ABB
ASEA BROWN BOVERI

April 15, 1993
LD-93-065

Docket No. 52-002

Attn: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: System 80+™ Safety Analysis Report Revisions

Dear Sirs:

The attachments to this letter provide markups to CESSAR-DC, as required by NRC staff, for closure of the corresponding DSER issues. These markups, and others discussed with staff in the near future, will be printed as part of Amendment O in May, 1993.

Attachment 1 provides additional revisions to I&C systems descriptions, in Chapter 7, which are related to hard-wired backup instrumentation and controls.

Attachment 2 provides revisions to the fire protection system description in Section 9.5.1.

Attachment 3 provides revisions to the Chemical and Volume Control System to reflect the increased power level for System 80+, minor technical revisions, and responses to RAIs related to materials and water chemistry controls. Also included in this attachment are materials and chemistry related revisions to Sections 4.5, 9.2, and 10.3.

Attachment 4 provides revisions to the human factors engineering program descriptions in Chapter 18.

Attachment 5 provides revisions to Sections 19.10 to show the PRA's low sensitivity to the assumed reactor coolant pump seal failure probability. This attachment also provides PRA Levels 2 and 3 sensitivity studies, Section 19.14, and the summary of PRA insights, Section 19.15.

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Attachment 6 provides minor changes to the text and power distribution figures of Section 4.3.

Finally, Attachment 7 provides the CESSAR-DC revision for DSER open item 20.2-11 (ECGB) on reactor vessel thermal stress.

If you have any questions, please call me or Mr. Stan Ritterbusch at (203) 285-5206.

Very truly yours,

COMBUSTION ENGINEERING, INC.

C. B. Brinkman
Acting Director
Nuclear Systems Licensing

CBB/ser
cc: J. Trotter (EPRI)
T. Wambach (NRC)
P. Lang (DOE)

ATTACHMENT 1

TABLE 7.2-2

REACTOR PROTECTIVE SYSTEM MONITORED PLANT VARIABLE RANGES

Monitored Variable	Minimum	Nominal(d) (full power)	Maximum
Neutron flux power, % of full power	1×10^{-7}	100	200
Cold leg temperature, °F	465	558	615
Hot leg temperature, °F	525	615	675
Pressurizer Pressure (high range), psia	1,500	2,250	2,500
Pressurizer Pressure (mid range), psia	600	(c)	1,650
Pressurizer pressure (low range), psia	0	(c)	~~1,600~~ 750
CEA positions	full in	NA	full out
Reactor coolant pump speed, rpm	100	1,190	1,200
Steam generator water level (wide range), %(a)	0	76.8	100
Steam generator water level (narrow range), %(b)	0	59.1	100
Steam generator pressure, psia	0	1,000	1,524
Containment pressure, psig	-5	0	60
Steam generator primary pressure differential, psid	0	43	47

NOTES:

a. % of the distance between the wide range level instrument nozzles (above the lower nozzle).

b. % of the distance between the narrow range instrument nozzles (above the lower nozzle).

c. The high and low pressurizer pressure sensor ranges are combined electronically within the PPS bistable for wide range applications.

d. Nominal values given are typical. These values may be adjusted during the final design process.

TABLE 7.2-3

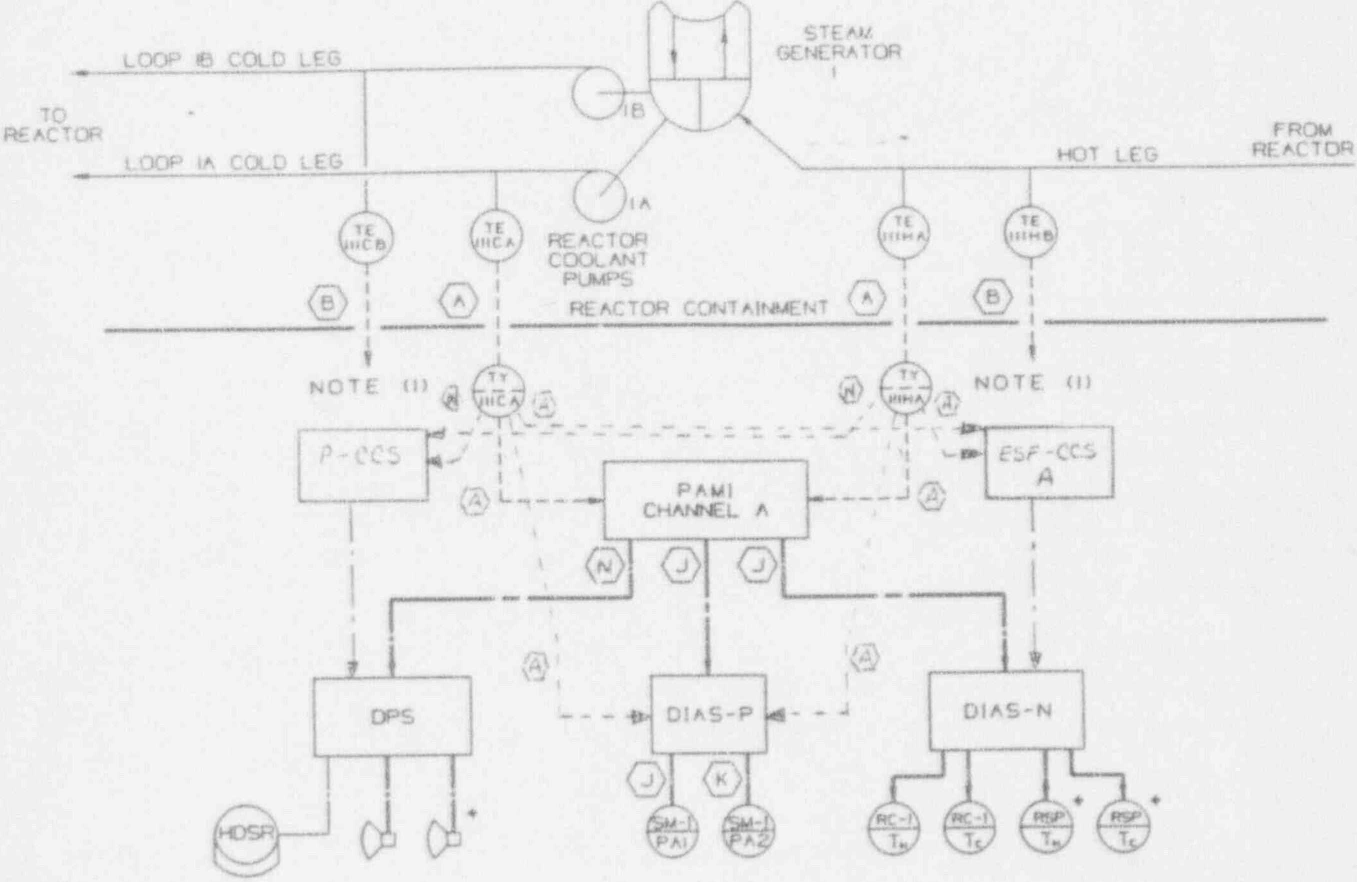
REACTOR PROTECTIVE SYSTEM SENSORS

Monitored Variable	Type	Number of Sensors	Location
Neutron flux power	Fission chamber	12(b)	Biological shield
Cold leg temperature	Precision RTD	8(b)	Cold leg piping
Hot leg temperature	Precision RTD	8(b)	Hot leg piping
Pressurizer pressure (high range)	Pressure transducer	4(a)(b)	Pressurizer
pressurizer pressure (mid range)	" "	4	Pressurizer
Pressurizer pressure (low range)	Pressure transducer	4(b)	Pressurizer
CEA positions	Reed switch assemblies	2/CEA(b)	Control Element Drive Mechanism
Reactor coolant pump speed	Proximity device	4/pump(b)	Reactor coolant pump
Steam generator level	Differential pressure transducer	8/steam generator(a)(c)	Steam generators
Steam generator pressure	Pressure transducer	4/steam generator(a)(b)	Steam generators
Containment pressure	Pressure transducer	4(a)	Containment structure
Steam generator primary differential pressure	Differential pressure transducer	4/steam generator	Steam generators

NOTES:
a. Common with Engineered Safety Feature Actuation System.
b. Common with control systems.
c. Only narrow range common with control systems.

CALMA FILE:
(.SYS80)FIGURE7222A

8



NOTES:

(1) REDUNDANT CHANNEL B SAME AS CHANNEL A SHOWN, WITH EXCEPTION THAT CHANNEL B SIGNAL TO DIAS-P PROVIDES ISOLATION FROM CHANNEL A.

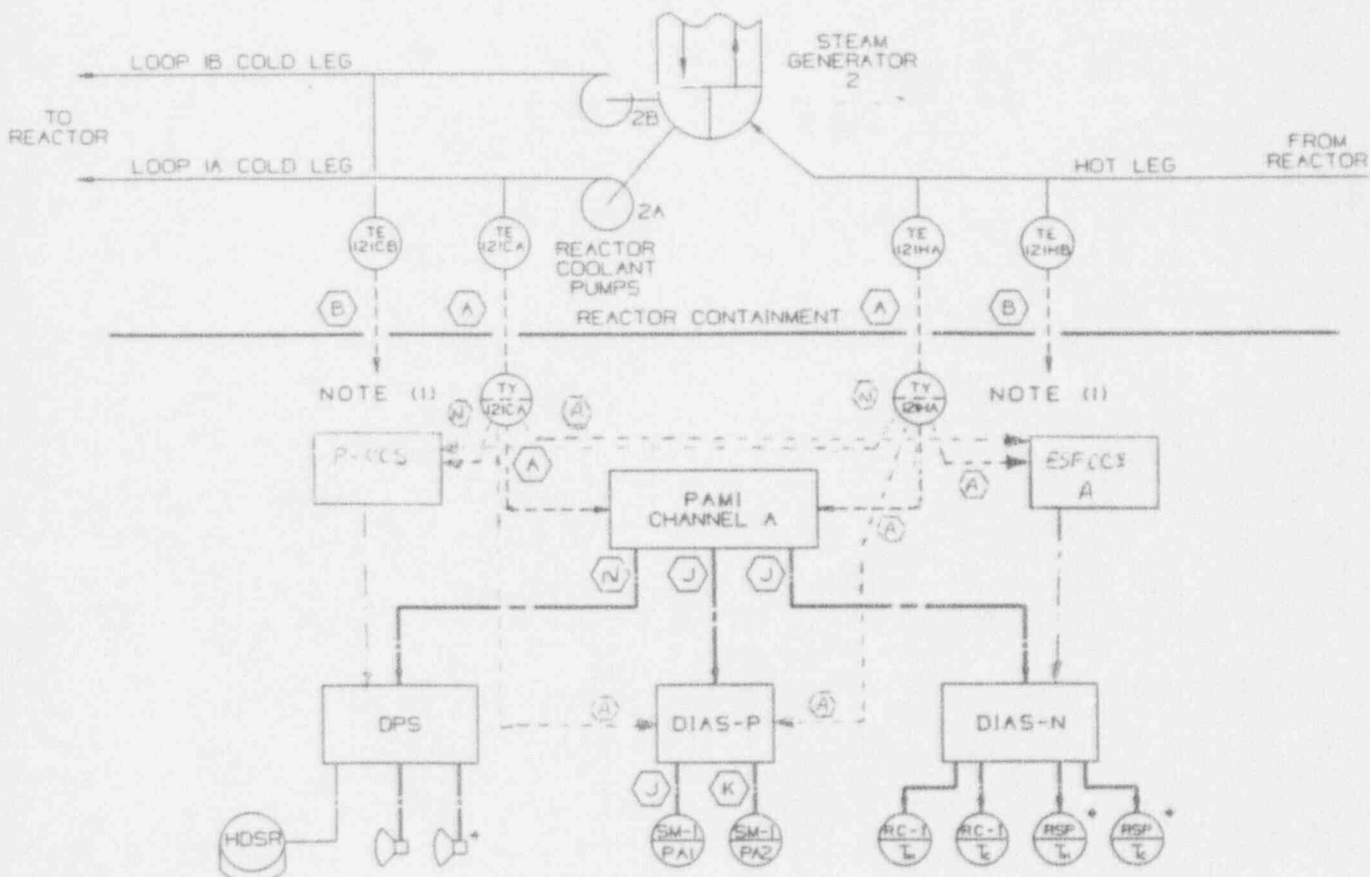
(2) DIAS INCLUDES LTOP ALARM LOGIC.

(3) PAMI CHANNEL CALCULATES SMM VALUE.

Amendment I
December 21, 1990

SYSTEM 80+™	RCS LOOP 1 TEMPERATURES (WIDE) MCBD	Figure 7.2-22a

CALMA FILE:
I.SYS80IFIGURE7222B



(1) REDUNDANT CHANNEL B SAME AS CHANNEL A SHOWN, WITH EXCEPTION THAT CHANNEL B SIGNAL TO DIAS-P PROVIDES ISOLATION FROM CHANNEL A

NOTES:

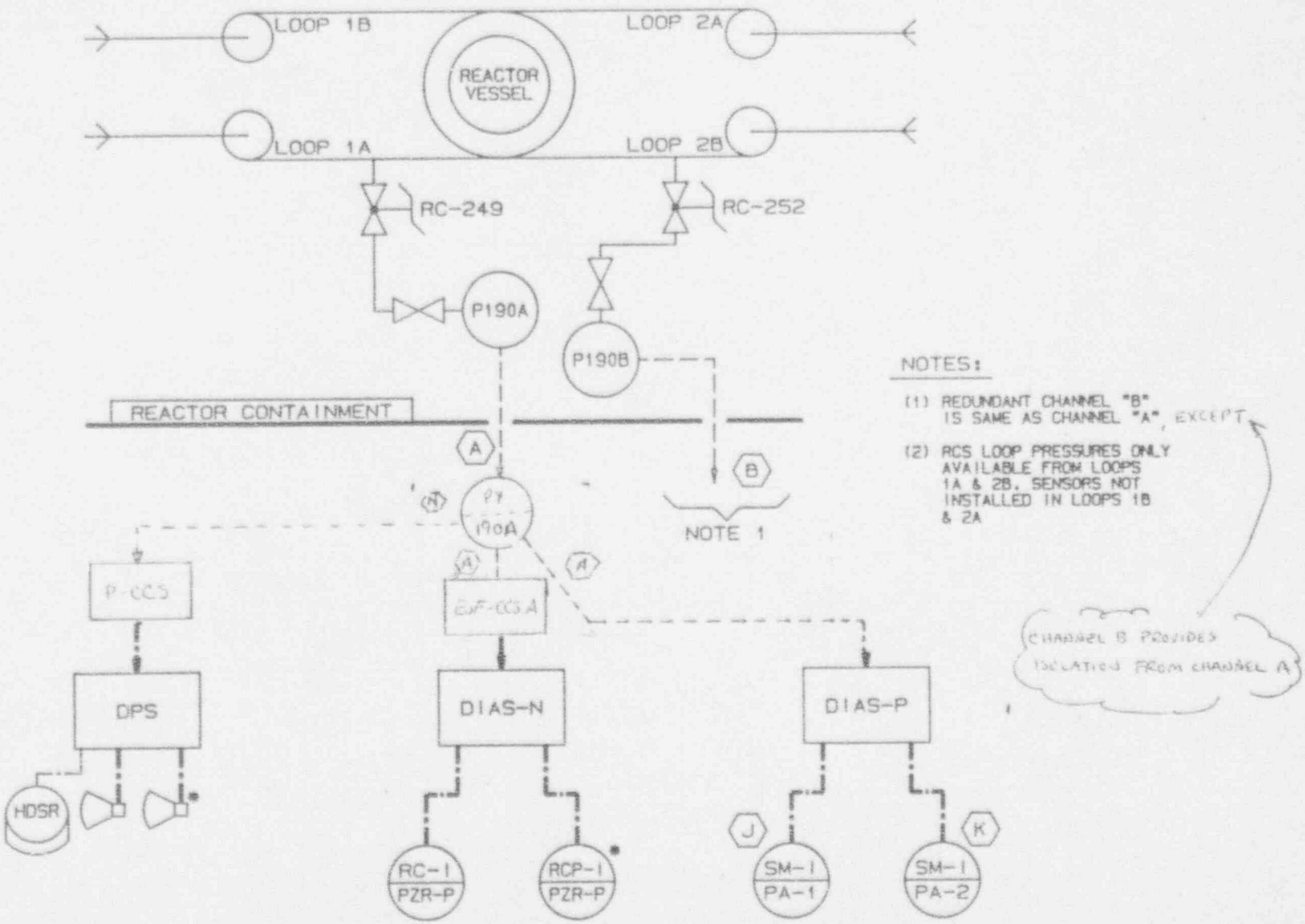
(1) ~~REDUNDANT CHANNEL "B" SAME AS CHANNEL "A" SHOWN.~~

(2) DIAS INCLUDES LTOP ALARM LOGIC.

(3) PAMI CHANNEL CALCULATES SMM VALUE.

Amendment I
December 21, 1990

SYSTEM 80+™	RCS LOOP 2 TEMPERATURES (WIDE) MCBD	Figure 7.2-22b



FILE: MCBD1

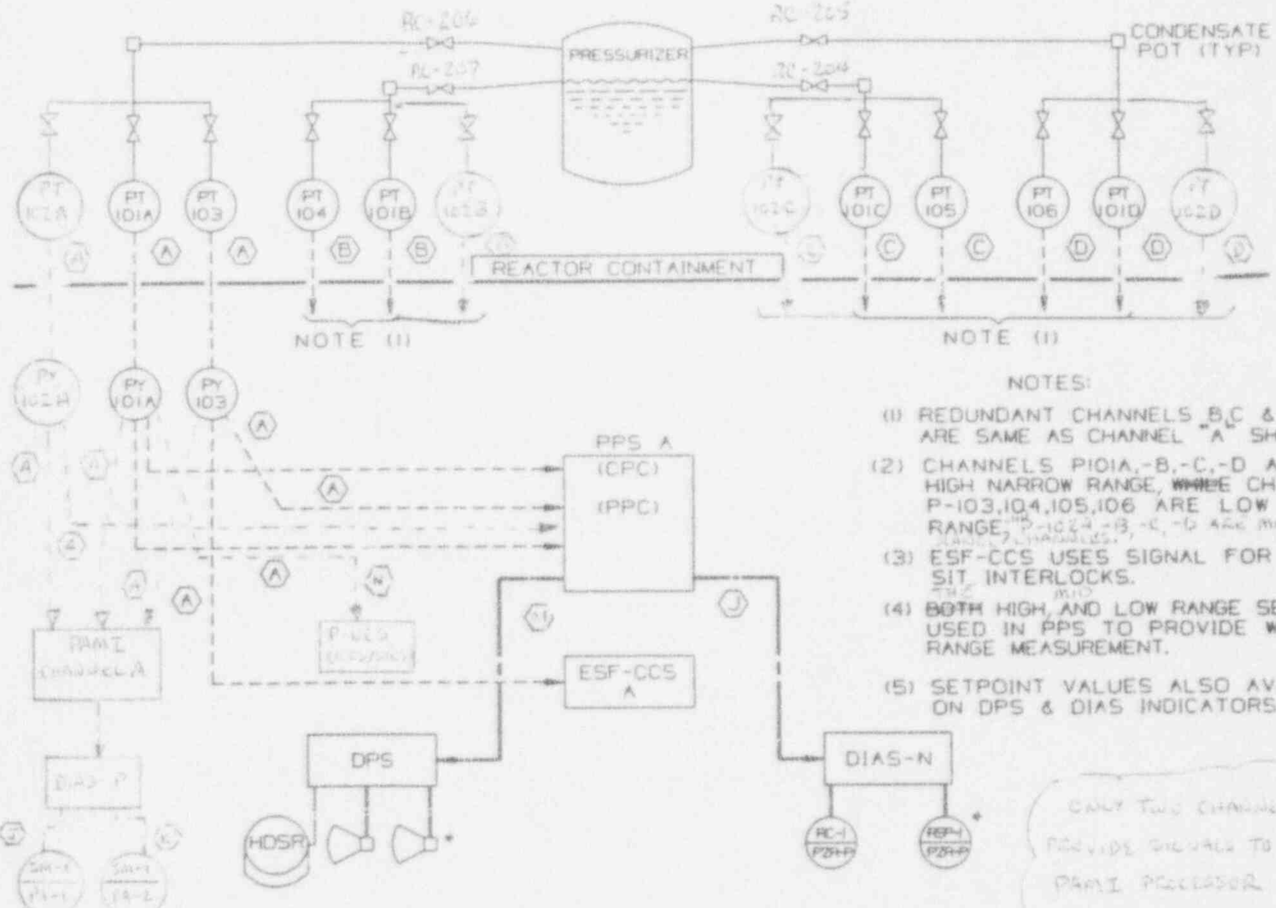
SYSTEM 80+™

REACTOR COOLANT PUMP PRESSURE MCBD

Figure 7.2-23a

Amendment E
December 30, 1988

CALMA FILE:
[.SYS80]FIGURE7224



NOTES:

(1) REDUNDANT CHANNELS B,C & D ARE SAME AS CHANNEL "A" SHOWN, EXCEPT

(2) CHANNELS P101A,-B,-C,-D ARE HIGH NARROW RANGE, ~~WHILE~~ CHANNELS P-103,104,105,106 ARE LOW NARROW RANGE, P-102A,-B,-C,-D ARE MID NARROW RANGE CHANNELS.

(3) ESF-CCS USES SIGNAL FOR SCS & SIT INTERLOCKS.

(4) ~~BOTH~~ THE HIGH, MID AND LOW RANGE SENSORS USED IN PPS TO PROVIDE WIDE RANGE MEASUREMENT.

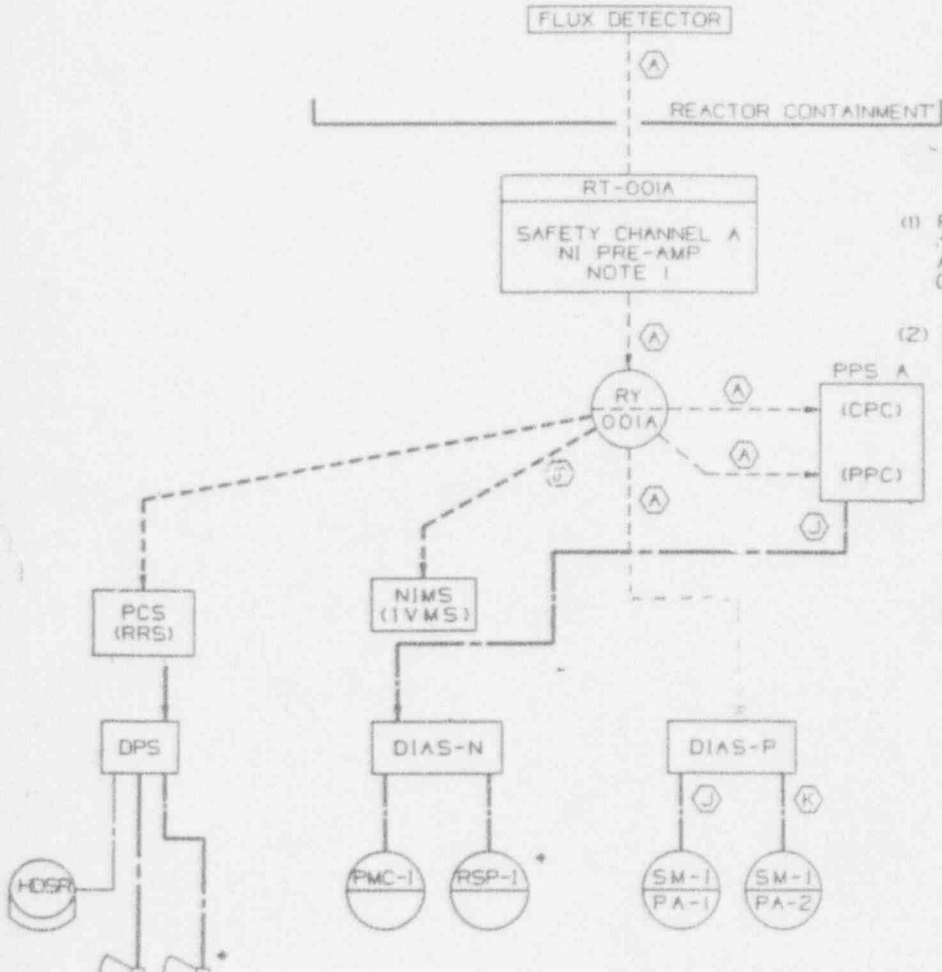
(5) SETPOINT VALUES ALSO AVAILABLE ON DPS & DIAS INDICATORS.

ONLY TWO CHANNELS (A,C) PROVIDE SIGNALS TO THE PAMI PROCESSOR TO CALCULATE SMM VALUE

Amendment I
December 21, 1990

SYSTEM 80+™	PRESSURIZER PRESSURE MCBD	Figure 7.2-24

CALMA FILE:
I.SYS80IFIGURE7225



NOTES:

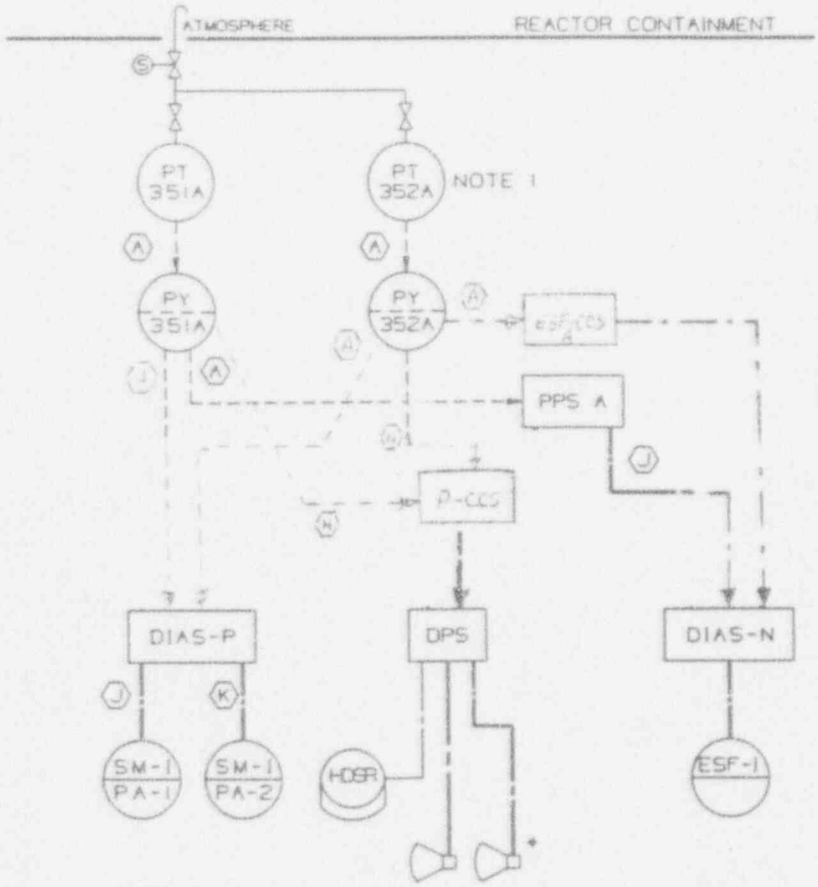
(1) REDUNDANT CHANNELS B,C & D SAME AS CHANNEL A. ASSOCIATED CIRCUITS ARE K,L & M, RESPECTIVELY; EXCEPT ONLY TWO CHANNELS (A,B) ~~PAMI~~ TO DIAS-P

(2) REDUNDANT ~~PAMI B. IDENTICAL TO PAMI A. SHOWN.~~ SIGNAL TO DIAS-P FOR CHANNEL B PROVIDES ISOLATION FROM CHANNEL A.

Amendment I
December 21, 1990

SYSTEM 80+™	NUCLEAR INSTRUMENTATION MCBD	Figure 7.2-25

CALMA FILE:
[.SY80]FIGURE7226



NOTES:

(1) REDUNDANT CHANNELS B,C & D SAME AS CHANNEL A, ASSOCIATED CIRCUITS ARE K,L & M, RESPECTIVELY; EXCEPT ONLY TWO CHANNELS (A,B) ~~OF PAM~~ To DIAS-P

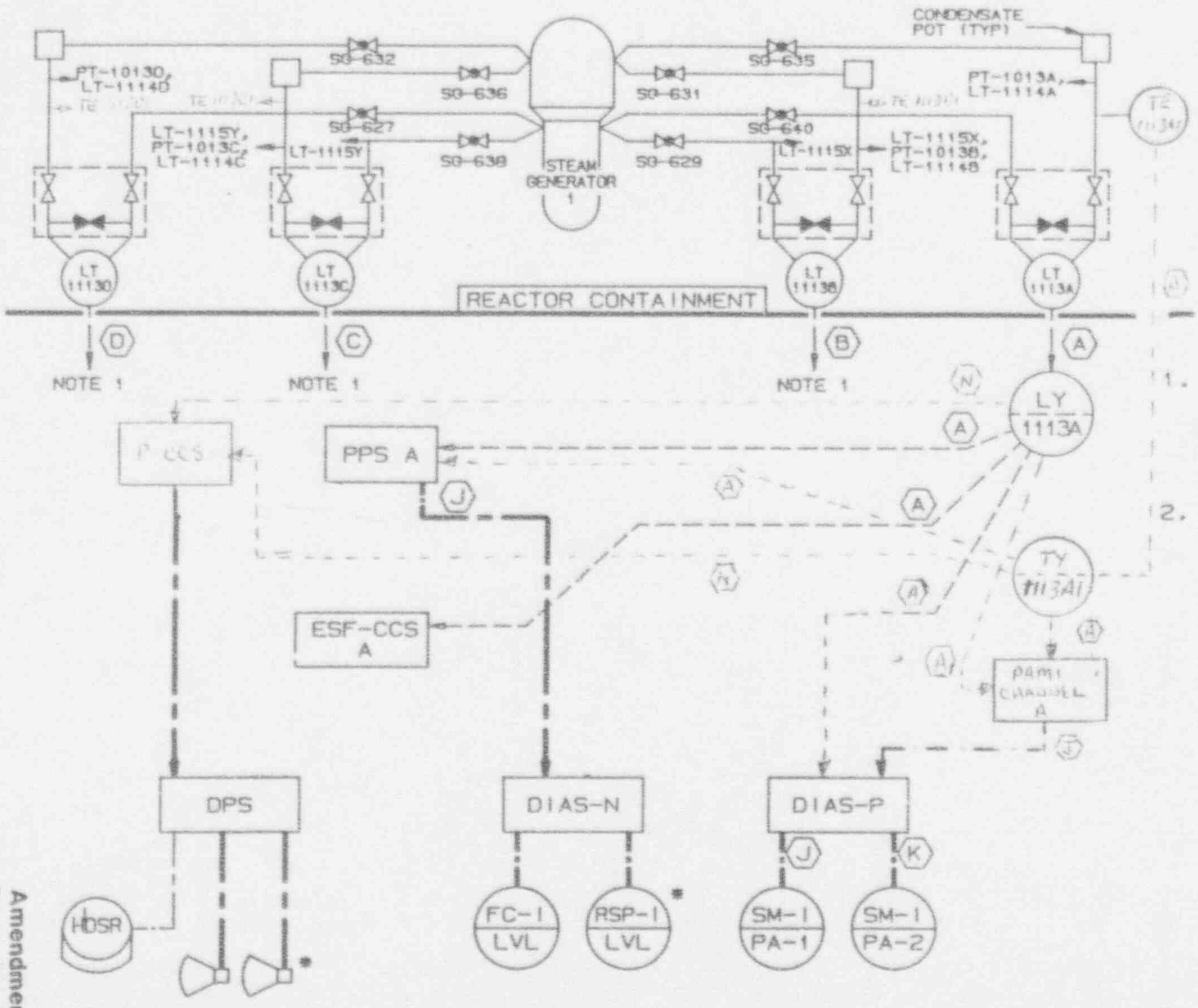
(2) PT-352A IS WIDE RANGE.

