) and IRWST capability to allow wetting of the bottom of the reactor vessel appears to contradict staff understanding based on severe accident discussions with ABB-CE representatives. The staff's understanding is that the CFS is specifically designed to <u>not allow</u> lower head wetting so as to prevent thermal shock to the vessel. Please clarify in this DA discussion the CFS flood capability.

<u>**Response:**</u>

Section 5.8 has been corrected and expanded as noted below:

The current arrangement for the IRWST will not allow wetting of the reactor vessel. The elevation of the IRWST was selected to ensure that wetting of the vessel would not occur should the holdup volume and cavity flood valves inadvertently open during power operation. This will prevent thermal shock of the vessel. However, water can be induced into the reactor cavity for exterior vessel cooling from external sources such as the Boric Acid Tank which provides a makeup source to the IRWST or by inducing water through the temporary hookup on the containment spray line discussed in Design Alternative 5.1 above and cost $1.5M. However, to utilize this option it must first be demonstrated that the reactor vessel will not breach

do to thermal shock of the vessel from the cold water. The analysis to demonstrate this is estimated to cost \$1M. This is based on the FERC prudence hearings for Yankee Atomic Electric Co. where it was reported that demonstration of vessel integrity would be a "multi-million dollar cost"[14]. The total cost would be \$2.5 M.

Given such a design modification was licensable, an inadvertent wetting of the reactor vessel during power, and no actual failures occurred, the event would require extensive testing and inspection before the plant would be permitted to startup. Such costs and additional economic risks have not been quantified but it is believed that these risks would outweigh any advantage of vessel flooding.

21. The design alternative related to reactor vessel exterior cooling does not appear to be a very cost-effective way to achieve the desired objective. For example, a small amount of water for pre-flooding the reactor cavity might be stored in an elevated tank and supplemented in the longer term by modifying the containment drainage system to route the drain flow to the lower reactor cavity. Provide further justification that lower cost of flooding the vessel externally are not possible, especially since this is a key strategy being pursued for passive reactors.

<u>Response:</u>

This DA has been reassessed and the response is included in RAI 20, above.

22. Provide additional discussion of the bases for the \$20 million cost estimate for the alternative high pressure safety injection system.

<u>Response:</u>

As stated in the text, the cost of \$20M was for two diesels to improve the existing four SIS pumps. In retrospect, this was not an effective way to improve the SIS system. The second paragraph of Section 5.10 will be replaced with:

As shown in Table 19.6.3.6-5 of the PRA, the dominant failure mode (80% of the total for small break LOCA) is common cause failure of the four check valves or four motor operated isolation valves. The Alternative SIS would have eight additional valves, each one with piping to parallel the existing valves. The estimated cost for this modification is \$2.2M. It is assumed that these valves are not subject to

common cause failures. Testing and maintenance has been neglected.

23. Explain why the costs of the MSSV and ADV scrubbing option cannot be significantly reduced by relocating these valves to within containment, thereby reducing the associated piping runs.

Response:

The original costing was for piping the discharge of the MSSV and ADV into containment. This DA will be modified to reflect the original intent, scrubbing the secondary side valve discharge with a spray system. A new DA5.27 addresses venting the MSSVs into containment. Section 5.13 will be modified as follows:

5.13 MSSV AND ADV SCRUBBING

The discharges of the main steam safety valves (MSSVs) and atmospheric dump valves (ADVs) could be scrubbed by routing the discharges through a structure with a water spray condense the steam and remove most of the fission products. This DA was introduced to specifically address steam generator tube rupture (SGTR) where isolation fails (the largest three RCs). Table 5-14 gives the risk reduction of this DA. The risk reduction is worth $544 dollars in capital to be cost beneficial.

This modification would require building structure over the valve discharges and installing a header system to distribute water. In addition, a pump, piping, water supply and instrumentation and drain system would be needed. Conceptually, this system is similar to a containment spray system for which a cost estimate of $9.5M was give in the Commanche Peak SAMDA analysis[15] and that cost estimate will be used in this analysis.

24. Provide a basis for the $10 million cost estimate for the additional diesel generator. This should include a comparison to typical costs incurred at those operating plants where diesel generators have been added.

Response:

The last paragraph of DA 5.14 will be revised as follows:

Addition of a third diesel generator to lower the probability of station blackout would require the addition of a 6.4 MW

do to thermal shock of the vessel from the cold water. The analysis to demonstrate this is estimated to cost $1M. This is based on the FERC prudence hearings for Yankee Atomic Electric Co. where it was reported that demonstration of vessel integrity would be a "multi-million dollar cost"[14]. The total cost would be $2.5 M.

Given such a design modification was licensable, an inadvertent wetting of the reactor vessel during power, and no actual failures occurred, the event would require extensive testing and inspection before the plant would be permitted to startup. Such costs and additional economic risks have not been quantified but it is believed that these risks would outweigh any advantage of vessel flooding.

21. The design alternative related to reactor vessel exterior cooling does not appear to be a very cost-effective way to achieve the desired objective. For example, a small amount of water for pre-flooding the reactor cavity might be stored in an elevated tank and supplemented in the longer term by modifying the containment drainage system to route the drain flow to the lower reactor cavity. Provide further justification that lower cost of flooding the vessel externally are not possible, especially since this is a key strategy being pursued for passive reactors.

Response:

This DA has been reassessed and the response is included in RAI 20, above.

22. Provide additional discussion of the bases for the $20 million cost estimate for the alternative high pressure safety injection system.

Response:

As stated in the text, the cost of $20M was for two diesels to improve the existing four SIS pumps. In retrospect, this was not an effective way to improve the SIS system. The second paragraph of Section 5.10 will be replaced with:

As shown in Table 19.6.3.6-5 of the PRA, the dominant failure mode (80% of the total for small break LOCA) is common cause failure of the four check valves or four motor operated isolation valves. The Alternative SIS would have eight additional valves, each one with piping to parallel the existing valves. The estimated cost for this modification is $2.2M. It is assumed that these valves are not subject to

common cause failures. Testing and maintenance has been neglected.

23. Explain why the costs of the MSSV and ADV scrubbing option cannot be significantly reduced by relocating these valves to within containment, thereby reducing the associated piping runs.

Response:

The original costing was for piping the discharge of the MSSV and ADV into containment. This DA will be modified to reflect the original intent, scrubbing the secondary side valve discharge with a spray system. A new DA5.27 addresses venting the MSSVs into containment. Section 5.13 will be modified as follows:

5.13 MSSV AND ADV SCRUBBING

The discharges of the main steam safety valves (MSSVs) and atmospheric dump valves (ADVs) could be scrubbed by routing the discharges through a structure with a water spray condense the steam and remove most of the fission products. This DA was introduced to specifically address steam generator tube rupture (SGTR) where isolation fails (the largest three RCs). Table 5-14 gives the risk reduction of this DA. The risk reduction is worth $544 dollars in capital to be cost beneficial.

This modification would require building structure over the valve discharges and installing a header system to distribute water. In addition, a pump, piping, water supply and instrumentation and drain system would be needed. Conceptually, this system is similar to a containment spray system for which a cost estimate of $9.5M was give in the Commanche Peak SAMDA analysis[15] and that cost estimate will be used in this analysis.

24. Provide a basis for the $10 million cost estimate for the additional diesel generator. This should include a comparison to typical costs incurred at those operating plants where diesel generators have been added.

Response:

The last paragraph of DA 5.14 will be revised as follows:

Addition of a third diesel generator to lower the probability of station blackout would require the addition of a 6.4 MW

diesel generator, its associated support systems, additional component cooling water piping to and from the diesel generator cooling water heat exchanger, an addition of a swing bus, additional cabling for connecting the diesel generator to the electrical distribution system, an additional diesel generator building to house the diesel, an additional fuel oil storage tank and storage tank structure, and additional HVAC systems for the diesel generator building and fuel oil storage tank structure. A study conducted for Duke Power Company's McGuire Nuclear Station estimates the cost of adding a similar swing diesel to be in excess of $25 Million. This McGuire study investigated the cost that other utilities incurred in installing additional diesel generators. Pennsylvania Power and Light installed a swing diesel at their Susquehanna plant. This job was originally bid at $30 Million; however, final installation ended up costing $130 Million. Northern States Power added additional diesel generators at the Prairie Island site. The initial bid for the project was $60 Million; however the final price was around $78 Million. The cost estimates for an additional diesel was $18.4M to $19M in the Comanche Peak SAMDA[15]. For this analysis the additional diesel will be estimated to cost $25 million.

25. Please provide the cost estimate for the ATWS injection system. This value was omitted in the CE document.

<u>Response:</u>

The omitted cost of this DA was $300,000 and represented instrumentation costs and has been added to the report.

26. Upgraded low pressure piping and components should be added as a design alternative for reducing the risk from interfacing system LOCAs. ABB-CE should provide a design alternatives discussion for those systems that have not been upgraded in the resolution of ISLOCA. For example, the portions of the CVCS letdown line that are protected by the automatic isolation feature (new pressure controller and auto closure of containment isolation valve) could be potential candidates for upgrade in the design alternatives discussion. The rationale provided to the staff for not upgrading these systems (such as hydrogen lines, nitrogen lines, ion exchanges) appeared technically valid and a potential justification as a cost-prohibited option.

Response:

The following section has been added to Section 5.

5.35 Automatic Overpressure Protection

ABB-CE conducted an extensive evaluation of the System 80+ standard design to respond to interfacing system LOCA challenges, to address Staff concerns raised in SECY-90-016 and SECY-93-087. ABB-CE and the Staff worked closely in the development of an acceptance criteria and performance of a system-by-system evaluation of ISLOCA challenges. The evaluation was documented in an ABB-CE special report which has been incorporated in CESSAR-DC as Appendix 5E. Table 2-1 of Appendix 5E summarizes the design changes made to achieve ISLOCA responses acceptable to the Staff. Section 4 of Appendix 5E presents the evaluation of design alternatives and rationale for the selected design approach for each potential ISLOCA pathway. Since this issue has been designated by the Staff as technically resolved, no further evaluation or reporting will be provided.

27. Given the digital control system, a potentially attractive design alternative is to develop a computer-based system to identify and isolate interfacing LOCAs, similar to that developed by the French. This system would use the existing network of radiation, temperature, and water level instrumentation to provide a diagnosis. Please provide an assessment of the feasibility of such a system.

Response:

Such design features have been included in the System 80+ design and are described in response to RAI 26 (above).

28. DA 5.16 - Diverse PPS - System 80+ has an alternate protection system (APS) in order to meet the ATWS rule. The APS contains an alternate scram system and a diverse emergency feedwater actuation system (DEFAS). Please clarify what the actual DA is as compared to the engineered system to meet the ATWS rule and revised design of the PPS to resolve I&C diversity concerns.

Response:

A better description of this DA has been added to the report making it clear that it is a third protection system and not the current APS.

29. DA 5.16: ABB-CE should also address the potential for a common mode failure (CMF) of the digital I&C system (software programming and hardware failures) (SECY 93-087 issue). ABB-CE evaluated the CMF in conjunction with SSAR Chapter 15 design-basis accidents. The two Chapter 15 events that were not evaluated were the large break LOCA and the MSLB inside containment. ABB-CE claimed use of leak-before-break application to shutdown the plant prior to the event. ABB-CE should quantify the low probability of a CMF in the hardware of the digital I&C and the software for the PPS and provide a cost-benefit analysis for not installing a diverse scram system for LBLOCA and MSLB. The APS could be modified as a potential design alternative to provide an alternate reactor scram on a low pressurizer pressure signal and a diverse ESF actuation signal for the MSLB or LBLOCA type events. ABB-CE should discuss potential DAs for the I&C diversity issue.

Response:

5.38 Digital LBLOCA Protection

The following section has been added to Section 5:

The likelihood of Plant Protection System (PPS) or Engineered Safety Feature (ESF) component system failure has been made extremely low through redundancy, hardware qualification, and a rigorous quality assurance program which has been reviewed by the NRC (see CESSAR-DC Section 7.2.1.1.2.5). Large Break LOCAs represent only 6.6% of the CDF and steam line breaks represent 0.5% of the CDF. These events are not major contributors to offsite risk because they tend to be in containment. Therefore only minor benefits in terms of public risk would be expected.The Large Break LOCA (LBLOCA) and steam line break within containment events can be assured through operator action in response to symptoms of precursor leakage (Leak Before Break, LBB). The instrumentation available to detect the leakage includes:

- Acoustic leak monitoring system alarm and trending
- Containment Temperature Level
- Containment Radiation
- Containment Humidity

The capacity of Nuplex 80+ makes possible tracking of leakage within containment and correlation of multiple symptoms. In addition to increased costs and complexity of additional trips and ESF actuation paths, the additional trips could decrease plant availability and increase the potential for equipment challenge (false actuation leading to transients) for a negligible

improvement in plant safety. Because of the small public risk associated with the LBBLOCA and the sophistication of the current protection system, this DA will not be further considered.

30. Provide additional discussion of the reactor cavity cooling design alternative. This should include an explanation of the cooling surfaces and heat transport paths (i.e., whether this modification would serve to remove heat from the reactor vessel sides, reactor vessel bottom head, or reactor cavity basemat). This modification would appear to be worth pursuing further based on its low cost and potential effectiveness.

    Response:
    This DA is only a enlargement of the existing piping from the IRWST to the holdup tank and then to the reactor cavity. The description will be clarified.

31. The costs estimates for the alternative startup system considered by CE appear to be driven by piping costs, and do not reflect the cost of an additional pump. Please justify why the costs for this modification could not be reduced through minor modifications to allow use of existing piping, or by reducing the distances involved.

    Response:

    Design Alternative 5.22 will be revised as follows:

    The System 80+ startup feedwater system has been modified such that it can be utilized as a back up to the Emergency Feedwater System. The System 80+ startup feedwater pumps are powered from the Combustion Turbine such that they are available on a loss of offsite power event. The condensate storage tank provides the water source for the startup feedwater pumps. Since, the startup feedwater system is non-safety the water from the startup feedwater pump is supplied upstream of the main feedwater isolation valves. Should the transient cause the main feedwater isolation valves to close on a Main Steam Isolation Signal, the signal can be bypassed and the valves reopened. The instrument air compressors are also powered from the Combustion Turbine. Therefore, they will be available to provide the air source for reopening the main feedwater isolation valves.

32. A system description and cost estimate is needed for design alternatives #46 [Vacuum Bld] and #47[Ribbed Containment]. these options were only identified in Table 5-1, and were not described in the text.

Response:

The following sections will be added to Section 5

5.36 Vacuum Building

ABB-CE developed a conceptual design for a vacuum building which was designed to reduce emissions from severe accidents and described is in Reference 11. The cost was estimated as $30 M in 1983. A separate IDCOR sponsored study (IDCOR Technical Report 19.1, July, 1983) also estimated the cost to be $30M (approximately $42M in 1993 dollars). Because of the high costs, and because most of the significant releases are bypass events for which the vacuum building would not help, this DA was not quantified.

5.37 Ribbed Containment

A ribbed containment was proposed to address failure of the containment from buckling during a seismic event coupled with an inadvertent actuation of the containment spray. This combination of events might lead to a vacuum in containment and some potential buckling. The ribs would not increase the maximum containment overpressure strength because the containment is assumed to fail at a weak point in the containment located between the ribs. Therefore since none of the RCs have containment failure due to a vacuum, no benefits were quantified for this DA. The cost of this DA is in the $10s of millions because the ribs complicate manufacturing and construction and would require field heat treating. Given that this DA has a high cost and no quantifiable benefit, it will not be further quantified.

33. Provide an assessment of major contributors to risk from external events, and design alternatives considered and/or implemented by CE to reduce risk from each of these contributors.

Response:

The following sections have been added to Section 5.

5.39 Seismic Capability

The System 80+ Plant is designed for a Safe Shutdown Earthquake (SSE) of 0.3g acceleration. The Seismic margins analysis (Section 19.7.5 of CESSAR-DC) addresses the margins associated with the seismic design and demonstrates that the plant High Confidence of Low Probability of Failure (HCLPF) value is 0.6g acceleration. Therefore, there is a 95%

confidence that existing equipment has less than a 5% probability of failure at twice the SSE level. To meet this stringent design goal, the containment design and SG support design may be modified. Recent Commission policy decisions state that ALWRs need to only demonstrate a HCLPF of 0.5g. The seismic capability is considered adequate for the System 80+ design and no additional changes are considered.

5.40 <u>Fire and Flood Capability</u>

The System 80+ Plant is designed with four quadrants, two in each of two divisions with permanent barriers between the divisions. Also sources of flooding were reduced in the annex building and drains were specifically designed to reduce flooding potential. These design features are described in Sections 9.5 (Fire Protection) and 3.4 (Flood Design) of CESSAR-DC. This capability is considered adequate for the System 80+ design and no additional changes are considered for fire and flood.

ATTACHMENT TO RAI-4

ANNEX B
ALWR REFERENCE SITE

The ALWR reference site is expected to conservatively represent the consequences of most potential sites. Characteristics of 91 U.S. reactor sites are tabulated in the NRC document, *Technical Guidance for Siting Criteria Development* (NUREG CR-2239). Below are listed several of these characteristics which are correlated with high off-site consequences. The values for the ALWR reference site are shown, as well as the approximate percentile for the values:

PARAMETER	ALWR VALUE	PERCENTILE
Population density 0-200 miles	182/sq. mi.	80
Population density 0-20 miles	370/sq. mi.	90
Population center 5-10 miles	1600/sq. mi.	90
Population center 10-20 miles	2700/sq. mi.	95
Rainfall – hours annually	540 hours	80

The following ALWR "reference site" characteristics are required as input to the CRAC2 computer code:

- Meteorological Data (see Table A.B-1);
- Population Data (see Table A.B-2);
- Evacuation and Sheltering Data (see Table A.B-3).

This annex provides a summary of these site characteristics. The actual data set is available on diskette and can be obtained from EPRI.

Meteorological Data

CRAC2 requires a file of hourly meteorological data consisting of wind speed, wind direction, atmospheric stability category, and intensity of precipitation. A CRAC2 meteorological data file contains data for one year, which consists of 8760 entries for a 365-day year. The weather data assessment is done by sorting the file into weather categories. The categories must provide a realistic representation of the year's weather without overlooking those kinds of weather that are instrumental in producing major consequence impacts. A set of 29 weather categories has been selected for the CRAC2 model to reflect these requirements.

ANNEX B
ALWR REFERENCE SITE

The entire year of data, 8760 hourly recordings, are sorted into the 29 weather categories. Each sequence is examined to determine (1) the first occurrence of rain within 30 miles of the site, or (2) the first occurrence of a wind speed slowdown within 30 miles of the accident site, or (3) the stability category and wind speed at the start of the sequence. The first of these conditions that is satisfied by the sequence determines the weather category to which it is assigned. Following the assessment process, the start hour of each weather sequence will have been assigned to one and only one weather category. Each of the weather categories then includes a set of weather sequences representing the corresponding weather type. The probability of occurrence of that weather type is the ratio of the total number of weather sequences in the year's data set.

The sampling procedure now has two key items of information available to it: (1) the category of each weather sequence and (2) the probability of occurrence of each category of weather. A sample consists of a set of weather sequences selected from each of the categories. Four sequences are selected from each category by the "Latin hypercube" sampling scheme [1]. With this sampling method, random samples are drawn from sets evenly spaced within the weather category. This assures that the model uses an event representation of the weather data over the full year.

Rather than present the entire file in CRAC2 input format, the summary tables are attached for review. These tables give statistics for 29 bins derived from the 8760 hours of data.

Bins 1 through 7 represent cases where rain occurs over the distance intervals 0 (site), 0-5, 5-10, 10-15, 15-20, 20-25, and 25-30 miles, respectively.

Bins 8 through 12 represent cases where slowdowns (periods of low wind speed) occur over the distance intervals 0-10, 10-15, 15-20, 20-25, and 25-30 miles, respectively.

Bins 13 and 14 represent cases with stability class A, B, or C and initial wind speeds of ≤ 3 and > 3 meters/sec, respectively.

Bins 15 through 19 represent cases with stability class D and initial wind speeds of < 1, 1-2, 2-3, 3-5, and > 5 meters/sec, respectively.

[1] Iman, R.L. and Conover, W.J. *Sensitivity Analysis Techniques: Self-Teaching Curriculum.* U.S. Nuclear Regulatory Commission Report, NUREG/CR-2350, June 1982.

Bins 20 and 24 represent cases with stability class E and initial wind speeds of < 1, 1-2, 2-3, 3-5, and > 5 meters/sec, respectively.

Bins 25 and 29 represent cases with stability class F and initial wind speeds of < 1, 1-2, 2-3, 3-5, and > 5 meters/sec, respectively.

All bins are further divided to provide statistics for the 16 different wind directions corresponding to 22.5-degree sectors. The first of these sectors is centered on due north, the second 22.5 degrees east of north, and so on.

TABLE A.B-1.
CRAC2 METEOROLOGICAL BIN SUMMARY

METEOROLOGICAL DATA FILE CONTAINS 513 HOURS OF OBSERVED RAIN DATA.

ACCUMULATED RAIN MEASUREMENTS TOTALED 47.64 INCHES FOR THE YEAR.

HOLZWORTH AFTERNOON MIXING HEIGHT 1500 METERS.