(3) PT-351A IS NARROW RANGE.

(4) REDUNDANT SIGNALS TO DIAS-P FOR CHANNEL B PROVIDES ISOLATION FROM CHANNEL A

Amendment N
April 1, 1993

SYSTEM 80+™	CONTAINMENT PRESSURE MCBD	Figure 7.2-26



NOTES:

1. REDUNDANT CHANNELS B, C & D SAME AS CHANNEL A, ASSOCIATED CIRCUITS ARE K, L & M, RESPECTIVELY; EXCEPT ONLY TWO CHANNELS (A, B) ~~PAMI~~. TO DIAS-P

2. REDUNDANT ~~PAMI B, IDENTICAL TO PAMI A, SHOWN.~~ SIGNAL TO DIAS-P FOR CHANNEL B PROVIDES ISOLATION FROM CHANNEL A.

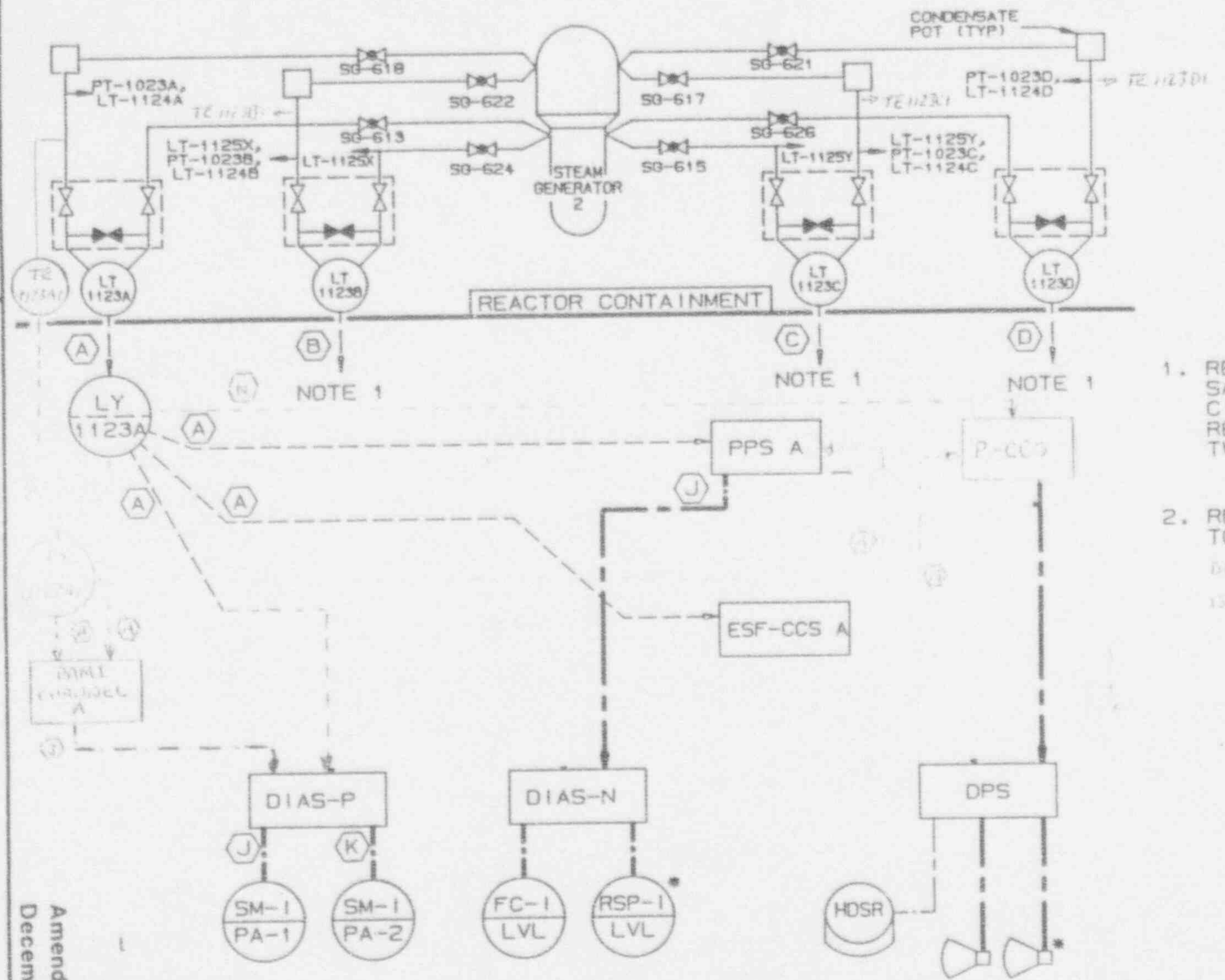
FILE: MCBD1

SYSTEM 80+™

STEAM GENERATOR-1 LEVEL (WIDE) MCBD

Amendment E
December 30, 1988

Figure 7.2-27a



NOTES:

1. REDUNDANT CHANNELS B, C & D SAME AS CHANNEL A, ASSOCIATED CIRCUITS ARE K, L & M, RESPECTIVELY; EXCEPT ONLY TWO CHANNELS PAMI (A, B) TO DIAS-P
2. REDUNDANT PAMI B, IDENTICAL TO PAMI A, SHOWN. SIGNAL TO DIAS-P FOR CHANNEL B PROVIDES ISOLATION FROM CHANNEL A.

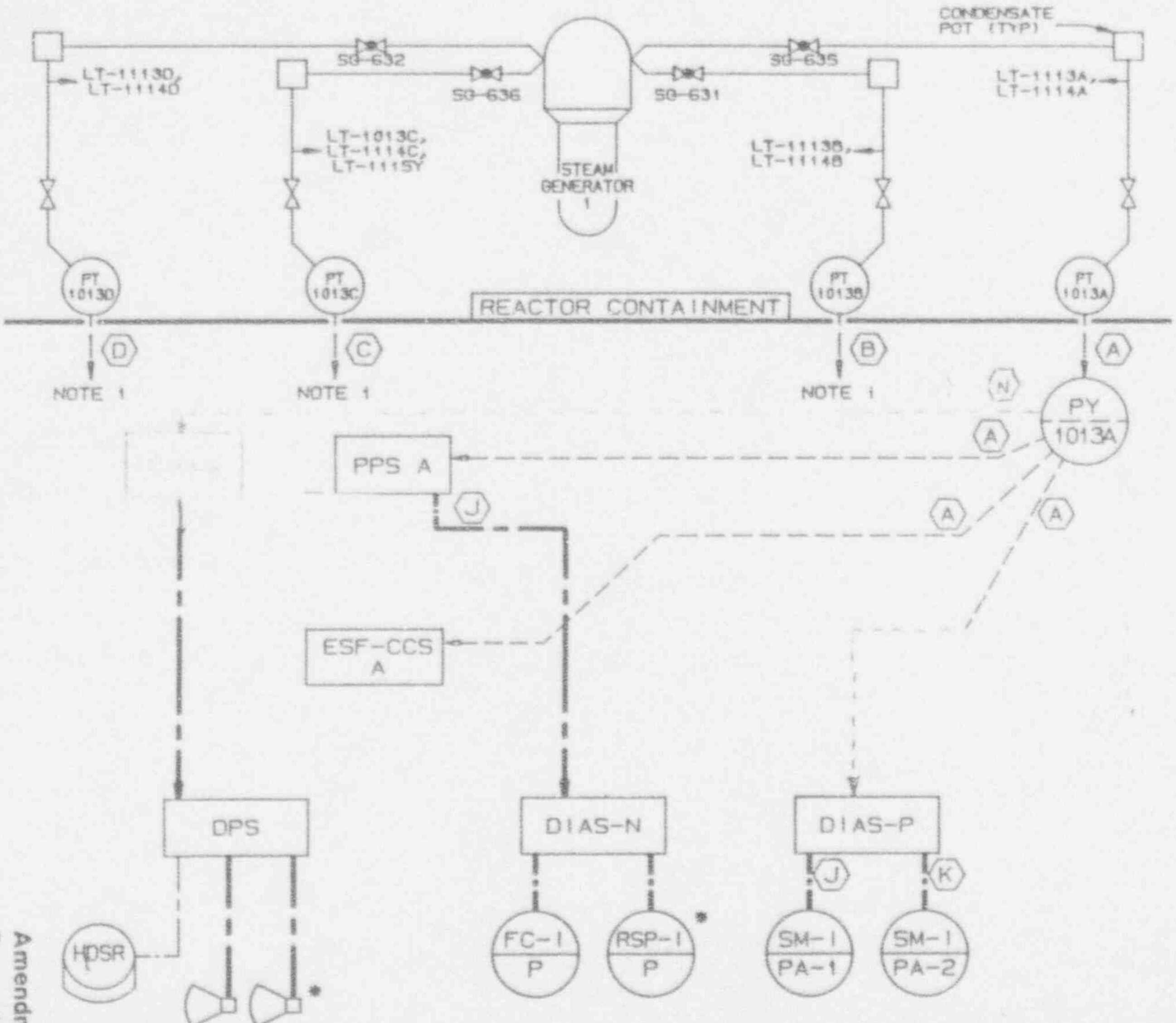
FILE: MCBD1

SYSTEM 80+™

STEAM GENERATOR-2 LEVEL (WIDE) MCBD

Figure 7.2-27b

Amendment E
December 30, 1988



NOTES:

1. REDUNDANT CHANNELS B, C & D SAME AS CHANNEL A, ASSOCIATED CIRCUITS ARE K, L & M, RESPECTIVELY; EXCEPT ONLY TWO CHANNELS (A, B) ~~PAMI~~ to DIAS-P

2. REDUNDANT ~~PAMI B, IDENTICAL TO PAMI A, SHOWN.~~ SIGNAL TO DIAS-P FOR CHANNEL B PROVIDES ISOLATION FROM CHANNEL A.

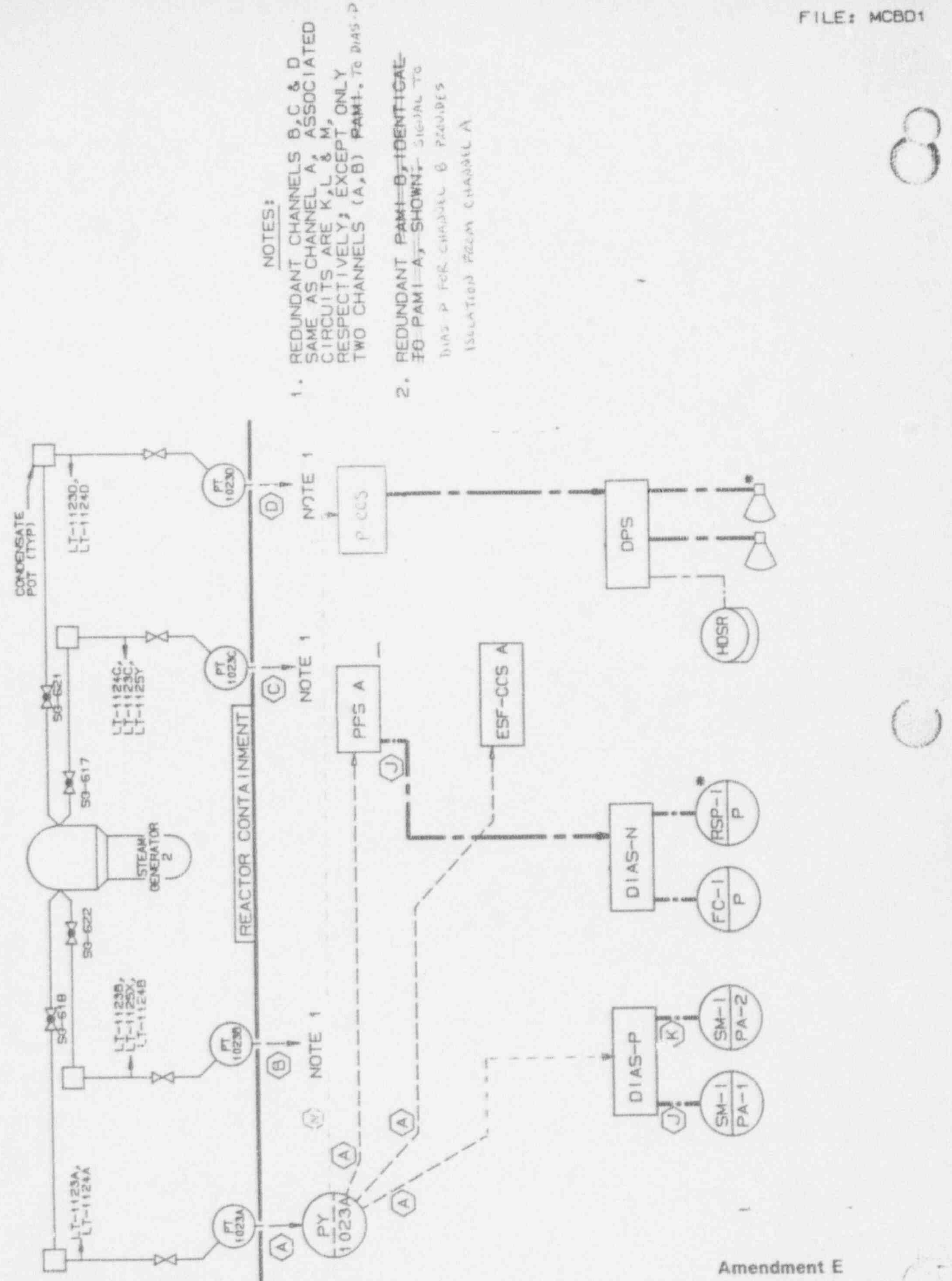
SYSTEM 80+™

STEAM GENERATOR-1 PRESSURE MCBD

Amendment E
December 30, 1988

Figure 7.2-28a

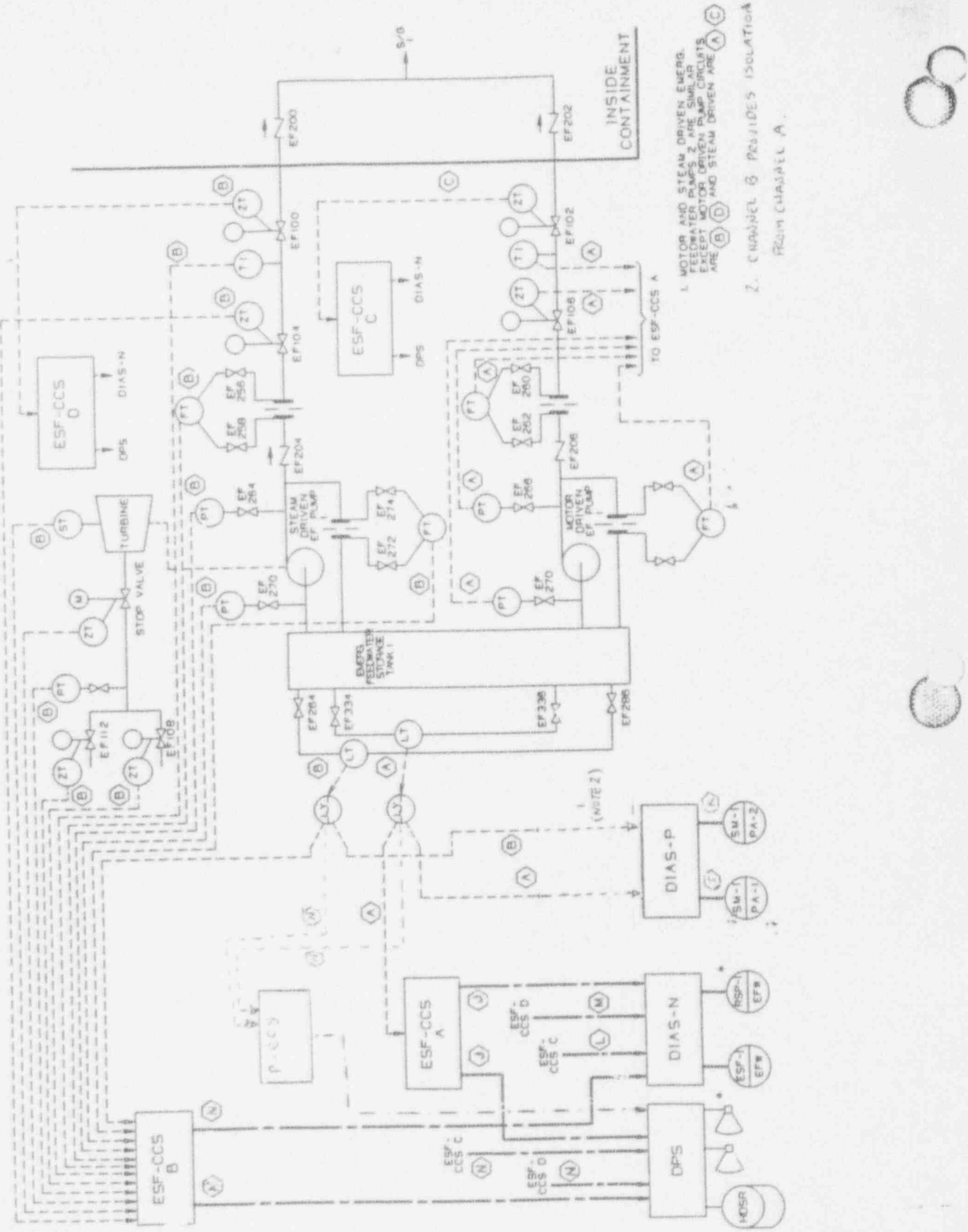
FILE: MCBD1



Amendment E
December 30, 1988

SYSTEM 80+™	STEAM GENERATOR-2 PRESSURE MCBD	Figure 7.2-28b

CALMA FILE:
I.SYS80\FIGURE739



Amendment I
December 21, 1990

SYSTEM 80+™	EMERGENCY FEEDWATER MCBD	Figure 7.3-19

7.5.1.1.7.1.1 Saturation Margin Sensors

Saturation Margin Monitoring (SMM) provides information to the reactor operator on:

(A) The approach to and existence of saturation.

(B) The existence of core uncovery.

The SMM utilizes inputs from the RCS cold and hot leg temperatures measured by RTDs, the maximum temperature of the top three Unheated Junction Thermocouples (UHJTC), and pressurizer pressure sensors. The UHJTC input comes from the output of the Heated Junction Thermocouple (HJTC) processing units. In summary, the sensor inputs are as follows:

Input	Range
Pressurizer Pressure	0-750 ~~0-1600~~ psia
Pressurizer Pressure Pressurizer Pressure	600-1650 psia 1500-2500 psia
RCS Pressure	0-4000 psia
Cold Leg Temperature	50-750°F
Hot Leg Temperature	50-750°F
Maximum UHJTC Temperature of top three sensors (from HJTC processing)	32-2300°F D
Representative CET Temperature	32-2300°F

7.5.1.1.7.1.2 Heated Junction Thermocouple (HJTC) Probe Assembly

The HJTC probe assembly measures reactor coolant liquid inventory above the fuel alignment plate with discrete HJTC sensors located at different levels within a separator tube ranging from the top of the fuel alignment plate to the reactor vessel head. The basic principle of operation is the detection of a temperature difference between adjacent heated and unheated thermocouples.

As pictured in Figure 7.5-2 the HJTC sensor consists of a Chromel-Alumel thermocouple near a heater (or heated junction) and another Chromel-Alumel thermocouple positioned away from the

TABLE 7.5-1

(Sheet 1 of 2)

SAFETY-RELATED PLANT PROCESS DISPLAY INSTRUMENTATION

Parameter	Number of Sensed Channels	Sensor Ranges (4)	Minimum Indicated Range(2)(3)	Location(1)
Pressurizer Pressure	4	0-750 psia		
Pressurizer Pressure	4	1500-2500 psia		
Pressurizer Pressure	4	~~0-1600~~ psia 600-1650	0-4000 psia	Control Room
RCS Pressure	2	0-4000 psia		
Steam Generator Differential Pressure (RCS)	4/SG	0-70 psid	0-70 psid	Control Room
Coolant Temperature (Hot)	8	525-675°F	50-750°F	Control Room
	4	50-750°F		
Coolant Temperature (Cold)	8	465-615°F	50-750°F	Control Room
	4	50-750°F		
Containment Pressure (Wide Range)	2	-5 psig to + 4 times design psig	-5 psig to + 4 times design psig	Control Room
Containment Pressure (Narrow Range)	4	-5 psig to + 1 times design psig	-5 psig to + 1 times design psig	Control Room
Steam Generator Pressure	4/SG	0-1524 psia	0-1524 psia	Control Room
Steam Generator Level (Wide Range)	4/SG	0-100%	0-100%	Control Room
Steam Generator Level (Narrow Range)	4/SG	0-100%		
Pressurizer Level	2	0-100%	0-100%	Control Room

TABLE 7.5-3

(Sheet 1 of 7)

POST-ACCIDENT MONITORING INSTRUMENTATION

Parameter	Number of Sensed Channels(5)	Minimum Sensor Ranges(6,3)	Minimum Indicated Range	Location(1,2)	Reg. Guide 1.97 Category
RCS Pressure	2	0-4000 psig	0-4000 psig	Control Room	1,2
Primary Safety Valve Position (Acoustic Leak Detector)	1/Valve	N/A	Closed/Not Closed	Control Room	2
In-containment RWST Level	2	0-100%	0-100%	Control Room	2
In-containment RWST Temperature	2	50-250°F	50-250°F	Control Room	2
Coolant Temperature (Hot)	4	50-750°F	50-750°F	Control Room	1
Coolant Temperature (Cold)	4	50-750°F	50-750°F	Control Room	1,3
Containment Pressure (Wide Range)	2	-5 psig to 4 times design psig	-5 psig to 4 times design psig	Control Room	1
Containment Pressure (Narrow Range)	4	-5 psig to 1 times design psig	-5 psig to 1 times design psig	Control Room	1
Steam Generator Pressure	2/SG	0-1524 psia	0-1524 psia	Control Room	1,2
Steam Generator Level (Wide Range)	2/SG	0-100%	0-100%	Control Room	1
Pressurizer Level	2	0-100%	0-100%	Control Room	1
Pressurizer Heater Status	1 pair/ heater bank	N/A	On/Off	Control Room	2
Pressurizer Pressure (high range) (high range)	4	1500-2500 psia(4)	Note 4	control Room	1
Pressurizer Pressure (mid range)	4	600-1650 psia(4)	Note 4	"	1
Pressurizer Pressure (low range)	4	0-750 psia(4)	Note 4	"	1

TABLE 7.5-3 (Cont'd)

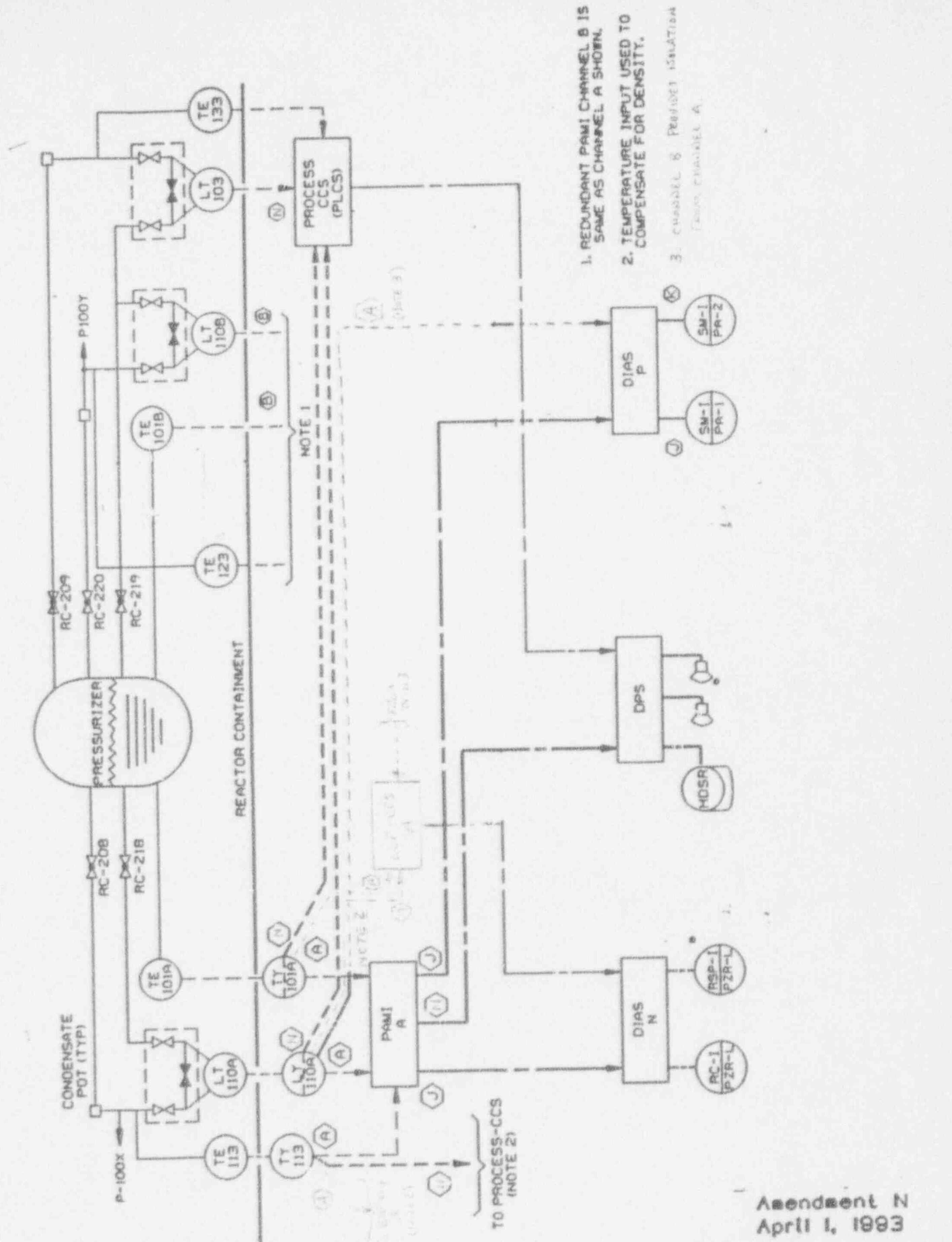
(Sheet 7 of 7)

POST-ACCIDENT MONITORING INSTRUMENTATION

Parameter	Number of Sensed Channels(5)	Minimum Sensor Ranges(6,3)	Minimum Indicated Range	Location(1,2)	Reg. Guide 1.97 Category

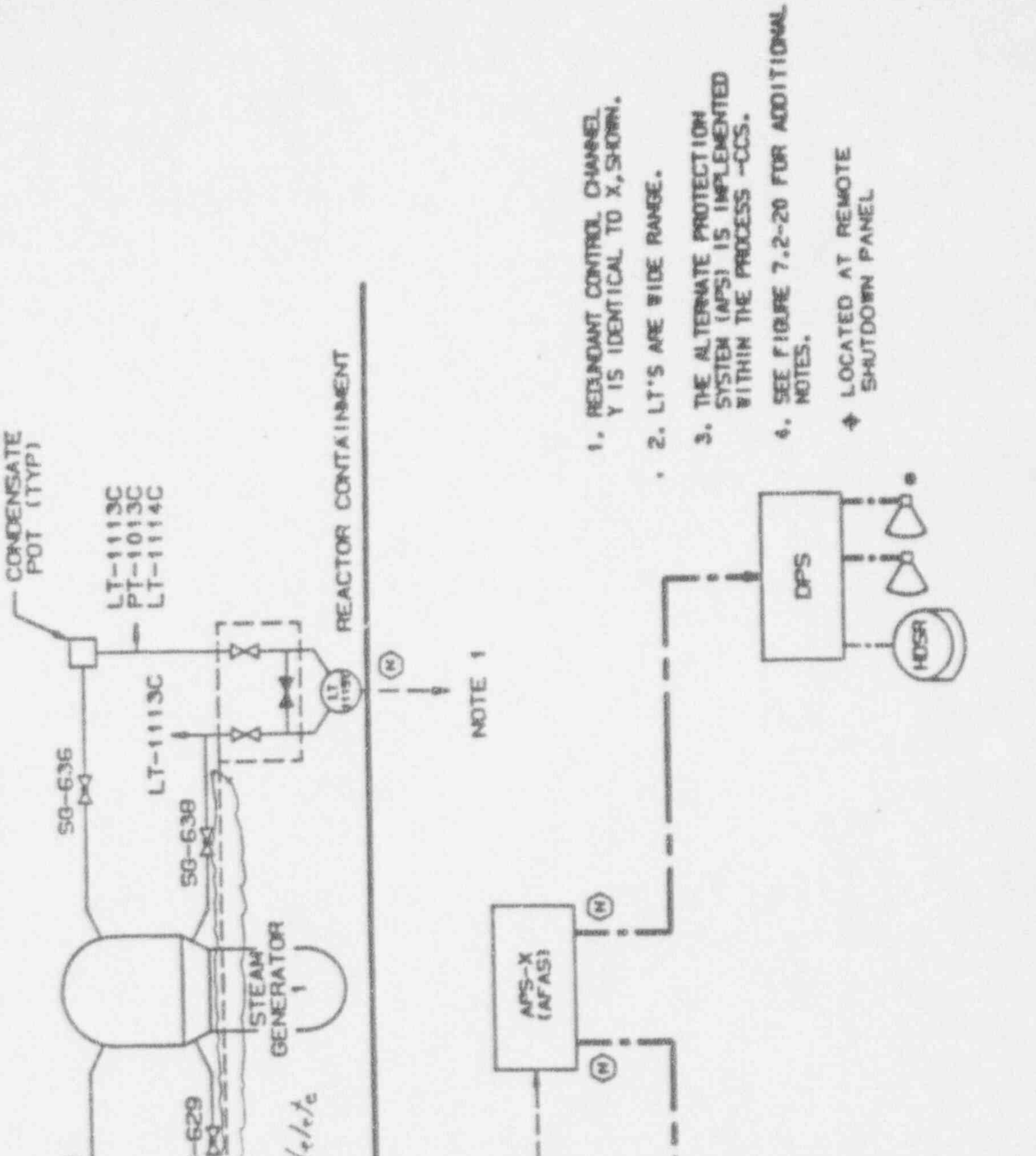
NOTES:

1. See Chapter 18 for type of readout.
2. All Category 1 variables are also recorded via the DPS Historical Data Storage and Retrieval program.
3. Post-accident monitoring instrumentation is qualified for the appropriate environmental conditions (refer to Section 3.11).
4. Degree of subcooling is calculated from ~~RCS~~ Pressurizer pressure, RCS temperature and core exit temperature parameters.
5. MCBDs are provided in appropriate sections of Chapter 7.
6. Post-accident channel accuracy is a time dependent function of post-accident environmental conditions.