* * * METEOROLOGICAL BIN SUMMARY * * *

BIN PRIORITIES

R - RAIN WITHIN INTERVALS

S - SLOWDOWNS WITHIN INTERVALS

C D E F – STABILITY CATEGORIES

1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (GT 5) – WIND SPEED INTERVALS (M/S)

	WIND DIRECTION																	
METBIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	TOTAL	PERCENT
1 R 0	0.136	0.111	0.090	0.047	0.041	0.021	0.049	0.097	0.078	0.107	0.068	0.029	0.029	0.019	0.035	0.041	513	5.8562
2 R 5	0.114	0.086	0.043	0.014	0.114	0.029	0.100	0.114	0.057	0.086	0.029	0.043	0.057	0.029	0.071	0.014	70	0.7991
2 R 10	0.075	0.075	0.082	0.034	0.075	0.062	0.110	0.068	0.062	0.137	0.082	0.027	0.027	0.007	0.041	0.034	146	1.6667
4 R 15	0.076	0.101	0.076	0.059	0.067	0.042	0.084	0.092	0.109	0.118	0.050	0.050	0.008	0.017	0.017	0.034	119	1.3584
5 R 20	0.054	0.045	0.116	0.045	0.027	0.009	0.107	0.098	0.089	0.116	0.089	0.027	0.036	0.018	0.045	0.080	112	1.2785
6 R 25	0.080	0.070	0.090	0.060	0.040	0.070	0.080	0.110	0.140	0.100	0.070	0.020	0.010	0.010	0.030	0.020	100	1.1416
7 R 30	0.063	0.116	0.074	0.0	0.063	0.063	0.084	0.074	0.147	0.126	0.116	0.042	0.011	0.0	0.011	0.011	95	1.0845
8 S 10	0.085	0.136	0.068	0.051	0.0	0.017	0.0	0.0	0.017	0.153	0.051	0.051	0.017	0.119	0.102	0.136	59	0.6735
9 S 15	0.175	0.050	0.100	0.075	0.0	0.025	0.0	0.025	0.025	0.050	0.075	0.075	0.175	0.050	0.025	0.075	40	0.4566

TABLE A.B-1.
CRAC2 METEOROLOGICAL BIN SUMMARY

BIN PRIORITIES

R - RAIN WITHIN INTERVALS
S - SLOWDOWNS WITHIN INTERVALS
C D E F – STABILITY CATEGORIES
1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (GT 5) – WIND SPEED INTERVALS (M/S)

WIND DIRECTION

METBIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	TOTAL	PERCENT
10 S 20	0.184	0.041	0.061	0.0	0.0	0.020	0.0	0.020	0.061	0.122	0.041	0.061	0.061	0.020	0.184	0.122	49	0.5594
11 S 25	0.087	0.109	0.109	0.0	0.043	0.022	0.022	0.022	0.022	0.065	0.217	0.022	0.022	0.0	0.109	0.130	46	0.5251
12 S 30	0.058	0.038	0.038	0.0	0.058	0.019	0.0	0.058	0.038	0.135	0.096	0.058	0.038	0.019	0.135	0.212	52	0.5936
13 C 3	0.052	0.058	0.067	0.052	0.069	0.053	0.076	0.073	0.067	0.091	0.131	0.056	0.059	0.044	0.023	0.028	1126	12.8539
14 C 4	0.057	0.079	0.052	0.026	0.018	0.002	0.019	0.031	0.029	0.116	0.246	0.085	0.039	0.063	0.072	0.066	1136	12.9680
15 D 1	0.065	0.043	0.087	0.065	0.098	0.087	0.043	0.033	0.033	0.065	0.054	0.065	0.054	0.065	0.065	0.076	92	1.0502
16 D 2	0.056	0.062	0.074	0.042	0.076	0.064	0.048	0.052	0.095	0.110	0.099	0.064	0.056	0.027	0.031	0.035	484	5.5251
17 D 3	0.048	0.081	0.116	0.048	0.057	0.032	0.041	0.057	0.063	0.154	0.125	0.063	0.027	0.032	0.023	0.032	559	6.3813
18 D 4	0.128	0.137	0.117	0.023	0.011	0.0	0.007	0.036	0.069	0.200	0.144	0.034	0.016	0.007	0.029	0.040	554	6.3242
19 D 5	0.063	0.215	0.018	0.0	0.0	0.0	0.004	0.031	0.094	0.135	0.229	0.045	0.004	0.0	0.072	0.090	223	2.5457
20 E 1	0.123	0.058	0.076	0.035	0.058	0.064	0.076	0.047	0.070	0.047	0.088	0.076	0.041	0.047	0.047	0.047	171	1.9521
21 E 2	0.070	0.041	0.038	0.019	0.035	0.019	0.041	0.086	0.089	0.153	0.152	0.072	0.064	0.038	0.029	0.053	627	7.1575

TABLE A.B-1.
CRAC2 METEOROLOGICAL BIN SUMMARY

BIN PRIORITIES

R - RAIN WITHIN INTERVALS
S - SLOWDOWNS WITHIN INTERVALS
C D E F – STABILITY CATEGORIES
1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (GT 5) – WIND SPEED INTERVALS (M/S)

WIND DIRECTION

METBIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	TOTAL	PERCENT
22 E 3	0.135	0.055	0.045	0.022	0.012	0.010	0.015	0.075	0.077	0.185	0.137	0.055	0.050	0.035	0.035	0.057	401	4.5776
23 E 4	0.155	0.082	0.034	0.010	0.007	0.0	0.0	0.082	0.103	0.258	0.117	0.021	0.003	0.0	0.052	0.076	291	3.3219
24 E 5	0.081	0.210	0.032	0.0	0.0	0.0	0.016	0.032	0.355	0.145	0.065	0.016	0.0	0.0	0.032	0.016	62	0.7078
25 F 1	0.078	0.073	0.065	0.039	0.057	0.035	0.071	0.043	0.092	0.082	0.086	0.057	0.065	0.049	0.057	0.051	510	5.8219
26 F 2	0.103	0.057	0.021	0.006	0.013	0.005	0.025	0.057	0.112	0.149	0.159	0.113	0.072	0.042	0.021	0.034	793	9.0525
27 F 3	0.107	0.020	0.008	0.004	0.004	0.0	0.0	0.055	0.091	0.154	0.154	0.091	0.134	0.059	0.043	0.075	253	2.8881
28 F 4	0.213	0.115	0.016	0.016	0.0	0.0	0.0	0.0	0.213	0.016	0.049	0.0	0.033	0.082	0.098	0.148	61	0.6963
29 F 5	0.0	0.250	0.0	0.0	0.0	0.0	0.0	0.0	0.250	0.500	0.0	0.0	0.0	0.0	0.0	0.0	16	0.1826
30 ALL	0.085	0.078	0.063	0.031	0.037	0.024	0.040	0.059	0.079	0.131	0.138	0.062	0.046	0.036	0.041	0.050	8760	

TABLE A.B-1.
CRAC2 METEOROLOGICAL BIN SUMMARY

BIN PRIORITIES
- R - RAIN WITHIN INTERVALS
- S - SLOWDOWNS WITHIN INTERVALS
- C D E F – STABILITY CATEGORIES
- 1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (GT 5) – WIND SPEED INTERVALS (M/S)

WIND DIRECTION

METBIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	TOTAL	PERCENT
1 R 0	70	57	46	24	21	11	25	50	40	55	35	15	15	10	18	21	513	5.8562
2 R 5	8	6	3	1	8	2	7	8	4	6	2	3	4	2	5	1	70	0.7991
3 R 10	11	11	12	5	11	9	16	10	9	20	12	4	4	1	6	5	146	1.6667
4 R 15	9	12	9	7	8	5	10	11	13	14	6	6	1	2	2	4	119	1.3584
5 R 20	6	5	13	5	3	1	12	11	10	13	10	3	4	2	5	9	112	1.2785
6 R 25	8	7	9	6	4	7	8	11	14	10	7	2	1	1	3	2	100	1.1416
7 R 30	6	11	7	0	6	6	8	7	14	12	11	4	1	0	1	1	95	1.0845
8 S 10	5	8	4	3	0	1	0	0	1	9	3	3	1	7	6	8	59	0.6735
9 S 15	7	2	4	3	0	1	0	1	1	2	3	3	7	2	1	3	40	0.4566
10 S 20	9	2	3	0	0	1	0	1	3	6	2	3	3	1	9	6	49	0.5594
11 S 25	4	5	5	0	2	1	1	1	1	3	10	1	1	0	5	6	46	0.5251

TABLE A.B-1.
CRAC2 METEOROLOGICAL BIN SUMMARY

BIN PRIORITIES

R - RAIN WITHIN INTERVALS

S - SLOWDOWNS WITHIN INTERVALS

C D E F – STABILITY CATEGORIES

1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (GT 5) – WIND SPEED INTERVALS (M/S)

WIND DIRECTION

METBIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	TOTAL	PERCENT
12 S 30	3	2	2	0	3	1	0	3	2	7	5	3	2	1	7	11	52	0.5936
13 C 13	59	65	76	58	78	60	86	82	75	103	148	63	66	49	26	32	1126	12.8539
14 C 4	65	90	59	29	21	2	22	35	33	132	279	97	44	71	82	75	1136	12.9680
15 D 1	6	4	8	6	9	8	4	3	3	6	5	6	5	6	8	7	92	1.0502
16 D 2	27	30	36	25	37	31	23	25	46	53	48	31	27	13	15	17	484	5.5251
17 D 3	27	45	65	27	32	18	23	32	35	86	70	35	15	18	13	18	559	6.3813
18 D 4	71	76	65	13	6	0	4	20	38	111	80	19	9	4	16	22	554	6.3242
19 D 5	14	48	4	0	0	0	1	7	21	30	51	10	1	0	16	20	223	2.5457
20 E 1	21	10	13	6	10	11	13	8	12	8	15	13	7	8	8	8	171	1.9521
21 E 2	44	26	24	12	22	12	26	54	56	96	95	45	40	24	18	33	627	7.1575
22 E 3	54	22	18	9	5	4	6	30	31	74	55	22	20	14	14	23	401	4.5776
23 E 4	45	24	10	3	2	0	0	24	30	75	34	6	1	0	15	22	291	3.3219

TABLE A.B-1.
CRAC2 METEOROLOGICAL BIN SUMMARY

BIN PRIORITIES (Page 6 of 7)

R - RAIN WITHIN INTERVALS
S - SLOWDOWNS WITHIN INTERVALS
C D E F – STABILITY CATEGORIES
1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (GT 5) – WIND SPEED INTERVALS (M/S)

WIND DIRECTION

METBIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	TOTAL	PERCENT
24 E 5	5	13	2	0	0	0	1	2	22	9	4	1	0	0	2	1	62	0.7078
25 F 1	40	37	33	20	29	18	36	22	47	42	44	29	33	25	29	26	510	5.8219
26 F 2	82	45	17	5	10	4	20	45	89	118	134	90	57	33	17	27	793	9.0525
27 F 3	27	5	2	1	1	0	0	14	23	39	39	23	34	15	11	19	253	2.8881
28 F 4	13	7	1	1	0	0	0	0	13	1	3	0	2	5	6	9	61	0.6963
29 F 5	0	4	0	0	0	0	0	0	4	8	0	0	0	0	0	0	16	0.1826

TABLE A.B-1.
CRAC2 METEOROLOGICAL BIN SUMMARY

BIN PRIORITIES
- R - RAIN WITHIN INTERVALS
- S - SLOWDOWNS WITHIN INTERVALS
- C D E F – STABILITY CATEGORIES
- 1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (GT 5) – WIND SPEED INTERVALS (M/S)

WIND DIRECTION

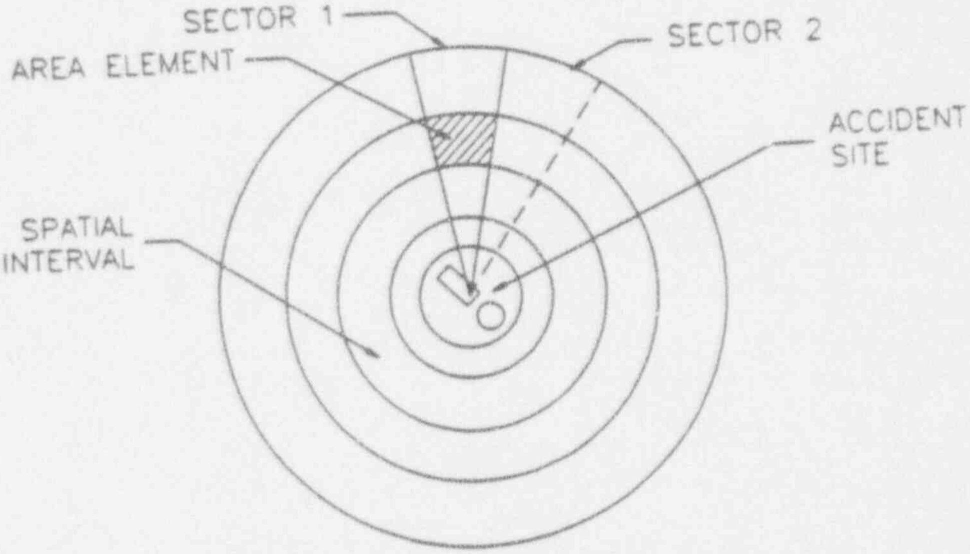
METBIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	TOTAL	PERCENT
* * * SUMMARIES * * *																		
R	118	109	99	48	61	41	86	108	104	130	83	37	30	18	40	43	1155	13.1849
S	28	19	18	6	5	5	1	[illegible]	8	27	23	13	14	11	28	34	246	2.8082
C	124	155	135	87	99	62	108	117	108	235	427	160	110	120	108	107	2262	25.8219
D	145	203	178	71	84	57	55	87	143	286	254	101	57	41	66	84	1912	21.8265
E	169	95	67	30	39	27	46	118	151	262	203	87	68	46	57	87	1552	17.7169
F	162	98	53	27	40	22	56	81	176	208	220	142	126	78	63	81	1633	18.6415
1	70	51	55	34	53	38	57	34	64	57	69	50	47	41	43	41	804	9.1781
2	174	114	104	63	107	75	107	163	230	301	306	184	146	92	56	88	2310	26.3699
3	143	124	133	72	73	53	73	118	123	267	278	123	111	72	58	81	1902	21.7123
4	174	163	130	42	27	2	24	76	105	296	323	101	49	60	75	92	1739	19.8516
5	39	99	11	4	2	0	4	12	56	70	128	32	8	20	62	57	604	6.8950

ANNEX B
ALWR REFERENCE SITE

Population Data

The population data which describes the ALWR reference site is contained in the Site Data file. The population distribution around the reactor site was assigned to elements of a grid defined by sixteen 22.5-degree sectors and twenty-five annuli. The first of these sectors is centered on due north, the second 22.5 degrees east of north, and so on. These directions correspond to the wind rose generated from the meteorological file, with the wind blowing **toward** the given directions. The annuli have the following radii in miles: 0.25, 0.75, 1.0, 2.0, 3.0, 4.0, 5.0, 6.0, 7.0, 8.0, 9.0, 10.0, 11.0, 12.0, 13.0, 14.0, 15.0, 16.0, 17.0, 18.0, 19.0, 20.0, 30.0, 40.0, 50.0.

Attached is the population distribution for the ALWR reference site. Information on format can be obtained from the CRAC2 Computer Code Users Manual.



Representation of the CRAC2 Geometry

TABLE A.B-2.
ALWR CRAC2 REFERENCE SITE – POPULATION DATA

Sector	#1	#2	#3	#4	#5	#6	#7	#8
Distance intervals (miles)								
0.0 – 0.25	0	0	0	0	0	0	0	0
0.25 – 0.75	1	3	0	1	1	7	0	0
0.75 – 1.0	2	3	0	2	2	8	0	0
1.0 – 2.0	44	30	35	41	11	75	27	7
2.0 – 3.0	76	31	38	19	50	935	229	256
3.0 – 4.0	819	113	70	89	156	566	726	465
4.0 – 5.0	435	461	100	139	219	146	413	777
5.0 – 6.0	255	161	178	71	376	300	406	1279
6.0 – 7.0	223	189	173	87	140	603	2025	4563
7.0 – 8.0	237	188	52	59	688	2762	414	6780
8.0 – 9.0	435	377	25925	25409	472	2188	254	4277
9.0 – 10.0	537	542	1054	257	1108	852	255	6276

TABLE A.B-2.
ALWR CRAC2 REFERENCE SITE – POPULATION DATA

Sector Distance Intervals (miles)	#1	#2	#3	#4	#5	#6	#7	#8
10.0 – 11.0	731	704	1587	1634	1156	216	661	2530
11.0 – 12.0	2305	783	2160	5780	2508	525	752	1300
12.0 – 13.0	4946	1586	4516	8019	2037	556	503	697
13.0 – 14.0	7747	2001	8474	9310	399	577	935	431
14.0 – 15.0	5996	2542	15120	10564	205	224	1738	771
15.0 – 16.0	6818	2955	17177	8195	436	417	217	304
16.0 – 17.0	6422	5506	21995	12552	2217	444	231	323
17.0 – 18.0	2761	4247	22467	12366	1729	471	245	343
18.0 – 19.0	3071	3052	23250	12254	783	497	260	362
19.0 – 20.0	1717	2452	23709	12438	1101	524	274	382
20.0 – 30.0	29136	25042	143872	104941	56858	18654	51951	2771
30.0 – 40.0	27439	59969	132594	21792	42640	14732	30022	15879
40.0 – 50.0	48856	40643	64239	24214	17771	20822	19065	3685

TABLE A.B-2.
ALWR CRAC2 REFERENCE SITE – POPULATION DATA

Sector	#9	#10	#11	#12	#13	#14	#15	#16
Distance Intervals (miles)								
0.0 – 0.25	0	0	0	0	0	0	0	0
025 – 0.75	4	0	1	3	1	0	0	0
0.75 – 1.0	5	0	2	3	2	0	0	0
1.0 – 2.0	11	31	0	0	15	68	61	30
2.0 – 3.0	113	236	73	39	0	15	27	30
3.0 – 4.0	290	265	184	39	60	69	36	119
4.0 – 5.0	595	392	85	39	90	74	262	80
5.0 – 6.0	834	386	126	130	103	100	180	[illegible]
6.0 – 7.0	2156	607	271	157	120	145	163	279
7.0 – 8.0	2317	432	201	115	140	255	333	350
8.0 – 9.0	3278	105	260	265	275	498	290	343
9.0 – 10.0	4199	353	110	2146	375	2263	238	215
10.0 – 11.0	2479	530	160	3135	320	2037	150	3232

TABLE A.B-2.
ALWR CRAC2 REFERENCE SITE – POPULATION DATA

Sector	#9	#10	#11	#12	#13	#14	#15	#16
Distance Intervals (miles)								
11.0 – 12.0	1053	220	225	1427	389	171	451	2241
12.0 – 13.0	629	175	250	340	346	230	1567	2046
13.0 – 14.0	512	215	190	197	215	290	1265	7624
14.0 – 15.0	331	177	155	133	200	339	2111	11128
15.0 – 16.0	257	325	116	247	225	107	1507	13046
16.0 – 17.0	274	345	124	263	239	114	1465	15289
17.0 – 18.0	290	366	132	279	254	121	2517	7189
18.0 – 19.0	307	387	139	295	269	127	1694	4992
19.0 – 20.0	323	408	147	310	283	134	8411	3369
20.0 – 30.0	4453	37878	5618	3593	14417	34231	47823	35411
30.0 – 40.0	4145	3906	35154	16059	59503	75906	29496	56468
40.0 – 50.0	19643	5506	17736	44895	126121	54872	16930	113123

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ANNEX B
ALWR REFERENCE SITE

Evacuation and Sheltering Data

For comparison to the ALWR requirement regarding serious release as it is defined in Section 5.1.1, evacuation for all groups in the population should be delayed for 24 hours, and sheltering factors for normal activity should be used.

Evacuation and sheltering data are provided to allow other calculations of offsite consequences to be made for the reference site (e.g., to compare to NRC safety goal. The ALWR off-site consequences analysis requires ~~six distinct~~ two evacuation schemes in order to adequately represent evacuation time estimates for the ~~permanent resident population, the transient population, and the special facility population (schools, hospitals, etc.)~~ 10-mile EPZ. The evacuation data includes an evacuation scheme that assumes ~~5~~ 0.5 percent of the population would delay evacuation for 24 hours after being warned to evacuate. ~~This very conservative assumption is used so that the ALWR risk estimates can be compared with the IDCOR and NUREG-1150 analyses which both use this assumption.~~ This evacuation model is representative of that used in the analyses for NUREG-1150.

Cloud and ground shielding factors are based on information given in WASH-1400. Breathing rate data is obtained from the PRA Procedures Guide.

ATTACHMENT 5

FAX

TO: **Nick Saltos**
USNRC - NRR
Mail Stop 10E4
Phone (301)504-1072
Fax (301)504-2260

FROM: **David Finnicum**
ABB-CE
Phone (203)285-3926
Fax (203)285-5881

XC: J. J. Herbst (w/o)
R. E. Jaquith (w/o)
J. Longo Jr.
S. E. Ritterbusch
9424 Files
9612 files

DATE: August 26, 1993

NUMBER: OPS-93-0641

SUBJECT: Transmittal of Response to Follow-on Question for DSER Open Item 19.1.2.1.1.3-2

I am providing ABB-CE's response to the follow-on question to DSER Open Item 19.1.2.1.1.3-2 as documented in your fax of June 14, 1993. If you have any questions on this information, please call me at (203)285-3926.

DSER Open Item 19.1.2.1.1.3-2

NRC Comments and Follow-up Questions: This open item concerns the feasibility of using the damaged steam generator for ASC to reduce RCS pressure below the SCS pump shutofff head before the core is uncovered. It also requires an assessment of the radioactivity release through the ADV during the cooldown period.

The applicant cited a few references in the updated System 80+ PRA (References 9-11 of System 80+ PRA) to argue that if ASC is initiated within about 15 minutes of an SGTR event, the core will remain covered and the SCS can successfully provide RCS inventory control. The applicant also used the the analysis results performed in the PRA for a small LOCA evant as additional justification. As for the radioactivity release, the applicant used the results in CESSAR-DC for a SGTR event with a loss of offsite power and stuck open ADV to argue that the radiological release for the present case would be within 10CFR100 limits.