Amendment N
April 1, 1993

SYSTEM 80+™	PRESSURIZER LEVEL MCBD	FIGURE 7.5-6



Amendment N
April 1, 1993

SYSTEM 80+™	ALTERNATE PROTECTION SYSTEM (AFAS-1) MCBD	Figure 7.7-25a

ATTACHMENT 2

9.5 OTHER AUXILIARY SYSTEMS

9.5.1 FIRE PROTECTION SYSTEM

9.5.1.1 Design Basis

The design basis of the System 80+ Standard Design fire protection system employs defense-in-depth systems approach in combination with an integrated program including operational surveillance, testing, maintenance, administrative controls, and Quality Assurance to provide a fire safe plant consistent with NUREG-0800, Section 9.5.1, "Standard Review Plan", and SECY-90-16 "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements." I J

9.5.1.1.1 Goals

A. Prevent release of radioactive contamination in excess of 10 CFR Part 100 limits.

B. Prevent loss of ability to achieve safe shutdown following fire.

C. Prevent fire from threatening more than any one ~~electrical channel or mechanical~~ division of equipment or components required to achieve cold shutdown.

D. Prevent fire from damaging more than any one ~~electrical channel or mechanical~~ division of safety related structures, equipment, or components.

E. Mitigate the potential of personnel injury due to fire. I

F. Preserve unit availability by limiting potential fire damage to an acceptable level.

G. Protect capital investment in the facility.

9.5.1.1.2 Objectives

A. Station design and layout to prevent the possibility of fire affecting redundant ~~channels and~~ divisions of equipment required for cold shutdown. Safe shutdown as defined in the Standard Review Plan pertains to cold shutdown as part of the System 80+ design philosophy. This includes potential interaction with other plant systems and to prevent a fire induced LOCA.

B. Plant layout to assure adequate access and egress routes for personnel protection.

C. Outside Containment and the Annulus: Provision of three-hour fire rated barriers between redundant divisions of safety-related equipment. Exceptions are control room and the remote shutdown panel room which are physically separated and electrically isolated from each other as described herein.

Safe cold shutdown can be achieved following fire in any area assuming all equipment in the fire area (or inside containment; at the specific location) is rendered inoperable and that reentry into the fire area for repairs and operator actions is not possible.

INSERT #1

~~Inside Containment and Annulus: Separation of redundant divisions by quadrant to provide sufficient spatial separation, as proven by engineering analysis in the annulus and at containment penetrations. Another option for separation is through use of mineral insulated jacketed cables which qualify as either a three hour rated barrier or a radiant heat shield. If it qualifies as a radiant heat shield, engineering analysis will verify that the heat shield coupled with minimum 20 ft. separation between redundant divisions with no intervening combustibles, and/or augmented with sprinklers and automatic fire detectors, will withstand any credible fire occurrence. Where redundant divisions of equipment normally used to achieve cold shutdown, by necessity, converge, an engineering analysis will be conducted (when sufficient design detail is available) to assure that cold shutdown can be achieved utilizing equipment which would not be affected by fire at the specific location. The engineering analysis will be maintained as part of the System 80+ design basis.~~

D. Fire detection and alarm systems to provide prompt detection and notification of fire.

E. Fixed automatic sprinkler systems to assure prompt fire suppression consistent with design objectives.

F. Manual fire fighting equipment for early fire suppression and for structural fire fighting.

G. HVAC systems to ~~mitigate~~ keep smoke migration beyond the area of fire origin.

H. A fire prevention program including housekeeping control of combustible materials, control of potential ignition sources, and a program of management inspections, audits and reviews.

I. A fire response program consisting of well trained and equipped plant personnel prepared at all times to assume fire fighting responsibilities.

J. Operations and maintenance programs for surveillance, testing and maintenance of fire protection systems and features. I

K. A Quality Assurance program to assure design methods and features are properly implemented. The Quality Assurance program also verifies that operations, maintenance, and surveillance programs are properly implemented.

L. Sufficient fire area compartmentation to preclude the presence of Category 1 risks. A Category 1 risk is defined in the Fire Hazards Assessment as an area where equipment or component damage and electrical faulting are unacceptable. An example would be a location where redundant equipment and components required for safe shutdown are susceptible to damage by a single fire. K

M. A fire protection program that complies with NUREG 0800 Standard Review Plan and CMEB 9.5-1, Rev. 2, July 1981: "Guidelines for Fire Protection of Nuclear Power Plants" and SECY Letter 90-16, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements." Specific deviations and technical justification are included in the fire hazards analysis.

The Design Basis Goals and Objectives as stated above will mitigate the potential of fire, provide for prompt detection should fire occur, provide automatic suppression and/or manual fire suppression capabilities as determined by the Fire Hazards Analysis, provide fire resistant barriers to mitigate fire propagation, protect redundant safety related trains of equipment from damage due to a common fire exposure, and preclude the potential release of radioactivity to the environment. I

9.5.1.2 General Design Guidelines

A. Outside containment redundant divisions of safety related electrical equipment are separated from each other by three-hour fire rated fire barriers. J Exceptions are control room and remote shutdown panel room which are physically separated, electrically isolated, and provide redundant shutdown capability. I ~~Individual~~ Transfer switches which transfer control from the Control Room to the Remote Shutdown Panel, are ~~separated from each other by three-hour fire rated barriers~~ located in the control room. K

INSERT #2

B. ~~Separation of redundant divisions by quadrant provide sufficient spatial separation in the annulus and at containment penetrations as proven by engineering analysis. Another option for separation is through use of mineral insulated jacketed cables which qualify as either a three hour rated barrier or a radiant heat shield. If it qualifies as a radiant heat shield, engineering analysis will verify that the heat shield coupled with minimum 20 ft. separation between redundant divisions with no intervening combustibles, and/or augmented with sprinklers and automatic fire detectors, will withstand any credible fire occurrence. Where redundant divisions of equipment normally used to achieve cold shutdown, by necessity, converge, an engineering analysis will be conducted (when sufficient design detail is available) to assure that cold shutdown can be achieved utilizing equipment which would not be affected by fire at the specific location. The engineering analysis will be maintained as part of the System 80+ design basis.~~ J

C. A fire protection water supply is installed, with redundancy and reliability to meet provisions of BTP CMEB 9.5-1.

D. Fixed automatic suppression systems are installed, engineered for the specific hazard to be protected in accordance with the design objectives as determined by the Fire Hazards Analysis. I

E. Portable fire extinguishers, fire hydrants, fire hose stations and supporting equipment are provided to facilitate manual fire fighting.

F. Ventilation systems are installed, including provisions for controlling spread of fire and smoke beyond the area of origin. HVAC systems are division specific; therefore, there are no dampers in barriers which separate redundant divisions of safety-related equipment. The exception to this is that there is a single opening in the divisional fire wall that separates the redundant air handling units. K An air intake duct which supplies make-up air to the redundant Control Room Systems passes through this opening. This arrangement enables make-up air to be drawn from either side of the facility and is necessary for nuclear safety reasons. This opening is protected with a combination fire and smoke damper. Smoke control capability is provided as part of HVAC system design to mitigate smoke spread beyond the area of origin.

G. Smoke control capability is provided as part of HVAC system design to mitigate smoke migration beyond the area of origin. INSERT #11

H. Multiplexed instrument and control signals by fiberoptic cables are provided to minimize the quantity of combustible exposed cable insulation in the plant.

I. Where fixed fire suppression systems are installed, provisions for control and drainage of water are included.

J. Systems and equipment are designed to mitigate the potential for fire due to equipment failure during the design basis seismic event. An example is the reactor coolant pump motor oil collection and drain system.

K. Fire protection (suppression and detection) equipment is designed to mitigate possible interaction with safety related equipment during the design basis seismic event. Interaction includes the potential for water spray or flood due to pipe failure. I

L. Fire hose standpipe system and a water supply in the Nuclear Annex are designed to withstand a design basis earthquake. Provide a water supply (250 gpm) to at least one hose station for at least 2 hours.

M. Respond to specific requirements of BTP CMEB 9.5-2, Rev. 1, July 1981.

N. ~~9.5.1.2 Safety-Related Fire Areas, Rooms, and Zones~~

Fire areas, rooms, zones and other areas containing equipment important to safety are identified in fire barrier drawings to establish the scope of the Fire Hazards Analysis and to assure compliance with Regulatory Guidance (see Figures 9.5.1-2 to 9.5.1-9. K

~~9.5.1.2.1 Description~~

~~A. Elevation 50+0 (Figure 9.5.1-2)~~

~~1. Nuclear Annex~~

~~a. Vital Instrumentation and Equipment Rooms, Division I, Channels A&C; Division II, Channels B&D.~~ I

9.5.1.3 Safe Shutdown Analysis

{See following pages}

9.5.1.3 FIRE PROTECTION SAFE SHUTDOWN ANALYSIS

9.5.1.3.1 ASSUMPTIONS

- The Fire Protection Safe Shutdown Analysis includes the effects of one worst case spurious actuation.
- Fire is postulated with or without loss of offsite power (which ever is the most severe challenge to the ability to achieve safe shutdown).
- Inside containment, cables for safe shutdown valve motor operators and instruments are three hour fire rated.
- Fire is not postulated concurrent with simultaneous, coincidental failures of safety systems, other plant accidents or the most severe natural phenomena.

9.5.1.3.2 FIRE PROTECTION SAFE SHUTDOWN DESIGN BASIS GOALS

- Achieve and maintain subcritical reactivity conditions in the primary system.
- Maintain reactor coolant inventory.
- Achieve primary system temperature and pressure conditions.
- Maintain Reactor Coolant System (RCS) process variables within those predicted for loss of AC power.
- Prevent fuel clad damage, failure of the primary system pressure boundary, or rupture of the containment boundary.

9.5.1.3.3 FIRE PROTECTION SAFE SHUTDOWN DESIGN BASIS OBJECTIVES

The following Design Basis Objectives are met in order to assure the Design Basis Goals stated above are satisfied:

- Maintain RCS pressure boundary integrity (i.e., reactor coolant pump seal integrity, CVCS letdown isolation, Safety Depressurization System isolation and RCS sample line isolation).
- Assure the reactivity control function maintains the available shutdown margin at greater than 1% $\Delta k/k$ with the highest worth control element assembly (CEA) fully withdrawn.
- Assure reactor coolant make up is available to maintain

reactor coolant in the pressurizer within prescribed limits.

- Maintain RCS decay heat removal function and cool down the RCS to cold shutdown conditions.

- Provide direct reading of process variables necessary to perform and control negative reactivity, reactor coolant pressurizer level and decay heat removal.

- Maintain support functions (process cooling, lubrication, etc.) for equipment required for safe shutdown.

9.5.1.3.4 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

- The RCS provides reactivity control by control element assembly (CEA) insertion and also removes decay heat from the core through natural circulation.

- Emergency Feedwater System (EFW) provides secondary side decay heat removal capability.

- Atmospheric Dump Valves provides secondary side pressure control capability.

- Shutdown Cooling System (SCS) provides residual heat removal function for cooldown from hot shutdown to cold shutdown conditions.

- Safety Injection System (SIS) provides makeup capability for inventory control and boron addition for reactivity control.

- Safety Depressurization System (SDS) provides primary system pressure control capability.

- Essential Chilled Water System (ECWS) provides chilled water for HVAC heat removal to all safety related room recirculation cooling units.

- Component Cooling Water System (CCWS) provides decay heat removal capability and equipment cooling for the Shutdown Cooling System, Safety Injection System, Essential Chillers, Emergency Diesel Generator Coolers, etc., as well as other non safe shutdown functions.

- Station Service Water System (SSWS) takes suction from the ultimate heat sink and provides cooling water flow to the CCWS heat exchangers for cooling and decay heat removal.

- The Control Building, Nuclear Annex, Subsphere and Diesel Generator Building Ventilation Systems provide ambient temperature control within parameters required to assure components function as intended to achieve safe shutdown conditions.

- Reactor coolant pump seal cooling is provided by either seal injection from the CVCS charging pumps or direct cooling from the CCWS.

- The Pool Cooling and Purification System provides decay heat removal from the spent fuel pool.

- The onsite Emergency Diesel Generators provide power for 1E busses for equipment power, control and instrumentation required to achieve safe shutdown conditions.

- The Combustion Turbine (AAC) provides onsite power to the permanent non-safety busses which provide power to the CVCS Charging Pumps and associated valves and controls.

Each of these systems includes adequate controls and instrumentation in the Control Room and at the Remote Shutdown Panel to assure safe shutdown can be achieved.

9.5.1.3.5 SYSTEMS WHICH REQUIRE ISOLATION

- SCS pressure isolation valves until RCS is cooled and depressurized to SCS entry conditions.

- SDS to prevent uncontrolled blowdown of the RCS.

- CVCS letdown to prevent uncontrolled letdown of the RCS.

- RCS sample lines to prevent uncontrolled letdown of the RCS.

- Main Steam System to prevent uncontrolled blowdown of the steam generators.

- Atmospheric Dump Valves to prevent uncontrolled blowdown of the steam generators.

- Main Feedwater System to prevent uncontrolled blowdown of the steam generators and steam generator over fill.

- Steam Generator Blowdown System and steam generator sample lines to prevent uncontrolled blowdown of the steam generators.

- EFW to prevent steam generator over fill.

Each of these systems includes adequate controls and instrumentation in the Control Room and at the Remote Shutdown Panel.

9.5.1.3.6 ASSOCIATED CIRCUITS

The potential for electrical interaction due to fire mandates that

a study be conducted to assure that redundant safe shutdown systems are not damaged by a single fire. Generic Letter 81-12 Rev. 1 defines Associated Circuits and provides guidance for documenting the Associated Circuits Study.

Outside of containment the System 80+ plant configuration provides complete separation of redundant safety related divisions by three hour fire rated barriers. Division 1 is located (plan) north of column line 17. Division 2 is located (plan) south of column line 17. An exception is the Control Room and the Remote Shutdown Panel room which are physically separated and electrically isolated and provide redundant shutdown capability. Transfer switches which transfer control from the Control Room to the Remote Shutdown Panel are located in the Control Room. Transfer switches are arranged such that when power is transferred from the Control Room to the Remote Shutdown Panel, manual operations in all four Vital Switchgear rooms are required to return control capability to the Control Room. Thus associated circuit interaction in the Control Room will not affect the ability to achieve safe shutdown from the Remote Shutdown Panel.

9.5.1.3.7 SAFE SHUTDOWN FOLLOWING FIRE OUTSIDE OF CONTAINMENT

As discussed in section 9.5.1.3.7, "Associated Circuits", redundant safe shutdown divisions are separated by column line 17. Each fire area is enclosed in three hour fire rated barriers. Three hour fire rated barrier walls are located along Column Line 17, except at elevations 115+6 and 130+6 where the Control Room is located. The exception to complete divisional separation is the Control Room and the Remote Shutdown Panel room which have redundant control function capability. They are physically separated and electrically isolated from each other. CESSAR-DC Figure 9.5.1 depicts the separation of redundant electrical divisions outside of containment.

Thus a fire in any fire area outside of containment will not affect redundant safe shutdown systems, equipment, or components.

9.5.1.3.8 SAFE SHUTDOWN FOLLOWING FIRE INSIDE CONTAINMENT

The Containment and Annulus are a single fire area. The only components inside the Containment and Annulus which are required for safe shutdown are motor operated valves and instruments associated with safe shutdown systems.

Inside the Annulus and Containment, three hour fire rated cable protective systems (i.e., mineral insulated cables) are used for cables associated with safe shutdown functions. An exception to the three hour fire resistance rating may be containment penetrations which are currently commercially available with a one hour fire resistance rating. Three hour fire rated containment penetrations will be purchased if available.

The only in situ combustible material inside containment that may be exposed to a fire is insulation of cables that are not associated with safe shutdown functions. Redundant trains of valves and instruments analyzed as an assured method of achieving safe shutdown are physically separated such that a potential fire will not affect redundant equipment as stated in section 9.5.1.3.9.

In situ combustible material inside containment is limited to those materials which are essential for unit operation (i.e., cable insulation, lubricants, etc.). The largest quantity of combustible materials is RCP motor lubrication oil. All potential leak points are enclosed in a seismically designed oil collection system which drains to a seismically designed oil collection tank. If oil were to escape from any reactor coolant pump, it would drain into the containment holdup volume. There are no safe shutdown components located in the containment holdup volume which may be damaged due to a fire at this location.

Transient combustible material will be administratively controlled to avoid unacceptable fire hazards.

9.5.1.3.9 PROTECTION OF REDUNDANT FUNCTIONS

1. OBJECTIVE: Maintain primary system pressure boundary integrity (i.e., reactor coolant pump seal integrity, CVCS letdown isolation , SCS isolation, SDS isolation and RCS sample line isolation).

   ANALYSIS:

   A. RCP seal integrity is maintained by either seal injection from the CVCS charging pumps or direct cooling from the CCWS. The CVCS is discussed in CESSAR-DC Section 9.3.4, and is shown in Figure 9.3.4-1. The CCWS is discussed in CESSAR-DC Section 9.2.2 and is shown in Figure 9.2.2-1. The RCP seals are discussed in CESSAR-DC Section 5.4.1 and are shown in Figure 5.1.2-2.

      Outside containment the two divisions of CCWS are separated by a three hour fire rated barrier. In addition, the redundant CVCS charging pumps are separated by a three hour fire rated barrier. However, each division of CCWS provides seal cooling for two of the four RCPs. Should one CCWS division be lost from a fire outside of containment, RCP seal integrity of the two RCPs cooled by the CCWS division is maintained through seal injection from the CVCS charging pump in the unaffected division. The seal injection line penetrating containment is located 90 degrees apart from each containment penetration for the CCWS supply and return line to the RCPs. Each of the CCWS supply and return lines to the RCPs has two isolation valves, (For division 1, the isolation valve located inside containment has

control power supplied from channel B and the isolation valve located outside of containment has control power supplied from channel A. For division 2, the isolation valve located inside containment has control power supplied from channel A and the isolation valve located outside of containment has control power supplied from channel B.). There is an isolation valve in the seal injection line located outside of containment. This valve has control power supplied from Channel C. Thus a fire outside containment cannot simultaneously isolate both seal cooling means.

Inside containment isolation and control valves on the CVCS seal injection, RCP controlled seal bleedoff and CCWS supply and return lines for the RCP seal coolers are protected such that spurious signals from a fire inside containment can not simultaneously isolate both RCP seal cooling means. Seal injection isolation valves on each side of the high pressure seal coolers are normally open with power removed from the valve operator. The CCW supply and return line isolation valves to each RCP are also normally open with power removed from the valve. The seal injection flow control and controlled seal bleedoff line valves are located near each associated RCP inside the Reactor Building crane wall. These valves are powered from the permanent non-safety electrical power busses. The containment isolation valves for the RCP seal cooler CCWS supply and return lines and the RCP controlled seal bleedoff line are powered from Class 1E busses. These valves are located close to the containment vessel and outside of the Reactor Building crane wall. Therefore, the containment isolation valves associated with the CCWS supply and return lines to the RCPs and the primary RCP controlled seal bleedoff line will not spuriously close from a fire which causes the seal injection flow control valves and controlled seal bleedoff valves near the RCPs to close and vise versa. Should both the RCP controlled seal bleedoff and CCWS containment isolation valves spuriously close, a relief valve located inside containment opens and allows continued RCP controlled seal bleedoff to the reactor drain tank.

B. The CVCS letdown line is discussed in CESSAR-DC Section 9.3.4 and is shown in Figure 9.3.4-1. The letdown line has two power operated valves in series. Each isolation valve is powered from a different division of Class 1E power and is separated and protected such that a fire inside containment can not prevent both isolation valves from closing.

C. Each division of SCS has two RCS pressure isolation valves in series. These pressure isolation valves are shown in CESSAR-DC Figure 6.3.2-1C and are discussed in

Section 5.4.7. Each valve has power supplied from a different Class 1E channel. In addition, the power is removed from one of the valves in series. This prevents spurious opening of both valves from a fire inside containment.

D. Each division of SDS from the pressurizer to the In Containment Refueling Water Storage Tank (IRWST) has two power operated valves in series. Each valve has power supplied from a different Class 1E channel. In addition, power is removed from one of the valves in series. Each of the redundant valves are located above the pressurizer where there is no potential fire exposure. Each division of SDS from the pressurizer to the reactor drain tank and from the top of the reactor vessel to the reactor drain tank has two power operated valves in series. Each valve has power supplied from a different Class 1E channel. The valves and cables are adequately separated and protected to prevent spurious opening of any two valves in series from a fire inside of containment. The SDS is discussed in CESSAR-DC Section 6.7 and is shown in Figure 5.1.2-3.

E. Primary sampling lines have a flow reducing orifice which restricts the flow to less than the normal makeup capacity. In addition, each sample line has a normally closed isolation valve inside containment and a normally closed isolation valve outside containment. Each containment isolation valve associated with a sample line penetration is powered from a different division of Class 1E power. Thus, a fire inside containment can only affect the operation of one of these valves. The sample system is discussed in CESSAR-DC Section 9.3.2, Containment Isolation is discussed in CESSAR-DC Section 6.2.4, and the containment isolation valves are shown on Figure 6.2.4-1.

2. <u>OBJECTIVE</u>: Assure the reactivity control function maintains the available shutdown margin at greater than 1% $\Delta k/k$ with the highest worth CEA fully withdrawn.

<u>ANALYSIS</u>: Reactivity control is maintained by the CEAs and by boration. The Safety Injection System (SIS) is the primary method of injecting boron into the primary system. The SIS is discussed in CESSAR-DC Section 6.3 and is shown in Figure 6.3.2-1. The SIS pumps and power operated valves are located outside containment. There are no power operated valves inside containment. Thus a fire inside containment will not affect the ability to maintain reactivity control. Each division of SIS outside of containment is separated by a three hour fire rated barrier.

3. <u>OBJECTIVE:</u> Assure reactor coolant make up is available to maintain reactor coolant in the pressurizer within prescribed

limits.

ANALYSIS: The Safety Injection System (SIS) is used for make up to the RCS. See item 2, "reactivity control", above for description and protection.

4. OBJECTIVE: Maintain reactor coolant decay heat removal function and cool down the RCS to cold shutdown conditions.

ANALYSIS:

A. Emergency Feedwater System (EFWS) provides decay heat removal from hot standby to hot shutdown conditions by supplying feedwater to each steam generator. The EFWS is discussed in CESSAR-DC Section 10.4.9 and is shown in Figure 10.4.9-1. Each division has a motor driven and a steam driven EFW pump. Each of these pumps is sized for full capacity so that only one pump per division is necessary to achieve safe shutdown. Each pump discharge line has two motor operated valves in series. The motor driven and steam driven EFW pump of each division feed into a common supply header. All pumps and power operated valves are located outside of the containment. Thus a fire inside of containment will not affect the EFW function. Outside of containment each EFW train is separated by a three hour fire rated barrier. In order to prevent steam generator over fill, the motor operated control valve at the discharge of each pump has power supplied from a different Class 1E channel compared to the associated pump controls. The valve and pump along with associated cables are located and routed through different fire areas to prevent losing both pump and valve control due to spurious signals.

B. Steam Generator pressure control is maintained by the atmospheric dump valves which are part of the Main Steam Supply System. These valves are discussed in CESSAR-DC Section 10.3 and are shown on Figure 10.3.2-1. Each of the four main steam lines has an atmospheric dump valve (ADV) and its associated block valve located upstream of the main steam isolation valves. These valves are located outside containment in the main steam valve houses (MSVH). Thus a fire inside containment cannot affect their operation. Each MSVH, which contains two of the four ADVs, is located on opposite sides of the Reactor Building and is separated by a three hour fire rated barrier. Only one steam generator and one of the ADVs associated with that steam generator are required for decay heat removal and cooldown. Each ADV in a division has power supplied from a different Class 1E channel in its respective division. Therefore, a fire outside containment can only affect the operation of the ADVs located in the division in which the fire occurs. Thus, the ADVs in the unaffected division will be

available to control pressure in the steam generator performing the cooldown function.

C. In order to prevent uncontrolled blowdown of the steam generator and steam generator overfeed, the main steam, main feedwater, and steam generator blowdown systems and the steam generator sample lines require isolation.

The Main Steam System is discussed in CESSAR-DC Section 10.3 and is shown in Figure 10.3.2-1. Each steam generator has two main steam lines. Each main steam line has a main steam isolation valve. Each main steam isolation valve has redundant solenoids powered from different Class 1E channels. In addition, these valves fail closed on loss-of-power. The main steam isolation valves are located outside of containment in their associated main steam valve house. Thus, a fire inside containment does not affect the operation of these valves.

The Main Feedwater System is discussed in CESSAR-DC Section 10.4.7 and is shown in Figure 10.4.7-1. Each steam generator has a economizer feedwater line and a downcomer feedwater line. Uncontrolled blowdown of a steam generator is prevented by two check valves in series on each of these lines. Steam generator over feed is prevented by closing the two feedwater isolation valves located in series on each of these lines. Each feedwater isolation valve in series has power supplied from a different Class 1E channel. In addition, these valves fail closed on loss of power. The feed water isolation valves are located outside of containment in their associated main steam valve house. Thus, a fire inside containment does not affect the operation of these valves.

The Steam Generator Blowdown System and the Process Sample System are discussed in CESSAR-DC Sections 10.4.8 and 9.3.2 respectively. The Steam Generator Blowdown System is shown in Figure 10.4.8-1. Each steam generator blowdown line and each steam generator sample line can be isolated by their associated containment isolation valves. Each containment penetration has a containment isolation valve located inside containment and a containment isolation valve located outside of containment. Each valve associated with a containment penetration is powered from a different division of Class 1E power. Thus, a fire inside containment can only affect the operation of one valve.

D. The Shutdown Cooling System (SCS) provides decay heat removal and cooldown after the primary system is cooled and depressurized to the point that allows opening of the RCS pressure isolation valves. The SCS cools the RCS

from hot shutdown to cold shutdown conditions. The SCS is described in CESSAR-DC Section 5.4.7 and is shown in Figure 6.3.2-1. The SCS has redundant divisions. Each division takes suction from a different RCS hot leg and returns the RCS after it is cooled directly to the reactor vessel. The majority of the SCS system is located outside of containment and the redundant divisions outside of containment are separated by a three hour fire rated barrier. Only the motor operated RCS pressure isolation valves are located inside containment. There are two valves in series in each of the redundant flow paths which are located by division 180 degrees apart. These valves are located near the crane wall on either side of containment such that they are over 100 feet apart. They are located at elevation 101+8 which is 10 feet above the floor elevation of 91+6. This distance is sufficient to ensure that one division of SCS is available to perform the cooldown function.

In addition, one of the valves in series in each flow path is normally deenergized during power operation to prevent potential spurious actuations from opening both valves in a division prior to reaching SCS entry conditions.

E. The Component Cooling Water System (CCWS) and the Station Service Water System (SSWS) transfer decay heat from the SCS to the ultimate heat sink. In addition, they provide process cooling to equipment and components required for safe shutdown. These systems are discussed in CESSAR-DC Sections 9.2.2 and 9.2.1 respectively and are shown in Figures 9.2.2-1 and 9.2.1-1 respectively. Each of these systems are located outside of containment and would not be affected by a fire inside containment. Outside of containment each division is separated by a three hour fire rated barrier. An exception is the CCWS cooling to the RCP seals which has valves located inside containment. See item 1A above for analysis of this item.

5. <u>OBJECTIVE</u>: Provide depressurization of the RCS to allow Shutdown Cooling System to be placed in service to obtain cold shutdown conditions.

<u>ANALYSIS</u>: RCS depressurization is accomplished utilizing the Safety Depressurization System (SDS). The SDS is described in CESSAR-DC Section 6.7 and is shown in Figure 5.1.2-3. Depressurization is accomplished by opening the valves and controlling flow from the pressurizer to the reactor drain tank. These valves are located inside containment. Two divisions of valves located in parallel are provided. Each division is powered from a different Class 1E division. Each division of SDS from the pressurizer to the reactor drain tank has two power operated valves in series. Each valve has power

supplied from a different Class 1E channel. One of the redundant lines is routed inside the pressurizer cavity, the other is routed outside of the pressurizer cavity. The valves and cables are adequately separated and protected (i.e. one division is inside the pressurizer cavity and one division is outside of the pressurizer cavity) to ensure one division of SDS is available for RCS depressurization in the event of a fire inside of containment.

6. OBJECTIVE: Provide direct reading of process variables necessary to perform and control negative reactivity, reactor coolant pressurizer level and decay heat removal.

   ANALYSIS: Instrumentation (Incore instrumentation, T-Hot, T-Cold, S\G Pressure, S\G Level, Pressurizer Pressure, Pressurizer Level, Neutron Flux): Cables for all of these instruments are three hour fire rated.

   A. Neutron Flux instrumentation, T-Hot and T-Cold are located inside the primary system and are not susceptible to fire damage.

   B. Pressurizer Pressure and Level instruments are located at the pressurizer. There are four channels of pressurizer pressure and level instrumentation. Each channel is located in a different quadrant around the pressurizer.

   C. S\G Pressure and Level instruments: Fire at either steam generator may damage instruments associated with that steam generator. However the other steam generator would not be affected and would be available to achieve safe shutdown. In addition, there are four channels of steam generator pressure and level instrumentation and each channel is located in a different quadrant around the steam generator.

7. OBJECTIVE: Maintain support functions (process cooling, lubrication, etc.) for equipment required for safe shutdown.

   ANALYSIS:

   A. Component Cooling Water System (CCWS) and Station Service Water System (SSWS) are discussed in , item 4E above.

   B. Lubrication: There is no equipment inside containment which requires lubrication for safe shutdown. Lubrication requirements outside of containment are divisionalized and separated by a three hour fire rated barrier.

   C. Ambient cooling:

      - The Essential Chilled Water System (ECWS) provides cooling water to area room coolers located outside

containment. These coolers are contained in the Control Complex, Reactor Building Subsphere, and Nuclear Annex Ventilation Systems. These systems are discussed in CESSAR-DC Sections 9.2.9, 9.4.1, 9.4.5, and 9.4.9 respectively, and are shown in Figures 9.2.9-1, 9.4-2, 9.4-5, and 9.4-8 respectively. Each of these system has two divisions which are entirely located outside of containment and are separated by a three hour fire rated barrier.

- The Diesel Generator Building Ventilation System maintains the ambient conditions within the diesel generator rooms to ensure operation of the diesel generators and controls. This system is discussed in CESSAR-DC Section 9.4.4 and is shown in Figure 9.4-7. This system is located outside of containment and each division is separated by a three hour fire rated barrier.

- Equipment located inside containment is qualified for high post accident temperatures. Therefore, containment cooling is not required to ensure operation of safe shutdown equipment following a fire.

8. <u>OBJECTIVE</u>: Remove decay heat from the spent fuel pool.

   <u>ANALYSIS</u>: Decay heat is removed from the spent fuel pool by the Pool Cooling and Purification System. The Pool Cooling and Purification System is discussed in CESSAR-DC Section 9.1.3 and is shown in Figure 9.1-3. All components associated with the spent fuel pool cooling function are located outside of containment and each division is separated by a three hour fire rated barrier.

9. <u>OBJECTIVE</u>: Provide an assured source of on-site electrical power to equipment and components required for safe shutdown.

   <u>ANALYSIS</u>: The assured source of electrical power is either of the emergency Diesel Generators for equipment and components powered from the Class 1E busses. The electrical distribution system is discussed in CESSAR-DC Section 8.3 and is shown in Figures 8.3.1-1 and 8.3.1-2. The emergency Diesel Generators and associated Class 1E busses are located outside containment and each division is separated by a three hour fire rated barrier. The Class 1E busses are separated from the non-1E busses by two isolation breakers in series. The CVCS charging pumps are powered from the permanent non-safety busses. Emergency on-site power is supplied to these busses by the combustion turbine. The permanent non-safety busses are located in the turbine building. The Turbine Building is separated from the Nuclear Annex by a three hour fire rated barrier. The combustion turbine is located in its own

structure which is separated from the Turbine Building and Nuclear Annex. Cables from the permanent non-safety busses are separated by the divisional three hour fire rated barrier after they enter the Nuclear Annex.

b. Vital Battery Rooms, Division I, Channels A&C; Division II, Channels B&D. I

K

c. Instrument Air Rooms.

d Unassigned Equipment Rooms. I

e Charging Pump Rooms. K

f. Chemical and Volume Control Equipment Room.

g. Component Cooling Water Pumps.

2. Reactor Building Subsphere

a. Turbine Driven Emergency Feedwater Pump Rooms, Divisions I and II.

b. Motor Driven Emergency Feedwater Pump Rooms, Divisions I and II. I

c. HVAC equipment areas (located in each quadrant).

NOTE: Fire rated walls are located along azimuths 0° to 180° and 90° to 270° to provide fire separation for the four quadrants.

3. Diesel Generator Buildings

a. Diesel Generator Rooms, Divisions I and II. K

B. Elevation 70+0 (Figure 9.5.1-3)

1. Nuclear Annex

a. Essential Battery Rooms, Division I, Channels A&C; Division II, Channels B&D. I

b. Electrical Equipment Rooms, Division I, Channels A&C; Division II, Channels B&D.

K

c. Remote Shutdown Panel Room. I

K

d. Cable chase, Division I, Channels A&C; Division II, Channels B&D. I

e. Essential Chiller Room, Division I, Channels A&C; Division II, Channels B&D.

f. Emergency Feedwater Tank Rooms, Division I, Channels A&C; Division II, Channels B&D. I

2. Reactor Building Subsphere

1. Fuel Pool Coolant Pump Rooms, Divisions I and II.

C. Elevation 91+9 (Figure 9.5.1-5) K

1. Nuclear Annex

a. 125 VDC Battery Rooms, Non-Essential N1 and N2 Equipment Rooms I

b. Unassigned Equipment Rooms.

c. Diesel Generator Building Vent 1 & 2 Rooms. K

d. Motor Control Centers.

e. RCP Switchgear. I

f. Emergency Feedwater Tank Rooms. K

2. Reactor Building I

a. Annulus.

b. Reactor Coolant Pumps Motor Oil Drain Tank.

c. Reactor Drain Tank. K

d. Cable concentrations, Division I, Channels A&C; Division II, Channels B&D.

e. Incore thermocouple cable concentrations.

f. Control Rod Drive Cable Concentrations.

D. Elevation 115+6 (Figure 9.5.1-6) I

1. Nuclear Annex

a. Control Room. K

b. Document Room.

c. HVAC (storage) Room.

d. Unassigned Equipment Room. I

e. Main Steam Valve House, Division I & II. K

f. Spent Fuel Pool. I

g. Subsphere Exhaust Equipment Room, Division I & II. K

h. Computer Room.

2. <u>Reactor Building</u> I

a. Control Element Drive Mechanism (CEDM) K

b. Steam Generators, I and II. I

c. Reactor Coolant Pumps A, B, C, D K

E. <u>Elevation 130+6 (Figure 9.5.1-7)</u>

1. <u>Nuclear Annex</u> I

a. Control Room HVAC Room, Division I. K

b. Control Room HVAC Room, Division II.

c. New Fuel Storage Area. I

d. Unassigned Equipment Room.

e. Annulus Exhaust Equipment Room, Division I & II.

f. Nuclear Annex Vent. Equip. Room, Division I & II. K

g. Nuclear Annex Exhaust Equip. Room, Division II.

F. <u>Elevation 146+0 (Figure 9.5.1-8)</u>

1. <u>Nuclear Annex</u> I

a. Unassigned Equipment Room. K I

b. Hot Tool Crib Rooms. K

2. <u>Reactor Building</u> I

a. Pressurizer. K

b. Containment Auxiliary Carbon Filter Units. I

c. CEDM Cooling Units.

d. Reactor Vessel.

G. Elevation 170+0 (Figure 9.5.1-9)

1. Nuclear Annex

a. Containment Purge Equipment Room.

b. Nuclear Annex Exhaust Equipment Room, Division I.

c. Fuel Pool Ventilation Equipment Room, Division I.

d. Fuel Pool Ventilation Equipment Room, Division II.

e. CCW Surge Tanks.

9.5.1.4 Fire Rated Barriers

9.5.1.4.1 Components of Fire Barriers

Fire barriers consist of architectural and structural features (walls, floors, and ceilings), assemblies to seal openings in fire barriers (doors, dampers, and penetration seal assemblies), and fire rated insulation material. ~~(cable wrap and radiant energy shields).~~ Each fire barrier component is tested in accordance with nationally recognized codes and standards to assure adequate fire resistance rating. INSERT #3

9.5.1.4.2 Architectural and Structural Features

Walls, floors, and ceiling assemblies designated as fire barriers meet the acceptance criteria of ASTM E119, "Fire Tests of Building and Construction Materials."

9.5.1.4.3 Door Units

Door units installed in designated fire barriers meet the acceptance criteria of ASTM E152, "Fire Tests of Door Assemblies" and NFPA 80, "Fire Doors and Windows." Door units include components (i.e., door leafs, frames, latches, closures, hinges, astragal strips, and kick plates).

Door units which are provided to meet multiple design criteria such as fire, flood, pressure, and security are reviewed and analyzed to assure they will withstand the potential fire exposure. Where possible, fire barrier doors with security (or other speciality) hardware are Listed or Approved as a fire rated door unit with the hardware as part of the door unit.

9.5.1.4.4 Fire Dampers

Ventilation dampers installed in designated fire barriers meet the acceptance criteria of UL 555, "Fire Dampers" and NFPA 90A, "Installation of Air Conditioning and Ventilation Systems."

To allow damper inspection and maintenance, access opening in ductwork is provided adjacent to each fire barrier.

9.5.1.4.5 Penetration Seals

Penetration seals in fire barriers for electrical and mechanical systems meet the acceptance criteria of ASTM E814, "Fire Tests of Through Penetration Fire Stops." Conduits which penetrate fire barriers are sealed in accordance with Edison Electric Institute, "Conduit Fire Protection Research Program," report dated 6/1/87. Where cable trays and HVAC ducts penetrate fire barriers, hangers on each side (or top side) of the barrier are designed to restrain the cable tray and ducts so that failure of hangers and collapse of cable trays or ducts on either side of the barrier will not pull the penetration seal assembly out of the opening.

9.5.1.4.6 Fire Insulating Material

The design philosophy of the System 80+ is to provide system/equipment ~~channel and~~ division separation to preclude the need for fire rated insulating material and radiant energy heat shields. In the course of detailed design and development of the Fire Hazards Analysis, it may be necessary to use these materials to assure fire safety in accordance with the Standard Review Plan. ~~If necessary, fire rated insulating material will be rated in accordance with ASTM E119 for architectural features. Components which may be protected by fire rated insulating material include structural steel, redundant safety related cables, and safety related components.~~

INSERT

INSERT #4

Electrical components protected by fire insulating material have ampacity derated based on insulating material property.

place

9.5.1.5 FIRE MITIGATING FEATURES

space

9.5.1.5.1 Isolation/Containment of Flames, Heat, Smoke, and Hot Gases

Isolation/containment of fire and products of combustion are achieved by implementing elements of the defense-in-depth concepts.

The System 80+ minimizes the available quantity of combustible material by use of fiber optic cable which reduces the number of control and signal cables (by an estimated order of magnitude

from that which would otherwise be required). Equipment location and separation by fire barriers as stated above serves to provide inherent containment of fire spread. Penetrations in fire barriers are designed to contain combustion products as well as prevent fire spread. Ventilation systems are designed to provide smoke control capabilities which are necessary to preclude the possibility of redundant safety related equipment from being damaged by fire and spread of products of combustion. The ventilation system for each area is arranged to ventilate products of combustion without spread to other areas.

The control building ventilation system is provided with separate outside air intakes for the control room separate from the remainder of the control complex including the remote shutdown room. Separate ductwork is utilized for the control room and the remote shutdown room to eliminate smoke migration between the two areas.

INSERT #5

~~The Control Complex has a smoke control system which utilizes dedicated smoke exhaust fans, smoke dampers and 100% outside air supplied by the Control Complex air-handling units. The smoke purge fans are sized to exhaust three cfm per sq. ft. The smoke purge system is manually activated by the control room operator.~~

In the subsphere, electrical equipment rooms A, B, C and D on elevation 50+0 are separated by channel with 3 hour fire resistance barriers. The two channels within a division share a common ventilation system, but are separated by fire dampers. Smoke purge fans are utilized to prevent smoke migration from one channel to the other in the same division.

Smoke migration between divisions in the nuclear annex is prevented by providing a 3 hour fire resistance wall between divisions with all penetrations sealed to maintain the 3 hour fire resistance barrier. No HVAC ducts will penetrate the divisional wall. Separate HVAC systems are provided for each side of the divisionally separated building. The stairwells are pressurized to prevent smoke from entering and migrating between elevations.

The ventilation systems handle smoke purge by isolation of supply air in the area in which the fire occurred. The normal exhaust system for the area will purge the smoke providing a slight negative pressure to the area in relation to surrounding areas still receiving supply air. The exhaust filter unit is bypassed in the smoke purge mode. This mode of operation is manually activated ~~by~~ the control room. The recirculation cooling units in an area with smoke will need a maintenance check to see if the prefilter needs replacing and the cooling coils need to be cleaned after the smoke purge is completed.

from

9.5.1.5.2 Interior Finish Materials

Structural materials are classified as noncombustible or fire resistive.

Interior finish, exposed thermal insulation, radiation shielding, and acoustical materials meet the following criteria in the installed configuration:

A. Flame spread of 25 or less

B. Smoke development of 450 or less

Floor coverings meet the following criterion in the installed configuration:

- Minimum critical radiant flux of $0.45 W/cm^2$

Flame spread and smoke developed are measured in accordance with ASTM E-84, "Test for Surface Burning Characteristics of Building Materials." Critical radiant flux is measured in accordance with ASTM E-648, "Test for Critical Radiant Flux of Floor Covering Systems Using a Radiant Heat Energy Source."

If it is necessary to select a specific material which does not meet or has not been tested to the above qualifications (in the installed configuration), an engineering analysis will confirm that the General Design Guidelines are met and there is no reduction in the fire safe quality of the plant.

9.5.1.5.3 Means of Egress

Personnel egress in the Nuclear Annex is arranged to meet provisions of NFPA 101, "Life Safety Code" or NFPA 101m, "Alternative Approaches to Life Safety."

There are stairs in each quadrant of the Nuclear Annex enclosed by two-hour fire rated walls. Each stair tower is pressurized by a dedicated fan mounted at the top of the tower. Exit pathways are clear and unobstructed, allowing personnel egress/access.

Access/egress into the Containment Building is through two personnel air locks, one located on elevation 115+6 and one located on elevation 146+0.

Sealed beam, battery powered emergency lighting units are installed to illuminate emergency egress paths in accordance with standards of NFPA 101, "The Life Safety Code."

INSERT #6

Sealed beam, 8-hour minimum battery powered emergency lighting units are provided for all areas and access to areas that must be occupied for safe shutdown of the plant following a fire. J

9.5.1.6 ~~Safe Shutdown Following Fire~~

The System 80+ plant arrangement and layout provides inherent separation of safety related systems, equipment and components, divisions and channels. The plant arrangement permits the unit to be taken to cold shutdown following a fire without the need to implement repairs or for operators to perform extraordinary manual actions outside of the control room or remote shutdown panel room. I

In the Nuclear Annex, each division of safety related equipment are separated by three-hour fire rated barriers. Exceptions are the control room and the remote shutdown panel room which contain safety related equipment of each division and channel. The control room and the remote shutdown panel room are essentially redundant to each other so that fire in either room will not affect the ability to achieve cold shutdown from the unaffected control system. K I

Electrical power, control, and instruments are separated and electrically independent to preclude electrical interaction and associated circuit failures in accordance with IEEE 384-1, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits." Associated circuits, as defined in Revision 1 to Generic Letter 81-12, will be avoided. J

~~INSERT #6~~ ~~INSERT #6~~

Cables of redundant safety related divisions and channels enter the Reactor Building on elevation 91+9. Division 1, which consists of Channels A and C, enters the Reactor Building from opposite sides, as does Division 2, which consists of Channels B and D. Channels A and B, which enter the Reactor Building in close proximity, are separated by a three-hour fire rated barrier. Likewise, Channels C and D, which enter the Reactor Building in close proximity, are separated by a three-hour fire rated barrier. These cables then transgress the annulus. Each safety related channel enters the annulus in a separate quadrant and is separated from the other safety related channels by at least 20 feet, without intervening combustibles. Where it is not possible to maintain 20 feet without intervening combustibles, cables are enclosed in three-hour fire rated barriers or heat shields until 20 feet separation is achieved. Heat shields and separation distance will be justified by engineering analysis. In addition, separation will be augmented with sprinklers and automatic fire detectors if required by the analysis. Cable ampacity will be derated in accordance with the characteristics of the insulating material. I J I J

~~Inside containment, safety related cables generally are confined to their respective quadrants. Where redundant divisions of safety related cables normally used for cold shutdown converge inside containment, an engineering analysis confirms that cold shutdown can be achieved utilizing systems and equipment which would not be affected by fire at that location. In each potential Reactor Building fire scenario, cold shutdown is assured by operation from the control room or remote shutdown panel without repairs or extraordinary operator action outside of the control room or remote shutdown panel room.~~

9.5.1.7 Fire Protection/Detection/Alarm Systems

The fire ~~penetration~~ protection water supply and distribution system configuration is as shown in Figure 9.5.1-1, "Fire Protection Water Distribution System."

9.5.1.7.1 Fire Pumps and Water Supply

The fire protection water supply is provided by two, 300,000 gallon ground level storage tanks designed in accordance with NFPA 22, "Standard for Water Tanks for Private Fire Protection." Each tank is equipped with a roof vent, roof access hatch, inside and outside ladder, overflow pipe, and a water level indication instrument.

Each tank has an automatic fill system supplied from the plant treated water system. Use of treated water will preclude potential system problems and deterioration associated with raw water (i.e., biological organism invasion such as Asiatic clams and microbiologically induced corrosion). The fill system is designed to refill either tank within eight hours. Water storage tanks are heated as necessary to preclude freezing. Where heating is not practical, pumps are designed to automatically begin recirculation through tanks when ambient temperature approaches the point where freezing obstructs water flow to the pumps. I

There are two full capacity fire pumps. Each fire pump is sized to provide both the maximum water volume and system pressure demand. One pump is electric motor driven and one is diesel engine driven. Each pump is arranged to take suction from either tank. Each fire pump unit is UL Listed for the specific application. Fire pumps are designed in accordance with NFPA 20, "Standard for Installation of Centrifugal Fire Pumps." Each fire pump includes an air release valve, set of suction and discharge gauges, and main relief valve. The electric motor driven pump has a recirculation relief valve. The diesel engine driven pump is equipped with an adjustable speed governor, overspeed protection, and redundant battery units including chargers. Controllers for each pump are

UL Listed and include adjustable time delay starters and a mercury pressure switch with high and low pressure settings. The following alarms are provided:

A. Electric driven pump

1. Pump running

2. Loss of power

B. Diesel driven pump

1. Engine running

2. Controller in "manual position"

3. Low engine oil pressure

4. High engine coolant temperature

5. Failure to start

6. Shutdown from overspeed

7. Battery failure

8. Battery charger failure

The diesel fuel oil storage tank is sized to provide an eight-hour fuel supply to the diesel engine driven fire pump. The tank is located in the diesel driven pump room for environmental control and to assure fuel quality.

The motor driven fire pump and the diesel engine driven fire pump are separated by a three-hour fire rated barrier to assure that both pumps would not be damaged by a single fire. Each fire pump room is protected by an automatic sprinkler system to further reduce fire exposure.

Discharge piping of each pump is interconnected so that either pump can supply either connection to the underground water distribution system.

The fire pump test header includes a flow meter to facilitate fire pump testing and a hose header to facilitate system flushing.

Fire pump and storage tank piping is designed to provide a fully adequate water supply to sprinkler and fire hose standpipe

I

systems with one fire pump and one water storage tank out of service.

The electric motor driven fire pump is powered by the unit auxiliary power supply. Back-up power is provided by the site alternate AC power supply combustion turbine and electrically protected so that fire in the power house will not interrupt pump operation. I

A jockey pump provides system pressure maintenance to avoid starting of the main fire pumps under nonfire conditions. K

9.5.1.7.2 Water Distribution System, Hydrants, and Hose Houses

Underground water distribution piping is cement lined ductile iron or plastic which is UL Listed or Factory Mutual Approved for fire service. Interior and above ground pipe is galvanized carbon steel which complies with ASTM A53, "Standard Specification for Welded Pipe, Steel, Black and Hot Dipped, Zinc Coated and Seamless Steel Pipe." Piping is "looped" around the power block and cross- connected within the Nuclear Annex so that sprinkler systems have redundant water supply flow paths. Two pipes penetrate containment, to provide redundant water supplies to primary and back-up fire protection systems.

Piping is sized based on water flow with the shortest flow path out of service. Calculations are based on anticipated internal pipe roughness after 60 years of service. I

Sprinkler systems and hose station connections to the water distribution system are arranged so that a single impairment will not isolate primary and secondary protection for any area.

Fire hydrants are located about 250 feet apart around the yard loop. Hydrants are provided with individual isolation valves so that they can be individually isolated for repair. Each hydrant has 2-2½ inch outlets individually controlled by gate valves. Fire hose houses are located near alternate fire hydrants. Hose house equipment includes:

A. 350 ft of 2½ inch fire hose

B. 150 ft of 1½ inch fire hose

C. 2-2½ inch x 1½ inch gated wye connectors

D. 2-2½ inch adjustable spray nozzles

E. 2-1½ inch adjustable spray nozzles

F. 2-2½ inch hose coupling gaskets

G. 2-1½ inch hose coupling gaskets

H. 4 coupling spanner wrenches I

I. 1 hydrant wrench

Where hydrants or hose houses are subject to damage by vehicle damage, appropriate guards and barriers are provided for protection.

Sectional isolation valves are located throughout the water distribution system to assure that any portion of the distribution system that serves buildings containing safety related systems, equipment, and components can be ~~impaired~~ repaired without isolating primary and secondary fire protection. J

INSERT # 7

The fire protection water distribution system complies with NFPA 24, "Standard for Private Fire Service Mains."

Piping, valves, fittings, and fire hydrants are designed for 175 psi operating pressure.

9.5.1.7.3 Automatic Sprinkler Systems

A. Description

Automatic preaction sprinkler systems are utilized for fixed fire protection in the Nuclear Annex, Reactor Building, and the alternate AC source - Combustion Turbine, as determined by the Fire Hazard Analysis. Wet pipe automatic sprinkler systems are used where preaction type systems are not mandated by the plant Design Basis. I

A preaction sprinkler system consists of a piping distribution system which supplies water to sprinkler heads which are located based on engineering analysis and requirements of NFPA 13, "Standard for Installation of Automatic Sprinkler Systems," to assure adequate water distribution and to preclude the possibility of interference with the water distribution pattern due to obstruction by other plant equipment and components. Sprinkler heads are normally closed and are actuated by heat sensitive elements. Actuation temperatures of these elements are based on the individual location and application. Distribution piping between the system control station and sprinkler heads is

normally dry and supervised with air or nitrogen. Water is held at a speciality "preaction valve" at the system control station. The system includes a fire detection subsystem activated by fire or smoke detection devices, selected by engineering analysis for the specific location and application based on the Fire Hazards Analysis. Upon activation of a fire detection device, the automatic preaction control valve opens, allowing water into the piping system. J

Water is then discharged only through sprinkler heads in which the heat sensitive element has actuated, thereby applying water only to the area involved in fire. Each preaction sprinkler system has a manual control valve immediately upstream of the preaction control valve and mechanical trim to accommodate testing and maintenance. Alarms monitor system air pressure and water flow. Each system alarms and annunciates locally and in the control room to alert station personnel to actuation. The main control station is located outside of the protected area. I

Two inch drains and inspector's test connections are provided to test system water flow and alarm operation. Each drain and test connection is arranged to discharge into a station drain for water control.

B. Coverage

INSERT # 8

~~The following areas are protected by automatic preaction sprinkler systems:~~

~~1. Elevation 50+0, Nuclear Annex~~

~~a. Janitorial/Health Physics Storage/Work Area.~~

~~b. Maintenance Work Areas.~~

~~c. Personnel Aisles.~~

~~d. General Storage Areas.~~ K

~~e. Diesel Generator Buildings.~~

~~f. Primary Chemistry Lab Area.~~

~~g. Chemical and Volume Control Pump Room.~~

~~h. Chemical and Volume Control Equipment Room.~~

2. Elevation 50+0, Reactor Building

   a. Turbine Driven Emergency Feedwater Pump Rooms, Divisions I and II.

3. Elevation 70+0, Nuclear Annex

   a. Channel A, B, C, D Personnel Access Aisles.

   b. Essential Chilled Water Areas.

4. Elevation 91+9, Nuclear Annex

   a. Personnel Access Areas.

   b. Radiation Access Control.

   c. Maintenance Areas - (Hot Machine Shop, Decontamination Room and Truck Bay).

5. Elevation 115+6, Nuclear Annex

   a. Storage Room.

   b. Maintenance Area.

   c. Tool Storage Room.

   d. Personnel Decontamination Areas.

   e. Break Room.

6. Elevation 115+6, Reactor Building

   a. Reactor Coolant Pump Motors.

7. Elevation 130+6, Nuclear Annex

   a. Technical Support Center.

   b. Maintenance Area.

8. Elevation 146+0, Nuclear Annex

   a. Maintenance Area.

   b. Hot Tool Crib Rooms.

   c. Personnel Decontamination Areas.

K

Sprinkler system design specifications (i.e., design density over the designed operating area) is determined by the Fire Hazards Analysis. Each system is designed based on the available water supply with 750 gpm reserved for hose streams.

C. Systems Interaction

1. Sprinkler system piping is seismically restrained to avoid interaction with systems, equipment, and components which must function following the design basis seismic event.

2. Sprinkler head locations are selected and analyzed to assure that water spray does not expose redundant equipment required to achieve cold shutdown or high voltage electrical equipment which may result in a personnel hazard. I

3. Sprinkler systems are analyzed to assure that pipe break/water spray does not potentially expose redundant equipment required for cold shutdown.

4. Sprinkler system drains and test connections are routed to unit drains to control water discharge.

5. In areas where equipment is subject to damage by water accumulation, floor drains are provided or equipment is installed on elevated platforms to avoid damage.

6. Sprinkler heads are located as required by NFPA 13. Other plant equipment and components are located so that they do not obstruct the designed sprinkler water discharge pattern. If obstruction is unavoidable, additional sprinkler heads are installed to assure proper water distribution.

7. Where installed, automatic sprinkler systems are considered primary protection. Portable extinguishers and fire hose stations are provided for back-up protection.

INSERT #9

Sprinkler system components including manual isolation valves, preaction control valves, pipe, fittings, hangers, sprinkler heads, and detectors are Underwriter's ~~Laboratory~~ Laboratories Listed or Factory Mutual Approved for use in fire protection systems. An exception to Listed or Approved equipment is containment isolation valves which are not available as Listed or Approved.

tank supplying a 150gpm Seismically designed pump

9.5.1.~~7.3~~ 7.4 Fire Hose and Standpipe Systems

Fire hose and standpipe systems consists of piping connections to the water distribution system, manual isolation valves, and 1½ inch fire hose. These systems are installed in accordance with NFPA 14, "Installation of Fire Hose and Standpipe Systems."

18,000

Fire hose and standpipe systems are designed to be operational following the design basis earthquake. The primary water supply to the standpipe system is from the fire protection water distribution system. Each connection of the standpipe system to the fire protection water distribution system includes a manual isolation and a back flow prevention check valve which are seismically qualified. A ~~20,000~~ gallon seismically designed ~~pressurized~~ water storage located ~~on the roof of~~ in the Nuclear Annex is connected to the fire hose standpipe system downstream of the check valves. The ~~20,000 gallon pressurized~~ water storage tank will provide ~~250 gpm~~ at a minimum of 65 psi to any fire hose stations in the Nuclear Annex or Reactor Building for two hour duration. In the event of loss of the fire protection water distribution system following a seismic event, the fire hose standpipe system can supply the specified volume and pressure to two ~~one~~ fire hoses in the safety related portions of the station.

tank supplying a 150gpm seismically designed pump

75gpm Pump and two

Fire hose stations are designed for Class III service (for use by building occupants and a fully trained structural fire brigade) as defined by NFPA 14, "Fire Hose and Standpipe Systems." I

Standpipe system piping is sized to supply 500 gpm at a minimum 65 psi pressure from the primary water supply and ~~250~~ 150 gpm from the seismically designed back-up water supply. Each hose connection to the standpipe includes a 1½ inch and a 2½ inch connection. Connections have pressure reducing orifices if necessary to maintain a maximum system pressure at 100 psi for firefighter safety.

Hose stations are located so that any location where safety related equipment may be damaged by fire can be reached with at least one effective hose stream.

Hose stations are equipped with 1½ inch fire hoses which are a maximum of 100 feet long. Hose stations which protect electrical equipment have adjustable spray nozzles qualified for use on energized electrical equipment.

Laboratories

Fire hoses, isolation valves, and hose nozzles are Underwriter's ~~Laboratory~~ Listed or Factory Mutual Approved for use in fire service. An exception is containment isolation valves which are not available as listed or Approved.

9.5.1.~~5~~7.5 Portable Fire Extinguishers

Portable fire extinguishers are located and arranged in accordance with NFPA 10, "Standard for Installation and Use of Portable Fire Extinguishers." An exception is that fire hose stations are utilized for Class A fires except in the control room and computer room where a water based extinguisher rated at 2A is installed. I J

Portable extinguishers are located such that extinguisher can be reached with a maximum of 75 feet of travel from any protected location. An exception is that in high radiation areas where the Fire Hazards Analysis determines that there is a minimum of combustible materials and a minimum of risk to safety related equipment or equipment necessary to maintain unit availability, fire extinguishers are located outside of the area where responding fire brigade members can obtain an extinguisher and carry it into the area for use. This is consistent with ALARA principles.

Due to the potential for chemical corrosion of safety related equipment and components, dry chemical extinguishers are not installed in safety related portions of the station. Dry chemical extinguishers are located in the fire brigade equipment room and are used at the discretion of the fire brigade captain. I

Inside containment, during power operation, fire extinguishers are located near the personnel access portals (rather than throughout containment). During maintenance outages, additional fire extinguishers will be moved into containment to support maintenance activities.

Fire extinguishers are located to be accessible. Locations are clearly marked to be prominently visible.

Fire extinguishers are Underwriter's ~~Laboratory~~ Laboratories Listed or Factory Mutual Approved for use in fire protection service.

9.5.1.~~6~~7.6 Fire Detection and Alarm System

INSERT #10

~~A fixed automatic fire detection system is installed in the Nuclear Annex and portions of the Reactor Building. Areas covered by the fire detection system are established by the Fire Hazards Analysis based on the potential hazard risk to safety related equipment and equipment necessary to maintain unit availability, potential detector effectiveness (based on engineering technique of NFPA 72, "Fire Detection and Alarm Systems"), and ALARA concerns. The fire detector system design~~

philosophy is to cover areas which contain major electrical equipment and components (such as control rooms, system transfer switches, computers, switchgear, motor control centers, battery, inverters, and technical cabinets), major safety related pumps, ventilation equipment areas, and areas containing substantial quantities of combustible material such as change room storage, contaminated area step off pads, and laundry areas. I

The type of fire detectors considered for use in the System 80+™ are as follows: K

A. Heat detectors - designed to operate at predetermined ambient temperature.

B. Ionization and photoelectric smoke detectors - designed to operate in the presence of particles of combustion.

C. Flame detectors - designed to operate by detection of infrared, visible, or ultraviolet radiation.

D. Continuous line type detection - designed to operate when exposed to a predetermined ambient temperature. ~~rate of rise~~.

Detectors are specifically selected for each location based on potential fire hazard, need for timely actuation, ambient conditions, ventilation and ceiling height, as determined in the Fire Hazards Analysis. I

~~Detectors~~ Spot type detectors are "addressable." The central control panel is a microprocessor based "intelligent" system. This arrangement allows detector sensitivity and function to be determined at the control panel.

Manual pull stations are addressable and are located as determined by the Fire Hazards Analysis. Either manual pull stations or individual fire detectors can activate the central control panel which initiates alarm and annunciation in the control room and locally in the vicinity of the activated device.

The control panel is located in the control room for operator convenience.

The fire detection and alarm system is powered from the station auxiliary, safety grade, power distribution system. The control panel contains back-up batteries capable of supplying power to detection system for 24 hours consistent with requirements of NFPA 72, "Fire Detectors and Alarm Systems."

Failure of the fire detection and alarm system would not affect operation of other plant systems.

Laboratories

Fire detectors, control panels, and manual pull stations are Underwriter's [redacted] Listed or Factory Mutual Approved for fire protection service.

9.5.1.[redacted] 8 System Interfaces

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9.5.1.[redacted] 8.1 Emergency Lighting

Sealed beam, battery powered lights are located, as determined by the Fire Hazards Analysis, for personnel egress [redacted] 115-6 in accordance with NFPA 101, "Life Safety Code," as well as in the control room, Technical Support Center, Operations Support Center, the Remote Shutdown Panel Room, and the stairway which provides access from the Control Room on elevation ~~70+00~~ and to elevation [redacted] 70' where the ~~reactor trip switchgears are located. Emergency lights will also be provided along the pathways between the Control Room and the transfer switches which are used to transfer~~ [redacted] Remote Shutdown Panel Room is located.

K

Batteries of these emergency lights are designed for eight hours continuous operation following loss of station auxiliary power. Bulbs are located and directed so that adequate illumination is provided and is not obstructed by plant equipment and components.

Battery powered, emergency lighting units are Underwriter's Laboratory Listed.

9.5.1.[redacted] 8.2 Ventilation Systems

Fire and smoke control are recognized as important elements of the overall fire protection program. The ventilation systems are designed in accordance with NFPA 90A, "Air Conditioning and Ventilation Systems" and NFPA 204M, "Smoke Control Systems."