Although the accident events whose analysis results are used in the above arguements are not the same as that of interest here, they do seem to provide some relevant information for the issues raised here. However, as can be seen from the referenced small LOCA analysis (Figure 1 of Response to Open Item 19.1.2.1.1.3-1), the water level in the core region drops to near the top of the active fuel soon after the RCPs are tripped. The top of the active fuel remains barely covered throughout the rest of the transient even after the contents of the SITs are discharged (at RCS pressure of about 600 psia and t=1100seconds). This may not be a big concern for a small LOCA event, but for an SGTR event, significantly larger radioactivity may be released directly to the environment if part of the core is uncovered, even for a short period of time. Furthermore, the performance of a damaged SG may not be as effective as an intact SG, the ASC used in an SGTR event may not be as effective as that in a small LOCA event, and it is likely that much longer time will be required for RCS pressure to drop to 600 psia when the SITs can start to inject. The main concern, therefore, is that whether the core can remain covered for the whole time period. To eliminate the above concerns, an analysis specifically prepared for SGTR is desired. We also noted that, based on figure 3 (of the Response to Open Item 19.1.2.1.1.3-1), core exit fluid temperature drops from 565 ^{0}F to 415 ^{0}F in about 17.5 minutes after ASC is started. this raises the question about whether the reactor will become critical again.

The success of ASC for SGTR requires the operator to properly diagnose the need of ASC and initiate ASC actions within about 15 minutes of accident initiation. The probability used in the updated System 80+ PRA for the operator to fail to perform ACS for SGTR is 0.07. Although SGTR recovery procedures were mentioned in the PRA, details of the procedures (or whether the procedures had already been prepared), are not provided. Since ASC involves actions that are contradictory to standard actions required for SGTR recovery (e.g., isolation), and also involves the opening of a release path of core radioactivity to the environment, the SGTR recovery procedure must provide sufficient information for the operator to make a correct and timely decision, consistent with the failure probability used in the PRA. This point should be emphasized in the preparation of the procedure. From the data presented in the updated PRA, ASC seems to play a very important role in the

determination of the total CDF of SGTR events. The leading SGTR sequence (SGTR-17) involves the failure of ASC. Its frequency of 2.73E-7/year constitutes 95% of total SGTR CDF (2.87E-7/year).

__Response:__ An SGTR is essentially a small LOCA in which the RCS inventory is discharged to the ruptured SG rather than to the containment. Thus, at initiation, an SGTR provides indications equivalent to those of a standard small LOCA. The initial system and operator responses are essentially the same for an SGTR as they are for a small LOCA. Safety injection would be initiated for RCS inventory control, reactor trip would be initiated for reactivity control, and the emergency feedwater system (or startup feedwater system) would be actuated for secondary side heat removal. The operators would initially verify reactor trip, safety injection actuation and emergency feedwater actuation. They would also verify that a plant cooldown using both SGs was established. They would then begin the break identification procedure to determine whether the break was a small LOCA or an SGTR. Once it is determined that an SGTR has occurred and the ruptured generator is identified, the operators will cool the RCS to a temperature at which the MSSVs in the ruptured generator would not lift when the ruptured generator is isolated. This would typically occur approximately 30 minutes after the SGTR initiation. (Note: the emergency procedures stipulate that the ruptured generator is not to be isolated if the RCS pressure is less than 50 psi greater than the pressure in the ruptured generator. If the differential pressure is greater than or equal to 50 psi, the generator can be isolated when the other temperature and pressure conditions are established.) At this point, the ruptured SG would be isolated by closing the MSIVs, MFIVs and the blowdown valves for the ruptured SG. The operators would then proceed with an orderly cooldown and depressurization of the RCS using the intact SG. During this cooldown, the RCS pressure would be maintained just above the pressure in the ruptured SG to minimize leakage to the ruptured SG.

Failure of safety injection actuation (Loss of RCS Inventory Control) would be identified during the initial phase of the response to the event, prior to determining whether the event was a standard small LOCA or an SGTR. The operator response to Loss of RCS Inventory Control would be the same as for a small LOCA because at this point in time the operators have not determined that an SGTR has occurred. The operator responses would therefore be performed within the same time frame. Thus ASC would be initiated within approximately 15 minutes after the initiation of the event. While performing the ASC, the operators would continue with the break identification. Once it is determined that an SGTR has occurred and the ruptured generator is identified, the operators would isolate the ruptured generator if the temperature and pressure conditions were appropriate and the RCS pressure is 50 psi greater than the pressure in the ruptured generator. Once the ruptured generator is isolated, the operator would continue with the cooldown using only the good generator. If the ruptured generator can not be isolated, the operators would continue the cooldown using both generators until such time as isolation conditions can be established or shutdown cooling is established and the transient event is terminated. As for the small LOCA case, all four SITs were assumed available for injection to provide short term inventory control.

A transient analysis was performed to demonstrate that ASC could be successfully accomplished for an SGTR. The SGTR occurred at time zero, and HPSI failed. At 15

minutes, an aggressive secondary cooldown was initiated using both steam generator. Each generator recieved the flow from one EFW pump, and one ADV on each generator was used for steam removal. Starting at 30 minutes, conditions in the the ruptured steam generator were monitored to determine if the ruptured generator could be isolated. As shown in the attached plots, the ruptured generator could not be isolated during the cooldown because the RCS pressure was less than 50 psi greater than the pressure in the ruptured generator. The low differential pressure did, however, minimize the primary to secondary leakage. Consistent with the emergency procedures, the operators continued the ASC using both steam generators. As is shown by the attached plots, the RCS was successfully cooled and depressurized using ASC while maintaining core covery and cooling.

The radioactivity released during the transient was calculated using the standard chapter 15 dose calculation methodology and and the conservative assumptions used for the SGTR dose calculation for chapter 15. The calculated 2 hour GIS thyroid dose was 15 Rem and the whole body dose was 0.585 Rem. The calculated 2 hour PIS thyroid dose was 43.7 Rem and the PIS whole body dose was 0.5 Rem.. Both of these calculated releases are well within the10CFR100 limit of 300 Rem..

ABB-CE is revising the text in section 19.4.4.3.3 to reflect the analysis discussed above. The insights in section 19.15..2.1.2 will be revised to address the importance of ASC with respect to procedures. Marked up copies of the revised CESSAR-DC pages are attached.