I

Ventilation Systems are division-specific so that fire or smoke in an area containing a safety related division of equipment cannot migrate through the ventilation ducts to an area containing the redundant division of safety related equipment. Fire dampers are installed in fire rated barriers and have the same fire resistance rating as the barrier. Exceptions are the Containment Purge and Pressure Control Systems and Annulus Ventilation System which must function following some plant design basis accidents to prevent release of radioactivity. Fire dampers are not installed in these systems because failure or spurious actuation would interfere with system safety function. [redacted] The Nuclear Annex Control Complex Smoke ~~Control~~ Purge System has motor operated Combination three hour fire rated [redacted] and smoke control dampers [redacted] installed in lieu of thermally operated, automatic closing fire dampers as described below.

The smoke control design philosophy is to allow for smoke venting from any plant area without spreading to adjacent areas, to maintain plant habitability for operator protection and to ensure protection of the public. The containment, subsphere, fuel pool, nuclear annex and two diesel buildings are each served by 100% outside air and 100% exhaust ventilation systems.

Smoke control and exhaust is accomplished by aligning the ventilation to supply 100% outside air and to exhaust directly to the outside. Smoke and gases containing radioactive materials are routed through a filter train to the unit vent if a radioactive signal is received. The control complex has smoke exhaust fans to remove smoke from specific areas as determined by control operators utilizing signals from smoke detectors located in exhaust and return air ducts. The control operator aligns dampers to exhaust an area where fire occurs while isolating exhaust and return air in adjacent areas while supply dampers remain open to create a slight positive pressure in adjacent areas. I

During the smoke purge mode of operation, the filter units are isolated and the smoke is bypassed around the filter units to the atmosphere. The smoke purge is manually activated by the control room after the fire is extinguished completely. Recirculation cooling units in the smoke filled areas will need a maintenance check to see if the prefilters need replacing and the cooling coils need to be cleaned after the smoke purge is completed. J

A moisture eliminator is provided in each exhaust filter unit upstream of the charcoal and HEPA filters to remove entrained particulate water in the airstream. Electric heaters are provided downstream of the moisture eliminators to vaporize the water particles not removed by the moisture eliminators.

Fresh air intakes are located remote from the ventilation system exhaust to preclude the possibility of contaminating the intake air with products of combustion.

Stairwells in the Nuclear Annex are individually pressurized with roof-mounted fans to preclude smoke infiltration.

Carbon and high energy particulate air (HEPA) do not represent a potential exposure fire hazard to nearby safety related components. Carbon, used in carbon filters, has a minimum ignition temperature of 620°F. HEPA filters have a minimum ignition temperature of 600°F. Normal heating system air temperature is about 105°F. If the air temperature approaches 200°F, carbon will begin to release any adsorbed radioactive iodine. If an air temperature excursion occurs in the safety related ventilation system with carbon or HEPA filters, the heat sensor will cut off the filter train fan and the redundant fan I

serving the redundant division will begin to serve the area involved; therefore, the fire will be isolated. I

8.3

9.5.1.[redacted] Equipment Water Shields

Protection of equipment susceptible to water damage required for safe shutdown of the plant from inadvertent or advertent discharge of water from fire protection systems will be through use of water shields, conduit seals, curbs and drains, and equipment pedestals.

[redacted] It is not expected that shielding from the effects of water spray from overhead sprinkler systems will be necessary. Sprinklers in safety related areas will be of the automatic pre-action type that requires the activation of an automatic fire detector and fusing of a sprinkler head prior to releasing water. A pipe break downstream of the pre-action control valve will not release water. Shielding from spray from manual fire fighting operations will not be required outside of containment. Redundant safety related equipment is separated with 3-hour fire rated barriers which will confine the fire and fire fighting operations to a single area. From a safe shutdown standpoint it is assumed that the fire will render the equipment in the affected area inoperable, and safe shutdown will be of no consequence. All penetration seals in floors and walls up to a height of 24 inches will be waterproofed to prevent water from J the affected area from migrating to adjacent areas.

Safety related equipment in close proximity to fittings in the standpipe and interior fire hose system will be shielded as necessary to prevent damage from inadvertent discharge. Shielding location will be finalized following as-built walkdowns.

Inside containment, where redundant division equipment is located in close proximity, (i.e, within 20 feet of each other), such as the motor operated depressurization valves located at the pressurizer, shielding will be provided as deemed necessary following interaction review during detailed design and as-built walkdowns.

[redacted] The open ends of all vertical conduit, and the open ends of all horizontal conduit that terminate within 18 inches of a floor, will be sealed to prevent water infiltration.

8.4

9.5.1.[redacted] Curbs and Drains

Where fixed fire protection systems are installed, floor drains are provided, sized to collect water discharge. In areas where I drains are not installed due to pressure boundary constraints,

equipment susceptible to water damage is installed on six-inch elevated curbs.

Floor drains installed in areas where radioactive material may be entrained in water discharge are routed to the radioactive water sump so that it can be analyzed and treated if necessary before release to the environment.

In areas containing combustible liquids, floor drains are designed with water seal traps so that burning liquids cannot flow into adjacent safety related areas through the drainage system.

9.5.1.8.5 Reactor Coolant Pump Motor Oil Collection System

Each reactor coolant pump motor contains about 250 gallons of oil used as a heat exchanger medium for motor cooling and for bearing lubrication. To preclude the potential for oil escaping from the motor, an oil collection shroud is installed. When combustible oil is used, the oil collection shroud is designed to withstand the design basis earthquake. Where fire resistant oil (similar to that commonly used in turbine governor control systems) is used, the system is not seismically qualified but is seismically restrained to prevent falling on other safety related equipment. The shroud encloses the upper and lower oil reservoirs and related piping so that any potential pressurized and nonpressurized leakage points are contained. The shroud is drained through a collection pipe to the reactor coolant pump motor oil drain tank, located in the lowest level of containment elevation 91+9. Each drain tank is located within a dike, sized to contain the full inventory of the motor oil. The vent for each tank has a flame arrestor to prevent the possibility of burning oil vapor propagating into the tank. Each tank will be provided with inventory level indication which is alarmed and annunciated in the control room. I J

9.5.1.8.6 Fire Brigade Radios

The station radio system includes a dedicated frequency for fire brigade use. Dedicated radio units for fire brigade use are located in the fire brigade equipment storage room. Radios are stored in the charger base to assure they are fully charged when needed. The frequency is selected to assure that plant security communication and protective relay systems are not affected. There are an adequate number of units for at least five fire brigade members, leaders, and spare units for additional brigade members and operators. I

The fire brigade radio system has fixed repeaters located so that fire brigade members can communicate with each other and the control room from any location of the plant. Fixed repeaters are

located and wiring routed so that radio communication is available following fire in any area of the plant.

9.5.1.8.7 Fire Brigade Breathing Air System

Fire brigade personnel protective equipment includes breathing air cylinders. There is an adequate quantity of cylinders for each fire brigade member (and a quantity of spare cylinders as determined appropriate by the Fire Brigade Leader) located in the Fire Brigade Equipment Storage room. In addition, a breathing air compressor is provided in an area which would not be susceptible to fire in a safety related area, and that is free of airborne contaminants under normal conditions. The compressor will be oil free or equipped with high temperature and carbon monoxide alarms in accordance with 29 CFR 1910.134 (OSHA) Section 1910.134, Respiratory Protection. The breathing air compressor is powered from the Alternate AC Source - Combustion Turbine. Power and control cables for the breathing air compressor are routed and protected to assure that fire in a safety related portion of the station which requires the use of fire brigade Self-Contained Breathing Air (SCBA) units will not interrupt operation of the breathing air compressor.

9.5.1.9 Startup and Recurring System Tests and Inspections

9.5.1.9.1 Fire Pumps

A. Acceptance Test Criteria

1. Hydrostatic Tests

   Pump suction piping (except short lengths between suction tanks and pumps) and discharge piping (up to the pump discharge isolation valve) are pressure tested at 200 psi or at 50 psi in excess of the maximum static pressure if the maximum static pressure is in excess of 150 psi for two hours. Maximum allowable leakage is two quarts per hour per 100 gaskets or joints.

2. Performance Tests

   Fire pumps are performance tested in accordance with NFPA 20, "Standard for Centrifugal Fire Pumps."

   a. Pumps are tested at minimum flow, rated flow, and 150% of rated flow. Performance shall be within ± 5% of the manufacturer's characteristic performance curve for flow and pressure. Voltage shall be within 5% below or 10% above the rated nameplate voltage.

b. Pumps are started and brought up to speed without interruptions under rated flow conditions.

c. Fire pump controllers shall perform at least 10 automatic and 10 manual starts, with the pump driver operating for at least five minutes at full speed during each operation. Controller operation is initiated by each starting feature (i.e., pressure switch, manual start button, remote start switch). Test of controllers of diesel engine driven pumps shall be divided between redundant battery sets. Each pump operates continuously for at least one hour without overheating or excessive vibration.

d. Local and remote alarms are verified during acceptance testing.

e. The transfer switch which aligns the motor driven fire pump to the Alternate AC Source - Combustion Turbine shall be verified. At least half of the manual and automatic pump operations during the acceptance test are performed with the pump power source aligned to the Alternate AC Source - Combustion Turbine bus. I

f. Proper operation of the jockey pump, motor controller, and pressure switch is confirmed.

B. Recurring Test

1. Annual Test

Fire pumps are performance tested in accordance with NFPA 20, "Standard for Centrifugal Fire Pumps."

a. Fire pump controllers each perform at least one start by each automatic and manual starting feature. The fire pump driver operates at full speed for at least five minutes.

b. Pumps are flow tested at minimum flow, rated flow, and 150% of rated flow. Flow and pressure performance shall be within 5% of the manufacturer's characteristic pump curve.

c. Pressure settings for relief valves and pressure switches are tested to assure performance at set points.

d. The diesel engine driven fire pump controller is started with both sets of batteries. The motor driven fire pump is transferred under load from the primary to the back-up power supply.

e. Test relief valves for actuation at the proper setting.

f. Proper operation of the jockey pump unit is verified.

2. Weekly Tests

Pumps are tested weekly to assure automatic starting upon system pressure drop. The diesel engine driven pump runs for at least 15 minutes, and the motor driven pump operates at least five minutes without excessive vibration or leakage at the packing. Diesel fuel tank levels are checked to assure an adequate supply. I

9.5.1 9.2 Water Distribution System

The water distribution system is tested in accordance with NFPA 24, "Standard for Private Fire Service Underground Mains."

A. Acceptance Tests

1. Hydrostatic Tests

The water distribution system is hydrostatically tested at 200 psi or at 50 psi in excess of the maximum static pressure if the maximum static pressure is in excess of 150 psi for at least two hours. Allowable leakage is up to 2 quarts per hour per 100 pipe joints. J

2. Flow Tests

a. Flow tests are conducted to assure adequate and unobstructed flow through each flow path of the water distribution system. The minimum acceptable flow rates are as follows: I

- 12 inch pipe 3520 gpm
- 10 inch pipe 2440 gpm
- 8 inch pipe 1560 gpm
- 6 inch pipe 880 gpm

These flow rates result in flow velocity of at least 10 feet per second.

Each fire hydrant is operated to assure that distribution piping is unobstructed.

b. Water flow is conducted through each flow path of the water distribution system to assure that the minimum calculated flow and pressure is available.

c. Manual control valves are cycled to assure proper operation.

d. Hose house equipment is inspected to assure there is no visible damage to equipment and hose hydrostatic tests are current.

B. Annual Tests

Flow tests are conducted for each flow path of the water distribution system to assure the minimum flow and pressure (based on engineering calculations) is available.

9.5.1 9.3 Automatic Sprinkler Systems

Automatic sprinkler systems are tested in accordance with NFPA-13, "Standard for Installation of Sprinkler Systems." I

A. Acceptance Tests

1. Hydrostatic Tests

Automatic sprinkler systems are hydrostatically tested at 200 psi or 50 psi above the maximum operating pressure for at least two hours with no visible leakage.

2. Performance Tests

a. Preaction valves are functionally tested by a simulated signal on a detector actuation device and by remote and local manual actuation devices.

b. Air pressure maintenance devices and flow switches are tested to assure proper operation.

c. Inspector's test connections and two inch drains of each system are flow tested to assure piping is unobstructed and remote and local alarms operate properly.

d. Each branch line and cross main is flushed to assure that piping is unobstructed.

B. Annual Tests

1. Control valves are cycled to verify proper operation.

2. Inspector's test connections are flow tested to assure that piping is unobstructed and remote and local alarms operate properly.

3. Preaction valves are actuated by local manual actuation device which is part of the valve body trim.

4. Detector circuits which actuate preaction systems are functionally tested by simulated signal to assure circuit continuity.

C. Monthly Tests

Two inch drains are flow tested to assure that piping is unobstructed and remote and local alarms operate properly.

NOTE: Automatic sprinkler systems in the containment building (and other areas which are inaccessible during I
power operation) are inspected and tested during maintenance and refueling outages.

9.5.1.9.4 Hose Station and Standpipe Systems

Hose station and standpipe systems are tested in accordance with NFPA 14, "Standard for the Installation of Hose Station and Standpipe Systems."

A. Acceptance Tests

1. Hose station and standpipe system piping is hydrostatically tested at 200 psi or at least 50 psi above normal system pressure for at least two hours with no visible leakage.

2. Hose stations and standpipe systems are flushed with a sufficient volume of water so as to remove all construction debris and trash that may accumulate during installation.

3. A flow test is conducted at the hydraulically most remote outlet to assure that at least 500 gpm is available at 65 psi.

B. Triennial Tests

Interior fire hoses are hydrostatically tested in accordance

C. Annual Tests

1. Exterior fire hoses are hydrostatically tested in accordance with manufacturer's instructions.

2. Hose stations are visually inspected to assure that there is no visible degradation.

9.5.1.9.5 Fire Detection and Alarm Systems

Fire detection and alarm systems are tested in accordance with NFPA 72, "Fire Detector and Alarm Systems."

A. Acceptance Test

1. Detectors are actuated by simulated signal to assure that sensitivity is within Listed or Approved tolerances.

2. The system is tested to assure that each manual pull station and detector device and activation circuit properly actuates local alarm and remote alarm and annunciation.

3. Loss of primary system power is simulated to assure that the back-up battery power supply assumes the load. I Alarm conditions are simulated while the system is powered by the back-up battery to assure proper operation.

4. System trouble alarm is simulated by removing detectors from the circuit. The test assures that a trouble alarm does not incapacitate the unaffected portions of the system.

B. Annual Tests

Detectors are actuated by simulated signal to assure that sensitivity is within Listed or Approved tolerances.

9.5.1.9.6. Portable Fire Extinguishers

Portable fire extinguishers are tested in accordance with NFPA 10, "Portable Fire Extinguishers."

A. Acceptance Tests

Portable fire extinguishers are visually inspected to assure there is no obvious defects and that the extinguisher is properly charged.

B. Five Year Intervals

Water fire extinguishers are discharged and cylinders are hydrostatically tested.

C. Twelve Year Intervals

Carbon dioxide fire extinguishers are discharged and cylinders are hydrostatically tested.

9.5.1.9.7 Smoke Control

Smoke control features of the HVAC system are tested in accordance with NFPA 90A, "Installation of Air Conditioning and Ventilating Systems" and NFPA 204M, "Smoke and Heat Venting."

A. Acceptance Tests

1. Fire dampers are drop tested under anticipated air flow conditions to assure proper operation.
2. The ventilation system for each area is aligned for smoke ventilation (i.e., 100% fresh air intake and 100% exhaust) to assure damper controls function properly.
3. Smoke dampers in return air ducts are actuated to assure proper operation. I

9.5.1.9.8 Emergency Lighting

A. Acceptance Tests

1. Lighting units are inspected to assure that each bulb is properly directed and unobstructed.
2. Ten percent of the lighting units are tested utilizing battery power to assure operation for the designated duration.

B. Annual Tests

1. Lighting units are inspected to assure that each bulb is properly directed and unobstructed.
2. Ten percent of the lighting units are tested utilizing battery power to assure operation for the designated duration.

9.5.1.9.9 Fire Brigade Radios

A. Fire brigade radios are functionally tested, during preoperation testing, in each area of the plant to assure proper operation.

B. Fire brigade radios are tested as part of station fire drills to assure continued proper operation.

9.5.1.10 Control of Combustible Materials

A program is established control of storage, use and disposal of combustible material. Combustible materials are defined as those materials which will ignite, burn, support combustion, or release combustible vapors when exposed to fire or heat in the installed configuration. I

9.5.1.10.1 Structures, Equipment, and Components

A. Structures

Structures are comprised of noncombustible material. Some interior finish materials are of limited combustible construction with the following fire resistive characteristics:

1. Maximum flame spread of 25
2. Maximum smoke development of 450
3. Minimum critical radiant flux of .45W/cm^2

Notes 1 and 2 are acceptance criteria of ASTM E-84, "Test for Surface Burning Characteristics of Building Materials" Note 3 is obtained from ~~NFPA 253~~ ASTM E-648, "Test for Critical Radiant Flux of Floor Covering Systems Using a Radiant Heat Energy Source." K

B. Equipment

Some plant equipment contains synthetic materials such as neoprene plastic and nylon parts. These quantities are not present in concentrations which would create a significant fire hazard. Locations containing significant quantities of plastic material such as cable insulation are evaluated in the Fire Hazards Analysis to consider the potential affects of combustion such as heavy smoke production and generation of corrosive and toxic gases. I

Bulk hydrogen storage cylinders are located outside of the Nuclear Annex within the protected area. In safety related areas of the plant, hydrogen piping is designed to Seismic Category 1 requirements.

Reactor coolant pump motors each contain about 250 gallons of oil for cooling and lubrication. Potential leak points are enclosed in a seismically designed oil collection shroud which drains to a full capacity, seismically designed tank in the basement of the Reactor Building. Thus, oil escaping from the reactor coolant pump motor would not create a potential fire hazard. (An option under consideration is use of fire retardant oil similar to that commonly used in turbine governor systems, which would reduce the potential for ignition and severity of a fire. An oil collection and drain system would be provided but would not be seismically qualified).

Some safety related pumps contain small quantities of lubricating oil or grease. These pumps are reviewed on an individual basis in the Fire Hazards Analysis. Fire protection features are provided as determined appropriate. I

C. Components

The majority of in situ combustible materials in safety related areas of the plant consists of plastic insulation of power, control, and instrumentation cables. Use of fiber optic cables from the control room and individual multiplexer panels in designated train-specific areas, reduces the quantity of combustible cable insulation by an estimated order of magnitude. Further, locations containing significant quantities of combustible materials are investigated in the Fire Hazards Analysis to consider the potential affects of burning, such as heavy smoke production and generation of corrosive and toxic gases.

Some piping and HVAC insulation consists of synthetic rubber type products, where moisture control is a significant concern.

9.5.1.[redacted] 10.2 Flammable and Combustible Liquids

Two above ground diesel fuel oil storage tanks (typically 67,500 gallons each) are located on either side of the Nuclear Annex, Diesel Generator Rooms. Storage complies with NFPA 30, "Flammable and Combustible Liquids Code."

There is a diesel fuel oil day tank (typically 900 gallons) in each diesel generator room. Each tank is surrounded by a full I height (of the tank) concrete dike sized to contain 110% of the tank capacity. Penetrations in the dike are sealed. Drains are K provided within the dike to remove spillage to a safe location. Tank vents are routed outside of the room.

The Alternate AC Source - Combustion Turbine (CT) is located remote from the Nuclear Annex such that fire involving the CT will not affect nuclear safety related equipment. Fire protection features are provided for the Combustion Turbine, consistent with the Fire Protection Design objectives as determined appropriate by the Fire Hazards Analysis.

Storage of flammable and combustible liquids complies with NFPA 30, "Flammable and Combustible Liquids." Cleaning fluids and solvents are normally used in quantities of one gallon or less.

9.5.1.10.3 Combustible Contents

9.5.1.10.3.1 Combustible Furnishings

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In areas designated as personnel work stations, change rooms, break rooms and combustible material storage areas, combustible furnishings, and work related material are present. In these areas, the Fire Hazards Analysis assesses the potential for fire ignition, growth, and consequences.

Based on this assessment, fire protection features are provided to assure that the Fire Protection Design Basis Goals and Objectives are met.

9.5.1.10.3.2 Transient Combustible Material

An administrative control program assures the amount of transient combustible material in safety related areas are properly managed and that additional fire protection features provided as appropriate. When specific tasks are completed or at the end of each shift, combustible waste material is collected and moved to the designated waste collection area.

Portable cylinders of flammable and combustible gases are used in the Nuclear Annex and Reactor Building. An administrative control program implements a permit system to assure control of use and storage of these cylinders.

Storage and disposal of anticontamination clothing at radiation control zone (RCZ) step-off pads is recognized as a potentially significant transient combustible fire hazard. Therefore, anticontamination clothing is stored in enclosed storage cabinets. Cabinet doors are normally closed as required by station directives. Used anticontaminated clothing is placed in metal drums which have fusible link actuated or otherwise Listed or Approved covers. Fire protection and detection features are provided for step-off pad areas based on conclusions of the Fire Hazards Analysis.

9.5.1.11 Fire Protection Program

9.5.1.11.1 Fire Prevention

A. Control of Hot Work

Cutting, welding, and grinding operations are governed by a permit system as required by station administrative controls. Each task is reviewed and an adequate number of trained and qualified fire watch patrols established to assure that hot slag or sparks do not ignite nearby in situ combustible material and that transient combustible materials are relocated outside the vicinity. Fire watch is maintained for at least 30 minutes after completion of hot work to assure that residual hot material does not ignite nearby combustible material.

B. Housekeeping

Station directives, developed based on the Fire Hazards Analysis, determine an appropriate quantity of combustible material that can be located in any area of the plant. Where it is necessary to exceed the allowable quantity of combustible material in an area, a permit system is established to determine appropriate additional fire protection features and the allowable duration of the variation.

Designate plant functional groups have material responsibility for specific plant areas and are responsible for housekeeping in these designated plant areas.

Plant management conducts regular housekeeping inspections to assure that the housekeeping program is being properly implemented and that violations are promptly corrected.

9.5.1.[redacted] 11.2 Personnel Qualifications

A. Fire Protection Engineer

The individual responsible for developing and implementing the overall fire protection program is designated as the Fire Protection Engineer. The Fire Protection Engineer is a [redacted] graduate of an [redacted] engineering curriculum of acceptable standing with at least six years of engineering experience, three of which have been in responsible charge of fire protection engineering activities.

B. Fire Chief

The individual designated as Fire Chief has certification as a firefighter training instructor. In addition, the Fire Chief has experience in organizing, instructing, training, drilling, and critiquing an industrial fire brigade.

C. Fire Brigade Members I

Fire brigade members have completed an initial 40 hours training consisting of a 40-hour course which includes classroom instruction and practical fire fighting training. Each member has passed a physical examination to assure ability to participate in fire brigade activities.

Fire brigade members receive annual requalification training and physical examination.

D. Fire Protection System Operation, Testing, and Maintenance

Functional groups responsible for fire protection system operation, maintenance, and testing are qualified by training and experience and understand functions of the system.

9.5.1.[redacted] 11.3 Fire Brigade Organization, Training, and Records

The plant fire brigade is fully qualified for structural fire fighting. There are at least five fire brigade members on duty at all times. J

Fire brigade members receive annual physical examinations to assure ability to perform fire fighting activities.

Fire brigade members are provided with the following personnel protective equipment:

A. Turnout coats

B. Boots

C. Gloves

D. Helmets

E. Self-contained breathing apparatus (SCBA) with full-face, positive pressure mask rated for 60 minute duration.

In addition, the fire brigade organization is provided with the following equipment:

A. Breathing air compressor

B. Radio communication system

C. Portable battery powered lights

D. Portable smoke ejectors and/or positive pressure ventilation fans

E. Portable fire extinguisher J

F. Additional lengths of 2-1/2 inch and 1-1/2 inch fire hose with nozzles, couplings, fitting, gaskets, spanner wrenches, etc.

G. Spare breathing air cylinders

H. First aid kit

There are at least 10 SCBAs reserved for fire brigade use. Each has two 60-minute reserve cylinders. The breathing air compressor is powered from the station emergency power combustion turbine.

Fire brigade training consists of initial classroom and practical training. Initial classroom training consists of:

A. Instruction concerning the fire fighting plan and member's responsibility.

B. Review of the prefire plan which includes type and location of fire hazards.

C. Instruction of potential effects of fire, flame, hot gases, and products of combustion.

D. Familiarization with plant layout, equipment functions and potential hazards, location of fire protection equipment, location of power supply controls, operation of ventilation and smoke ~~control~~ removal systems, and access/egress routes for each area.

E. Use of available fire fighting equipment and correct method of fighting fires in energized electrical equipment, fires in cables, and cable trays, hydrogen fires and other types of flammable and combustible liquids, and fires involving ordinary combustible materials.

F. Use of fire brigade radios, portable emergency lighting, smoke control equipment including portable smoke ejectors, and other manual fire fighting equipment.

G. Procedures for fire attack in buildings and confined spaces.

H. Instruction regarding fire fighting strategy for each fire area, room, or zone.

I. Fire fighting activities are coordinated with the local ~~volunteer~~ fire department to assure adequate back-up fire fighting capability can be provided if necessary.

J. Operational precautions for fighting fire on nuclear power sites including radiological protection and special hazards associated with a nuclear power plant. J

Refresher and requalification training consists of the following activities:

A. Meetings with the local fire department are held annually to review significant plant modifications and changes to fire fighting strategies.

B. Periodic refresher training sessions are held so that each brigade member participates in training at least every two years.

C. Practice sessions are held for each brigade member in proper fire fighting techniques and use of fire brigade equipment. Each fire brigade member participates in at least one drill per year.

D. Drills are performed in the plant at least once per quarter for each shift. Each fire brigade member participates in at least two drills per year. At least one drill per year, per

shift, is unannounced, and at least one drill per year for each shift occurs on the back shift. At least once per year, the local fire department participates in station drills.

E. At least every three years, drills are critiqued by qualified individuals independent of the corporate staff. J

F. Drill critiques include: fire alarm effectiveness, time required for notification, fire brigade response, fire fighting strategies, use of fire fighting equipment and suppression techniques, assessment of members' knowledge of roles and equipment use, and strategies and equipment use.

The station prefire plan details fire fighting strategies for each area of the station including known hazards, location of fire fighting equipment, location of controls for power supply and ventilation systems, and other pertinent information. I

Drill scenarios are based on realistic potential fire events in various areas of the plant. Scenarios include fire growth, effect on safety-related and safe shutdown functions, and availability of ventilation.

Records of fire brigade member physical examination, training drills, and critiques are maintained on file. J

NFPA 600, "Standard for Private Fire Brigades" is used as guidance in organization and training of the fire brigade.

9.5.1.12 Fire Hazards Analysis

A Fire Hazards Analysis is conducted for each room area or zone of the plant. Containing safety related equipment or equipment important to safety. It considers the function of major equipment in the area, location and number of redundant equipment or functions, known and anticipated quantity and configuration of combustible material, ventilation and smoke control, presences of predetermined fire protection features, and consequences of fire with and without fire protection features functioning properly. Where the Fire Hazards Analysis determines that Design Objectives are met in accordance with Section 9.5.1.1, fire protection is considered adequate. The Fire Hazards Analysis is maintained as part of the Plant Design Basis. I

9.5.1.13 Fire Protection Quality Assurance Program

The Fire Protection Quality Assurance Program implements a "graded" approach focusing attention to features that assure that design, procurement, installation, testing, operation,

maintenance, and repair are conducted as appropriate. The program assures that systems, equipment, components, and procedures produce the fire protection function as intended. The program complies with the intent of NUREG 0800, Section 9.5.1, "Standard Review Plan."

The program applies to features addressed in the Fire Hazards Analysis as follows:

A. Features provided to separate or protect redundant systems and equipment required to achieve cold shutdown.

B. Features which provide defense-in-depth for protection of safety related systems, equipment, and components.

Program objectives are to assure that fire protection features, including mechanical and electrical systems, fire barrier components and fire insulating material, are properly designed, installed, operated, and maintained in accordance with regulatory requirements, industry standards, and National Fire Protection Association codes and standards. Objectives are achieved as follows: I

A. Fire protection specialty items are tested and approved by a nationally recognized testing laboratory.

B. Control design documents, procurement, and installation of fire protection features.

C. Receipt inspection of specialty items to assure receipt of proper materials.

D. As-built inspections to assure proper material installation and layout.

E. Operational tests of completed installations to assure that systems function as intended.

Records of Quality Assurance activities are maintained on file for future review.

Annual fire protection audits are conducted by a team of off-site personnel including a QA Qualified Lead Auditor, fire protection engineer, and an individual knowledgeable of plant systems. Every third year, the Quality Assurance audit includes an independent fire protection engineer who is not a direct employee of the licensee.

REFERENCE FOR SECTION 9.5.1

1. NFPA 10, "Portable Fire Extinguishers, Installation, Maintenance, and Use"

2. NFPA 13, "Sprinkler Systems"

3. NFPA 14, "Standpipe and Hose Systems"

4. NFPA 15, "Water Spray Fixed Systems"

5. NFPA 20, "Centrifugal Fire Pumps"

6. NFPA 22, "Water Tanks for Private Fire Protection"

7. NFPA 24, "Private Fire Service Mains"

8. NFPA 26, "Supervision of Valves"

9. NFPA 30, "Flammable Combustible Liquids Code"

10. NFPA 51B, "Cutting and Welding Processes"

11. NFPA 70, "National Electric Code"

12. NFPA 72, "Fire Detection and Alarm Systems" I

13. NFPA 80, "Fire Doors and Windows"

14. NFPA 92M, "Waterproofing and Draining of Floors"

15. NFPA 101, "Life Safety Code"

16. NFPA 204M, "Smoke and Heat Venting Guide"

17. NFPA 220, "Types of Building Construction"

18. NFPA 232, "Protection of Records"

19. NFPA 251, "Fire Tests, Building Construction and Materials"

20. NFPA259, "Test Method for Potential Heat of Building Materials"

21. NFPA 600, "Private Fire Brigades"

22. NFPA 802, "Recommended Fire Protection Practice for Nuclear Reactors"

23. NFPA 803, "Fire Protection of Nuclear Power Plants"

24. NUREG-0050, "Recommendations Related to Browns Ferry Fire," Report by Special Review Group, February 1976.

25. WASH-1400 (NUREG-75/014), "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," October 1985.

26. NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

    Section 9.5.1, "Fire Protection Program."

    Section 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

    Section 6.4, "Habitability Systems."

27. Appendix A "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities" General Design Criterion 3, "Fire Protection."

28. Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems."

29. Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants."

30. Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

31. Regulatory Guide 1.75, "Physical Independence of Electrical Systems."

32. Regulatory Guide 1.88, "Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records."

33. Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants."