FIGURE 1

SYS 80+ : STEAM GENERATOR TUBE RUPTURE

RCS and SG PRESSURES

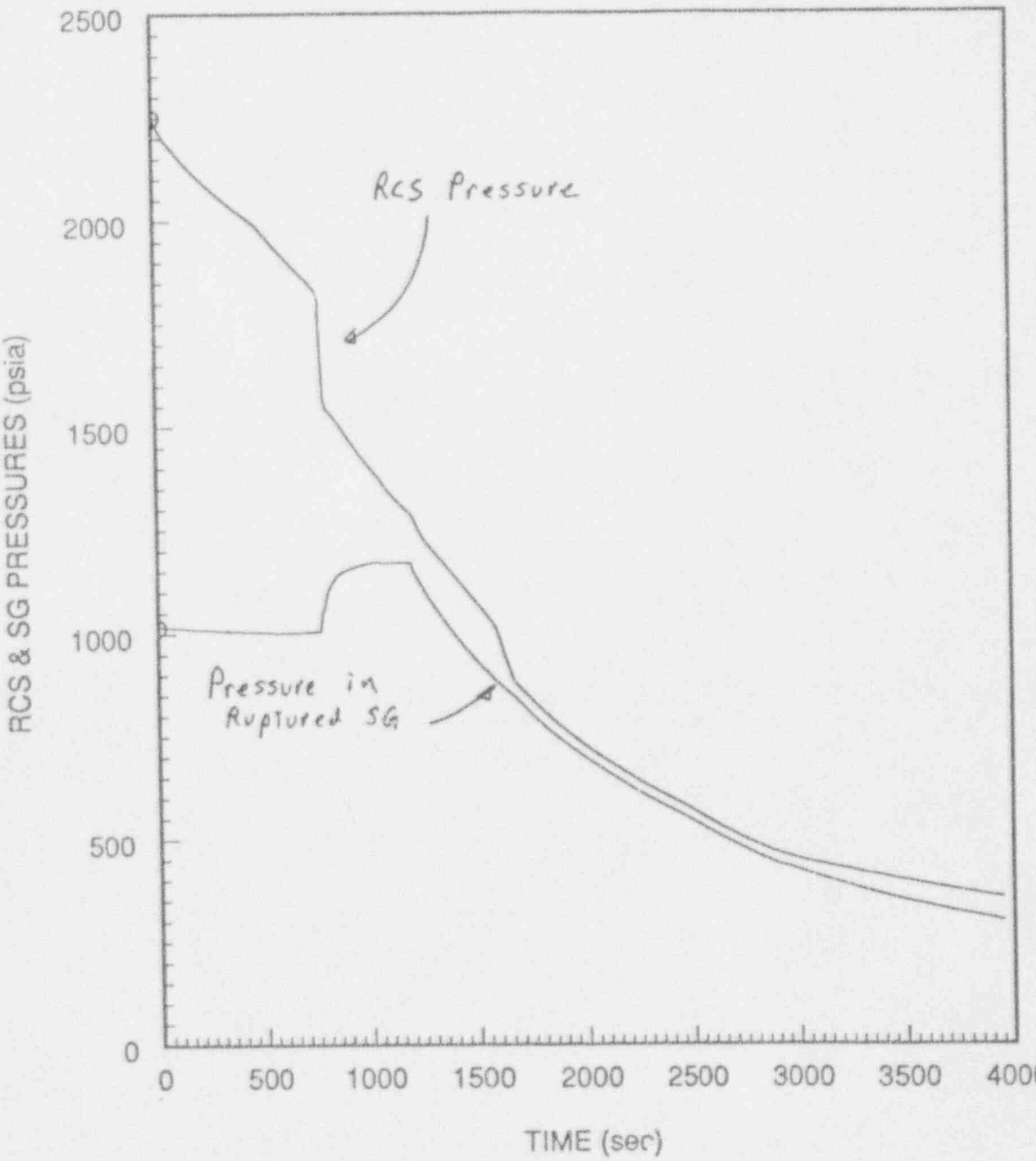


FIGURE 2

SYS 80+ : STEAM GENERATOR TUBE RUPTURE
STEAM GENERATOR LEVELS

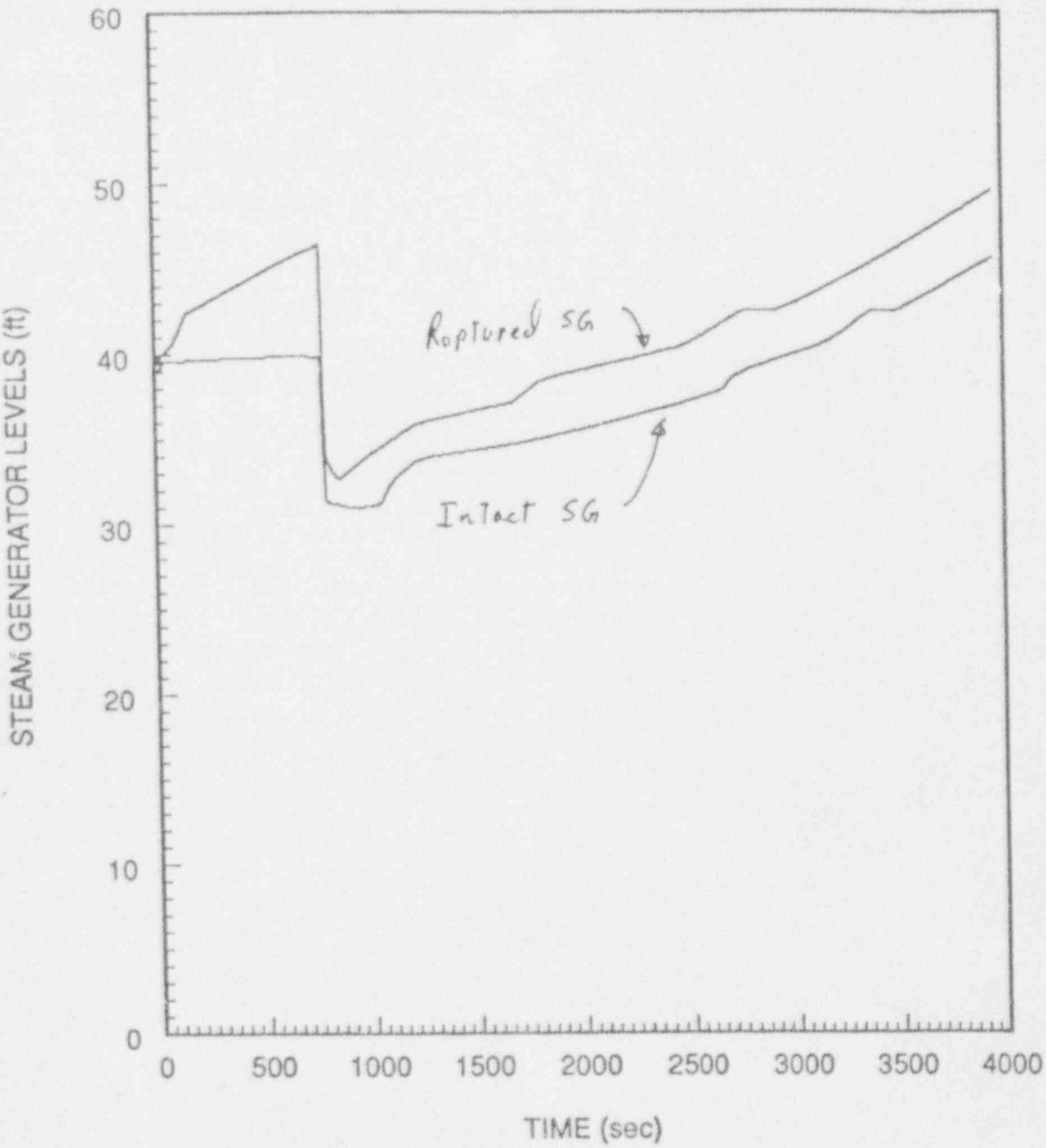


FIGURE 3

SYS 80+ : STEAM GENERATOR TUBE RUPTURE
RV UPPER HEAD LIQUID LEVEL

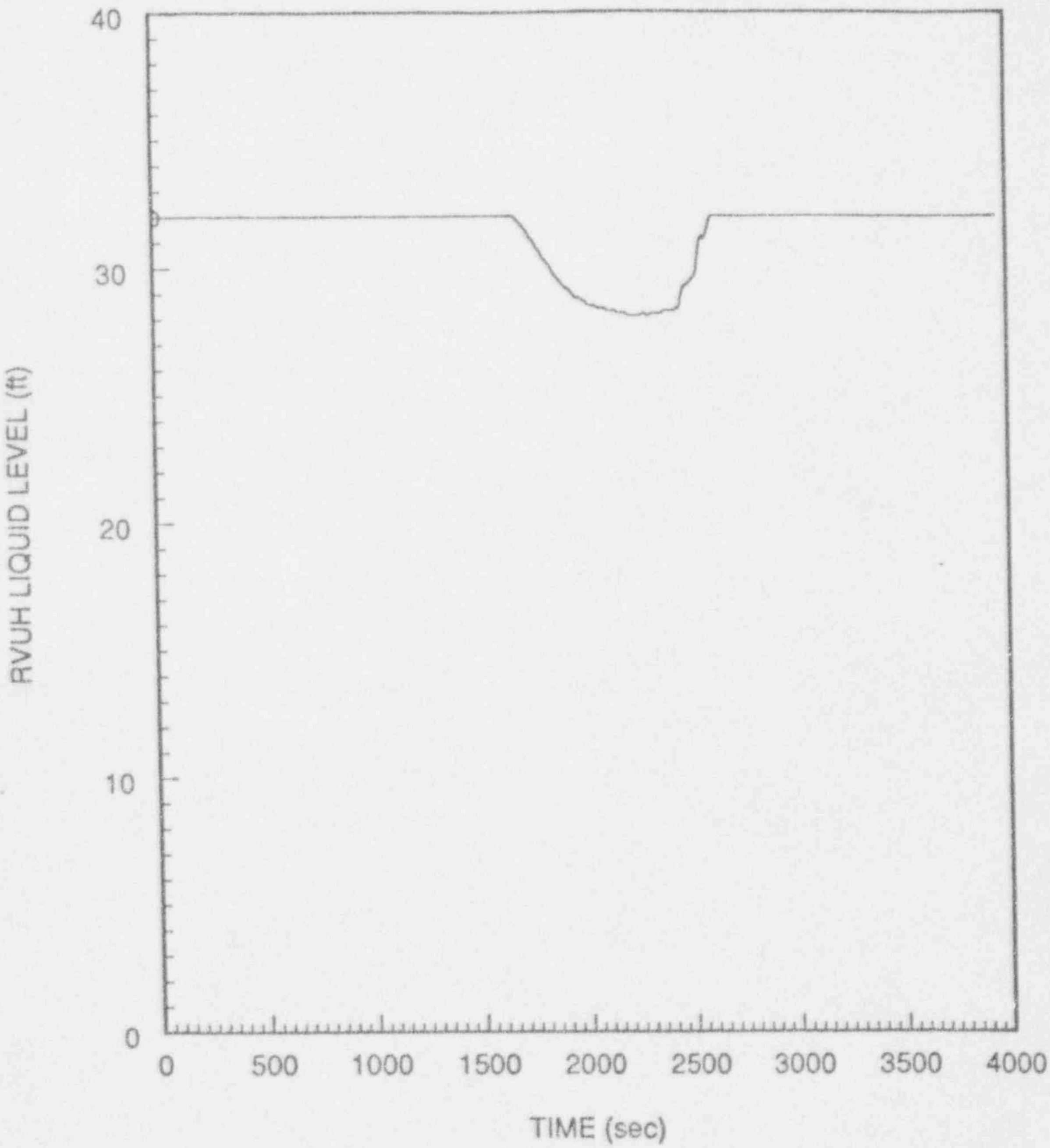


FIGURE 4

SYS 80+ : STEAM GENERATOR TUBE RUPTURE
COLD AND HOT LEG TEMPERATURE

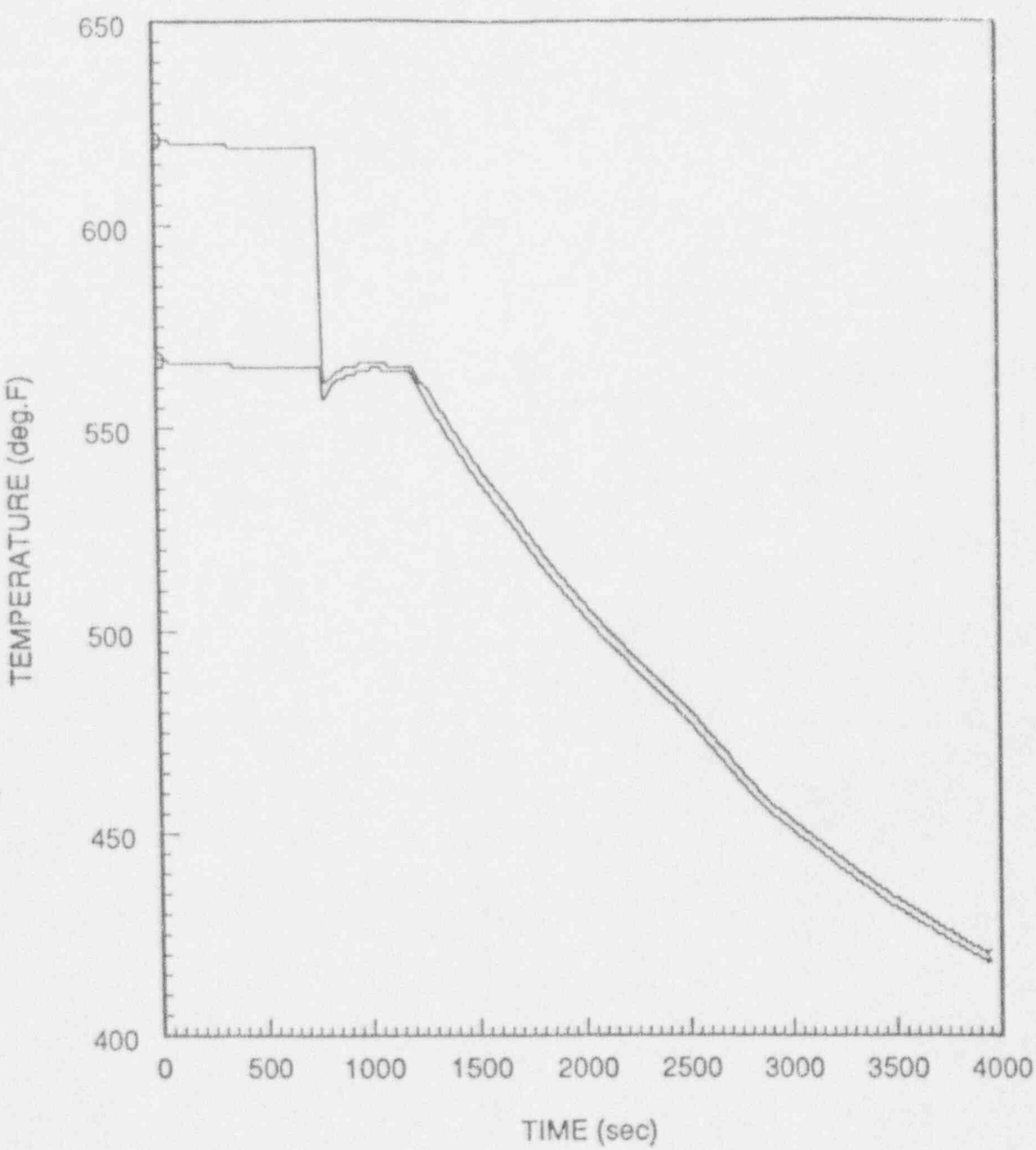


FIGURE 5

SYS 80+ : STEAM GENERATOR TUBE RUPTURE
STEAM GENERATOR STEAM FLOW

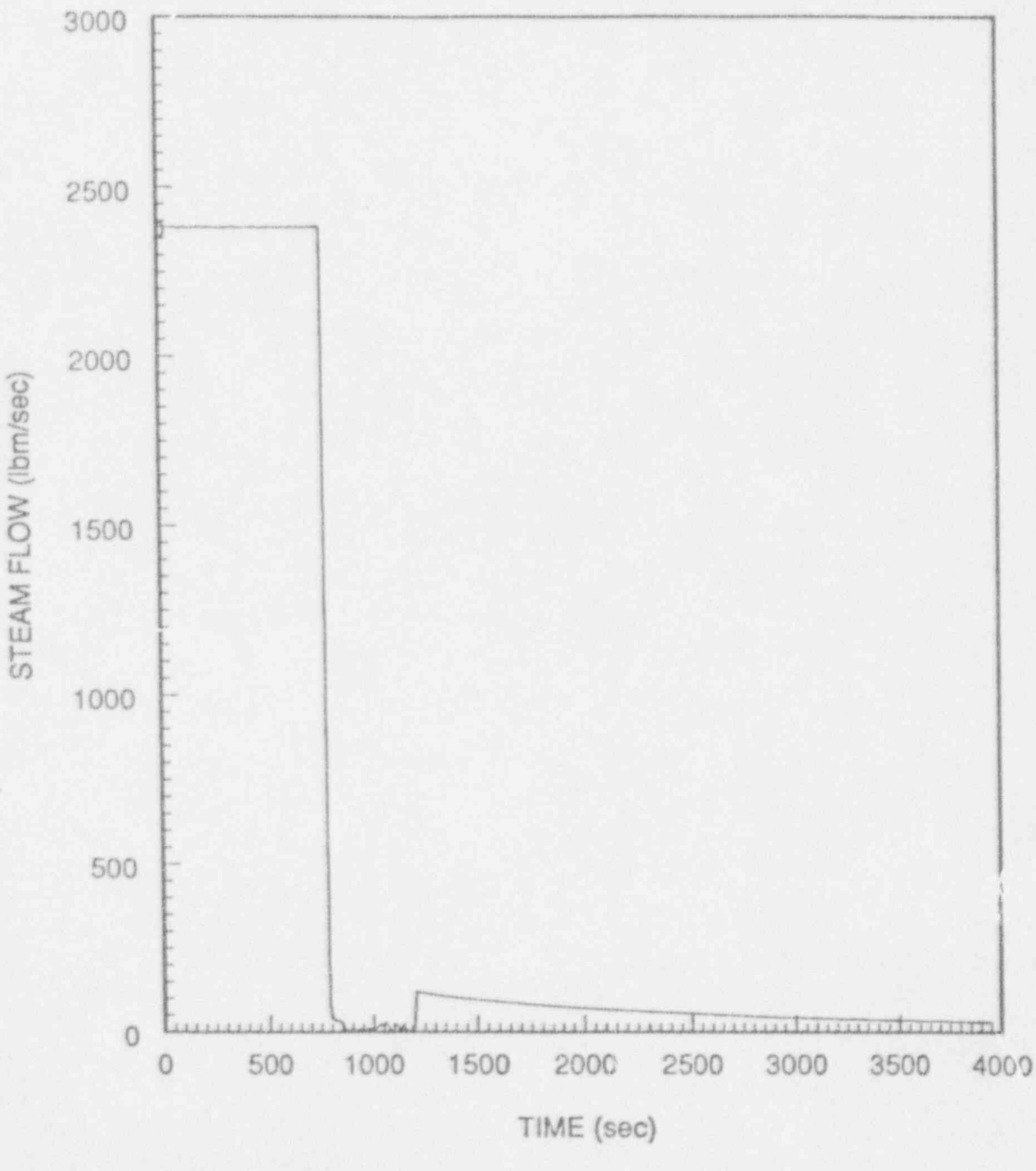


FIGURE 6

SYS 80+ : STEAM GENERATOR TUBE RUPTURE
STEAM GENERATOR PRESSURE

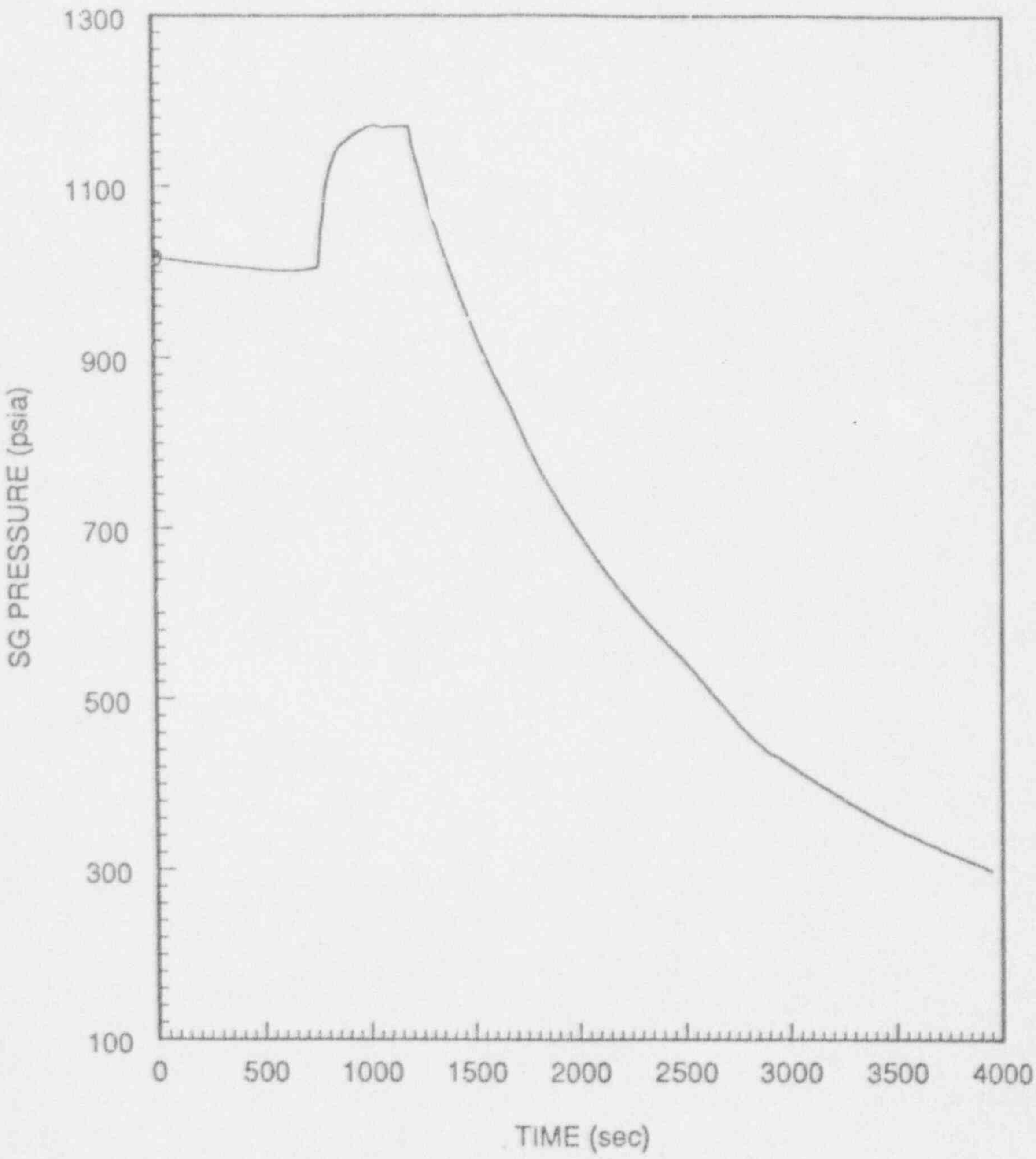


FIGURE 7

SYS 80+ : STEAM GENERATOR TUBE RUPTURE

PRESSURIZER PRESSURE

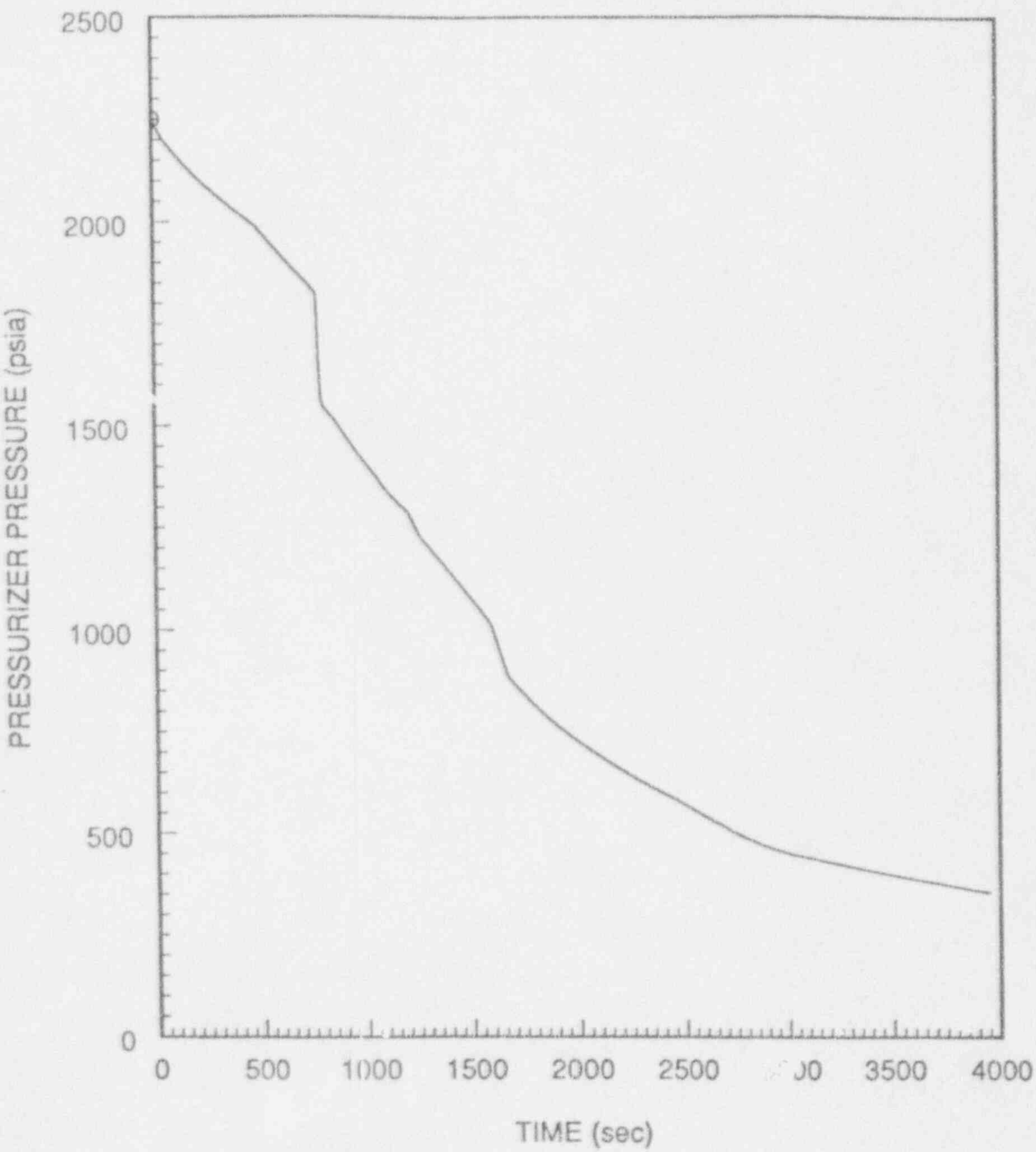
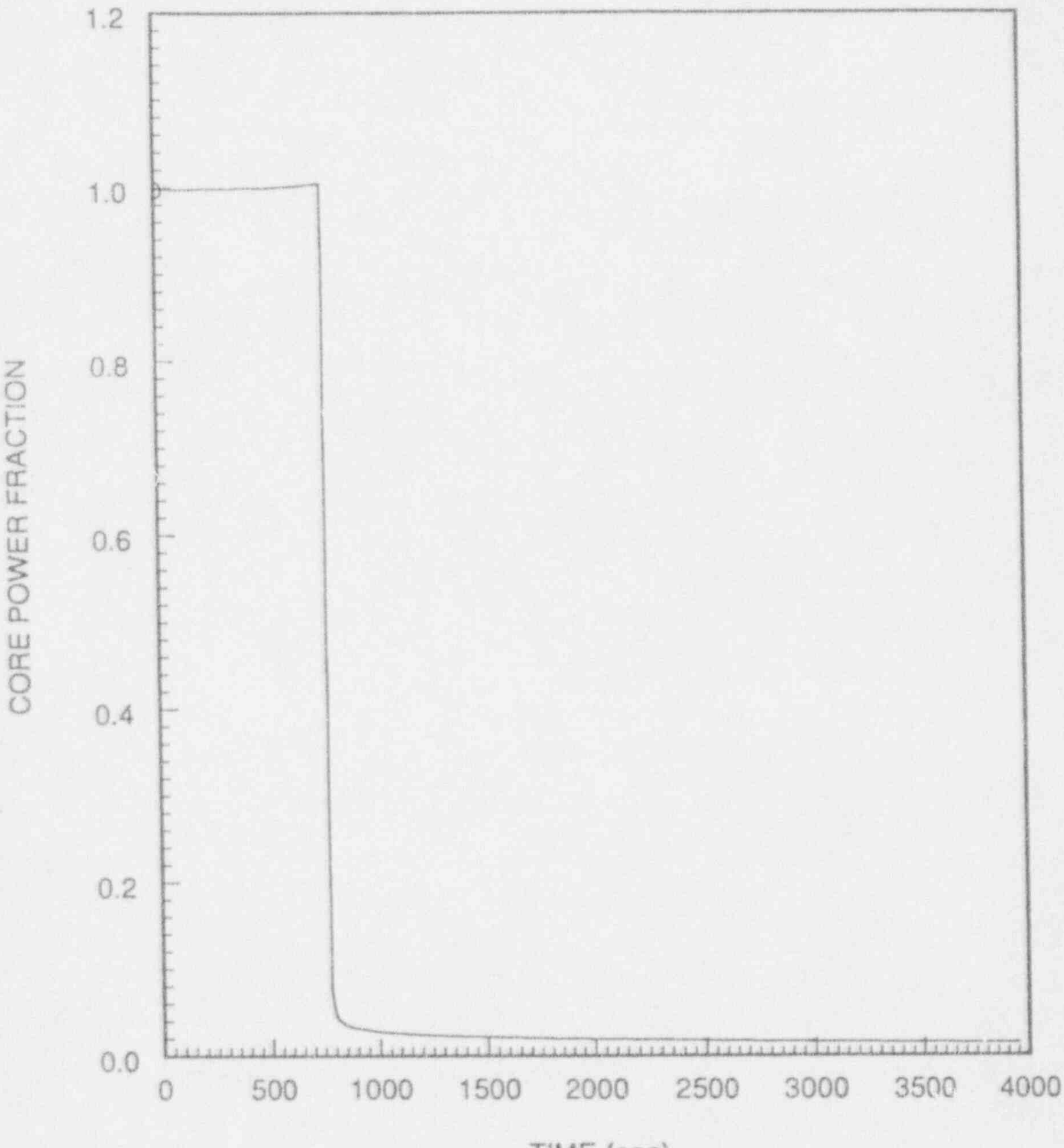


FIGURE 8

SYS 80+ : STEAM GENERATOR TUBE RUPTURE
CORE POWER FRACTION



There are four SIS pumps and success is defined as one of these four SIS pumps injecting water into the RCS from the In-Containment Refueling Water Storage Tank (IRWST).

If SIS fails to deliver flow to the RCS following a SGTR event, the Shutdown Cooling System (SCS) can be actuated to inject water for reactor coolant inventory control if the RCS is depressurized to a pressure below the SCS pump shutoff head. Sufficient time is available to remove heat via both steam generators using Aggressive Secondary Cooldown. This results in reduction of the RCS pressure to a point at which the SCS System can be placed in service. This is described in the following sections.

19.4.4.3.3 Aggressive Secondary Cooldown

For a SGTR with a failure of the SIS, the SCS System can be used to provide injection for the RCS inventory control if the primary system can be depressurized below the SCS pump shutoff head before the core is uncovered and core damage begins. Depressurization of the primary system is achieved by aggressively cooling the primary system using the secondary system. ~~Analyses for System 80 plants have shown that if the aggressive cooldown is initiated within approximately 15 minutes of a SGTR event, the SCS System can successfully provide RCS inventory control. A confirmatory analysis was performed for System 80+. This analysis is described in Section 19.4.5.1.3 (Small LOCA).~~

Replace with Insert A5G

~~Aggressive Secondary Cooldown is performed by delivering the emergency feedwater to both steam generators and removing steam from the steam generators using one of two Atmospheric Dump Valves (ADVs) on each generator.~~

~~The success criteria for Aggressive Secondary Cooldown are that each Emergency Feedwater System (EFWS) train must deliver the flow of one of its two pumps to its associated steam generator from its Emergency Feedwater Storage Tank (EFWST), one ADV on each steam generator must be available to remove steam, and all four of the safety injection tanks must inject water into the RCS during the primary side depressurization. (NOTE: The ruptured steam generator must be used for aggressive secondary cooldown.)~~

The main feedwater or the startup feedwater may be used for Aggressive Secondary Cooldown. However, it is not credited in the analysis.

The top logic for the element "Fail to Perform Aggressive Secondary Cooldown" is presented in Figure 19.4.4-2.

19.4.4.3.4 Shutdown Cooling System Injection

For a SGTR with a failure of the SIS, the SCS System can be used to provide injection for the RCS inventory control if the primary

Insert A-SG

An SGTR is essentially a small LOCA in which the RCS inventory is discharged to the ruptured SG rather than to the containment. Thus, at initiation, an SGTR provides indications equivalent to those of a standard small LOCA. The initial system and operator responses are essentially the same for an SGTR as they are for a small LOCA. Safety injection would be initiated for RCS inventory control, reactor trip would be initiated for reactivity control, and the emergency feedwater system (or startup feedwater system) would be actuated for secondary side heat removal. The operators would initially verify reactor trip, safety injection actuation and emergency feedwater actuation. They would also verify that a plant cooldown using both SGs was established. They would then begin the break identification procedure to determine whether the break was a small LOCA or an SGTR. Once it is determined that an SGTR has occurred and the ruptured generator is identified, the operators will cool the RCS to a temperature at which the MSSVs in the ruptured generator would not lift when the ruptured generator is isolated. This would typically occur approximately 30 minutes after the SGTR initiation. (Note: the emergency procedures stipulate that the ruptured generator is not to be isolated if the RCS pressure is less than 50 psi greater than the pressure in the ruptured generator. If the differential pressure is greater than or equal to 50 psi, the generator can be isolated when the other temperature and pressure conditions are established.) At this point, the ruptured SG would be isolated by closing the MSIVs, MFIVs and the blowdown valves for the ruptured SG. The operators would then proceed with an orderly cooldown and depressurization of the RCS using the intact SG. During this cooldown, the RCS pressure would be maintained just above the pressure in the ruptured SG to minimize leakage to the ruptured SG.

Failure of safety injection actuation (Loss of RCS Inventory Control) would be identified during the initial phase of the response to the event, prior to determining whether the event was a standard small LOCA or an SGTR. The operator response to Loss of RCS Inventory Control would be the same as for a small LOCA because at this point in time the operators have not determined that an SGTR has occurred. The operator responses would therefore be performed within the same time frame. Thus ASC would be initiated within approximately 15 minutes after the initiation of the event. While performing the ASC, the operators would continue with the break identification. Once it is determined that an SGTR has occurred and the ruptured generator is identified, the operators would isolate the ruptured generator if the temperature and pressure conditions were appropriate and the RCS pressure is 50 psi greater than the pressure in the ruptured generator. Once the ruptured generator is isolated, the operator would continue with the cooldown using only the good generator. If the ruptured generator can not be isolated, the operators would continue the cooldown using both generators until such time as isolation conditions can be established or shutdown cooling is established and the transient event is terminated. As for the small LOCA case, all four SITs were assumed available for injection to provide short term inventory control.

A transient analysis was performed to demonstrate that ASC could be successfully accomplished for an SGTR. The SGTR occurred at time zero, and HPSI failed. At 15 minutes, an aggressive secondary cooldown was initiated using both steam generator. Each generator recieved the flow from one EFW pump, and one ADV on each generator was used

for steam removal. Starting at 30 minutes, conditions in the the ruptured steam generator were monitored to determine if the ruptured generator could be isolated. As shown on Figure 19.4.4-4, the ruptured generator could not be isolated during the cooldown because the RCS pressure was less than 50 psi greater than the pressure in the ruptured generator. The low differential pressure did, however, minimize the primary to secondary leakage. Consistent with the emergency procedures, the operators continued the ASC using both steam generators. As is shown on Figures 19.4.4-4 through 19.4.4-11, the RCS was successfully cooled and depressurized using ASC while maintaining core covery and cooling. These plots indicate that there was no return to power following the cooldown.