34. ANSI Standard B31.1-1973, Power Piping."

I

35. ASTM D-3286, "Test for Gross Calorific Value of Solid Fuel by the Isothermal-Jacket Bomb Calorimeter (1973)

36. ASTM E-84, "Surface Burning Characteristics of Building Materials (1976)."

37. ASTM E-119, "Fire Test of Building Construction and Materials (1976)."

39. IEEE Std. 383-1974, "IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations," April 15, 1974

40. IEEE Std. 384-1981, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits."

41. MAERP-NELPIA, "Specifications for Fire Protection of New Plants."

42. Factory Mutual System Approval Guide - Equipment, Materials, Services for Conservation of Property. I

43. "International Guidelines for the Fire Protection of Nuclear Power Plants," National Nuclear Risks Insurance Pools, 2nd Report (IGL).

44. NFPA Fire Protection Handbook.

45. SFPE Handbook of Fire Protection Engineering.

46. Underwriters Laboratories Rating List.

47. Underwriters Laboratories, "Building Materials Directory."

38. ASTM E-648, "Test for Critical Radiant Flux of Floor Covering Systems Using a Radiant Heat Energy Source"

ALWR CESSAR INSERTS

INSERT #1: SECTION 9.5.1.1.2.C (Third paragraph) Delete existing, Insert: Inside Containment and the Annulus: Cables used for safe shutdown functions will be three hour fire rated cable protective systems (i.e., mineral insulated cable or equivalent). A potential exception is containment penetrations installed in the containment vessel to transition between inner and outer containment. These penetrations are currently available as one hour fire rated. Three hour fire rated containment penetrations will be purchased if available.

INSERT #2: SECTION 9.5.1.2.B Inside containment: Column line 17 bisects containment. Division 1 components are located (Plan) north of column line 17. Division 2 components are located (Plan) south of column line 17. Thus, redundant divisions are generally located in separate hemispheres of containment. The Fire Protection Safe Shutdown Analysis (which will be maintained as part of the System 80+ design basis) will assure that fire at any specific location inside containment will not affect redundant safe shutdown components. It will also assure that redundant safe shutdown components such as instruments and valves will be separated to the extent practicable as stipulated in SECY-90-016," Evolutionary Light Water Reactor (LWR)Certification Issues and Their Relationship to Current Regulatory Requirements".

As stated in section 9.5.1.1.2.C, cables used for safe shutdown functions inside containment will be three hour fire rated cable protective systems (i.e., mineral insulated cables or equivalent). An exception to the three hour fire resistance rating may be containment penetrations which are currently commercially available with one hour fire resistance rating. Three hour fire rated containment penetrations will be purchased if available.

INSERT #3: SECTION 9.5.1.3.2.1: Where it is not practical to provide laboratory tested components, engineering analysis will assure that an acceptable quality of fire safety is provided. The engineering analysis will be submitted to NRC for review.

INSERT #4: SECTION 9.5.1.3.2.6, Third sentence: Components that may be protected by fire rated insulating material include structural steel, cables and safe shutdown components. If any of these features must be protected with fire rated insulating material, the material will be qualified by test to acceptance criteria that has been adopted by the Nuclear Regulatory Commission.

INSERT #5: SECTION 9.5.1.3.3 Fourth paragraph: The control complex has a dedicated smoke removal system. The smoke removal system serves the following areas:

Channels A, B, C, D Vital Instrument and Equipment Rooms

Divisions 1 & 2 Essential Electrical Equipment Rooms
Remote Shutdown Panel Room
Divisions 1 & 2 Non-Essential Equipment Rooms
CAS and Security Equipment Rooms
Computer Room and its Mechanical Equipment Room
Control Room
OSC Equipment Room
OSC
TSC
TSC Mechanical Equipment Room
Control Room Mechanical Equipment Room

The Control Complex Smoke Removal System utilizes dedicated smoke exhaust fans for each safety related division. Dampers are three hour fire rated, normally closed, motor operated smoke dampers. The dampers are remotely actuated from the control room. Motor operators and power and control cables are located on the opposite side of the fire barrier from which smoke is to be exhausted. The system uses 100% outside air supplied by the Control complex air-handling units. The smoke purge fans are sized to exhaust three cfm per square foot.

Insert #6: SECTION 9.5.1.4 Fourth paragraph: Delete existing: Inside Containment and the Annulus: As stated in Section 9.5.1.1.2.C, three hour fire rated cable protective systems (i.e., mineral insulated cable or the equivalent) are used for cables associated with safe shutdown functions. An exception to the three hour fire resistance rating may be containment penetrations that are currently commercially available with one hour fire resistance rating. Three hour fire rated containment penetrations will be purchased if available.

The only in situ combustible material inside containment that may be exposed to a fire is insulation of cables that are not associated with safe shutdown functions. Redundant trains of valves and instruments analyzed as an assured method of achieving safe shutdown are physically separated such that a potential fire will not affect redundant equipment.

The "Fire Protection Safe Shutdown Analysis" which will be maintained as part of the plant Design Basis provides a detailed description of separation of redundant safe shutdown functions.

Insert #7 SECTION 9.5.1.5.2 Fire protection control valves will be either locked or electrically supervised to assure they remain in the open position. Administrative controls will assure that the fire protection water supply system is not used for non fire protection purposes.

INSERT #8: SECTION 9.5.1.5.2.B: Delete existing: Automatic sprinkler systems will be installed in areas as stipulated in BTP CMEB 9.5-1. Sprinklers will be installed in other areas, as determined necessary to meet Fire Protection Program Design Basis Goals and Objectives as stated in Sections 9.5.1.1.1 & 9.5.1.1.2;

INSERT #9: SECTION 9.5.1.5.3.C: CESSAR-DC Section 3.4.4.1 contains a discussion of internal flood protection methods. These flood protection methods will protect safe shutdown equipment from internal flooding including flooding due to water released during fire suppression activities.

INSERT #10: SECTION 9.5.1.5.6: A fixed automatic fire detection system is installed in the Nuclear Annex and portions of the Reactor building. Fire detection will be installed in the following areas:

- Areas containing major cable concentrations
- Safe shutdown major pumps
- Switchgear
- Motor control centers
- Battery and Inverter areas
- Relay rooms
- Fuel areas

Fire detectors will also be installed in other areas containing appreciable in situ or potentially transient combustible materials such as change room storage contaminated area step off pads and laundry rooms.

INSERT #11: SECTION 9.5.1.2.G: The fans dedicated for smoke purge are sized to provide a minimum of 3 CFM/square foot of floor area. The ventilation systems are sized to provide an air flow of 1 CFM/square foot of floor area or more depending upon the area served. The layout of the ductwork is such that it ensures ventilation of all corners of the area as much as practical.

ATTACHMENT 3

Heat treat - Heat to 1800 °F +/- 25 °F, air cool and temper at 1125 °F minimum for 4 hours per Code Case N-4-11.

4.5 REACTOR MATERIALS

4.5.1 CONTROL ELEMENT DRIVE STRUCTURAL MATERIALS

4.5.1.1 Material Specifications

A. The materials used in the control element drive mechanism (CEDM) reactor coolant pressure boundary components are as follows:

1. Motor housing assembly

SA 182, Type 347 (austenitic stainless steel)

ASME Code Case N-4-11 (modified Type 403 martensitic stainless steel), and additional requirements of ASME SA-182

SB 166 (nickel-chromium alloy)

2. Upper pressure housing

SA 213, Type 316 (austenitic stainless steel)

SA 479, Type 316 (austenitic stainless steel)

The above listed materials are also listed in Section III of the ASME Boiler and Pressure Vessel Code. In addition, the materials comply with Sections II and IX of the ASME Boiler and Pressure Vessel Code. Code Case N-4-11 is acceptable per Regulatory Guide 1.85.

The functions of the above listed components are described in Section 3.9.4.1.

B. The materials in contact with the reactor coolant used in the CEDM motor assembly components are as follows:

1. Latch guide tubes

ASTM A269, Type 316 (austenitic stainless steel)

Chrome Oxide (plasma spray treatment)

2. Magnet and spacer

ASTM A276, Type 410 (martensitic stainless steel)

3. Latch and magnet housing

   ASTM A276, Type 316 (austenitic stainless steel)

   QQ-C-320, Class 2B (chrome plating)

   ASTM A276, Type 440C (martensitic stainless steel)

4. Spacer

   ASTM A240, Type 304 (austenitic stainless steel)

5. Alignment Tab

   ASTM A276, Type 410 (martensitic stainless steel)

6. Spring

   AMS 5698B, Inconel X-750 (nickel base alloy)

7. Pin

   Haynes Stellite No. 6B (cobalt base alloy) or an alternate material demonstrated to be functionally equivalent

8. Dowel pin

   300 Series ~~SST~~ (austenitic stainless steel)

9. Spacer and screw

   ASTM A276, Type 321 (austenitic stainless steel)

10. Stop

    ASTM A276, Type 304 (austenitic stainless steel)

11. Latch and pin

    Haynes Stellite No. 36 (cobalt base alloy) or an alternate material demonstrated to be functionally equivalent

12. Locking cup and screws

    Type 300 Series austenitic stainless steel

E. Bolt and pin material

ASTM-A-453 and ASTM-A-638, Grade 660 material (trade name A-286) is used for bolting and pin applications. This alloy is heat treated in accordance with the ASTM specifications by precipitation hardening at 1300-1400°F for 16 hours to a minimum yield strength of 85,000 psi. Its corrosion properties are similar to those of the Type 300 series austenitic stainless steels. It is austenitic in all conditions of fabrication and heat treatment. This alloy was used for bolting in previous reactor systems and test facilities in contact with primary coolant and has proven completely satisfactory.

F. Chrome plating and hardfacing

G. Special Purpose Material

Haynes Alloy

SA 479 S 21800 (Trade Name Nitronic 60) is used for special applications where anti-galling properties are desired.

Chrome plating or hardfacing are employed on reactor core support and internals structures, components or portions thereof where required by function. Chrome plating complies with Federal Specification No. QQ-C-320. The hardfacing material employed is ~~Stellite~~ 25 or an alternate material demonstrated to be functionally equivalent.

All of the materials employed in the reactor internals and in-core instrument support system have performed satisfactorily in operating reactors such as Palisades (Docket-50-255), Fort Calhoun (Docket-50-285) and Maine Yankee (Docket-50-309).

4.5.2.2 Welding Acceptance Standards

Welds employed on reactor internals and core support structures are fabricated in accordance with Article NG-4000 in Section III, and meet the acceptance standards delineated in article NG-5000, Section III, Division I, and control of welding is performed in accordance with Section III, Division I, and Section IX of the ASME Code. In addition, consistency with the recommendations of Regulatory Guides 1.31 and 1.44 is described in Section 4.5.2.3.

4.5.2.3 Fabrication and Processing of Austenitic Stainless Steel

The following information applies to unstabilized austenitic stainless steel as used in the reactor internals.

Exposure of completed or partially fabricated components to temperatures ranging from 800 to 1500°F is prohibited except as described in Section 4.5.2.3.1.5.

Duplex, austenitic stainless steel containing more than 5FN delta ferrite (weld metal, cast metal, weld deposit overlay) are not considered unstabilized since these alloys do not sensitize, i.e., form a continuous network of chromium-iron carbides. Specifically, alloys in this category are:

CF8M CF8	Cast stainless steel (delta ferrite controlled to 5FN-30FN)
308, 309 312, 316	Singly and combined stainless steel weld filler metals (delta ferrite controlled to 5FN-15FN as deposited)

In duplex austenitic/ferritic alloys, chromium-iron carbides are precipitated preferentially at the ferrite/austenite interfaces during exposure to temperatures ranging from 800-1500°F. This precipitate morphology precludes intergranular penetrations associated with sensitized Type 300 series stainless steels exposed to oxygenated or otherwise faulted environments.

4.5.2.3.1.4 Avoidance of Sensitization

Exposure of unstabilized austenitic Type 300 series stainless steels to temperatures ranging from 800 to 1500°F will result in carbide precipitation. The degree of carbide precipitation or sensitization depends on the temperature, the time at that temperature, and the carbon content. Severe sensitization is defined as a continuous grain boundary chromium-iron carbide network. This condition induces susceptibility to intergranular corrosion in oxygenated aqueous environments, as well as those containing halides. Such a metallurgical structure will readily fail the Strauss Test, ASTM A708. Discontinuous precipitates (i.e., an intermittent grain boundary carbide network) are not susceptible to intergranular corrosion in a PWR environment.

Weld heat affected zone sensitized austenitic stainless steels (which will fail the Strauss Test, ASTM A708) are avoided by careful control of:

A. Weld heat input to less than 60 kJ/in

B. Interpass temperature to 350°F maximum

C. Carbon content to ≤ 0.065%

Q/R 281.47

TABLE 9.2.3-1

PRIMARY AND SECONDARY MAKEUP WATER LIMITS

pH	6.0 to 8.0
Conductivity	Less than 0.2 μmhos/cm
Chloride	Less than ~~0.15~~ 0.005 ppm ~~Cl~~
Fluoride	Less than ~~0.10~~ 0.005 ppm ~~F~~
Suspended Solids	Less than ~~0.35~~ 0.05 ppm
Silica reactive ~~(SiO_2)~~	Less than 0.01 ppm
Sodium	Less than 0.003 ppm
Sulfate	Less than 0.005 ppm
Magnesium	Less than 0.04 ppm
Calcium plus Magnesium	Less than 0.08 ppm
Aluminum	Less than 0.08 ppm
Iron	Less than 0.02 ppm
Copper	Less than 0.002 ppm
Oxygen	Less than 0.1 ppm

B

Q/R 281.46

B. The CVCS is designed to supply makeup water or accept letdown due to power decreases or increases:

1. The system is designed for 10% step power increases between 15% and 90% of full power and 10% step power decreases between 100% and 25% full power, as well as for ramp changes of $\pm 5\%$ of full power per minute between 15 and 100% power.

2. The Volume Control Tank (VCT) is sized with sufficient capacity to accommodate the inventory change resulting from a full to zero power decrease with no makeup system operation, assuming that the VCT level is initially in the normal operating level band.

C. The CVCS provides a means for maintaining activity in the RCS within the appropriate technical specification limit, assuming a one percent failed fuel condition and continuous full power operation.

EPRI report NP-7077, "PWR Primary Water Chemistry Guidelines", Revision 2 (dated November 1990), and repeated in

D. The CVCS is designed to maintain the reactor coolant chemistry within the limits specified in Table 9.3.4-1. B

E. Letdown and charging portions of the CVCS are designed to withstand the design transients defined in Table 9.3.4-2 without any adverse effects, as applicable. B

F. The CVCS has the capacity to receive and process all excess reactor coolant generated during all normal and anticipated modes of operation. Excess coolant generated during typical plant operations is shown in Table 9.3.4-3. B

G. The CVCS is designed to provide 30 gpm of filtered flow to the reactor coolant pump seals and to accept a 20 gpm controlled bleedoff flow. I

H. Components of the CVCS are designed in accordance with applicable standards or codes as shown in Table 9.3.4-4. B Safety classes and seismic classes are shown on Figure 9.3.4-1 Sheets 1 through 4, and in Section 3.2. I

I. The CVCS active valves are given in Table 9.3.4-7. Refer also to Section 3.11 for environmental design criteria applicable to CVCS valves.

J. The CVCS is designed to operate with no boric acid concentration above the point where precipitation could occur. The boric acid batching tank and discharge lines, and the boric acid concentrator discharge line to the SWMS I

9.3.4 CHEMICAL AND VOLUME CONTROL SYSTEM

9.3.4.1 Design Bases

9.3.4.1.1 Functional Requirements

The Chemical and Volume Control System (CVCS) is designed as a non-safety-related system. As such, the CVCS is not required to perform any accident mitigation or safe shutdown function. In particular, the CVCS is not required to function in order to ensure the capability to shut down the reactor and maintain it in a safe shutdown condition, nor ensure the capability to prevent or mitigate the consequences of plant accidents. It is not required to show acceptable results for safety analysis. For the

System 80+ Standard Design, safety functions are performed by dedicated safety systems. Specifically, the safety injection system is credited for RCS inventory control and boration in Chapter 15 accident analyses, Chapter 6 Loss of Coolant Accidents (LOCA) events, and safe shutdowns. Pressure control during these events is accomplished via the safety depressurization and vent system. I

Although not required to perform any accident mitigation or safe shutdown functions, the chemical and volume control system is essential for the normal day-to-day operation of the plant. The CVCS has therefore been provided with a high degree of reliability and redundancy and has been designed in accordance with accepted industry standards and quality assurance commensurate with its importance to plant operations. B

The Chemical and Volume Control System is designed to perform the following functions:

A. Maintain the chemistry and purity of the reactor coolant during normal operation and during shutdowns.

B. Maintain the required volume of water in the RCS, compensating for reactor coolant contraction or expansion resulting from changes in reactor coolant temperature and for other coolant losses or additions.

C. Receive, store and separate borated waste for recycle, or discharge to the Liquid Waste Management System (LWMS).

D. Control the boron concentration in the RCS to obtain optimum Control Element Assembly (CEA) positioning, to compensate for reactivity changes associated with major changes in reactor coolant temperature, core burnup, and xenon variations, and to provide shutdown margin for maintenance and refueling operations.

E. Provide auxiliary pressurizer spray for (1) control of pressurizer pressure during the final stages of shutdown and (2) to allow for pressurizer cooling. I

F. Provide injection water at the proper temperature, pressure, and purity for the reactor coolant pump seals, and collect the controlled bleedoff from the reactor coolant pump seals.

G. ~~Leak test the RCS.~~ Provide RCS hydrotesting capability.

H. Provide a reactor makeup water supply to various auxiliary equipment.

I. Provide a means for sluicing ion exchanger resin to the Solid Waste Management System (SWMS).

J. Provide a means for continuous removal of noble gases from the RCS.

K. Provide borated makeup to the spent fuel pool.

L. Provide purification of shutdown cooling flow.

M. Provide makeup for losses from small leaks in RCS piping.

N. Provide a means to purify the contents of the In-containment Refueling Water Storage Tank (IRWST). B

O. Provide a means to add makeup and adjust the chemistry of the IRWST.

9.3.4.1.2 Design Criteria

The CVCS is designed in accordance with the following criteria:

A. The CVCS is designed to accept RCS letdown flow when the reactor coolant is heated at the maximum administrative rate of 75°F/hr and to provide the required makeup using one of the two charging pumps when the reactor coolant is cooled at the maximum administrative rate of 75°F/hr. B

B. The CVCS is designed to supply makeup water or accept letdown due to power decreases or increases:

1. The system is designed for 10% step power increases between 15% and 90% of full power and 10% step power decreases between 100% and 25% full power, as well as for ramp changes of $\pm 5\%$ of full power per minute between 15 and 100% power.

2. The Volume Control Tank (VCT) is sized with sufficient capacity to accommodate the inventory change resulting from a full to zero power decrease with no makeup system operation, assuming that the VCT level is initially in the normal operating level band.

C. The CVCS provides a means for maintaining activity in the RCS within the appropriate technical specification limit, assuming a one percent failed fuel condition and continuous full power operation.

D. The CVCS is designed to maintain the reactor coolant chemistry within the limits specified in Table 9.3.4-1. B

E. Letdown and charging portions of the ~~CVCS~~ RCPB are designed to withstand the design transients defined in Table 9.3.4-2 without any adverse effects, as applicable. B

F. The CVCS has the capacity to receive and process all excess reactor coolant generated during all normal and anticipated modes of operation. Excess coolant generated during typical plant operations is shown in Table 9.3.4-3. B

G. The CVCS is designed to provide 30 gpm of filtered flow to the ~~r~~eactor ~~c~~oolant ~~p~~ump seals and to accept a ~~20~~ 22 gpm controlled bleedoff flow. I

H. Components of the CVCS are designed in accordance with applicable standards or codes as shown in Table 9.3.4-4. Safety class~~es~~ and seismic ~~classes are shown on~~ category information is provided by Figure 9.3.4-1 Sheets 1 through 4, and ~~in Section 3.2~~ Tables 3.2-1 and 3.2-2. B I

I. The CVCS active valves are ~~given~~ identified in Table 9.3.4-7. Refer ~~also~~ to Section 3.11 for environmental design criteria applicable to CVCS valves.

J. The CVCS is designed to operate with no boric acid concentration above the point where precipitation could occur. The boric acid batching tank and discharge lines. ~~and the boric acid concentrator discharge line to the GWMS~~ I

are the only portions of the system requiring heat tracing to preclude boric acid precipitation. These portions of the system can contain fluid concentrated to 12 weight percent boric acid. The remaining portions of the system contain a lower boric acid concentration (less than 2.5 wt%), and heat tracing to prevent precipitation is not required. B

I

K. One charging pump has the capacity to replace the flow lost to the containment due to a break ~~leaks~~ in a small RCS line~~s~~, such as instrument and sample lines. These lines ~~typically~~ have 7/32-inch I.D. by 1-inch long flow restricting ~~devices~~ orifices installed in their RCS nozzles to limit leakage in the event of a line break.

L. The CVCS is designed to receive discharges from drains and relief valves ~~from~~ in the RCS, SIS and ~~Shutdown Cooling System (~~SCS~~)~~.

M. The CVCS provides for boron concentration adjustment in the RCS by feed and bleed. The maximum possible rate of boron dilution is limited, such that the operator has sufficient time to identify and terminate a boron dilution incident prior to reaching criticality during any refueling operations (see Section 7.7.1.1.10 for a description of the boron dilution alarm).

N. The CVCS concentrated boric acid reserve is sufficient to make the reactor subcritical in the cold shutdown condition with the most reactive CEA withdrawn.

9.3.4.1.3 System Functions

9.3.4.1.3.1 Reactor Coolant Inventory

The volume of water in the RCS is automatically controlled using level instrumentation located on the pressurizer. The pressurizer level setpoint is programmed to vary as a function of average RCS temperature in order to minimize the transfer of fluid between the RCS and CVCS during power changes. This linear relationship is shown in Figure 5.4.10-2. Reactor power is ~~determined for this situation using~~ directly proportional to the average reactor coolant temperature derived from hot and cold leg temperature measurements. A level error signal is obtained by comparing the programmed setpoint with the measured pressurizer water level. Volume control is achieved by automatic control of the charging and letdown flow control valves in accordance with the pressurizer level control program shown in Figure 5.4.10-4. B

B

Two parallel charging pump flow control valves, two parallel letdown flow control valves, and two parallel charging pumps are provided. During ~~all~~ reactor power operations, one charging pump is running

with one in standby. In addition, one of the letdown and one of the charging pump flow control valves are selected for use. The selected charging pump flow control valve is normally maintained by the pressurizer level control program at a preset position to ~~[illegible]~~ give~~s~~ a constant ~~[illegible]~~ flow rate at normal operating pressures. The position of the selected charging pump flow control valve is maintained constant by the pressurizer level control program, except in response to a high or low pressurizer level condition as shown in Figure 5.4.10-4. Fine control of pressurizer level is accomplished via letdown control. The position of the selected letdown flow control valve is varied by the pressurizer level control program in response to the level error in order to compensate for small changes in pressurizer level. ~~and to keep it within the programmed level band~~.

The level in the VCT is controlled automatically. VCT Letdown flow is diverted to the holdup tank via the pre-holdup ion exchanger and gas stripper when the ~~control band~~ setpoint high level is reached. The makeup system is normally set ~~up~~ for the automatic mode of operation, in which flow at a preset blend of boric acid from the Boric Acid Storage Tank (BAST) and demineralized water from the Reactor Makeup Water Tank (RMWT) is ~~initiated by~~ provided to the VCT upon a low level signal. A low-low VCT level signal automatically closes an outlet valve on the VCT, opens the boric acid flow control bypass valve, and starts the boric acid makeup pumps.

isolates the VCT from the charging pumps, switching the charging pump suction to the BAST.

9.3.4.1.3.2 Reactor Coolant System Corrosion Control via the CVCS

Two chemicals are added to the reactor coolant to control oxygen: (1) hydrazine during the precriticality period, or after a long shutdown; and (2) hydrogen during post-criticality. Hydrazine is maintained in the reactor coolant in the range of 30 to 50 ppm whenever the reactor coolant temperature is below 150°F and reactor coolant is circulating. This prevents halide-induced attack, which could occur if significant quantities of fluorides or chlorides and significant amounts of dissolved oxygen are present. During heatup, any dissolved oxygen is scavenged by the hydrazine, eliminating the potential for general corrosion. At higher temperatures, the hydrazine decomposes, forming ammonia. The resultant increase in pH aids in the development and maintenance of passive oxide films on reactor coolant system surfaces. It has been well established that the corrosion rates of Ni-Cr-Fe Alloy and 300-series stainless steels decrease with time when exposed to normal reactor coolant chemistry conditions, approaching low steady state values within approximately 200 days. A high pH minimizes corrosion product release and assists in the rapid development of the passive oxide film. Most of the film is established during pre-core operations within seven days at hot, high pH conditions.

To aid in maintaining the pH during system passivation, lithium in the form of lithium hydroxide, is added to the coolant and maintained within a 1-2 ppm lithium-7 range. I

At power, oxygen concentration is limited by maintaining excess dissolved hydrogen gas in the coolant. The excess hydrogen forces the water decomposition/synthesis reaction in the reactor core toward water synthesis, rather than hydrogen and oxygen ~~decomposition~~ formation. Oxygen added via makeup water is removed in this way.

In order to minimize the effect of crud deposition on the reactor core heat transfer surfaces, lithium-7 hydroxide additions are made. Lithium-7 hydroxide produces pH conditions within the reactor coolant at operating temperatures that reduce the corrosion product solubility and, hence, the dissolved crud inventory in the circulating reactor coolant. The elevated pH promotes conditions within the coolant for selective deposition of corrosion products on cooler surfaces (steam generators) rather than hotter surfaces (core). An additional advantage is the formation of a more stable and tenacious passive oxide layer on out-of-core system surfaces. The lithium concentration is maintained within a 0.2-2.2 ppm lithium-7 range during normal operation. I

9.3.4.1.3.3 Reactivity Control

Boron concentration is normally controlled by feed-and-bleed. To change concentration, the makeup system supplies either reactor makeup water or boric acid to the VCT, and the letdown stream is diverted to the holdup tank via the pre-holdup ion exchanger and the gas stripper. Toward the end of a fuel cycle, with low boric acid concentration in the coolant, feed-and-bleed to further reduce boron concentration becomes inefficient, and the deborating ion exchanger is used. The deborating ion exchanger contains an anion resin initially in the hydroxyl form, which is converted to a borate form as boron is removed from the reactor coolant.

9.3.4.2 System Description

9.3.4.2.1 System

The normal reactor coolant flow path through the CVCS is B
indicated by the heavy lines on the flow diagrams (Figure 9.3.4-1, Sheets 1 through 4). Design parameters for the major components are shown in Table 9.3.4-4. Normal operating parameters for the CVCS are listed in Table 9.3.4-5. Process flow data is shown in Table 9.3.4-6. I

Letdown flow from the RCS passes through the tube side of the regenerative heat exchanger where an initial temperature reduction takes place via heat transfer to cooler charging fluid on the shell side of the heat exchanger. The regenerative heat exchanger is designed to cool letdown flow to less than 450°F for all normal operations and to heat the charging flow by a minimum of 100°F. A final temperature reduction to the purification subsystem operating temperature is made by the letdown heat exchanger. The letdown heat exchanger is sized to cool inlet water from the maximum regenerative heat exchanger outlet temperature to 120°F (or lower) for most operating conditions. Both the letdown and the regenerative heat exchangers are designed for full RCS pressure and both are located inside containment. B I B

Letdown fluid pressure is reduced from RCS pressure to the operating pressure of the purification subsystem in two stages. The first pressure reduction occurs at the letdown orifices and the second occurs at the letdown control valves located downstream of the orifices. The letdown orifices are sized to pass the maximum letdown flow at full RCS pressure with ~~the~~ one control valve full open. The orifice provides the pressure reduction necessary to minimize erosion of the letdown control valve seating surfaces during normal RCS ~~reactor coolant pressure~~ operations. A bypass valve around the orifices is provided for low pressure operations. The process flow is then filtered via the purification filter, purified via a purification ion exchanger, and ~~then~~ sprayed into the VCT. An excess hydrogen inventory is ~~provided~~ maintained in the RCS by keeping a hydrogen overpressure ~~in~~ on the VCT contents. I B I

The charging pumps normally take suction from the VCT and discharge to the RCS. During normal operations, one charging pump is running and the other is in standby with power removed. normally One letdown and one charging pump flow control valve are selected for use. Seal injection water is supplied to the reactor coolant pumps (RCP) by diverting a portion of the charging flow ~~at a point in the system~~ just downstream of the charging pumps. This seal flow is then heated in the seal injection heat exchanger to approximately 125°F before filtering. Once the flow has been filtered, the seal injection fluid is distributed to the four RCPs. The undiverted charging fluid is sent to the regenerative heat exchanger where it is heated before injection into the RCS. I B

small "S"

A chemical addition tank and a chemical addition metering pump are used to transfer chemical additives to the charging line downstream of the seal injection takeoff connection. Sufficient connections exist between the CVCS and the IRWST to allow for purification, inventory adjustments, and boron adjustments to the contents of that tank.

The boron recovery portion of the CVCS accepts letdown flow diverted from the VCT as a result of feed and bleed operations for shutdowns, startups, and boron dilution over core life. The diverted letdown flow, which has passed through a purification filter and ion exchanger, also passes through the pre-holdup ion exchanger. The pre-holdup ion exchanger retains cesium, lithium, and other ionic radionuclides with high efficiency. The process flow then passes through the gas stripper, where hydrogen and fission gases are removed with high efficiency; thus (1) precluding the buildup of explosive gas mixtures in the holdup tank and (2) minimizing the release of radioactive fission product gases in aerated vents or liquid discharges. The degassed liquid is automatically pumped from the gas stripper to the holdup tank. I

Reactor coolant quality water from valve and equipment leakoffs, drains, and reliefs within the containment is collected in the Reactor Drain Tank (RDT) and scheduled for batch processing. Recoverable reactor coolant quality water outside the containment from various equipment and valve leakoffs, reliefs, and drains is collected in the Equipment Drain Tank (EDT) and scheduled for batch processing. Reactor coolant collected in either of these tanks is periodically discharged by the reactor drain pumps through the reactor drain filter and pre-holdup ion exchanger, and processed ~~according to the flow path~~ described above. This liquid is also pumped to the holdup tank.
in the same manner as diverted VCT flow, is

When a sufficient volume accumulates in the holdup tank, it is pumped by a holdup pump to the boric acid concentrator, where the bottoms are concentrated to within the range of 4000 to 4400 ppm boron. The boric acid concentrator bottoms are continuously monitored for proper boron concentration, and normally ~~the concentrator bottoms are~~ pumped directly to the BAST. In the event that abnormal quantities of radionuclides are present, the bottoms are ~~concentrated to 13 weight percent boric acid and are~~ discharged to the L~~S~~WMS. The boric acid concentrator distillate passes through a boric acid condensate ion exchanger, where boric acid carryover is removed. The distillate is collected in the RMWT for reuse in the plant. If recycle is not desired, the ~~condensate~~ distillate is diverted to the LWMS.