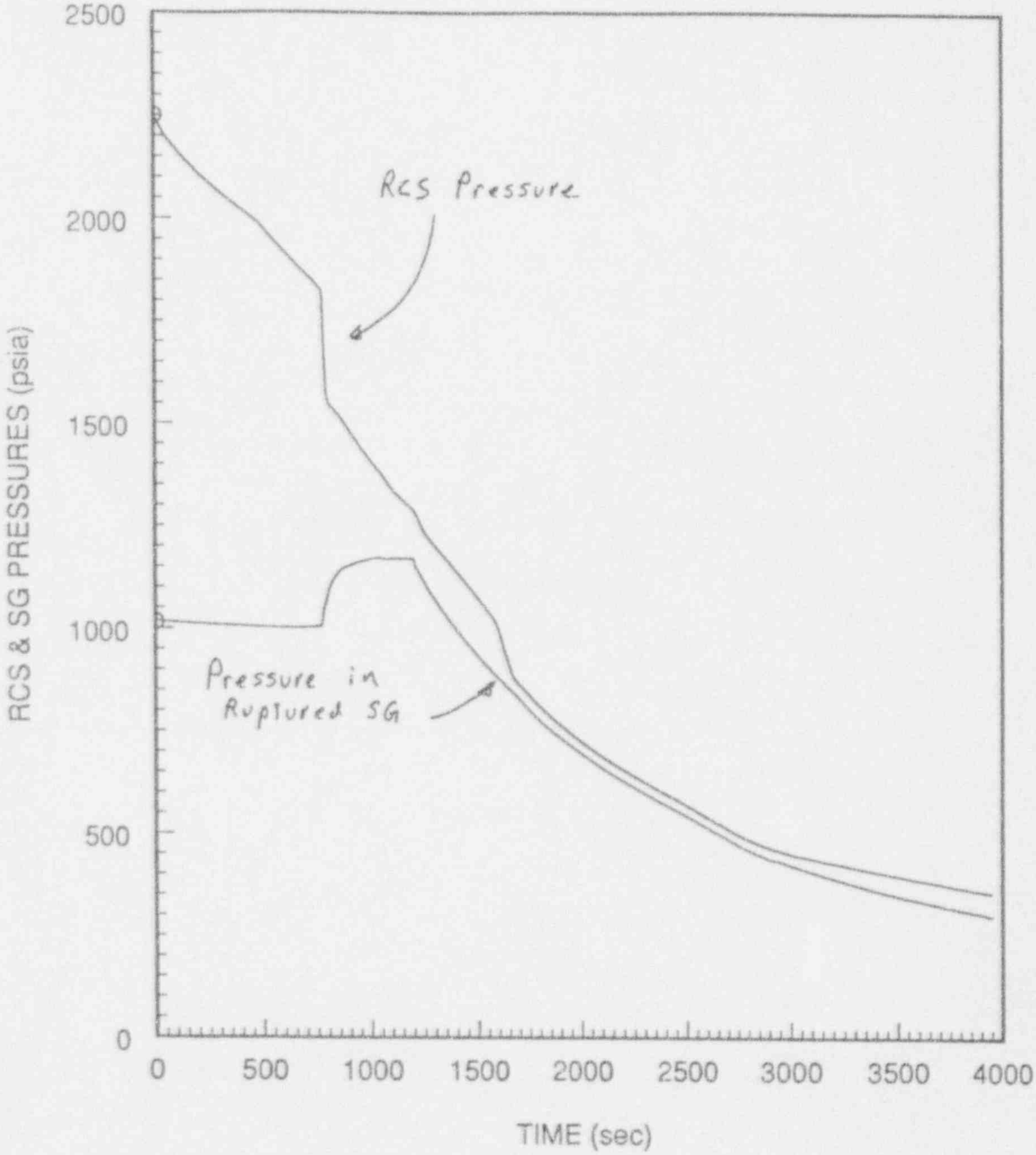
The radioactivity released during the transient was calculated using the standard chapter 15 dose calculation methodology and and the conservative assumptions used for the SGTR dose calculation for chapter 15. The calculated 2 hour GIS thyroid dose was 15 Rem and the whole body dose was 0.585 Rem. The calculated 2 hour PIS thyroid dose was 43.7 Rem and the PIS whole body dose was 0.5 Rem.. Both of these calculated releases are well within the10CFR100 limit of 300 Rem..

new

Figure 19.4.4-4

SYS 80+ : STEAM GENERATOR TUBE RUPTURE

RCS and SG PRESSURES

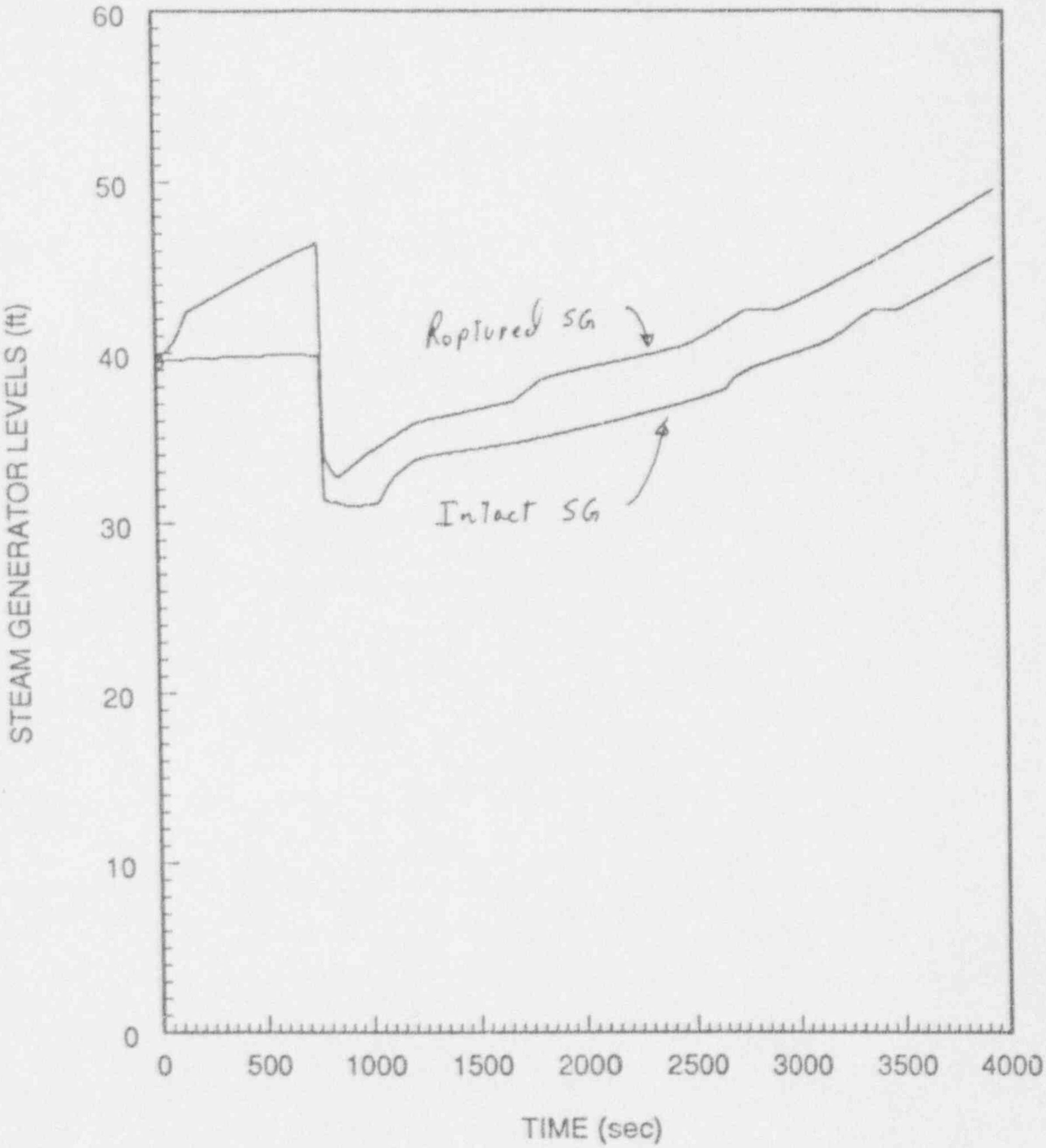


(New)

Figure 19.4.4-5

SYS 80+ : STEAM GENERATOR TUBE RUPTURE

STEAM GENERATOR LEVELS

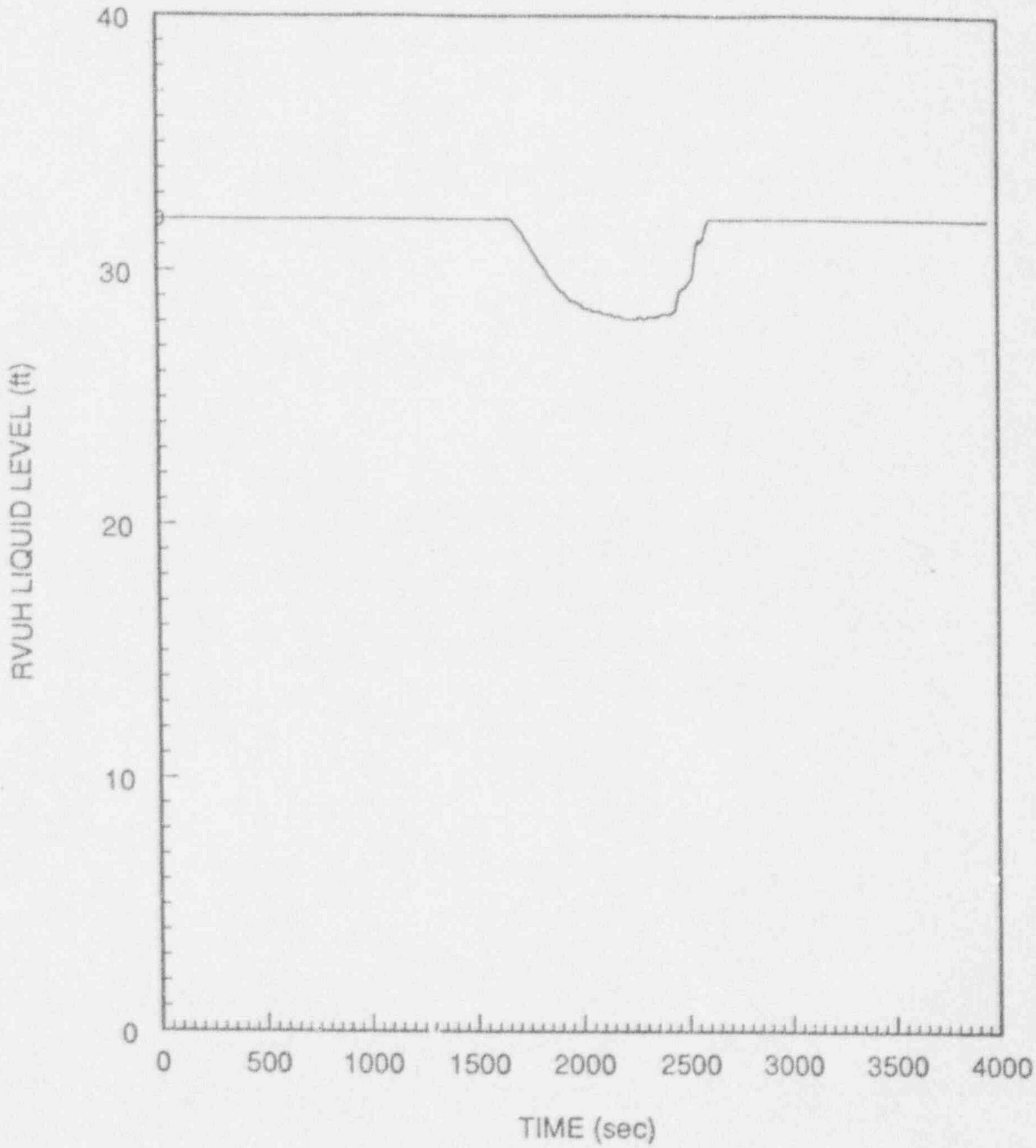


(new)

Figure 19.11.4-6

SYS 80+ : STEAM GENERATOR TUBE RUPTURE

RV UPPER HEAD LIQUID LEVEL

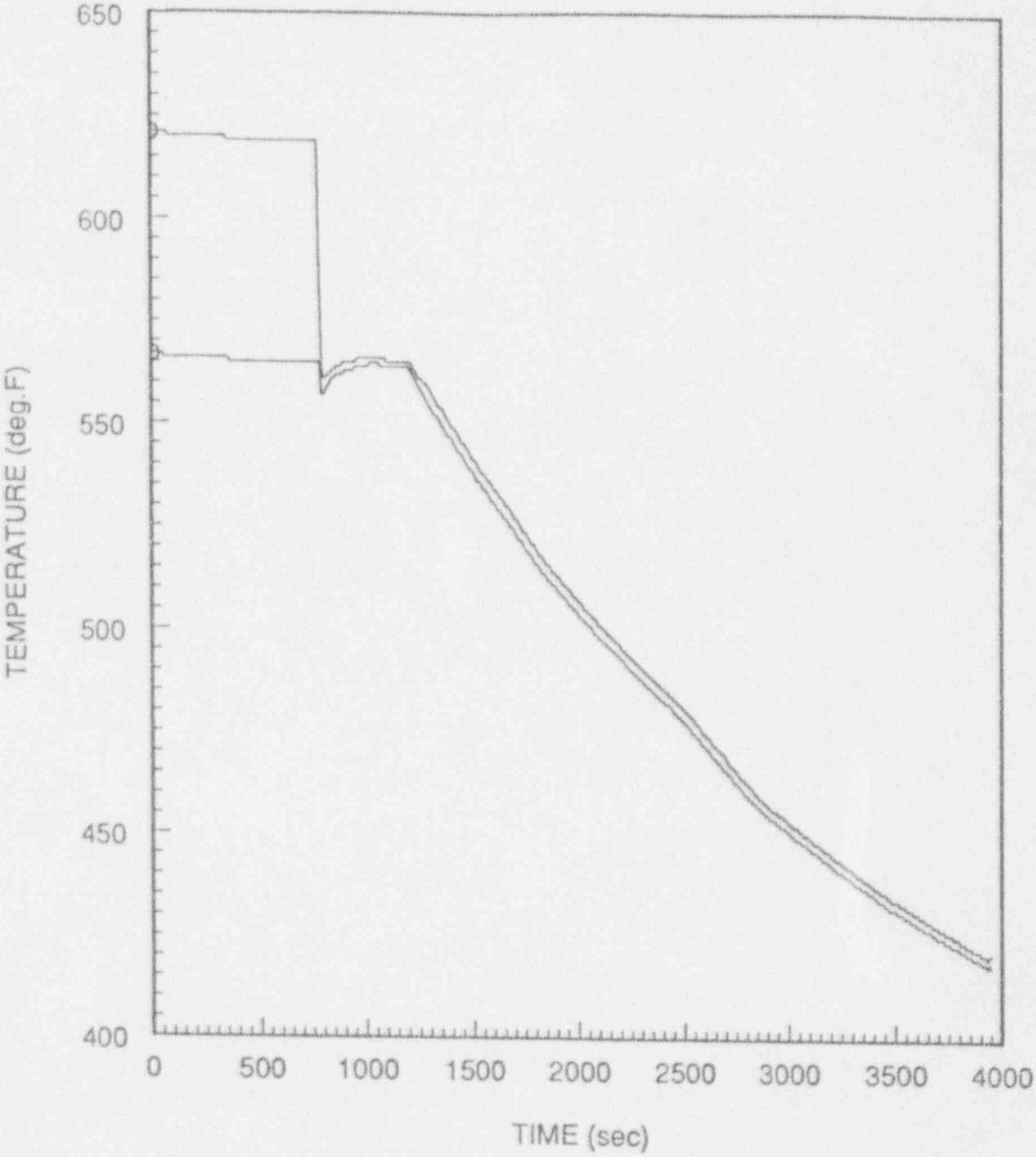


(New)

Figure 19.4.4-7

SYS 80+ : STEAM GENERATOR TUBE RUPTURE

COLD AND HOT LEG TEMPERATURE

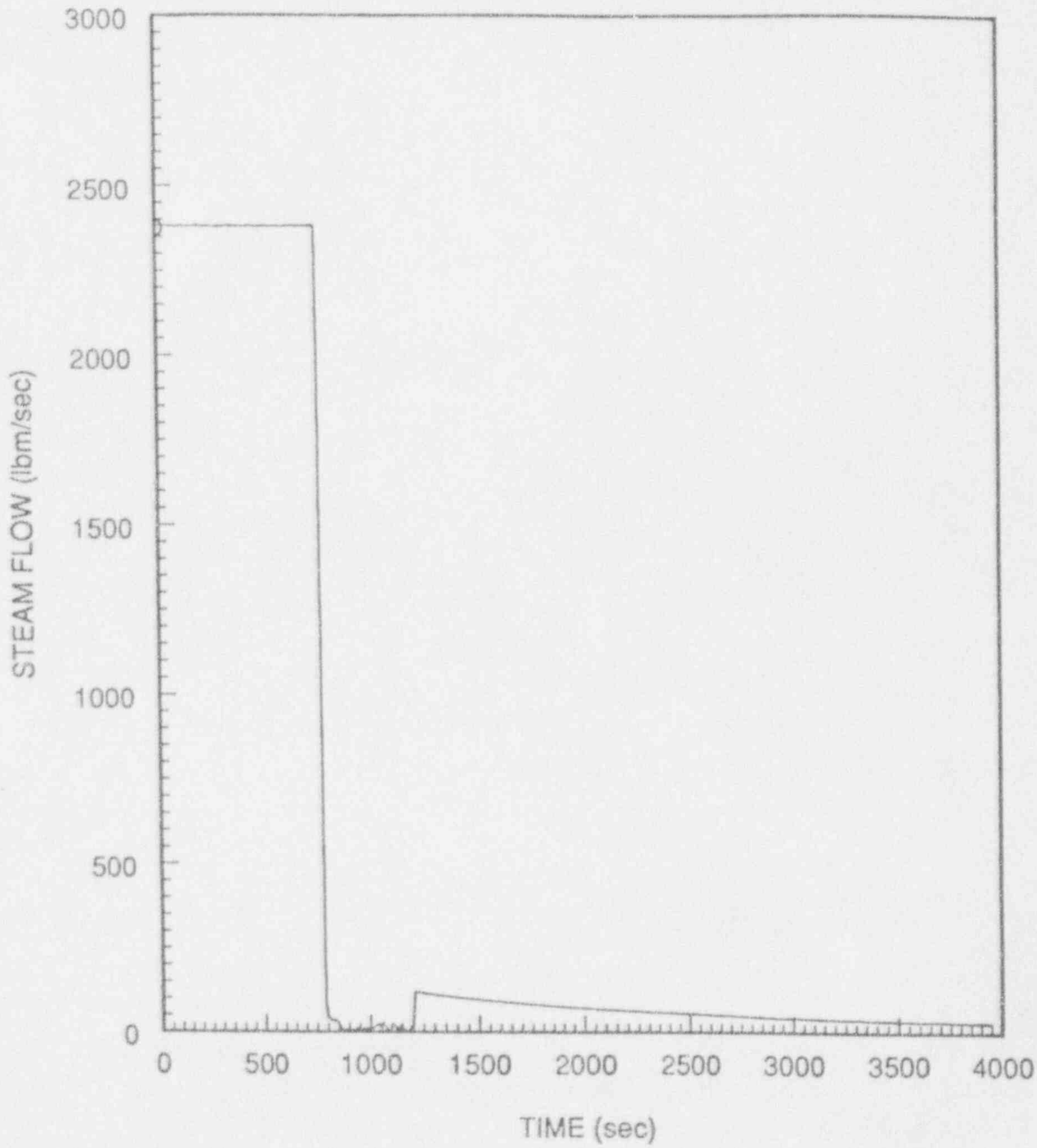


(New)

Figure 19.4.4-8

SYS 80+ : STEAM GENERATOR TUBE RUPTURE

STEAM GENERATOR STEAM FLOW

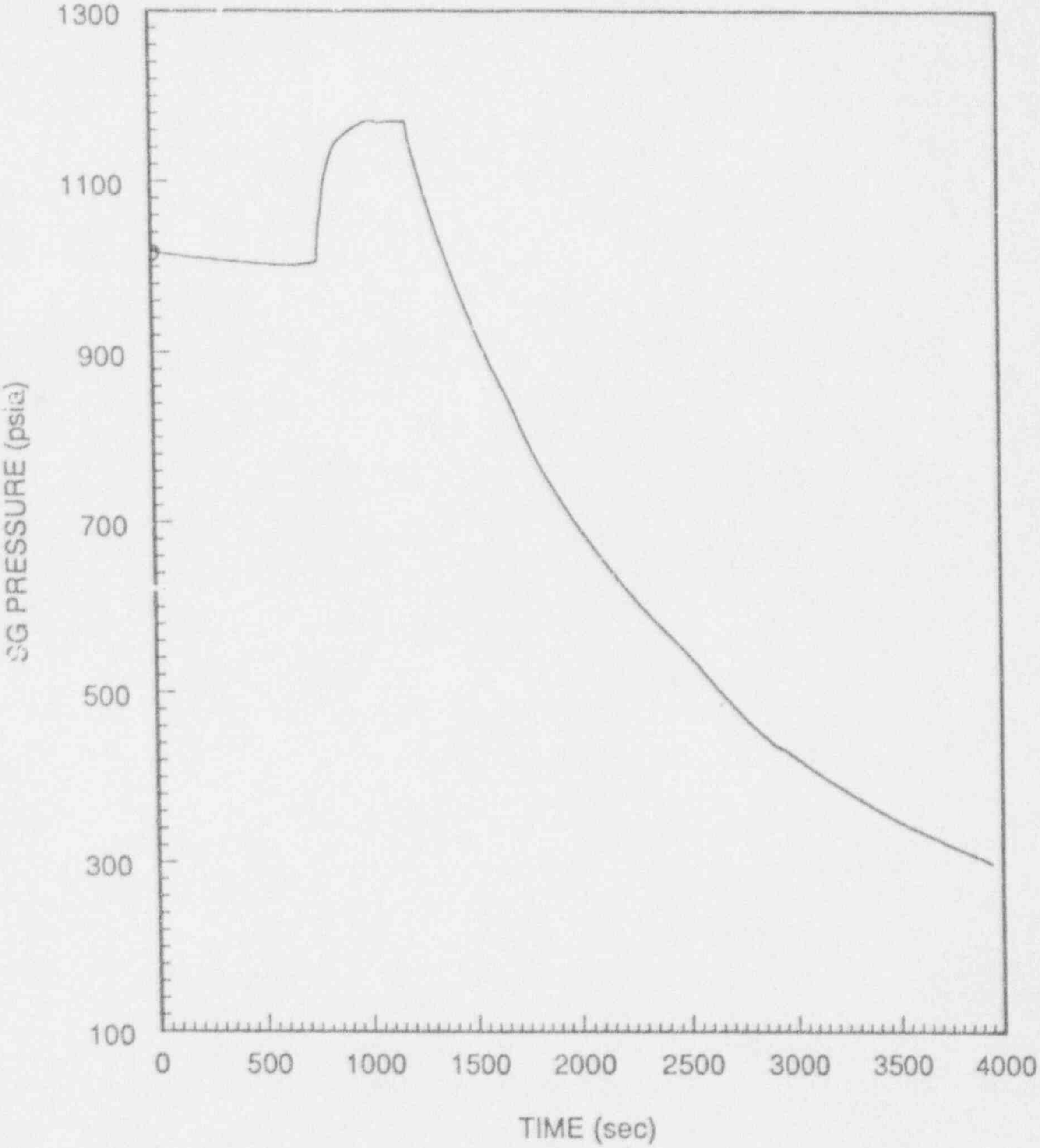


New

New

Figure 19.4.4-9

SYS 80+ : STEAM GENERATOR TUBE RUPTURE
STEAM GENERATOR PRESSURE

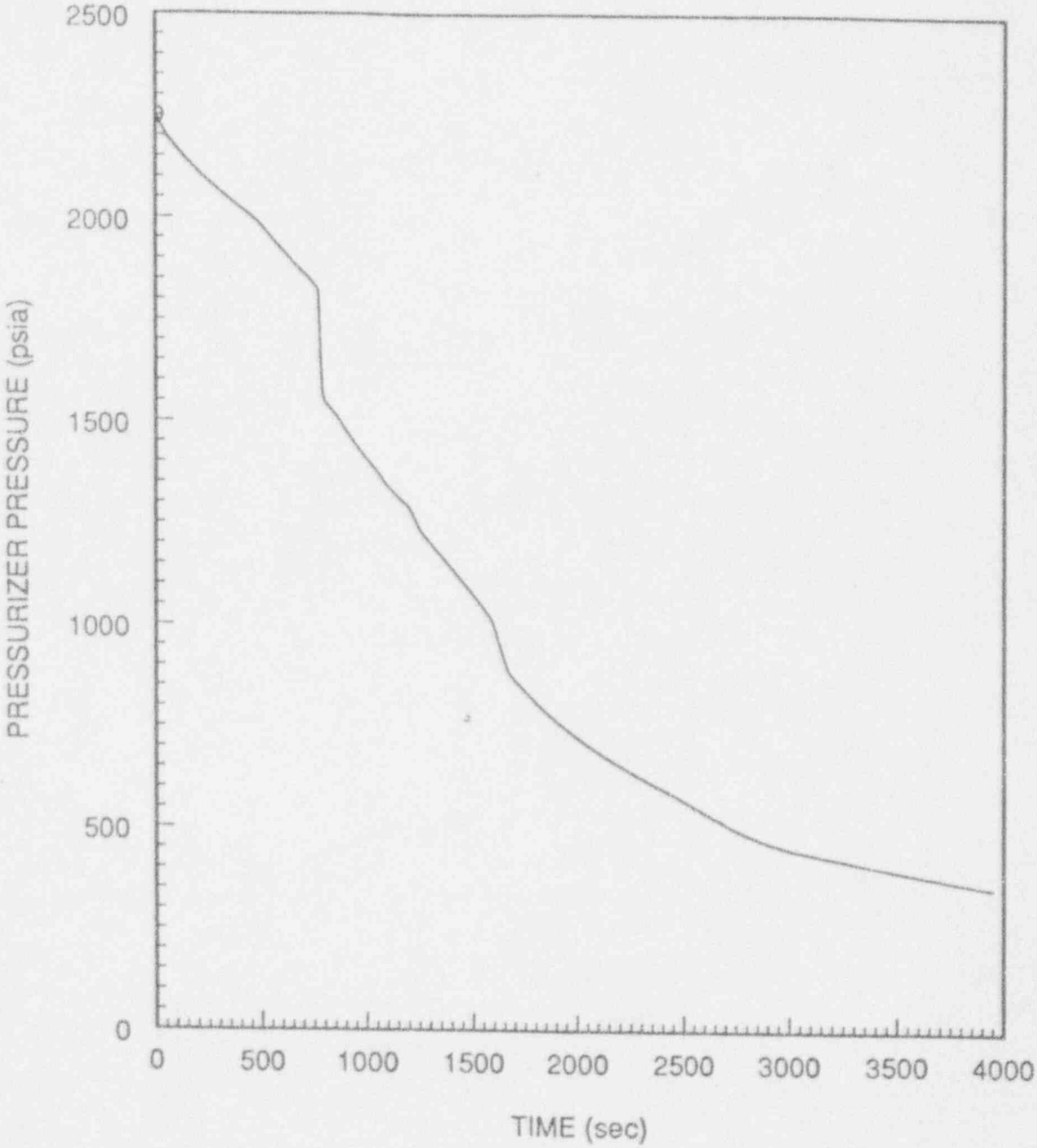


(new)

Figure 19.4.4-10

SYS 80+ : STEAM GENERATOR TUBE RUPTURE

PRESSURIZER PRESSURE

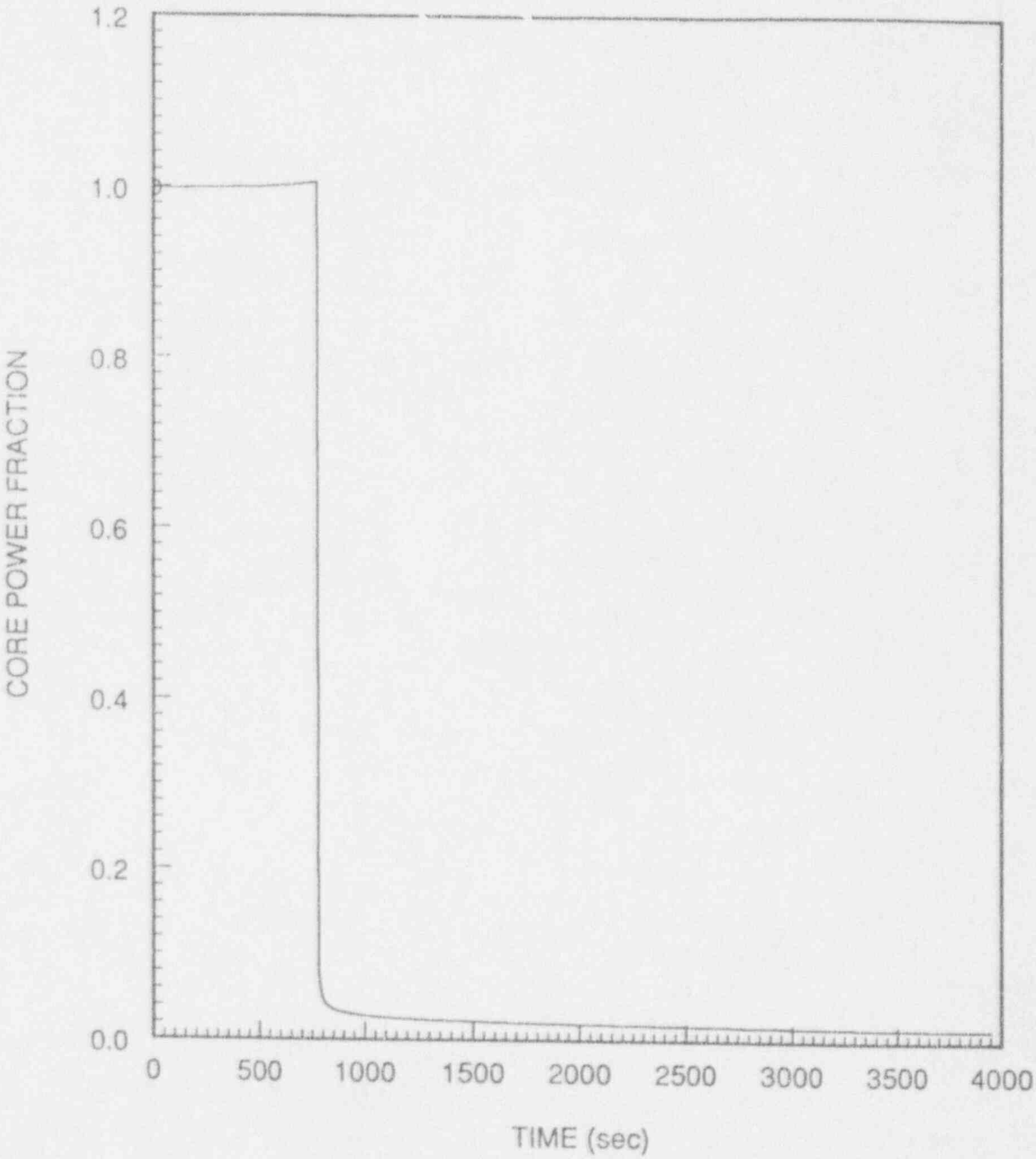


(New)

Figure 19.4.4-11

SYS 80+ : STEAM GENERATOR TUBE RUPTURE

CORE POWER FRACTION



Insert C-1

(This change is in response to open item 19.1.2.1.1.3-2)

Aggressive Secondary Cooldown (ASC) has a significant impact on the core damage frequency contribution for SGTR. Therefore, the emergency operating procedures for responding to an SGTR should specifically address ASC. This should include procedural steps for early identification of the failure of safety injection and specific steps for instituting the the cooldown by opening the ADVs and ensuring that EFW is being delivered to both generators. The procedure should also specify that even if ASC is in progress, the ruptured steam generator should be isolated when the RCS temperature and pressure have decreased to the point at which there is reasonable assurance that the MSSVs on the ruptured generator will not lift and the RCS pressure is at least 50 psi greater than the pressure in the ruptured steam generator. If the isolation conditions are met, the procedures should direct the operator to continue the ASC using only the good generator. Otherwise, the procedures should direct the operators to continue the ASC until low pressure injection can be established and the isolation conditions can be established or until shutdown cooling can be established. The procedures should also include all steps needed to align the SCS pumps for injection once the appropriate temperature and pressure limits have been reached.

Insert D-1

(This change is in response to open item 19.1.2.1.1.3-2)

Aggressive Secondary Cooldown (ASC) has a significant impact on the core damage frequency contribution for small LOCA. Therefore, the emergency operating procedures for responding to a small LOCA should specifically address ASC. This should include procedural steps for early identification of the failure of safety injection and specific steps for instituting the the cooldown by opening the ADVs and ensuring that EFW is being delivered to both generators. The procedures should also include all steps needed to align the SCS pumps for injection once the appropriate temperature and pressure limits have been reached.

FAX

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9424 Files
9612 files

DATE: September 23, 1993

NUMBER: OPS-93-0790

SUBJECT: Transmittal of PRA Operational Assumptions/Insights for CESSAR-DC

I am sending you a list of additional PRA assumptions/insights for CESSAR-DC. This is in response to our commitment made during the meeting on September 2, 1993 and in conjunction with the telephone conversation with D. J. Finnicum on September 17, 1993. The list includes operational assumptions/insights for power and shutdown operations, and severe accident conditions.

/RAI9308.wp1

CESSAR-DC PRA ASSUMPTION / INSIGHTS CROSS-REFERENCE

CESSAR-DC Section/Table/Figure	System/ Structure	No.	ASSUMPTION/INSIGHTS	S/R	REFERENCE
19.4.1.3.2, 19.4.2.3.2, 19.4.3.3.2, 19.4.4.5.2, 19.4.5.3.2, 19.4.6.3.2, 19.4.7.3.2, 19.4.8.2.3.2, 19.4.9.4.2, 19.4.10.3.2, 19.4.11.3.2, 19.4.12.3.2, 19.4.13.4.2	BG, BN	1.	Following containment failure due to failure of long-term containment heat removal some IRWST inventory (in the form of steam flashing) will be lost. The operator can align the CVCS to replenish the IRWST. The operator actions involved in replenishing the IRWST inventory include identifying the need to replenish the IRWST inventory, aligning the appropriate manual valves in the CVCS to transfer the contents of the Boric Acid Storage Tanks to the IRWST , and starting the Boric Acid Makeup Pumps to complete the transfer. Alignment procedures will be available and the operators have some experience in performing the alignment.	S	Specified by PRA
19.4.1.3.2, 19.4.2.3.2	BB, BH	2.	Following a large or a medium LOCA, hot leg and direct vessel injection (DVI) must be established to prevent crystallization. The basic operator actions involved in establishing simultaneous hot leg and DVI include throttling SI flow to the DVI nozzle in the SI trains that will provide the hot leg injection by closing the injection valves and establishing flow to the hot legs by opening the appropriate valves. Establishing simultaneous hot leg and DVI is covered in the Post-LOCA recovery procedures, and the operators have been trained in the use of the procedures.	S	Specified by PRA
19.4.1.3.2, 19.4.2.3.2, 19.4.3.3.2, 19.4.4.5.2, 19.4.5.3.2, 19.4.6.3.2, 19.4.7.3.2, 19.4.8.2.3.2, 19.4.9.4.2, 19.4.10.3.2, 19.4.11.3.2, 19.4.12.3.2, 19.4.13.4.2	BC, BH, AL, BK	3.	The operator can manually initiate the ESF actuation signals from the control room if the signals were not initiated automatically.	S	Specified by PRA
19.4.3.3.2, 19.4.4.5.2	BH, BC, AL, AB	4.	If the Safety Injection System (SIS) fails to provide RCS inventory control following a small LOCA or a Steam Generator Tube Rupture (SGTR), an aggressive cooldown of the secondary side can be performed so that the Shutdown Cooling System (SCS) can be aligned to provide RCS inventory control. The operator actions involved in performing an aggressive secondary side cooldown include identifying that an aggressive cooldown is required, ensuring that each steam generator has at least one EFW pump delivering flow to it, and that at least one ADV is open on each steam generator. The operator actions are addressed in the small LOCA and SGTR recovery procedures, and all actions can be performed from the control room. The operators have 15 minutes from the loss of safety injection to initiate aggressive secondary side cooldown.	S	Specified by PRA

CESSAR-DC PRA ASSUMPTION / INSIGHTS CROSS-REFERENCE

CESSAR-DC Section/Table/Figure	System/ Structure	No.	ASSUMPTION/INSIGHTS	S/R	REFERENCE
19.4.3.3.2, 19.4.4.5.2	BC, BN	5.	If the Safety Injection System (SIS) fails to provide RCS inventory control following a small LOCA or a Steam Generator Tube Rupture (SGTR), an aggressive cooldown of the secondary side can be performed so that the Shutdown Cooling System (SCS) can be aligned to provide RCS inventory control. The operator actions involved in aligning the SCS for injection include opening the cross-connect valves between the SCS pump suction line and the containment spray pump suction line from the IRWST, verifying that the SCS suction valves are closed, and recognizing when SCS entry conditions are met so that SCS pumps can be started and the discharge valves can be opened. It is assumed that the operator actions are addressed in procedures, and the operators are trained in the use of the procedures.	S	Specified by PRA
19.4.3.3.2, 19.4.4.5.2, 19.4.5.3.2, 19.4.6.3.2, 19.4.7.3.2, 19.4.8.2.3.2, 19.4.9.4.2, 19.4.10.3.2, 19.4.11.3.2, 19.4.12.3.2, 19.4.13.4.2	BH	6.	Following containment failure due to failure of long-term containment heat removal the containment pressure will rapidly decrease to atmospheric pressure and the IRWST inventory will start flashing as it re-establishes saturated equilibrium at atmospheric condition. As a result of the flashing, the safety injection pumps will lose NPSH and trip. Once equilibrium is re-established, the safety injection pumps can be restarted. To accomplish this action, the operator must recognize that the safety injection pumps have tripped and can be restarted. He must dispatch equipment operators to the safety injection pump rooms to bleed the pumps. When the control room operator receives notification that the pumps have been bled, the safety injection pumps can then be restarted from the control room. It is assumed that the bleed of the safety injection pumps will be covered in the maintenance procedures.	S	Specified by PRA
19.4.3.3.2, 19.4.4.5.2, 19.4.5.3.2, 19.4.6.3.2, 19.4.7.3.2, 19.4.8.2.3.2, 19.4.9.4.2, 19.4.10.3.2, 19.4.11.3.2, 19.4.12.3.2, 19.4.13.4.2	BC	7.	After shutdown cooling entry conditions are met, the operator must align the SCS for long-term cooling operation. This involves opening the SCS suction valves to take suction from the RCS, opening the SCS discharge valves, and starting the SCS pumps. The operator must also verify that the SCS pumps and SCS heat exchangers are being supplied with component cooling water. It is assumed that these actions are covered by procedures and the operators are trained in the use of the procedures.	S	Specified by PRA
19.4.3.3.2, 19.4.4.5.2, 19.4.5.3.2, 19.4.6.3.2, 19.4.7.3.2, 19.4.8.2.3.2, 19.4.9 4.2, 19.4.10.3.2, 19.4.11.3.2, 19.4.12.3.2, 19.4.13.4.2	AL, AP	8.	If the SCS cannot be aligned for long-term cooling, the Emergency Feedwater System (EFWS) can be used for continued decay heat removal. The inventory of the Emergency Feedwater Storage Tanks (EFWSTs) can support decay heat removal for approximately 16 hours, beyond which makeup to the EFWST must be provided from the condensate storage tank. The operator actions involved in aligning the Condensate Storage Tank (CST) to the EFWST involve recognizing the need for EFWST makeup, dispatching an operator to open the inlet valves to the EFWSTs and to open the CST discharge valve to the EFWSTs. These actions are assumed to be covered by procedures and the operator is trained to establish EFWST makeup.	S	Specified by PRA

CESSAR-DC PRA ASSUMPTION / INSIGHTS CROSS-REFERENCE

CESSAR-DC Section/Table/Figure	System/ Structure	No.	ASSUMPTION/INSIGHTS	S/R	REFERENCE
19.4.3.3.2, 19.4.4.5.2, 19.4.5.3.2, 19.4.6.3.2, 19.4.7.3.2, 19.4.8.2.3.2, 19.4.9.4.2, 19.4.10.3.2, 19.4.11.3.2, 19.4.12.3.2, 19.4.13.4.2	AL, BC	9.	If the SCS is successfully aligned and then fail during long-term cooling, it is assumed that the EFW pumps can be restarted to continue decay heat removal. It is also assumed that the EFW pumps are secured when SCS is successfully aligned. The operator actions involved in restarting the EFW pumps include recognizing that the SCS is not removing decay heat and then restarting the necessary number of EFW pumps to satisfy decay heat removal. These actions are assumed to be covered by procedures and the operator is trained in the use of the procedures.	S	Specified by PRA
19.4.3.3.2, 19.4.4.5.2, 19.4.5.3.2, 19.4.6.3.2, 19.4.7.3.2, 19.4.8.2.3.2, 19.4.9.4.2, 19.4.10.3.2, 19.4.11.3.2, 19.4.12.3.2, 19.4.13.4.2	BB	10.	If decay heat cannot be removed by the EFWS and the SCS, once through cooling or "Feed and Bleed" must be initiated. The operator actions involved in establishing "Feed and Bleed" include determining the need for "Feed and Bleed" and opening the Rapid Depressurization Valves. The operator must also confirm that the Safety Injection System has actuated automatically, and if not, must actuate it manually. To be successful, "Feed and Bleed" must be initiated at or before the time at which the primary safety valves lift. For a loss of feedwater event with no secondary side heat removal, the steam generators will dry out and the primary safety valves will lift within 45 to 60 minutes. Therefore, in order for "Feed and Bleed to be successful the operator must act within this time frame. The operator actions to initiate "Feed and Bleed" are assumed to covered by procedures and the operator is trained in the use of the procedures.	S	Specified by PRA
19.4.4.5.2, 19.4.13.4.2	BN, BG, BB	11.	If the ruptured steam generator cannot be isolated due to loss of RCS pressure control, the RCS must be depressurized using secondary side cooling. This is a slow process which may deplete the IRWST inventory and the IRWST inventory must then be replenished from the Boric Acid Storage Tanks (BASTs) to complete the cooldown. The operator actions involved in replenishing the IRWST inventory include identifying the need to replenish the IRWST inventory following a SGTR, aligning the appropriate manual valves in the CVCS to transfer the contents of the BASTs to the IRWST, and starting the boric acid makeup pumps to complete the transfer. The operator actions are assumed to be covered in the SGTR recovery procedures and the operator has some experience in performing the alignment to replenish the IRWST inventory.	S	Specified by PRA
19.4.4.5.2, 19.4.13.4.2	BH, BB	12.	Following a SGTR, RCS pressure control must be established in sufficient time to permit plant cooldown and stopping the leak before the inventory of the IRWST is depleted. The operator must accomplish two primary tasks to establish RCS pressure control. First, the safety injection pumps must be throttled to prevent them from "holding up" the RCS pressure near the shutoff head of the safety injection pump. Secondly, the RCS pressure must be reduced while decay heat is being removed by the secondary side. RCS pressure can be controlled using the Pressurizer Spray System, or, if spray system is unavailable due to loss of offsite power or mechanical failure, the Reactor Gas Vent System can be used. The operator actions involved in RCS pressure control include recognizing the need to throttle the safety injection pumps, and establishing a pressurizer spray or vent path by opening the appropriate valves from the control room. It is assumed that the operator actions to establish RCS pressure control is covered in the SGTR recovery procedures and the operator is trained in the use of the procedures.	S	Specified by PRA