When the SCS is operational, a flow path through the CVCS can be established for purification. This is accomplished by diverting a portion of the flow from the shutdown cooling heat exchanger to the letdown line upstream of the letdown heat exchanger. The flow then passes through the purification filter, purification ion exchanger, and letdown strainer, and is returned to the suction of the shutdown cooling pumps. B I

IF desired,

When continuous degasification of the RCS is desired, the letdown flow is diverted from the inlet ~~line~~ of the VCT to the gas stripper, bypassing the pre-holdup ion exchanger. The letdown flow is processed in the gas stripper and is then returned to the VCT via the normal spray nozzle. ~~Sufficient hydrogen absorption occurs via the~~ VCT hydrogen overpressure can be used to replace the hydrogen removed during the gas stripping process. The charging pumps take suction from the VCT, and return the processed fluid to the RCS. B

A makeup subsystem of the CVCS provides for changes in RCS boron concentration. Boron is initially added to the CVCS using the (BABT). ~~boric acid batching tank.~~ Reactor makeup water is added to the ~~boric acid batching tank~~ BABT via the makeup supply header, and the fluid is heated by immersion heaters. Boric acid powder is added to the heated fluid while ~~the~~ a mixer agitates the fluid. A boric acid concentration of as high as 12 weight percent can be prepared. Electric immersion heaters maintain the temperature of the solution in the boric acid batching tank high enough to preclude precipitation. The concentrated boric acid solution in the BABT is drawn into the boric acid batching eductor and diluted by fluid being circulated ~~through the boric acid batching eductor~~ (from the BAST) via the boric acid makeup pumps. The reactor makeup water pumps can also be used by taking suction from the reactor makeup water tank and ~~circulating~~ pumping the water through the eductor to the BAST. I

Boric acid solution stored in the BAST is normally supplied to the RCS via the boric acid makeup pumps, while the reactor makeup water stored in the RMWT is supplied via the reactor makeup water pumps. Four operational modes of CVCS makeup are provided: dilute, borate, manual and automatic. In the dilute mode, a preset quantity of reactor makeup water is introduced into the VCT, or directly into the charging pump suction header via the volume control tank bypass valve, at a preset rate. In the borate mode, a preset quantity of boric acid is introduced into the VCT, or directly into the charging pump suction header via the VCT bypass valve, at a preset rate. In the manual mode, the flow rates of the reactor makeup water and the boric acid can be preset to give any blended boric acid solution between zero and the boric acid solution concentration in the BAST (4000-4400 ppm). ~~The manual mode is primarily used for makeup to the Safety Injection Tanks.~~ In the automatic mode, a preset blended boric acid solution is automatically introduced into the VCT upon demand from the ~~volume control tank~~ VCT level controller. The preset solution concentration is adjusted periodically by the operator to match the boric acid concentration ~~being maintained~~ in the RCS. B

9.3.4.2.2 Component Description

The major components of the CVCS are described in this section. The principal component data summary, including component design code, is given in Table 9.3.4-4. Component seismic and safety classifications are discussed in detail in Section 3.2.

A. Regenerative Heat Exchanger

The regenerative heat exchanger is a vertically mounted, shell and tube (U-tube) heat exchanger. The regenerative heat exchanger conserves RCS thermal energy by transferring heat from the letdown ~~flow~~ fluid to the charging ~~flow~~ fluid. Heating the charging ~~flow~~ fluid serves to minimize charging nozzle thermal transients. The heat exchanger is designed to maintain a letdown outlet temperature below 450°F under all normal operating conditions.

B. Letdown Heat Exchanger

The letdown heat exchanger is a horizontally mounted, shell and tube heat exchanger. The letdown heat exchanger uses component cooling water to cool the letdown ~~flow~~ fluid from the outlet temperature of the regenerative heat exchanger to a temperature suitable for operation of the purification system. The letdown heat exchanger is sized to cool the letdown ~~flow~~ fluid from the maximum outlet temperature of the regenerative heat exchanger (450°F) to below the maximum allowable operating temperature of the ion exchange resins (140°F).

C. Purification Filters

Each of the two purification filters is designed to remove insoluble particulates from the letdown flow. Each filter is designed to pass the maximum letdown flow without exceeding the allowable differential pressure across the filter elements in the maximum fouled condition. Each filter is designed for efficient remote removal of filter cartridges due to the buildup of high activity levels during filter operation.

D. Purification Ion Exchangers

Each of the the two purification ion exchangers contains a mixed bed resin~~s~~ and is provided with the necessary connections to replace the resin~~s~~ by sluicing. Each ion exchanger is designed to pass the maximum letdown flow and is identical in mechanical design to the other. The volume of resin contained in one ion exchanger is sufficient to continuously remove impurities | B

and radionuclides from normal letdown flows. The other purification ion exchanger is used intermittently to control the lithium concentration in the reactor coolant.

E. Deborating Ion Exchanger

The deborating ion exchanger is identical to the purification ion exchangers in mechanical design. The deborating ion exchanger contains an anion resin in the hydroxyl (OH^-) form. The deborating ion exchanger is sized to reduce the reactor coolant boron concentration from 30 ppm to 0 ppm using two charges of anion resin.

F. Volume Control Tank

The VCT is designed to accumulate letdown water from the RCS, to provide for control of hydrogen concentration in the reactor coolant, and to provide a reservoir of reactor coolant for the charging pumps. The VCT has sufficient ~~volume~~ (provided for below the normal operating band and above a reserve volume for vortex prevention) to accommodate full charging flow for ten minutes which is with no makeup provided to the VCT. A VCT low level signal automatically actuates the boric acid makeup system to replenish VCT fluid for an extended period of time. The VCT has sufficient volume above the normal operating band to accumulate full ~~makeup~~ letdown flow for five minutes (with charging secured), plus an additional volume for a gas cushion sized to maintain VCT pressure in the normal operating range. The normal operating level band accommodates the maximum allowable RCS leakage for one hour without the need for makeup addition. The tank has hydrogen and nitrogen gas supplies, and a vent to the GWMS to enable venting of hydrogen, nitrogen, and fission gases. B I

G. Charging Pumps

The two charging pumps are multi-stage centrifugal type pumps. Each pump is provided with vent, drain, and flushing connections to minimize radiation levels during maintenance operations. B

H. Charging Pump Mini-flow Heat Exchanger B

The charging pump mini-flow heat exchanger is a horizontally mounted, shell and tube heat exchanger. The mini-flow heat exchanger uses component cooling water to cool the recirculation flow from an operating charging pump. I

I. Boric Acid Batching Tank

The boric acid batching tank allows the operator to conveniently mix boric acid. The tank is designed to permit handling of up to 12 weight percent boric acid. The tank is

insulated and has a reactor makeup water supply from the makeup supply header. Sampling provisions, a mixer, temperature controller, and electric immersion heaters are provided. I

J. Boric Acid Storage Tank

The BAST is sized to permit ~~a~~ one back-to-back shutdown to cold shutdown, followed by a shutdown for refueling at the most limiting time in core cycle with the most reactive control rod withdrawn. The maximum concentration of boric acid in the tank shall be 2.50 weight percent.

K. Holdup Tank

The holdup tank is sized to store all recoverable reactor coolant generated by ~~a~~ one back-to-back cold shutdown to five percent subcritical with the most reactive CEA withdrawn and subsequent startups at 90% core life.

L. Reactor Makeup Water Tank

The reactor makeup water tank capacity is based on providing dilution to allow total recycle. The tank also provides dilution for a back-to-back shutdown and subsequent startup at 90 percent core life.

M. Boric Acid Makeup Pumps

The two boric acid makeup pumps are single-stage, STET centrifugal pumps. The pump motors are induction, squirrel-cage motors. STET The capacity of each boric acid makeup pump is greater than the maximum charging capacity.

N. Reactor Makeup Water Pumps

The two reactor makeup water pumps are single stage, centrifugal pumps. The pump motors are induction, squirrel cage motors. The capacity of each reactor makeup water pump is greater than the maximum charging capacity.

O. Holdup Pumps

The two holdup pumps are single stage, centrifugal pumps. The pump motors are induction, squirrel cage motors.

P. Chemical Addition Package

The chemical addition package consists of a chemical addition tank, a chemical addition pump, and a strainer. The capacity of the chemical addition tank is based upon the

maximum anticipated amount of lithium to be added in one batch. The chemical addition pump is a positive displacement pump with a variable capacity.

Q. Boric Acid Filter

The boric acid filter is designed to remove insoluble particulates from the BAST and makeup flow. B

R. Reactor Makeup Water Filter

The reactor makeup water filter is designed to remove insoluble particulates from the reactor makeup water supply to the resin sluice supply header, makeup header, and makeup system.

S. Reactor Drain Pumps

STET

The two reactor drain pumps are single-stage, centrifugal pumps. The pump motors are induction, squirrel cage motors.

T. Reactor Drain Filter

The reactor drain filter is designed to remove insoluble particulates from the contents of the reactor drain tank, equipment drain tank, and holdup tank.

U. Reactor Drain Tank

The reactor drain tank is designed to: B

1. receive relief valve discharges from the shutdown cooling and safety injection systems,

2. receive gravity drains and leakage of reactor coolant ~~grade quality water~~ from components located within containment, and

stet

3. receive gravity drains from the RCS.

V. Equipment Drain Tank

The equipment drain tank receives gravity drains from the recycle drain header and the ion exchanger drain header. The equipment drain tank is also sized to accept gas stripper bypass flow for 30 minutes, and to accept discharge from miscellaneous relief valves.

W. Preholdup Ion Exchanger

The preholdup ion exchanger is identical to the purification ion exchangers in mechanical design. The preholdup ion exchanger contains a mixed bed resin and is designed to pass the maximum letdown flow. The volume of resin contained in the preholdup ion exchanger is sufficient to remove impurities and radionuclides from normal letdown flows.

X. Gas Stripper

The gas stripper achieves efficient gas stripping of reactor coolant by heating the process fluid and passing it ~~down~~ through a packed tower which employs steam as a stripping medium. ~~The degassed process fluid is then cooled.~~ The gas stripper package includes pumps ~~which~~ to transfer the degassed process fluid to the holdup tank, or to the VCT during continuous degassing of normal letdown flow. Noncondensible gases, along with trace quantities of fission gases and water vapor, flow to the GWMS.

Y. Boric Acid Concentrator Package

The boric acid concentrator concentrates the boric acid solution in the process flow by means of evaporation. The process flow enters the concentrator and is heated via recirculation through a steam heater. The vapor leaving the recirculation flow is stripped of entrained liquid by demisters, condensed, and pumped to the RMWT. The concentrate (bottoms) is cooled and pumped to either the BAST, or the LWMS. B

Z. Boric Acid Condensate Ion Exchanger

The boric acid condensate ion exchanger contains an anion resin of sufficient volume to remove boron carryover from the boric acid concentrator distillate, and is designed to pass the maximum boric acid concentrator bypass flow.

AA. Seal Injection Filters

These two redundant filters are designed to remove insoluble particles from the seal injection flow to the reactor coolant pumps. Each unit is designed to pass the maximum anticipated flow without exceeding the allowable differential pressure across the element in the maximum fouled condition.

A pressurizer steam bubble is formed by ~~heating the pressurizer~~ adjusting RCS pressure via the letdown control valves and then ~~heating the pressurizer to saturation~~ increasing the pressurizer temperature until it is heated to saturation. When reactor coolant is removed from the RCS, this causes flashing in the pressurizer steam space.

BB. Seal Injection Heat Exchanger

The seal injection heat exchanger is a vertical heat exchanger which uses steam (shell side) to heat the seal injection flow (tube side). The seal injection heat exchanger functions to maintain a relatively constant fluid temperature to minimize thermal transients to the RCP seals. B

9.3.4.2.3 System Operation

The Chemical and Volume Control System is designed to be operated as follows:

A. Plant Startup

Plant startup is the series of operations which bring the plant from a cold shutdown condition to a hot standby condition (normal operating pressure, zero power temperature, with the reactor critical at a low power level).

~~The~~ A charging pump~~s~~ and the letdown control valves are used during the initial phase of reactor coolant system heatup to maintain RCS pressure until the pressurizer steam bubble is established. ~~This is performed in the following manner.~~ Prior to establishing a pressurizer steam bubble, the RCS will be in a water solid condition with one charging pump, one letdown ~~flow~~ control valve, and one charging pump flow control valve in operation. The charging pump flow control valve will be held in its minimum automatic position by the pressurizer level control program. ~~The manual bypass around the letdown orifice will be full open.~~ RCS pressure will be automatically maintained by the letdown control valve. ~~Once a pressurizer steam bubble has been established, pressurizer level and system pressure are controlled by use of both pressurizer heaters and the letdown control valves.~~ B

The letdown orifices will be bypassed by diverting flow through the bypass line.

The VCT is initially purged with nitrogen, and a hydrogen overpressure is ~~then~~ established. The RCS boron concentration may be reduced during heatup in accordance with shutdown margin limitations. The makeup controller is operated in the dilute mode to inject a predetermined amount of reactor makeup water at a preset rate. Compliance with the shutdown margin limitations is verified by sample analysis and boronometer indication.

The letdown orifice bypass valve is closed to limit downstream pressure. Finally, the pressurizer level control system is placed in automatic.

B. Normal Operation

Normal operation includes hot standby operation and power generation (RCS operations at normal RCS pressure and

During the RCS heatup, pressurizer level and RCS pressure is maintained by adjusting the position of the letdown control valves, in conjunction with placing individual orifices in service.

The normal VCT hydrogen overpressure is replaced with nitrogen purges to maintain a low reactor coolant hydrogen concentration.

temperature). A description of normal operation is contained in Section 9.3.4.2.1.

C. Plant Shutdown

Plant shutdown is a series of operations which brings the plant from a hot standby condition to a cold shutdown condition for maintenance or refueling.

Prior to the plant cooldown, the gas space of the VCT is vented to reduce fission gas activity and the RCS dissolved hydrogen concentration to less than ~~10~~ 5 cc/kg cc(STP)/kg. The purification rate may be increased to accelerate the degasification, ion exchange, and filtration processes. Degassing the reactor coolant is accomplished by diverting letdown flow to the gas stripper and returning the process fluid to the VCT. Addition of other chemicals is not normally required during a plant shutdown.

Boron concentration in the reactor coolant system is normally increased concurrently with the cooldown by direct charging from the BAST. ~~as part of the inventory required to make up for coolant contraction.~~ Borating concurrently with the cooldown greatly reduces the amount of liquid waste generated during the shutdown process.

Once the required RCS boron concentration has been reached, the charging pump suction is switched from the BAST to the VCT. Following this switchover, the low level condition in the VCT will cause automatic makeup at the required shutdown boron concentration. Pressurizer level is maintained via positioning of the charging pump and letdown flow control valves. All or part of the charging flow may be used for auxiliary spray to cool the pressurizer and increase its boron content when RCS pressure is below that required to operate the reactor coolant pumps.

Prior to ~~an~~ scheduled refuelings, ~~outage,~~ the IRWST boron concentration is verified to be at the maximum operating limit of 4400 ppm. Any borating operations of the IRWST which may be necessary to achieve this concentration are completed prior to the scheduled shutdown. ~~any borating operations of the IRWST which may be necessary are completed prior to the scheduled shutdown.~~ As ~~After~~ the reactor vessel head is removed, the continuous spray of shutdown cooling system pumps take borated water from the IRWST, ~~and inject the water into the reactor coolant loops via the normal flow paths, thereby~~ to fill~~ing~~ the refueling pool during the shutdown. The resulting concentration of the refueling pool and the RCS is between the lower operating boron concentration limitation of the IRWST (4000 ppm) and the ~~IRWST~~ maximum operating boron concentration of the IRWST (4400 ppm). Thus, the contents of the refueling pool can be returned directly to the IRWST prior to plant startup without hindering plant operations.

During refueling shutdowns, the reactor makeup water supply piping is continuously monitored via flow switch F-250. ~~and~~ An alarm~~ed~~ is annunciated if ~~for~~ flow is detected in order to prevent dilution of the refueling pool.

9.3.4.3 Design Evaluation

9.3.4.3.1 Availability and Reliability

A high degree of functional reliability is assured by providing standby components and by assuring fail-safe responses ~~to~~ for the most probable modes of failure. Redundancy is provided as follows:

Component	Redundancy
Purification and Deborating Ion Exchangers	Three identical components
Charging Pumps	One operating, one in standby
Charging Pump Flow Control Valves	One operating and one parallel, standby valve B
Letdown Control Valves	One operating and one parallel, standby valve
Boric Acid Makeup Pumps	Two identical pumps in parallel, one operates on demand, one in standby
Gas Stripper Package	The gas stripper package includes redundant standby pumps
Seal Injection Filters	Two identical filters in parallel, one operating, one in standby
Purification Filters	Two identical filters in parallel, one operating, one ~~in~~ standby
Reactor Makeup Water Pumps	Two identical pumps in parallel, one operates on demand, one in standby
Boric Acid Concentrator Package	The concentrator package includes redundant standby pumps

In addition to component redundancy, it is possible to operate the CVCS in a manner such that some components are bypassed. It is possible to transfer boric acid to the charging pump suction header by bypassing the VCT. The letdown filter, and the purification and deborating ion exchangers can be bypassed. Controlled bleedoff flow can be routed to the RDT rather than the VCT.

Independent and redundant gravity feed lines from the BAST to the charging pump suction are provided to assure makeup. The charging pumps also have an alternate source of borated water from the spent fuel pool, which is maintained above 4000 ppm boron. ~~concentrations~~

9.3.4.3.2 Accident Response

The letdown line is isolated on receipt of a Safety Injection Actuation Signal (SIAS). A Containment Isolation Actuation Signal (CIAS) isolates the letdown line, the resin sluice supply header (RSSH) line to the RDT, and the reactor drain pump suction line.

A CIAS (or SIAS) does not isolate the charging line or stop the charging pumps. Maintaining charging flow following a CIAS continues to provide seal injection to the reactor coolant pump seals. A sufficient volume of fluid exists in the VCT to provide ample time to align alternate feed lines from the BAST to the charging pump suction header.

9.3.4.3.3 Overpressure Protection

In order to provide for safe operation of the CVCS, overpressure protection is provided throughout the system by relief valves. The following is a description of the relief valves that are located in the CVCS:

A. Low Pressure Letdown Relief Valve

The relief valve downstream of the letdown control valves protects the low pressure piping, purification filters, ion exchangers, and letdown strainer from overpressurization. The valve capacity is equal to ~~the capacity of the letdown orifice with the control valve full open at normal system pressure~~ maximum letdown flow. The set pressure is equal to the design pressure of the low pressure piping and components.

B. Volume Control Tank Relief Valve

The relief valve on the VCT is sized to pass a liquid flow rate equal to the sum of the following flow rates: the

maximum letdown flow rate of possible without actuating the high flow alarm on the letdown flow indicator; the normal controlled bleedoff flowrate; the design purge flow rate of the sampling system (SS); and, the maximum flow rate ~~that~~ the boric acid makeup system ~~can~~ ~~produce with relief pressure in the VCT.~~ (corrected for maximum VCT pressure). The set pressure is equal to the design pressure of the VCT.

C. Volume Control Tank Gas Supply Relief Valve

This ~~The~~ relief valve is sized to pass a flow rate greater than the combined maximum capacity of the nitrogen and hydrogen gas regulators. The set pressure is lower than the VCT design pressure.

D. Reactor Coolant Pump Controlled Bleedoff Header Relief Valve

The relief valve at the reactor coolant pump controlled bleedoff header allows controlled bleedoff flow to be rerouted to the RDT in the event that a downstream valve in the controlled bleedoff line to the VCT is closed. It does not serve an overpressure protection function. The valve is sized to pass the flow due to the failure of ~~one~~ two seals in one reactor coolant pump~~'s seals~~, plus the normal bleedoff from the other three reactor coolant pumps. ~~The maximum relief valve opening pressure is less than the controlled bleedoff high high pressure alarm setpoint.~~ B

E. Boric Acid Batching Eductor ~~Heat Traced Piping~~ Relief Valve~~s~~

A ~~R~~relief valve~~s~~ is ~~are~~ provided for the at ~~those~~ portion~~s~~ of the boric ~~acid~~ batching system that is ~~are~~ heat traced and ~~which~~ can be ~~individually~~ isolated. The set pressure is equal to the design pressure of the ~~corresponding portion of the~~ system piping. The ~~Each~~ valve is sized to relieve the maximum fluid thermal expansion rate that would occur if maximum duplicate heat tracing power were inadvertently applied to the isolated line.

F. Equipment Drain Tank Relief Valve

The EDT relief valve is sized to pass a liquid flow rate equal to the flow of the shutdown cooling return relief valve. The set pressure is equal to the design pressure of the EDT. B

G. Reactor Drain Tank Relief Valve

A relief valve which vents to the Holdup Volume Tank (containment sump) is provided for the RDT. The relief valve is sized to pass a liquid flow equal to the total flow rate of all discharges into the RDT. The set pressure is equal to the design pressure of the RDT.

H. Charging Pump Mini-flow Relief Valve

The relief valve down stream of the charging pump mini-flow orifice protects the charging pump minimum flow piping from overpressurization due to thermal expansion that might result from operating a charging pump with its discharge isolation valve closed. The relief valve is sized to pass the flow rate equal to the maximum fluid thermal expansion rate that would occur due to pump heat input. I

I. Seal Injection Heat Exchanger Thermal Relief

The tube side of the seal injection heat exchanger is protected by a thermal relief valve. This relief valve is sized to protect the heat exchanger from overpressurization because of ~~from~~ thermal expansion of trapped water due to inadvertent closure of the isolation valves with steam to the shell side.

STET on cross out lines, i.e. include J. section w/ noted change

J. Regenerative Heat Exchanger Thermal Relief

A thermal relief valve ~~A spring loaded check valve (CH-435)~~ is provided in the charging line [redacted] downstream of the regenerative heat exchanger to protect against overpressure from continued letdown operation with both charging and auxiliary spray isolated. [redacted] It is sized to pass full charging flow should CH-208 fail closed. B

9.3.4.3.4 Chemistry And Purity Control

The CVCS controls the chemistry and purity of the reactor coolant in order to:

A. minimize the corrosion of hardware, which includes minimizing the fouling of heat transfer surfaces;

B. control core reactivity throughout the life of the core (by adjusting the chemical shim);

C. limit the transport of radioactive corrosion products; and I

D. ensure that the quality of reactor coolant fluid is maintained within specific operating limits.

Table 9.3.4-1 describes the chemistry of the reactor coolant. B

The oxygen and chloride limits [redacted] presented in Table 9.3.4-1 of $\leq$ 0.10 ppm and $\leq$ 0.15 ppm, respectively, were established from the relationships between oxygen and chloride concentrations and their effect on the susceptibility to stress corrosion cracking I

of austenitic stainless steel. Current industry data reveals that no chloride stress corrosion occurs at oxygen concentrations below approximately 0.8 ppm. The oxygen limit in Table 9.3.4-1 was reduced by a factor of 8 to give a conservative concentration of 0.10 ppm oxygen. The maximum amount of oxygen from air dissolved in water at 77°F is approximately 8 ppm. At this concentration, a chloride concentration of less than approximately 1.50 ppm would preclude the possibility of chloride stress corrosion. This limit was reduced by a factor of 10 to provide a conservative chloride limit of 0.15 ppm.

The fluoride limit of 0.15 ~~0.10~~ ppm for reactor coolant is the result of the fluoride ion being identified as causing intergranular corrosion of sensitized austenitic stainless steels. Based on this, it is essential to minimize fluoride ions in the reactor coolant. Therefore, the concentration chosen as the maximum limit is the lowest concentration which can be both: (1) readily detected in bulk water and (2) be maintained by the action of the purification ion exchanger.

Chemistry control of the reactor coolant consists of preoperational removal of oxygen by hydrazine scavenging, degasification (via the gas stripper) of makeup water if necessary during startup, control of oxygen concentration by maintaining an excess hydrogen concentration during normal operation, and pH control by maintaining lithium within a specific control band. ~~during normal operation~~. A chemical addition tank and pump ~~is~~ are used to transfer hydrazine and/or lithium hydroxide to the discharge side of the charging pumps for injection into the RCS. I

Lithium is generated in significant quantities in the core region by the reaction $B^{10}(n,\alpha)Li^7$; therefore, ~~it~~ lithium hydroxide is the logical choice for a pH control agent. However, there exists a threshold for accelerated attack of Zircaloy at approximately 35 ppm lithium. Therefore, ~~lithium concentration limits are specified to provide a wide margin between~~ the upper operating limit and the threshold for accelerated attack in the event any concentrating phenomena exist. is specified Early in core life, periodic removal of lithium by ion exchange is required to control the lithium concentration below the upper limit. One purification ion exchanger is used intermittently to control the lithium concentration. Prior to refueling shutdown, when large ~~dilution~~ boration operations are necessary, lithium additions will be necessary to maintain the lithium concentration within the control band. The lower limit on lithium concentration ensures that sufficient lithium hydroxide is present during operation to provide the benefits noted in Section 9.3.4.1.3.2.

The control of other impurities is accomplished by the continuous operation of the second purification ion exchanger, which has been

converted to the lithium or ammonia lithium form and does not remove lithium. The resin beds remove soluble nuclides by an ion exchange mechanism, and insoluble particles by the impingement of these particles on the surface of the resin beads.

The normal method of adjusting boron concentration is by feed and bleed. To change concentration, the makeup portion of the CVCS supplies either reactor makeup water or boric acid to the VCT. ~~and~~ Upstream of the VCT, the incoming letdown stream is diverted ~~to~~ through the pre-holdup ion exchanger to the holdup tank, to avoid overfilling the VCT. Toward the end of the core cycle, the quantities of waste produced due to feed-and-bleed operations become excessive, and the deborating ion exchanger is used to reduce the reactor coolant system boron concentration. An anion resin, initially in the hydroxyl form, is converted to a borate form as boron is removed.

Boric acid recovery from the reactor coolant liquid waste is accomplished by the boric acid concentrator package at a processing rate of 20 gpm. Based on the waste estimates identified in Table 9.3.4-3, the concentrator once-through usage factor is less than 10 percent, thus resulting in an adequate opportunity for reprocessing of the RMWT contents, the BAST contents, or the IRWST contents if necessary. B I

Various reactions taking place within the reactor during power operation result in the production of tritium, which appears in the reactor coolant as tritiated water. See Section ~~11.1.5~~ 11.1.1.3 for a discussion of tritium.

9.3.4.3.5 ~~System~~ Containment Isolation

~~9.3.4.3.5.1 Containment Isolation~~

There are seven penetrations through the containment structure to accommodate CVCS piping. Four of these penetrations (charging flow to RCS, purification stream from the shutdown cooling heat exchanger to the letdown heat exchanger, seal injection flow to RCPs, and resin sluice supply header flow to the RDT) allow flow in the inward direction, and three of these penetrations (letdown line flow to purification ion exchangers, RCP controlled bleed-off flow, and RDT flow to reactor drain pumps) allow flow in the outward direction. B

The penetration for the charging piping to the RCS consists of a Safety Class 2 motor-operated valve (CH-524) outside containment and a Safety Class 2 check valve (CH-747) inside containment. CH-524 is operable from the control room and is provided with position indication in the control room. CH-524 does not receive an automatic close signal. The penetration for the purification I B

stream from the SCS heat exchanger to the letdown heat exchanger consists of a Safety Class 2 manual isolation valve (CH-307) outside containment and a Safety Class 2 check valve (CH-304) inside containment. CH-307 is a normally closed, locked closed valve that is only opened after the RCS has been shutdown and placed on shutdown cooling. The penetration for seal injection flow consists of a Safety Class 2 motor-operated valve (CH-255) located outside containment and a Safety Class 2 check valve (CH-835) located inside containment. CH-255 is operable from the control room, with position indication in the control room, and does not receive an automatic close signal. The penetration for the resin sluice supply header flow to the RDT consists of a Safety Class 2 pneumatic valve (CH-580) located outside containment and a Safety Class 2 check valve (CH-494) located inside containment. CH-580 is a failed close valve operable from the control room, and receives an automatic close signal on CIAS.

The penetration for the letdown line flow to the purification system consists of two Safety Class 1 pneumatic valves (CH-515 and CH-516) located inside containment, and a Safety Class 2 pneumatic valve (CH-523) located outside containment. All are fail-closed valves, operable from the control room, with position indication in the control room, and both CH-516 and CH-523 receive an automatic close signal on CIAS. The penetration for RCP controlled bleed-off flow consists of a Safety Class 2 pneumatic valve (CH-505) located outside containment and a Safety Class 2 pneumatic valve (CH-506) located inside containment. Both CH-506 and CH-505 are fail-closed valves, operable from the control room with position indication in the control room and both receive an automatic close signal on CSAS. The penetration for the RDT flow to the reactor drain pumps consists of a Safety Class 2 pneumatic valve (CH-560) located inside containment and a Safety Class 2 pneumatic valve (CH-561) located outside containment. Both CH-560 and CH-561 are fail-closed valves operable from the control room, with position indication in the control room. Both receive an automatic close signal on CIAS.

CH-516 also receives an automatic close signal on an SIAS, as does CH-515.

measuring devices

9.3.4.3.6 Leakage Detection and Control

Components in the CVCS are provided with welded connections wherever possible, to minimize leakage to the atmosphere. However, flanged connections are provided on all pump suction and discharge lines, on relief valve inlet and outlet connections, and on some flow ~~meters~~ to permit removal for maintenance. All valves larger than 2 inches, and all actuator-operated valves are provided with double-packing, lantern rings, and leakoff connections, unless the valves are diaphragm (packless) valves. Diaphragm valves are utilized around the VCT gas space to minimize activity release due to valve leakage.

is used to

The CVCS ~~can~~ also monitor the total RCS water inventory. If there is no leakage, ~~throughout the plant,~~ the level in the VCT and pressurizer should remain constant during steady-state operation. Therefore, a decreasing level in the VCT alerts the operator to a possible leak somewhere in the system. Increasing RDT or EDT levels may also be indicative of reactor coolant leakage.

RCS or CVCS

During refueling shutdowns the reactor makeup water piping is monitored to detect leakage past isolation valve CH-195 (which is locked shut during refueling shutdowns). If leakage occurs, an alarm is annunciated in the control room.

9.3.4.3.7 Failure Mode and Effects Analysis

Since the CVCS is not a safety-related system, a detailed failure mode and effects analysis is not ~~performed~~ required. I

9.3.4.3.8 Radiological Evaluation

areas other STET

Frequently used, manually-operated valves located in high radiation or inaccessible areas, are provided with extension stem handwheels which terminate in low radiation, ~~and~~ accessible control areas. Manually-operated valves are provided with locking provisions if unauthorized operation of the valve is considered a potential hazard to plant operation or personnel safety. Refer to Section 12.2 for further information.

STET

9.3.4.4 Testing and Inspection Requirements

Each component is inspected and cleaned prior to installation into the CVCS. A high-velocity flush using demineralized water is ~~will be~~ used to flush particulate material and other potential contamina~~tion~~nts from all lines in the system.

are are
Instruments ~~will be~~ are calibrated during preoperational testing. Automatic controls ~~will be~~ are tested for actuation at the proper setpoints, and alarm functions ~~will be~~ are checked for operability and proper setpoints. The relief valve settings ~~will be~~ are checked and adjusted as required. All sections of the CVCS ~~will be~~ are operated and tested initially with regard to flow paths, flow capacity and mechanical operability. Pumps ~~will be~~ are tested to demonstrate head and capacity.