CESSAR-DC PRA ASSUMPTION / INSIGHTS CROSS-REFERENCE

CESSAR-DC Section/Table/Figure	System/ Structure	No.	ASSUMPTION/INSIGHTS	S/R	REFERENCE
19.4.13.4.2	BB, BG	13.	Following an ATWS event, the RCS pressure may remain high enough so that the safety injection pumps cannot be used to borate the RCS. In this case, a charging pump may be used to deliver boron to the RCS for reactivity control. The operator actions involved in delivering boron to the RCS via a charging pump include verifying that at least one charging pump is operating, isolating the VCT by closing the VCT discharge valve, establishing a flow path from the Boric Acid Storage Tank to the charging pump suction by opening the appropriate valves and starting the boric acid makeup pumps if necessary. All these operator actions are assumed to be covered in the procedures for reactivity control. It is also assumed that the operator is trained in the use of the procedures.	S	Specified by PRA
19.4.4.3.8	AB	14.	Following the isolation of the ruptured or most affected steam generator, the level in the steam generator will increase if there is a large pressure differential between the primary and secondary sides. Without operator action, the ruptured or most affected steam generator could overfill and cause the Main Steam Safety Valves (MSSVs) to lift. The operator actions involved in preventing the ruptured or most affected steam generator from overfilling include recognizing the need to steam or drain the ruptured generator, establishing a path to relieve steam by opening the appropriate ADVs, or establishing a path to reduce the level by opening the appropriate blowdown valves generator. These operator actions are assumed to be covered in the SGTR recovery procedures and the operator is trained in the use of the procedures.	S	Specified by PRA
19.4.14.1	BC	15.	During startup operations, the SCS suction isolation valves inside the containment are closed by the operator and are verified closed when the RCS pressure increases to a certain limit. It is assumed that if any SCS suction valve inside the containment should fail to close, startup operations are suspended until the affected valve is closed. It is also assumed that the closure and the verification that the SCS valves are closed are covered in the startup procedures.	S	Specified by PRA
19.4.14.2	BH	16.	Redundant check valves are provided in the DVI lines. It is assumed that the check valves closet to the RCS are verified to have reseated after each refueling or prior to returning to power following each cold shutdown. If the back flow through any of these checks (i.e., check valves closes to the RCS) exceeds the limits of the technical specifications, it is assumed that the operator will shutdown the plant within the next 30 hours. Thus the second check valve may be exposed to RCS operating pressure for no more than 30 hours.	S	Specidied by PRA
19.4.1.4, 19.4.2.4, 19.4.3.4, 19.4.4.6, 19.4.5.4, 19.4.6.4, 19.4.7.4, 19.4.8.2.4, 19.4.8.3.3, 19.4.9.5, 19.4.10.4, 19.4.11.4, 19.4.12.4, 19.4.13.5	MD	17.	Following a plant trip, offsite power is the preferred source of power for plant equipment. If offsite power is lost, the emergency diesel generators or the alternate AC power source will provide power to shutdown the plant. Depending on the initiating event and the systems affected, various elapse time periods are considered for restoring offsite power. It is assumed that the emergency operating procedures will provide guidance for restoring offsite power with and without the emergency diesel generators being available. It is also assumed that the operators are fully trained in the use of the procedures.	S	Specified by PRA
19.6.3.3.2.1	EG, SA	1.	The non-essential component cooling water (CCW) loads are isolated during emergency operating conditions. It is assumed that if SIAS fails to automatically isolate the non-essential loads the operator can manually close the appropriate valves from the control room. The operator actions involved in isolating the non-essential CCW loads include verifying that SIAS has closed the isolation valves, and if not, manually close them from the control room. It is assumed that these operation actions are covered in the procedures for aligning the CCW to safety related equipment during emergency operating conditions.	S	Specified by PRA

CESSAR-DC PRA ASSUMPTION / INSIGHTS CROSS-REFERENCE

CESSAR-DC Section/Table/Figure	System/ Structure	No.	ASSUMPTION/INSIGHTS	S/R	REFERENCE
19.6.3.3.2.1	EG	2.	During normal operation, one CCW pump and one heat exchanger in each division is typically placed in operation while the other pump and heat exchanger is in standby. If the operating pump trips, the standby pump should start automatically. If the standby pump does not start automatically as required, it is assumed that operating procedures would direct the operator to try and start the standby pump. If the standby heat exchanger also fails, the standby heat exchanger would be realigned for use. The operator actions involved in realigning the heat exchanger are assumed to be covered by operating procedures.	S	Specified by PRA
19.6.4.2.1.3	BC, BK, BN	3.	During "Feed and Bleed" operation, the heat absorbed by the water in the IRWST must be removed to prevent a threat to the containment integrity. The removal of heat is referred to as "Cooling the IRWST", and can be accomplish by using components of the Shutdown Cooling System or the Containment Spray System. The operator actions involved in "Cooling the IRWST" include recognizing the need to remove heat from the IRWST, and establishing a flow path via the SCS or CCS heat exchanger by opening appropriate valves and starting the associated SCS or CCS pump. It is assumed that the operator actions are covered in the procedure for "Cooling the IRWST" and the operator is trained in the use of the procedure.	S	Specified by PRA
19.6.4.1.1.1.1.4	BC	4.	After the SCS entry conditions are met, the SCS must be manually aligned to continue long-term decay heat removal. The operator actions involved in aligning the SCS for long-term decay heat removal include recognizing that SCS entry conditions are satisfied, aligning the suction of the SCS pumps to the RCS by opening the SCS suction valves, aligning the SCS discharge to the RCS by opening the SCS discharge valves, establishing CCW flow to the SCS heat exchangers by opening the inlet/out CCW valves or verifying that these valves are open, and starting the SCS pumps. It is assumed that if the SCS pumps become unavailable for any reason, the operator will align the appropriate CCS pump as backup. The operator actions for aligning the SCS for long-term cooling is assumed to be covered in procedures and the operator is trained in the use of the procedures.	S	Specified by PRA
19.6.3.13.2.1	BC, BK	5.	Following a large or medium LOCA, the Containment Spray System (CSS) is used to provide containment heat removal. It is assumed that if a containment spray pump should fail for any reason the operator will use the associated shutdown cooling pump as backup. The operator actions involved in using the shutdown cooling pump as backup to the containment spray pump include recognizing that the containment spray pump is inoperable, isolating the affect containment spray pump, establishing a cross-connect path between the CSS and the SCS by opening the cross-connect valves on the suction and discharge sides of the appropriate shutdown cooling pump, and starting the shutdown cooling pump. The operator actions are assumed to be covered in the recovery procedures for medium or large LOCA and the oper_tor is trained in the use of the procedures.	S	Specified by PRA
19.6.3.15.1.4	BK	6.	If the preferred means of containment heat removal should fail, the Emergency Containment Spray Backup System (ECSBS) is manually actuated to provide containment heat removal. The operator actions involved in actuating the ECSBS include recognizing the need for actuating the ECSBS, dispatching a crew to remove the blind flange at the IRWST fill connection, connecting the standpipe to the IRWST fill connection, dispatching a crew to retrieve the pumping device, connecting the pumping device to the standpipe, connecting the suction of the pumping device to the ultimate heat sink (i.e., cooling pond) using a hose, and starting the pumping device to transfer water from the ultimate heat sink to the containment spray header.	S	Specified by PRA

CESSAR-DC PRA ASSUMPTION / INSI[illegible] [illegible]SS-REFERENCE

CESSAR-DC Section/Table/Figure	System/ Structure	No.	ASSUMPTION/INSIGHTS	S/R	REFERENCE
19.6.3.7.2.1	AL	7.	For a large secondary side break or SGTR, it is assumed that the emergency procedure guidelines will instruct the operator to maintain the level in the intact steam generator within the normal range. This action is performed on a continuous basis. Therefore, the probability that the operator fails to maintain the level of the intact steam generator is considered to be negligible and hence failure of this operator action is not explicitly modeled.	S	Specified by PRA
19.6.3.8.2.1, F19.6.3.8-2	AS	8.	The Startup Feedwater System (SFWS) pump is normally aligned to take suction from the deaerator storage tank. If the inventory of this tank is depleted during startup or shutdown operation, pump suction must be transferred to the condensate storage tank. It is assumed that adequate instrumentation is provided in the control room so that the operator can monitor the level of the deaerator storage tank, and the operating procedures will provide complete instructions regarding the transfer of the suction source for the SFWS pump. It is also assume that the operator is trained in the use of the procedures.	S	Specified by PRA
19.6.4.5.2.1	AB, AL, AE	9.	Following a SGTR event, the operator would act to stabilize the RCS by initiating cooldown then identifying and isolating the ruptured steam generator. The operator actions involved in isolating the ruptured steam generator include identifying the ruptured or most affect steam generator; and then isolating the ruptured generator by closing the appropriate MSIV and MSIV bypass valves, reclosing the ADVs on the ruptured if they were opened, terminating feedwater flow to the ruptured generator, and closing the appropriate steam supply valve to the emergency feedwater turbine driven pump. These operator actions are assumed to be covered in the SGTR recovery procedures and the operator is trained in the use of the procedures.	S	Specified by PRA
19.6.3.2.2.1	EF	10.	Typically during normal operation, one SSWS pump is each division is running and the other pump is in standby. If the operation pump trips, the standby pump should start automatically. If the standby SSWS pump does not start automatically, it is assumed that the operating procedures would direct the operator to try and start the standby SSWS pump.	S	Specified by PRA
19.6.3.16.2.1	BB	11.	If core damage occurs, the operator is required to use the Cavity Flooding System to flood the reactor cavity. The operator actions involved in flooding the cavity include recognizing the symptoms of the onset of core damage and then opening the holdup volume and the cavity spillway valves. The symptoms of the onset of core damage include loss of water above the core, superheated steam temperature, and core voidage. It is assumed that the operator actions are covered in the severe accident procedures and the operator is trained in the use of the procedures.	S	Specified by PRA
19.6.3.16.2.1	BB	12.	If core damage occurs, it is assumed that the operator will initiate cavity flooding before vessel failure. This gives the operator approximately one hour from the onset of core damage to vessel failure. The time required to open the cavity flooding valves is less than five minutes under very high stress conditions.	S	Specified by PRA
19.6.3.16.2.1	BB, MD, NE, PE	13.	It is assumed that the station blackout procedures will direct the operators to initiate cavity flooding just prior to battery depletion if the recovery of site power is not imminent and the onset of core damage has occurred.	S	Specified by PRA
19.6.2	TD	14.	The unreliabilities of system components are assumed to be consistent with values presented in Table 19.5.2	S	Specified by PRA

CESSAR-DC PRA ASSUMPTION / INSIGHTS CROSS-REFERENCE

CESSAR-DC Section/Table/Figure	System/ Structure	No.	ASSUMPTION/INSIGHTS	S/R	REFERENCE
19.6.2	TD	15.	Pipe breaks are not included in the fault tree models. Their contribution to system unavailability is insignificant when compared with contributions from other components.	S	Specified by PRA
19.6.2	TD	16.	Potential flow diversion paths that are isolated from the main flow paths by two or more normally closed valves and those potential flow diversion paths with piping significantly smaller (10% or less) than the piping of the main flow paths are not included in the fault tree models. The failure probabilities for these diversion paths are assumed to be insignificant.	S	Specified by PRA
19.6.2	TD	17.	The sensing and protective devices (i. e., thermal overload and over-current trip coils) and control circuitry for motor-operated valves are included within the boundary of the component and therefore not explicitly modeled.	S	Specified by PRA
19.6.2	TD	18.	The mission time to open or close a motor-operated valve is small. The probability that the valve breaker transfers open during the mission time is, therefore, insignificant when compared with the failure probability of the valve to operate. Transfer opening of the valve breaker prior to a demand would be detected immediately and corrective actions taken (Note that the positions of the motor-operated valves are continuously displayed in the control room and transfer opening of a valve breaker causes loss of valve position display for the affected valve). Hence, the fault exposure time of a motor-operated valve due to transfer opening of the breaker is assumed to be negligible. Because of the small mission time and the fault exposure time that are assumed to be negligible, transfer opening of motor-operated valve breaker is not modeled.	S	Specified by PRA
19.6.2	TD	19.	The unavailability of an ESF component due to maintenance is included directly in the fault tree models. This unavailability is affected by the appropriate limiting conditions for operation and the periodic surveillance of the component as specified by the Technical Specifications. It is assumed that an ESF component of concern may be inoperable for a certain period of time during normal plant operation without changing the plant's mode of operation. If the limiting conditions for operation cannot be met, then the plant's mode of operation must be changed within a specified time period. This may result in a plant shutdown. Furthermore, because ESF components are usually in a standby state during normal plant operation, periodic surveillance must be performed on these components to determine their operability.	S	Specified by PRA
19.6.2	TD	20.	Components of the ESF systems are tested periodically during power operation to determine their operability. These systems are designed so that they can be tested without being taken out of service. It is assumed that even if the system is in test, it will respond when demanded. Therefore, unavailability due to testing of components is not a significant contributor to system unavailability.	S	Specified by PRA
19.6.3.15.1.4	BK	21.	Following a severe accident, it is assumed that the pumping device of the Emergency Containment Spray Backup System must be capable, as a minimum, of delivering an initial flow of 1800 gpm against a containment back pressure of approximately 100 psia or 85 psig.	S	Specified by PRA

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CESSAR-DC Section/Table/Figure	System/ Structure	No.	ASSUMPTION/INSIGHTS	S/R	REFERENCE
F19.7.5A-8	TD	1.	It is assumed that seismic induce failure of the spent fuel heat exchanger will drain the component cooling water system if the heat exchanger is not isolated. The operator actions involved in isolating the spent fuel heat exchanger include recognizing the symptoms of the loss of component cooling water, and closing the valves in the spent fuel heat exchanger lines. The symptoms for loss of component cooling water include decreasing water level in the CCWS surge tank and indication of decreased flow in the spent fuel heat exchanger line. It is assumed that procedures exists to verify that the spent fuel heat exchangers are intact following a seismic event.	S	Specified by PRA
19.7.3.1.6	TD	2.	It is assumed that procedures are in place for maintaining fire barriers during power operation so that a fire will not propagate from one fire zone to the next.	S	Specified by PRA
19.7.4.1	TD	3.	It is assumed that procedures are in place for maintaining flood barriers during power operation so that a flood will not propagate from one flood zone to the next.	S	Specified by PRA
19.5.4	TD	1.	The inoperable time limit for all major components of the engineered safety features systems is assumed to be no more than 72 hours.	S	Specified by PRA
19.12.2.2.10.2.2, 19.11.4.3.2.1.6	BB	1.	It is assumed that the holdup volume will have provisions for controlling the pH of the RCS and the IRWST inventory (at a value of 7 or higher) that is discharged to the containment during a severe accident.	S	Specified by PRA
19.12.2.2.10.1.3.2	BB	2.	It was assumed that heat transfer from the RCS to its surrounding environment is low for all plant damage states because of the insulation of the reactor vessel and the RCS piping.	S	Specified by PRA
19.12.2.2.7.1.3.3, 19.11.4.1.3.1.5	ZC	3.	Low H_2 concentrations are assumed to occur if H_2 burn occurred in the containment during the early portion of the severe accident.	S	Specified by PRA
19.12.2.2.6.3.1.1.1.2.1.2, 19.11.3.4, 19.11.3.4.5, 19.11.3.4.6	ZC	4.	The H_2 Ignitor System is a standby system which must be manually actuated during a severe accident. It is assumed that procedures are in place and this action is performed at the same time the Cavity Flooding System would be actuated and under the same conditions.	S	Specified by PRA
19.12.2.2.6.1.1.3, 19.11.3.5, 19.11.3.5.3, 19.11.3.5.4, 19.11.3.5.5	BB	5.	It was assumed that "Rapid Depressurization" prior to reactor vessel failure can be successful provided the operator actuates the Rapid Depressurization System within one hour after the Primary Safety Valves (PSVs) lift. Optimum operation indicates actuation at PSV lift would extend the time to reactor vessel failure.	S	Specified by PRA
19.12.2.1.1.4, 19.11.4.1.3, 19.11.4.1.3.1.5	ZC	6.	For sequences defined as early, it is assumed that insufficient core concrete interaction (CCI) can occur so that the hydrogen contribution due to CCI is small. This resulted in a maximum hydrogen production during the core melt progression equivalent to 100% oxidation of the active cladding. For late hydrogen burns in the containment, hydrogen produced due to CCI was assumed to be potentially available for combustion.	S	Specified by PRA
19.11.3.8.3, 19.11.4.4.2.2.1.2	TD	7.	After 24 hours following core uncovery, the operator can still protect containment integrity by providing external spray to the containment (i.e., by using the Emergency Containment Spray Backup System).	S	Specified by PRA

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CESSAR-DC Section/Table/Figure	System/ Structure	No.	ASSUMPTION/INSIGHTS	S/R	REFERENCE
19.11.4.4.2.12	ZC	8.	Following a severe accident, sprays will be used to scrub containment atmosphere of fission products.	S	Specified by PRA
19.15.3.2	ZJ	1.	The control room is used to shut the plant down under all conditions, except when there is a fire in the control room and it becomes uninhabitable. In this case, the control room would be evacuated and the plant is placed in cold shutdown conditions from remote shutdown room. It is assumed that transfer of controls from the main control room to the remote shutdown room and the evacuation of the main control room are covered in an abnormal procedure. It is also assumed that the operators are trained in the use of the procedure.	S	Specified by PRA
19.15.3.1	EF	2.	It is assumed that procedures are in place to cover the removal debris from the station service intake structure [illegible]wing a tornado event.	S	Specified by PRA
19.15.2.1.4.2, 19.10.2	TD	3.	The capability of the operator to perform mitigating actions (such as aligning the CST to the EFWST) outside the control room during the progression of an accident is essential in achieving a low core damage frequency.	S	Specified by PRA
19.15.2.1.4.2, 19.10.1	TD	4.	Increasing the operator error rates by an order of magnitude would also cause the core damage frequency for internal events to increase. This implies that the core damage frequency for internal events is sensitive to capability of the operator to perform certain task during the progression of an accident.	S	Specified by PRA
19.15.2.1.4.2, 19.10.3	TD	5.	Increasing the failure rate of motor-operated valves by an order of magnitude would also cause the core damage frequency for internal events to increase. This implies that the core damage frequency for internal events is somewhat sensitive to the reliability of motor-operated valves.	S	Specified by PRA
19.15.2.1.4.2, 19.10.4	BH	6.	It is assumed that the Safety Injection Tanks (SITs) were not needed to prevent core damage following a medium LOCA event. Crediting the SITs for medium LOCA has no effect on the core damage frequency.	S	Specified by PRA
19.15.2.1.4.2, 19.10.5	TD, BH	7.	The core damage frequency for internal events is somewhat sensitive to the feasibility of establishing aggressive cooldown of the secondary side following a small LOCA or SGTR event and failure of the Safety Injection System.	S	Specified by PRA
19.15.2.1.4.2, 19.10.6	BB	8.	RCP seal failure following a station blackout event is a non-credible event for the System 80+ design. By assuming that an RCP seal failure may occur following a blackout event, there would be no significant increase in the core damage frequency for internal events.	S	Specified by PRA
19.15.2.1.4.2, 19.10.8	BB	9.	For 99% of core life, the moderator temperature coefficient (MTC) is below a critical value. If the MTC were adverse over a longer fraction of core life (i.e., 10% instead of 1%) there would be no significant change to the core damage frequency.	S	Specified by PRA
19.15.2.1.4.2, 19.10.9	MD	10.	The loss of offsite power (LOOP) initiating event is defined as loss of site power which requires the startup and loading of the emergency diesel generators. This takes into consideration the runback capability of the turbine/generator and the two separate switchyards for the site. These features have lowered the initiating event frequency of LOOP for the System 80+ design. By increasing the initiating event frequency for LOOP by an order of magnitude, the core damage frequency for internal events would increase and would be slightly sensitive to the event frequency of LOOP.	S	Specified by PRA

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CESSAR-DC Section/Table/Figure	System/ Structure	No.	ASSUMPTION/INSIGHTS	S/R	REFERENCE
19.15.2.1.4.1	TD	11.	The following human errors are the major contributors to the uncertainty of core damage frequency for internal events: (a) Failure to initiate "Feed and Bleed", (b) Failure to perform aggressive secondary side cooldown following a SGTR, and (c) Failure to perform aggressive secondary side cooldown following a small LOCA.	R	CESSAR-DC Table 19.15.2-3, CESSAR-DC Section 19.9.2
19.15.2.1.4.1	TD	12.	The majority of the major contributors to the dominant accident sequences have relatively small uncertainties (i.e., error factor less than 10) associated with them.	R	CESSAR-DC Table 19.15.2-3, CESSAR-DC Section 19.9.2
19.15.2.1.4.1	TD	13.	A few of the major contributors to the dominant accident sequences have relatively large uncertainties (i.e., error factor of 10 or greater). The contributors with large uncertainties include hardware failures such as common cause failure of the safety injection pumps, common cause failure of the diesel generator sequencers, independent failure of the CST manual makeup valve to the EFWST, and vessel failure.	R	CESSAR-DC Table 19.15.2-3, CESSAR-DC Section 19.9.2
19.15.2.1.4.3, 19.9.4.2	TD	14.	The capability of the operator to perform the following mitigating actions inside the control room is essential in achieving a low core damage frequency: (a) aggressive cooldown of the secondary side following a SGTR, (b) aggressive cooldown of the secondary side following a small LOCA, (c) "Feed and Bleed" operation, and (d) hot-leg injection.	S	Specified by PRA
19.15.2.1.4.3, 19.9.4.1	TD	15.	The reliability of certain system components is important in achieving a low core damage frequency. The systems that would adversely impact (increase) the overall core damage frequency the most, if component reliability decreased substantially, include: (a) the Electrical Distribution System, (b) the Emergency Feedwater System, (c) the Safety Injection System, (d) and the Component Cooling/Station Service Water Systems	S	Specified by PRA
19.15.2.1.4.3, 19.9.4.2	TD	16.	Because of the redundancy and diversity of the mitigating systems, independent hardware faults are not the most important events that would adversely impact the core damage frequency for internal events.	S	Specified by PRA
19.15.2.1.4.3, 19.9.4.2	BC	17.	A substantial decrease in the reliability of the shutdown cooling return line check valves and the shutdown cooling suction valves would be an important contributor to increasing the core damage frequency due to Interfacing System Loss of Coolant Accident (ISLOCA).	S	Specified by PRA
19.15.2.1.4.3, 19.9.4.2	PB, PE, PG, PH, PK	18.	A substantial increase in the common cause failure rate of electrical equipment would be an important contributor to increasing the core damage frequency. These equipment include (a) 125 VDC class 1E buses, (b) 480 VAC class 1E load center transformers, (c) 4.16 KV class 1E buses, (d) 480 VAC class 1E load centers, and (e) 480 VAC motor control centers.	S	Specified by PRA
19.15.2.1.4.3, 19.9.4.2	AL	19.	A substantial increase in the common cause failure rate of the EFWS distribution line check valves or EFW pump discharge check valves would be an important contributor to increasing the core damage frequency.	S	Specified by PRA
19.15.2.1.4.3, 19.9.4.2	AP, AL	20.	Failure of the CST makeup valves to the EFWST is the single most important independent fault that would cause the core damage frequency to increase.	S	Specified by PRA
19.15.2.2.4, 19.14.1.1	ZC	21.	The conditional probabilities for the various containment failure modes are insensitive to the availability of the hydrogen ignitors following a severe accident.	S	Specified by PRA

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CESSAR-DC Section/Table/Figure	System/ Structure	No.	ASSUMPTION/INSIGHTS	S/R	REFERENCE
19.15.2.2.4, 19.14.1.2	ZC	22.	The System 80+ containment characteristics do not favor deflagration to detonation transition and the release classes are not sensitive to deflagration to detonation transition.	S	Specified by PRA
19.15.2.2.4, 19.14.1.3, 19.14.1.4	BB, ZC	23.	Late containment failure releases are somewhat sensitive to low heat transfer rate from the corium to the cavity water. These release class is also very sensitive to the amount of water that is discharged to the cavity by the Cavity Flooding System following a severe accident.	S	Specified by PRA
19.15.2.2.4, 19.14.1.5	BK, ZC	24.	Late containment failure releases are sensitive to the reliability and capacity of the emergency containment heat removal system and the recovery of containment heat removal following a severe accident.	S	Specified by PRA
19.15.2.2.4, 19.14.1.6, 19.14.1.7	BB	25.	The conditional probabilities for System 80+ releases classes are not sensitive to temperature induced creep failure of the RCS piping and the depressurization of the RCS using the Safety Depressurization System.	S	Specified by PRA
19.15.2.2.4, 19.14.1.8	ZC	26.	The frequency of the containment isolation failure releases is strongly coupled to the reliability of the Containment Isolation System (CIS). A very reliable CIS would result in a very low frequency for containment isolation failure releases.	S	Specified by PRA
19.15.2.3.2, 19.14.2.1	TD	27.	The risk measures for whole body doses at 300 meters and one-half mile from the reactor are relatively insensitive to the location of the release point (i.e., whether the release occurs at the top of the containment building or at grade level).	S	Specified by PRA
19.15.2.3.2, 19.14.2.3	ZC, TD	28.	The overall risk of the System 80+ design is relatively insensitive to containment bypass releases that are not scrubbed prior to their release in to the environment.	S	Specified by PRA
19.15.2.3.2, 19.14.2.4	ZC, TD	29.	The reliability of the containment isolation function can have a significant impact on the overall risk of the System 80+ design.	S	Specified by PRA
19.15.2.3.2, 19.14.2.5	TD	30.	The risk measures at 300 meters and one-half mile are somewhat sensitive to basemat melt-through that occurs more frequently than currently anticipated.	S	Specified by PRA
19.15.2.3.2, 19.14.2.6	TD	31.	Because enhanced features and improvements are incorporated into the System 80+ design, the frequency of interfacing system LOCA is several orders of magnitude lower than existing PWRs. Because of this low frequency, containment bypass releases are not major contributors to the risk of the System 80+ design.	S	Specified by PRA
19.15.2.3.2, 19.14.2.2	TD	32.	The risk of the System 80+ design is sensitive to the isotopic content that is used to characterize the various release classes.	S	Specified by PRA

CESSAR-DC PRA ASSUMPTION / INSIGHTS CROSS-REFERENCE

CESSAR-DC Section/Table/Figure	System/ Structure	No.	ASSUMPTION/INSIGHTS	S/R	REFERENCE
19.15.3.4, 19.7.5.3	TD	33.	The System 80+ class 1E electrical distribution system is provided with protection schemes which conform to the requirements of IEEE STD-741-1986. The protective schemes ar designed to isolate faulted equipment from the rest of the system to minimize the effect of the fault and to maximize the availability of the remaining equipment. The basic schemes consist of ground fault protection, instantaneous overcurrent and timed overcurrent protection. In developing the Seismic Margin Assessment models, it was assumed that the seismic failure of equipment in the electrical distribution system were "open circuit" failures. Implicit within this assumption is the assumption that if a "hot short" failure were to occur, the appropriate circuit interrupter(s) would open on overcurrent to prevent "backward" propagation of the fault.	S	Specified by PRA
T19.8.2-2, T19.8A.2.4-3	MD	1.	Following a loss of offsite power during low power and shutdown operation, it is assumed that the alignment of any available power source is covered by procedures and the operator is trained in the use of the procedures.	R	CESSAR-DC Sections 19.8A.4.3.2, 19.8A.4.3.3
T19.8.2-2, T19.8A.2.4-3	BB	2.	Following a loss of RCS coolant during Mode 5 with RCS in reduced inventory, nozzle dams installed, and the IRWST is filled, it is assumed that maintaining shutdown cooling flow rate near the minimum require for decay heat removal is covered in the procedures and the operators are trained in the use of the procedures. Recovery from this event includes regaining RCS inventory control by using the shutdown cooling pumps, safety injection pumps, or containment spray pumps to inject IRWST inventory into the RCS.	R	CESSAR-DC Sections 19.8A.4.3.1.3.1, 19.8A.4.3.1.3.2, 19.8A.4.3.1.3.3, 19.8A.4.3.1.3.4
T19.8.2-2, T19.8A.2.4-3	BB	3.	Following a loss of RCS coolant during Mode 5 with RCS in reduced inventory, nozzle dams not installed, and the IRWST is filled, it is assumed that maintaining shutdown cooling flow rate near the minimum require for decay heat removal is covered in the procedures and the operators are trained in the use of the procedures. Recovery from this event includes regaining RCS inventory control by using the shutdown cooling pumps, safety injection pumps, or containment spray pumps to inject IRWST inventory into the RCS.	R	CESSAR-DC Sections 19.8A.4.3.1.3.1, 19.8A.4.3.1.3.2, 19.8A.4.3.1.3.3, 19.8A.4.3.1.3.4
T19.8.2-2, T19.8A.2.4-3	BB	4.	Following a loss of RCS coolant during Mode 5 with RCS open, nozzle dams not installed, and the IRWST is filled, it is assumed that maintaining shutdown cooling flow rate near the minimum require for decay heat removal is covered in the procedures and the operators are trained in the use of the procedures. Recovery from this event includes regaining RCS inventory control by using the shutdown cooling pumps, safety injection pumps, or containment spray pumps to inject IRWST inventory into the RCS.	R	CESSAR-DC Sections 19.8A.4.3.1.3.1, 19.8A.4.3.1.3.2, 19.8A.4.3.1.3.3, 19.8A.4.3.1.3.4
T19.8.2-2, T19.8A.2.4-3	BB	5.	Following a loss of RCS coolant during Mode 5 with RCS not in reduced inventory, nozzle dams installed, and the IRWST is filled, it is assumed that realigning one containment spray pump to regain decay heat removal capability is covered in the procedures and the operators are trained in the use of the procedures. If a containment spray pump cannot be realigned, decay heat removal can be established by "Feed and Bleed".	R	CESSAR-DC Sections 19.8A.4.3.1.3.1, 19.8A.4.3.1.3.2, 19.8A.4.3.1.3.3, 19.8A.4.3.1.3.4
T19.8.2-2, T19.8A.2.4-3	BB	6.	Following a loss of RCS coolant during Mode 5 with RCS water level above reduced inventory, nozzle dams not installed, and the IRWST is filled, it is assumed that realigning one containment spray pump to regain decay heat removal capability is covered in the procedures and the operators are trained in the use of the procedures. If a containment spray pump cannot be realigned, decay heat removal can be established by "Feed and Bleed".	R	CESSAR-DC Sections 19.8A.4.3.1.3.1, 19.8A.4.3.1.3.2, 19.8A.4.3.1.3.3, 19.8A.4.3.1.3.4

CESSAR-DC PRA ASSUMPTION / INSIGHTS CROSS-REFERENCE

CESSAR-DC Section/Table/Figure	System/ Structure	No.	ASSUMPTION/INSIGHTS	S/R	REFERENCE
T19.8.2-2, T19.8A.2.4-3	BB	7.	Following a loss of RCS coolant during Mode 6 with the refueling pool filled, the reactor vessel head not in place, the upper internals in place, and the IRWST empty, it is assumed that the decay heat removal capability can be regained by aligning an available containment spray pump if the shutdown cooling pumps are unavailable. It is assumed that procedures are in place to cover these operator actions.	R	CESSAR-DC Sections 19.8A.4.3.1.3.1, 19.8A.4.3.1.3.2, 19.8A.4.3.1.3.3, 19.8A.4.3.1.3.4
T19.8.2-2, T19.8A.2.4-3	BB	8.	Following a loss of RCS coolant during Mode 6 with the refueling pool filled, the reactor vessel head not in place, the upper internals not in place, and the IRWST empty, it is assumed that the decay heat removal capability can be regained by aligning an available containment spray pump if the shutdown cooling pumps are unavailable. It is assumed that procedures are in place to cover these operator actions.	R	CESSAR-DC Sections 19.8A.4.3.1.3.1, 19.8A.4.3.1.3.2, 19.8A.4.3.1.3.3, 19.8A.4.3.1.3.4
T19.8A.2.1-1	BB, TD	9.	Procedural guidance is provided to guard against unplanned draining of the RCS during shutdown modes of operation. This guidance includes: (a) Administrative control of major potential drain paths identified for shutdown modes to prevent unplanned draining of the RCS, (b) Monitoring instrumentation for RCS level, inventory and temperature controls to identify unplanned draining of the RCS, and (c) implementing immediate operator actions to mitigate the unplanned draining of the RCS. Unplanned draining of the RCS can be identified by monitoring: (a) refueling pool level, (b) containment and subsphere sump levels, (c) level indicators and alarms such as EDT, RDT, IRWST, HVT, VCT, (d) RCS operational leakage - Tech spec surveillance, (e) RCS level indicators and alarms such as pressurizer level instrumentation, wide range differential pressure based on refueling water level instrumentation, narrow range differential pressure based on refueling water level instrumentation, and heated junction thermocouple probes, (f) pressurizer pressure, (g) RCS temperature using core exit thermocouples, resistance temperature detectors (RTDs), and shutdown cooling system. Immediate operator actions to mitigate an unplanned draining of the RCS include identifying the leakage path, isolating the leakage path, and making up the lost inventory by using: (a) the Safety Injection System, (b) Shutdown Cooling System via the IRWST, (c) Containment Spray System via SCS lines, (d) Chemical and Volume Control System using charging pumps and boric acid storage tank, and (e) Safety Injection tanks.	R	CESSAR-DC Sections 19.8A.2.12.1, 19.8A.2.12.3.1, 19.8A.2.12.3, 19.8A.2.12.3.2.1, 19.8A.2.12.4, 19.8A.2.3.3.1, 19.8A.2.8, T19.8A.2.8-1
T19.8A.2.1-1	TD	10.	Procedural guidance is provided for maneuvering heavy loads during shutdown operations. The guidance covers dropping of transported equipment, dropping of fuel bundles, refueling pool seal integrity, and loads over the ICI table. Restrictions are specified for (a) lift height, (b) travel direction, and (c) systems lineup.	R	CESSAR-DC Section 19.8A.2.11.3
T19.8A.2.1-1	TD	11.	Procedural guidance is provided for outage maintenance during shutdown operations. The strategy for outage maintenance during shutdown operations includes: (a) define operating and operational divisions, and (b) limit maintenance activities to components and systems not included in (a).	R	CESSAR-DC Section 19.8A.2.4.3.2.2
T19.8A.2.1-1	KC	12.	Procedural guidance is provided for fire protection during shutdown operations. The guidance includes administratively requiring fire protection systems to remain operable during shutdown modes. The guidance will cover procedural control of: (a) combustible materials, (b) housekeeping, and (c) hot work. Fire planning will include: (a) outline fire fighting strategies, and (b) monitor status of fire barriers	R	CESSAR-DC Sections 19.8A.2.7.3.2, 19.8A.2.7.3.3

CESSAR-DC PRA ASSUMPTION / INSIGHTS CROSS-REFERENCE

CESSAR-DC Section/Table/Figure	System/ Structure	No.	ASSUMPTION/INSIGHTS	S/R	REFERENCE
T19.8A.2.1-1	BB	13.	Procedural guidance is provided for RCS cooling using "Feed and Bleed" (other systems not available). The guidance covers: (a) starting of safety injection pump, (b) reducing RCS pressure through the SDS venting to IRWST (maintain subcooled temperature in RCS), (c) securing the operating RCPs if applicable, (d) cycling safety injection "Feed" and SDS "Bleed" to reduce RCS pressure and temperature, (e) when RCS is depressurized, open SDS and run safety injection continuously, (f) aligning SCS heat exchangers for IRWST cooling, and (g) restoring normal SCS.	R	CESSAR-DC Sections 19.8A.2.4.3.1.3.1.1, 19.8A.2.4.3.1.3.2.1
T19.8A.2.1-1	BB	14.	A requirement is included in the Emergency Procedure Guides to maintain a positive primary to secondary pressure differential following a SGTR.	R	CESSAR-DC Table T19.8A.2.6-1
T19.8A.2.1-1	AE	15.	Procedural guidance is provided to administratively lock out main feedwater pumps during shutdown modes	R	CESSAR-DC Sections 19.8A.4.1.1, 19.8A.4.1.2
T19.8A.2.2-1	SF	16.	The limiting conditions of operation for Shutdown Margin (TS 3.1.1) is applicable for ≤500 °F. This extends the applicable modes for shutdown margin.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	SF	17.	The limiting conditions of operation for Shutdown Margin (TS 3.1.2) includes added requirements for K_{n}, and estimate critical position (ECP). This provides protection for ejected CEA and CEA group withdrawal in shutdown modes.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	SF	18.	The limiting conditions of operation for Shutdown Margin Test Exemption for CEDMs Testing (TS 3.1.10) allow CEDMs testing in Modes 4 and 5. This provides exceptions to Test Operability of CEDMs. Movement of only one CEA at a time is allowed.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	SB	19.	The limiting conditions of operation for RPS Instrumentation (TS 3.3.1) specify the modes of applicability in Table 3.3.1-1. Steam Generator Pressure - low is extended to Mode 3 and RC Flow - low is extended to Modes 3, 4, and 5 when the CESs can be moved. This provides reactor trip function for steam line break in shutdown modes.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	SC	20.	The limiting conditions of operation for Core Protection Calculators (TS 3.3.5) extend operability to Modes 3, 4, and 5 when CEAs can be moved. This provides reactor trip function for unplanned CEA group withdrawal.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	SA	21.	The limiting conditions of operation for ESFAS Instrumentation Automatic Actuation (TS 3.3.10) add Mode 4 to CSAS Mode applicability in Table 3.3.12-1. This ensures availability of automatic CSAS for mitigation of LOCA event in shutdown Mode 4.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	BB	22.	The limiting conditions of operation for RCS P/T Limits (TS 3.4.11) add minimum pressure restriction for RCS temperature between 483 °F and 543 °F. This provides an SIAS for steam line break and other increasing heat removal events initiated in this temperature regime.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	BB	23.	The limiting conditions of operation for LTOP (TS 3.5.3) change restriction on number of safety injection pumps operable to 2. Two divisions of safety injection are required to be operable in applicable modes.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	BH	24.	The limiting conditions of operation for Safety Injection System (TS 3.5.4) extend the requirements for two safety injection divisions to all of modes 4, 5, and 6. This allow RCS inventory makeup for LOCA events in lower operating modes.	R	CESSAR-DC Section 19.8A.2.2

CESSAR-DC PRA ASSUMPTION / INSIGHTS CROSS-REFERENCE

CESSAR-DC Section/Table/Figure	System/ Structure	No.	ASSUMPTION/INSIGHTS	S/R	REFERENCE
T19.8A.2.2-1	BN	25.	The limiting conditions of operation for the IRWST (TS 3.5.4) extend operability requirements to Modes 5 and 6. The maximum water temperature of 110 °F is also specified.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	MD, NA, NB	26.	The limiting conditions of operation for AC Sources - Shutdown (TS 3.8.2) require one circuit between the offsite transmission network to each onsite class 1E distribution system in Modes 5 and 6. This provides additional backup AC power source.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	BB	27.	The limiting conditions of operation for Boron Dilution Alarm (TS 3.3.15) specify that both boron dilution alarms shall be operable in Modes 3, 4, 5, and 6. This provides additional protection for prevention of an inadvertent boron dilution of the RCS.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	SC	28.	The limiting conditions of operation for Accident Monitoring Instrumentation (TS 3.3.14) require radiation monitoring instrumentation for (a) steam generator liquid blowdown, (b) steam line, (c) air ejectors, and (d) stack. The requirement is added to Table 3.3.14-1 and serves as a means of detecting steam generator tube rupture during shutdown modes.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	SF	29.	The limiting conditions of operation for Shutdown CEA Insertion Limits (TS 3.1.6) add special test exceptions and applicability to only critical conditions.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	SF	30.	The limiting conditions of operation for Regulating CEA Insertion Limits (TS 3.1.7) add special test exceptions and applicability to only critical conditions.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	SA	31.	The limiting conditions of operation for ESFAS Instrumentation Manual Actuation (TS 3.3.12) add Mode 4 to CSAS Mode applicability in Table 3.3.12-1. This ensures availability of automatic CSAS for mitigation of LOCA event in shutdown Mode 4.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	BB	32.	The limiting conditions of operation for LTOP (TS 3.4.11) no longer have requirements for the safety injection pumps.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	PE, PK	33.	The limiting conditions of operation for DC Sources - Shutdown (TS 3.8.5) provide the most reliable line up to prevent loss of operable diesel generator due to maintenance.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	PB	34.	The limiting conditions of operation for Distribution Systems - Shutdown (TS 3.8.8) provide the most reliable line up to prevent loss of operable diesel generator due to maintenance.	R	CESSAR-DC Section 19.8A.2.2
T19.8A.2.2-1	BC	35.	The limiting conditions of operation for Shutdown Cooling - Refueling operations (TS 3.9.4) require additional shutdown cooling division to be operable. This allows increased reliability for decay heat removal.	R	CESSAR-DC Section 19.8A.2.2

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DATE: September 24, 1993

NUMBER: OPS-93-0792

SUBJECT: Transmittal of Response to Part 2 of the Follow-on Question to DSER Open Item 19.1.2.1.2.8-1

I am providing ABB-CE's response to part 2 of the follow-on question for DSER Open Item 19.1.2.1.2.8-1as documented in your fax of August 13, 1993. If you have any questions on this information, please call me at (203)285-3926.

19.1.2.1.2.8-1 Impact Of Key Issues And Parameters On Risk Results
19.1.2.1.3.5-1 Source Term Uncertainty

PART 2. Although MAAP calculations were performed by the applicant for some transient events (Section 19.11.5.4) the fission product releases predicted from the MAAP calculations are not compared with the results from S80SOR calculations. Please provide such a comparison where possible.

Response: The attached table provides a comparison between the release fractions calculated using S80SOR and those calculated using MAAP for those cases that were directly comparable. This comparison includes one intact containment case, one late containment overpressure failure case, two early containment failure cases and one isolation failure case.

The MAAP results presented are point estimates of the releases. The release fractions for S80SOR are the weighted averages of the mean release fractions for the dominant PDSs that comprised a given release class. As shown on the comparison table, the MAAP release fractions and the S80SOR release fractions are generally consistent with each other.

Comparison of S80SOR and MAAP Release Fractions for Selected Release Classes									
Release Class MAAP Case	Fission Product Release Fractions								
	Noble Gases	Iodine	Cesium	Tellurium	Barium	Strontium	Ruthenium	Lanthanum	Cerium
RC1.1E	5.00E-03	2.30E-07	1.93E-07	9.61E-08	2.31E-08	4.24E-09	1.38E-09	6.12E-09	2.56E-08
MAAP - PDS3	5.00E-03	8.5E-07	8.5E-07	1.2E-10	5.0E-08	5.0E-08	1.1E-09	8.0E-10	8.0E-10
MAAP - PDS201	5.00E-03	5.6E-07	5.0E-07	2.2E-12	1.25E-07	1.0E-07	3.3E-07	1.6E-09	1.04E-08
RC2.2E	1.00E+0	1.09E-04	1.00E-03	1.86E-05	3.54E-06	9.03E-07	2.18E-07	1.01E-06	3.92E-06
MAAP	0.999	2.6E-04	5.3E-04	2.16E-05	9.38E-05	3.6E-06	1.2E-04	3.94E-07	2.8E-06
RC3.1E	1.00E+0	1.37E-02	1.15E-2	3.53E-03	1.10E-03	2.14E-04	6.10E-05	2.64E-04	1.25E-03
MAAP*	0.309	5.6E-04	[illegible]-04	3.3E-07	1.32E-03	1.24E-04	1.24E-05	3.17E-05	3.17E-05
RC3.4E	1.00E+0	2.26E-02	1.50E-02	7.11E-03	1.84E-03	2.29E-04	1.45E-04	3.65E-04	1.83E-03
MAAP	0.309	2.6E-04	1.23E-04	3.3E-04	1.32E-04	1.24E-04	1.24E-04	3.17E-05	3.17E-05
RC4.8E	1.00E+0	7.80E-03	6.92E-04	3.29E-04	8.65E-05	1.06E-05	6.78E-06	1.76E-05	8.55E-05
MAAP	0.146	3.3E-03	3.3E-03	2.9E-07	1.58E-04	1.15E-05	8.21E-06	3.1E-06	3.1E-06

* MAAP predicts that with sprays on, the containment goes subatmospheric after containment failure.