The CVCS is tested for integrated operation with the RCS during hot functional testing. Testing of heat exchanger performance, and the proper control of the letdown and charging pump flow control valves by the pressurizer level control program is included. The charging line ~~will be~~ is checked to assure that the piping is free of excessive vibration. Response of the makeup portion of the CVCS in the automatic, dilute, and borate modes ~~will be~~ is verified. Any defects in operation that could affect plant safety are corrected before fuel loading.

As part of normal plant operation, tests, inspections, data tabulation and instrument calibrations are made to evaluate the condition and performance of the CVCS equipment and instrumentation. Data ~~will be~~ is taken periodically during normal plant operations to confirm heat transfer capabilities and purification efficiency. Pump and valve leakage ~~will be~~ is monitored.

make sure capitalized in SSC of Safety Class

Appropriate vents, drains, and test connections are provided to permit ~~the site operator to perform~~ inservice testing of valves. Provisions are made to permit the inservice testing of Safety Class 2 and 3 pumps in accordance with Section XI of the ASME Code.

9.3.4.5 Instrumentation Requirements

9.3.4.5.1 Temperature Instrumentation

A. Holdup Tank and Reactor Makeup Water Tank Temperature

The temperature of the contents of these tanks is indicated in the main control room. An ~~low temperature~~ alarm annunciates in the main control room to warn the operator of low temperature in ~~the~~ either tank.

B. Boric Acid Storage Tank Temperature

Two instruments are installed in the BAST. One provides temperature indication in the control room, the other provides indication locally. Alarm Annunciation in the control room warns the operator of an abnormally low tank temperature. B

C. Boric Acid Batching Tank Temperature

The batching tank temperature measurement channel controls the tank's electric immersion heaters. Local indication is provided to facilitate batching operations. I

D. Letdown Line Temperature

The letdown line (tubeside)

~~R~~egenerative heat exchanger outlet temperature ~~on the letdown line~~ is indicated in the control room. ~~and local indication is provided outside of containment.~~ An alarm is provided to alert the operator to an abnormally high letdown temperature. The instrument also provides a signal that positions the letdown flow control valve automatically to minimum flow ~~at a setpoint above the~~ on high temperature ~~alarm setpoint~~ conditions. The valve must be manually reset to restore normal letdown flow. B

E. Letdown Heat Exchanger Outlet Temperature

This instrument is used to control the Component Cooling Water System (CCWS) flow through the letdown heat exchanger to maintain the proper letdown temperature for purification system operation. Letdown heat exchanger outlet temperature is indicated in the control room.

F. Ion Exchanger Inlet Temperature

This temperature instrument has ~~three~~ two control functions: (1) on ~~at~~ high temperature, it provides ~~a~~ signal which positions the letdown flow control valve automatically to minimum flow, and ~~(2) at high temperature, it~~ actuates isolation valves which bypass flow, around the purification and deborating ion exchangers; ~~and~~ (~~3~~ 2) on high-high temperature ~~indication~~, it shuts CH-516, thus isolating letdown. Flow to the ion exchangers must be manually restored when the temperature decreases below the high setpoint. Temperature indication, ~~a~~ high, and ~~a~~ high-high temperature alarm(s) are provided in the control room. ~~Temperature indication is also provided at the remote shutdown panel.~~ I

and

G. Volume Control Tank Temperature

The VCT is provided with temperature indication in the control room. A~~high temperature~~n alarm is provided to alert the operator to an abnormally high water temperature.

H. Charging Line Temperature

The charging line (shellside) regenerative heat exchanger outlet temperature ~~on the charging line~~ is indicated in the control room. This indication is used to monitor heat exchanger performance and verify that auxiliary spray initiation conditions are satisfied.

I. Pre-holdup Ion Exchanger Inlet Temperature

This channel indicates gas stripper influent temperature in the control room. A high temperature alarm annunciates in the control room ~~On high inlet temperature~~ and the flow is diverted to bypass the ion exchanger to preclude resin damage.

J. Reactor Drain Tank Temperature

The RDT is provided with temperature indication in the control room. An ~~high temperature~~ alarm is provided to alert the operator of abnormally high water temperature and the need for cooling of the tank contents.

K. Seal Injection Heat Exchanger Inlet and Outlet Temperature

~~A~~ Temperature instruments ~~controller~~ on the inlet and ~~at the~~ outlet of the seal injection heat exchanger provide~~s~~ input to the seal injection temperature controller to maintain the outlet side fluid temperature within acceptable limits. The seal injection temperature controller positions CH-231, which regulates the flow which bypasses the heat exchanger. The proper mix of thru flow and bypass flow, for any given inlet temperature, will ensure a properly regulated outlet temperature. Indication and alarms are provided in the control room. B

L. Equipment Drain Tank Temperature

The EDT is provided with temperature indication in the control room. An ~~high temperature~~ alarm is provided to alert the operator of abnormally high water temperature and the need for cooling of the tank contents.

M. Charging Pump Mini-Flow Heat Exchanger Outlet Temperature I

This instrument is used to control the CCWS flowrate through the charging pump mini-flow heat exchanger to maintain adequate charging pump~~s~~ motor heat removal.

9.3.4.5.2 Pressure Instrumentation

A. Letdown Line Pressure

~~The letdown line~~ A pressure instrument upstream of the letdown control valves measures letdown pressure, with ~~to maintain a constant RCS pressure during low system pressure operations.~~ Indication and alarms ~~are provided~~ in the control room. This instrument also controls the letdown control valve position to maintain proper upstream pressure regardless of flow variations during low pressure operations. B

B. Purification Filter, Ion Exchanger and Letdown Strainer Differential Pressures

Differential pressure ~~indicators~~ instruments are provided to indicate the pressure loss across the purification filters, and across the ion exchangers plus letdown strainer. Both differential pressure indicators have local readouts and control room high differential pressure alarms. Periodic monitoring of these instruments will indicate any progressive loading of the units.

C. Volume Control Tank Pressure

This channel indicates VCT pressure in the control room. High and low pressures are annunciated in the control room.

D. Charging Pump Suction Line Pressure Switches

A pressure switch on each charging pump suction manifold stops the associated charging pump on low suction line pressure, thus preventing damage due to cavitation.

E. Boric Acid Makeup Pump Discharge Pressures

The discharge pressure of each boric acid makeup pump is indicated in the control room and locally. Low pressure alarms, annunciating in the control room, are provided. If the pump has been manually turned off by the operator, the discharge pressure alarm is suppressed. A low discharge pressure stops the respective pump, and starts the alternate pump. B I

F. Boric Acid Filter Differential Pressure

A differential pressure ~~indicator~~ instrument with local readout is provided to indicate the pressure loss across the boric acid filter. ~~The~~ A high pressure alarm is provided in the control room.

G. Charging Line Pressure

The charging line pressure is indicated in the control room. ~~and at a location outside the control room.~~ A low pressure

alarm is provided in the control room. A low charging line pressure alarm during normal operation is indicative of charging pump failure. I

H. Seal Injection Filter Differential Pressure

A differential pressure ~~indicator~~ instrument with local indication and high differential pressure annunciation in the control room is provided to determine the pressure loss across the seal injection filters. Periodic readings of this instrument will indicate any progressive loading of the ~~unit~~. operating ~~[illegible]~~ filter.

I. Reactor Coolant Pump Controlled Bleedoff Header Pressure

A pressure measurement is designed channel is provided to measure the pressure at the reactor coolant pump controlled bleedoff header. Indication is provided in the control room, and the measuring device ~~has overpressure protection~~ for RCS design pressure. A high alarm and a high-high alarm are annunciated in the control room. The high alarm indicates that a valve in the line to the VCT has been closed. The high-high alarm indicates that the controlled bleedoff flow has ~~stopped~~. been isolated. B

as a means of overpressure protection.

J. Ion Exchanger Drain Header Strainer Differential Pressure

A local differential pressure indicator is provided with a local alarm. These instruments will indicate any progressive loading of the strainer.

K. Equipment Drain Tank Pressure

EDT pressure is indicated, ~~in the control room~~ and ~~is provided with~~ annunciated, a high-pressure alarm in the control room. This instrument also actuates valves to automatically isolate the equipment drain tank from the gas analyzer, gaseous waste management system, the recycle drain header, and the reactor drain pumps when the tank pressure exceeds the high-pressure alarm setpoint.

L. Reactor Drain Tank Pressure

This instrument provides pressure indication, alarm on in the control room and actuates ~~a~~ high pressure. ~~alarm.~~ This instrument closes the RDT isolation valve to the GWMS and the containment isolation valve (inside containment) on high RDT pressure.

~~[illegible]~~
The high pressure alarm is used to alert the operator that the tank has received a discharge from one or more relief valves inside containment. In order to prevent a potential radioactive release outside containment,

M. Reactor Drain Pump Discharge Pressure

~~The~~ Each pump(s) discharge pressure~~s~~ ~~are~~ is indicated locally and in the control room~~, and function~~ in order to monitor pump performance.

N. Reactor Drain Filter, and Pre-holdup Ion Exchanger and Strainer Differential Pressures

Differential pressure ~~indicators~~ instruments are provided to indicate the pressure loss across these components. Both differential pressures are indicated locally. High-pressure alarms are annunciated in the control room.

O. Holdup Pumps Discharge Pressure

Individual pump discharge pressures are indicated locally, in order ~~and function~~ to monitor pump performance.

P. Boric Acid Condensate Ion Exchanger and Strainer Differential Pressure

A local differential pressure indicator with a high alarm is provided. Periodic reading of this instrument will indicate any progressive loading of the ion exchanger and/or strainer.

Q. Reactor Makeup Water Pump Discharge Pressure

Reactor makeup water pump discharge pressure is indicated locally and in the control room. ~~The~~ a low pressure alarm annunciates in the control room. Low pressure on one pump stops that pump and starts the standby pump. If the pump has been manually turned off by the operator, the low discharge pressure alarm is suppressed. B

R. Reactor Makeup Water Filter Differential Pressure

A differential pressure ~~indicator~~ instrument, with local readout and a high differential pressure alarm in the control room, is provided to indicate excessive loading ~~, causing high pressure loss across~~ of the reactor makeup water filter.

9.3.4.5.3 Level Instrumentation

A. Holdup Tank and Reactor Makeup Water Tank Level

Level indication and alarms for these tanks are ~~indicated~~ provided in the control room. On low level in the HT and low-low level in the RMWT, the respective pumps are automatically stopped. A ~~The~~ low level alarm for the RMWT warns the operators ~~of~~ if they are entering the volume required for back-to-back cold shutdowns at 90% of core life. A high level alarm in the HT indicates

that tank processing should be commenced. ~~The~~ A high level alarm in the RMWT and a ~~the~~ high-high level alarm in the HT indicate~~s~~ that filling of the respective tank~~s~~ should be secured.

B. Volume Control Tank Level

Redundant, differential pressure type level instruments provide VCT level indication in the control room. ~~locally, and one provides level indication at a remote location.~~ One VCT level instrument controls the starting and stopping of the automatic makeup system on low and high level indications. ~~This~~ The other channel ~~also~~ automatically diverts letdown flow on high level to the gas stripper, via the pre-holdup ion exchanger. On low-low level, both channels redundantly isolate the VCT after realigning charging pump suction ~~from~~ to the BAST. ~~so that a constant source for charging flow is maintained.~~ Redundant high, low and low-low level alarms are provided in the control room. I

C. Equipment Drain Tank and Reactor Drain Tank Level

Differential pressure type level instruments indicate level for each tank in the control room. The transmitters also activate high and low level alarms in the control room, and automatically stop the reactor drain pump on low level.

D. Boric Acid Storage Tank Level

Two instruments are provided with indication and alarms in the control room. ~~and at a location outside of the control room, and alarms in the control room.~~ One transmitter also stops the boric acid makeup pumps on low-low level. B

9.3.4.5.4 **Flow Instrumentation**

A. Letdown Flow

An orifice-type flow meter indicates letdown flow in the control room. ~~and at a location outside of the control room.~~ This channel actuates a high flow alarm in the control room. I

B. Charging Flow

Charging flow rate indication and low flow annunciation are provided in the control room ~~and at a location outside the control room.~~ If the charging pump is manually turned off by the operator, the low flow alarm is suppressed. B

C. Seal Injection Flow Rate

Orifice-type flow meters indicate seal injection ~~supply~~ flow~~s~~ to each reactor coolant pump. These instruments control the seal injection flow control valves to maintain the desired flow to each pump. Control room indication ~~of~~ and high, high-high, and low flow annunciation is provided.

D. Volume Control Tank Hydrogen and Nitrogen Gas Flow

Local indications of nitrogen and hydrogen gas flow to the VCT are provided. The nitrogen flow meter is used during VCT purging operations. The hydrogen flow meter is used during operations where a hydrogen overpressure is desired in the VCT.

E. Reactor Makeup Water Flow

An orifice-type flow meter is provided to measure the reactor makeup water flow rate to the VCT makeup blending tee. This (CH-210X) channel controls the reactor makeup water control valve to obtain a preset flow rate. High and low flow alarms are delayed to allow the set flow rate to become established. A high-high alarm is provided to avoid exceeding design flow of the reactor makeup water filter. The flow rate is recorded and the total quantity of reactor makeup water added is indicated in the control room.

F. Concentrated Boric Acid Flow

An ultrasonic flow meter is provided to measure the BAST concentrated boric acid flow rate to the VCT makeup blending tee. This channel controls the boric acid control valve (CH-210Y) to obtain a preset flow rate. High and low flow alarms are delayed after initiation of the makeup signal to allow the set flow rate to become established. A high-high alarm is provided to avoid exceeding design flow of the boric acid filter. The flow rate is recorded and the total quantity of boric acid added is indicated in the control room.

G. Reactor Makeup Water Flow Switch

A flow switch located downstream of ~~the~~ makeup controller flow indicator F-210X is used to indicate and alarm in the control room if demineralized water flow occurs during refueling operations. During normal operations, the flow switch is not operational.

H. Boric Acid Batching Flow

This instrument indicates locally the flow of boric acid from the boric acid batching tank to the boric acid batching eductor. I

I. Ion Exchanger Drain Header Flow Switch

A flow switch is provided with local indication of flow. An ~~The~~ indicator light is on whenever draining is in progress. The light goes off when an ion exchanger draining operation is complete. When refilling an ion exchanger after changing resin, the light indicates overflow from the vent line drain, and therefore completion of the filling operation.

J. Resin Sluice Supply Header Air Flow

This instrument provides local indication of air flow to the resin sluice supply header.

K. Reactor Makeup Water Flow to Resin Sluice Supply Header

This instrument provides local indication of reactor makeup water flow to the resin sluice supply header.

9.3.4.5.5 Radiation Monitoring Instrumentation

9.3.4.5.5.1 Gas Stripper Effluent Radiation Monitor

This monitor provides ~~a~~ continuous indication ~~recording~~, in the control room~~,~~ of the gross gamma activity leaving the gas stripper and entering the holdup tank. A high alarm indicates improper operation of upstream purification equipment. Normally, however, an increasing activity trend will allow operators to take corrective measures (replace ion exchanger resin or filter cartridges) before significant activity increase occurs in the holdup tank. The radiation monitor consists of a logarithmic ratemeter which processes pulses from a shielded scintillation detector. I

THIS TABLE and other pages of this section HAVE BEEN REVISED BY applicable parts of SAR PKG ALWR-FS-137, Rev. 00. A COPY OF THIS PACKAGE IS ATTACHED.

TABLE 9.3.4-1

(Sheet 1 of 2)

OPERATING LIMITS

1.0 REACTOR COOLANT MAKEUP WATER

Analysis	Normal	
Chloride (Cl)	< 0.005 ppm	I
pH	6.0 - 8.0	B
Fluoride (F)	< 0.005 ppm	I
Suspended Solids	< 0.05 ppm	

2.0 PRIMARY WATER

Analysis	Pre Core Hot Functionals (1)	Initial Core Load and Criticality	Power Operation	
pH (77°F)	3.8 - 10.4	4.5 - 10.5	4.5 - 10.5	I
Conductivity	(2)	(2)	(2)	
Hydrazine	30-50 ppm (3)	30-50 ppm (3)	1.5 x Oxygen ppm (4) (max. 20 ppm)	B
Ammonia	0-50 ppm	0-50 ppm	0-2ppm	I
Dissolved Gas			(5)	
Lithium	1-2 ppm	0.2-2.2 ppm	0.2-2.2 ppm	
Hydrogen		(6)	25-50 cc (STP)/kg (H_2O) (7)	B
Oxygen	$\leq$0.1 ppm	$\leq$0.1 ppm (9)	$\leq$0.1 ppm	
Suspended Solids	<0.35 ppm, 2 ppm max. (8)	<0.35 ppm, 2 ppm max. (8)	<0.35 ppm, 2 ppm max. (8)	
Chloride	$\leq$0.15 ppm	$\leq$0.15 ppm	$\leq$0.15 ppm	
Fluoride	$\leq$0.15 ppm	$\leq$0.15 ppm	$\leq$0.15 ppm	I
Boron	$\leq$ Refueling Concentration	$\leq$ Refueling Concentration	$\leq$ Refueling Concentration	

SEE NOTE ON PRIOR PAGE.

TABLE 9.3.4-1 (Cont'd)

(Sheet 2 of 2)

OPERATING LIMITS

NOTES:

(1) Special hot conditioning limits:

Temperature >350°F for 7-10 days

(2) Consistent with pH additive concentration.

(3) Hydrazine is maintained at 30-50 ppm any time the RCS is less than 150°F.

(4) Prior to exceeding 150°F during heatup or below 400°F during cooldown.

(5) Prior to a depressurization shutdown, reduce total gas to <10cc(STP)/kg (H_2O) to limit the possibility for explosive mixtures.

(6) During the transition from post-core to operating, hydrogen should be maintained in the 15 to 25cc(STP)/kg (H_2O) range to minimize degassing requirements in case the reactor plant must be shutdown and depressurized.

(7) Hydrogen should be <5cc(STP)/kg (H_2O) before securing the reactor coolant pumps.

(8) The abnormal condition of 0.35 to 2.0 ppm is permitted for up to 14 hours to allow for crud burst conditions. B

(9) Not applicable during core load.

Q/R 281.47

TABLE 9.3.4-1

(Sheet 1 of ~~2~~ 3)
REACTOR COOLANT
OPERATING LIMITS

~~1.0 REACTOR COOLANT MAKEUP WATER~~

~~Analysis~~	~~Normal~~	
~~Chloride (Cl)~~	~~< 0.005 ppm~~	I
~~pH~~	~~6.0 - 8.0~~	B
~~Fluoride (F)~~	~~< 0.005 ppm~~	I
~~Suspended Solids~~	~~< 0.05 ppm~~	

~~2.0 PRIMARY WATER~~

INSERT NEXT PAGE HERE

Analysis	Pre Core Hot Functionals (1)	Initial Core Load and Criticality	Startup & Power Operation	
pH (77°F)	3.8 - 10.4	4.5 - 10.5	4.5 - 10.5	I
Conductivity	(2)	(2)	(2)	B
Hydrazine	30-50 ppm (3)	30-50 ppm (3)	1.5 x Oxygen ppm (4) (max. 20 ppm)	B
Ammonia (10)	0-50 ppm	0-50 ppm	0-2ppm	I
Dissolved Gas	---	---	(5)	B
Lithium	1-2 ppm	0.2-2.2 ppm	0.2-2.2 ppm	B
Hydrogen	---	(6)	~~25~~ 15-50 cc (STP)/kg (H_2O) (7)	B
Oxygen	≤0.1 ppm	≤0.1 ppm (9)	≤0.1 ppm	B
Suspended Solids (10)	<0.35 ppm, 2 ppm max. (8)	<0.35 ppm, 2 ppm max. (8)	<0.35 ppm, 2 ppm max. (8)	B
Chloride	≤0.15 ppm	≤0.15 ppm	≤0.15 ppm	B
Fluoride	≤0.15 ppm	≤0.15 ppm	≤0.15 ppm	I
Boron	< Refueling Concentration	< Refueling Concentration	< Refueling Concentration	I
Sulfate (10)	≤0.1 ppm	≤.05 ppm	≤.05 ppm	

Q/R 281.47

INSERT TO SHEET 1 OF TABLE 9.3.4-1

Analysis	Shutdown and Refueling
pH (77°F)	3.8 - 10.5
Conductivity	(2)
Hydrazine	---
Ammonia(10)	0 - 2 ppm
Dissolved Gas	<10 cc STP/kg(H_2O)
Lithium	---
Hydrogen	<5 cc STP/kg(H_2O)
Oxygen	---
Suspended Solids	---
Chloride	≤0.15 ppm
Fluoride	≤0.15 ppm
Boron	≤Refueling Concentration
Sulfate	≤0.1 ppm

Q/R 281.47

TABLE 9.3.4-1 (Cont'd)

(Sheet 2 of 3)
REACTOR COOLANT
OPERATING LIMITS

NOTES:

(1) Special hot conditioning limits:

Temperature >350°F for 7-10 days

(2) Consistent with pH additive concentration.

(3) Hydrazine is maintained at 30-50 ppm any time the RCS is less than 150°F.

(4) Prior to exceeding 150°F during heatup or below 400°F during cooldown.

(5) Prior to a depressurization shutdown, reduce total gas to <10cc(STP)/kg (H_2O) to limit the possibility for explosive mixtures.

(6) During the transition from post-core to operating, hydrogen should be maintained in the 15 to 25cc(STP)/kg (H_2O) range to minimize degassing requirements in case the reactor plant must be shutdown and depressurized.

(7) Hydrogen should be <5cc(STP)/kg (H_2O) before securing the reactor coolant pumps.

(8) The abnormal condition of 0.35 to 2.0 ppm is permitted for up to 14 hours to allow for crud burst conditions. B

(9) Not applicable during core load.

(10) This parameter is used for problem diagnosis.

Q/R 281.47

TABLE 9.3.4-1 (Sheet 3 of 3)

REACTOR COOLANT

DETAILED STARTUP AND POWER OPERATION SPECIFICATIONS[a]

Analysis	Range		
	Normal	Abnormal	Immediate Shutdown
pH	4.5 - 10.5	--	--
Conductivity	(b)	--	--
Hydrazine, ppm	1.5 x O_2 ppm[c] (max. 20 ppm)	--	--
Ammonia[d], ppm	0-2	--	--
Lithium, ppm	0.2 - 2.2[e]	--	--
Hydrogen CC (STP) H_2/kg H_2O			
Power Operation	25 - 50	15 - 25	≤5
Startup	15 - 25	--	--
Oxygen, ppm	≤0.1	>0.1	>1.0
Suspended Solids,[d] ppm	≤0.35	--	--
Chloride, ppm	≤0.15	>0.15	>1.5
Fluoride, ppm	≤0.15	>0.15	>1.5
Boron, ppm	<Refueling Concentration	--	--
Sulfate[d], ppm	≤.05	--	--

Notes:

b. Consistent with additive concentrations.

d. This parameter is used for rapid problem diagnosis.

e. Consistent with plant lithium management program.

a. This table expands upon operation specifications as depicted ~~in~~ on Sheet 1.

c. Prior to exceeding 150°F during startup.

delete and replace with SAR chg. pkg ALWR-FS-142, rev 00, Attachment 6 with markups as shown herein.

TABLE 9.3.4-2

(Sheet 1 of 4)

DESIGN TRANSIENTS

CVCS Code Class 2 Components* Which Are Part Of The Reactor Coolant Pressure Boundary

Event		Assumed number of occurrences during the 60-year design life of the plant	Affected Components(1)
1.	Plant heatup and cooldown	500	L,C,S
2.	Turbine power step changes of +10% power (15-100% power)	2000	L,C
3.	Turbine power step changes of -10% power (15-100% power)	2000	L,C
4.	Turbine power steps of +1 percent power (5-15% power)	2000	L,C
5.	Turbine power steps of -1 percent power (5-15% power)	2000	L,C
6.	Turbine power ramp changes of +5 %/min for 90% of core life (15-100% power)	15000	L,C
7.	Turbine power ramp changes of -5 %/min for 90% of core life (100-15% power)	15000	L,C
8.	Turbine power ramps of +1 %/min (5-15% power)	2000	L,C
9.	Turbine power ramps of -1 %/min (15-5% power)	2000	L,C
10.	Daily load cycle of 100-50-100% power	15000	L,C
11.	Shutdown Cooling System purification flow operations during HSD	1000	L
12.	Shift from normal to maximum CVCS purification flowrate	2000	L,C

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TABLE 9.3.4-2 (Cont'd)

(Sheet 2 of 4)

DESIGN TRANSIENTS

CVCS Code Class 2 Components* Which Are Part Of The Reactor Coolant Pressure Boundary

	Event	Assumed number of occurrences during the 60-year design life of the plant	Affected Components(1)
13.	Low-low volume control tank level and charging pump diversion to the BAST	40	L,C
14.	NSSS operations with the control systems in the manual mode (5-100% power)	2000	L,C,S
15.	Increase in feedwater flow rate	20	L,C
16.	Reactor trip during plant startup or power operation (5-100% power)	40	L,C
17.	Startup and coastdown of a reactor coolant pump at HSB	4000	C,S
18.	Loss of normal feedwater flow	20	L,C
19.	Loss of forced reactor coolant flow	20	L,C,S
20.	Uncontrolled CEA withdrawal from a subcritical or low power condition	10	L,C
21.	Uncontrolled CEA withdrawal at power	10	L
22.	Control rod misoperation, system malfunction, operator error or inadvertent RPCS operation	20	L,C
23.	Loss of component cooling water to the letdown heat exchanger	10	L,C
24.	Startup of an inactive reactor coolant pump	20	C,S

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TABLE 9.3.4-2 (Cont'd)

(Sheet 3 of 4)

DESIGN TRANSIENTS

CVCS Code Class 2 Components* Which Are Part Of The Reactor Coolant Pressure Boundary

Event	Assumed number of occurrences during the 60-year design life of the plant	Affected Components(1)
25. Closure of a single MSIV	5	L
26. Spurious actuation of the pressurizer heaters (shutdown or power operation)	40	L,C
27. Spurious actuation of the pressurizer spray	40	L,C
28. Loss of non-emergency AC power to the station auxiliaries	10	L,C,S
29. Inadvertent opening of a pressurizer safety valves	1	L,C
30. Increase in steam flow rate	20	L,C
31. Loss of condenser vacuum	20	L,C
32. Inadvertent opening of a steam generator safety valve	10	L,C
33. Feedwater system pipe break (FWLB)	1	L,C,S
34. CVCS malfunction that increases RCS inventory	20	C
35. Operation of the auxiliary spray system (included in the plant heatup and cooldown events)	500	C
36. Opening of the feedwater economizer valve	500	L,C
37. Inadvertent closure of one economizer feedwater valve	40	L,C

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Delete and replace with ~~SAR~~ change pkg. ALWR-FS-142, Rev 00 Attachment 6 with markups as shown herein.

TABLE 9.3.4-2 (Cont'd)

(Sheet 4 of 4)

DESIGN TRANSIENTS

CVCS Code Class 2 Components* Which Are Part Of The Reactor Coolant Pressure Boundary

Event	Assumed number of occurrences during the 60-year design life of the plant	Affected Components(1)
38. Steam system piping failure (SLB)	1	L,C,S
39. Reactor coolant pump rotor seizure	1	L,C,S
40. Reactor coolant pump shaft break	1	L,C,S
41. Steam generator tube rupture (SGTR)	1	L,C,S
42. Loss of coolant accidents resulting from postulated pipe breaks within the RCS pressure boundary (LOCA)	1	L,C,S
43. Decrease in feedwater temperature	20	C
44. Rod ejection accident	1	C

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NOTES: (1) Code for symbols:
- L - Letdown line (from RCS letdown nozzle to and including CH-523)
- C - Charging line (from and including CH-524 to RCS charging nozzle)
- S - Seal injection line (from and including CH-255 to the reactor coolant pump seals)

* Design transients for Code Class 1 components are listed in Section 3.9.1.1.

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DESIGN TRANSIENTS FOR CVCS CODE CLASS 2 COMPONENTS WHICH ARE PART OF THE REACTOR COOLANT PRESSURE BOUNDARY

	Event	Occurrences (See Note 1)	Affected Components (See Note 2
1.	Power operation with normal ± NSSS process parameter variations (5-100% power)	1,000,000. *	L,C
2.	Daily load cycle of 100-50-100 % power (2 hour ramps)	22,000.	L,C
3.	Frequency control	800,000.	L,C
4.	Turbine power steps of ± 10 percent (15-100% power)	2,000. *	L,C
5.	Turbine power steps of ± 1 percent (5-15% power)	2,000. *	L,C
6.	Turbine power ramps of ± 1 %/min (5-15 % power)	2,000. *	L,C
7.	Turbine load rejection up to 50 % (50-100% power)	40.	L,C
8.	Turbine generator runback to house load (15-100 % power)	40.	L,C
9.	Loss of a main feedwater pump without causing a reactor trip (50-100% power)	40.	L,C
10.	Uncomplicated reactor trips (5-100% power) ~~(See Note 3)~~	60.	L,C
11.	NSSS operations with the control systems in the manual mode (0-5% power)	2,000.	L,C,S
12.	NSSS operations with the control systems in the manual mode (5-100% power)	2,000.	L,C,S
13.	Opening of the FW economizer valve during power increasing operations	400.	L,C
14.	Startup and coastdown of a Reactor Coolant Pump at hot standby conditions	4,000. *	C,X

This is SAR chg pkg ~~[illegible]~~, rev 00, Attachment 6, ~~[illegible]~~ marked up

Event	Occurrences (See Note 1)	Affected Components (See Note 2)
15. Operation of the auxiliary spray system	300.	C
16. Tie line thermal backup (± 20 % power change)	60. *	L,C
17. Plant heatup	300.	L,C,S,B
18. Plant cooldown	300.	L,C,S,B
19. Shift from normal to maximum CVCS flow rate and return	2,000.	L,C
20. Low-low VCT level and charging pump diversion to the ~~IRWST~~ BAST	40.	L,C,S
~~21. CVCS resin sluicing operations ....~~	~~2,000.~~	~~XX~~
~~22. Boric acid mixing to the IRWST~~ BAST	~~2,000.~~	~~L,C~~
~~23. Boric acid concentrator makeup to the IRWST~~ BAST ...	~~2,000.~~	~~L,C~~
~~24. IRWST filtration and purification of the refueling pool makeup~~	~~2,000.~~	~~L,C~~
~~25. Letdown diversion to the preholdup ion exchanger and degassing operations~~	~~2,000.~~	L,C
26. SCS purification operations at HSD	1,000.	L,C
~~27. Starting a boric acid makeup pump against the pump shutoff head~~	~~40.~~	~~L,C~~
~~28. Reactor drain tank pumpdown after a pressurizer relief valve lift~~	~~40.~~	~~L,C~~
29. Spurious pressurizer spray actuation	40.	L,C
30. Spurious pressurizer heater actuation	40.	L,C
31. Inadvertent closure of one economizer feedwater valve	40.	L,C
32. Inadvertent isolation of one main feedwater heater	40.	L,C

This is SAR change
Package ~~[illegible]~~
Attachment 4,
marked up

TABLE 9.3.4-2
(3 of 4)

Event	Occurrences (See Note 1)	Affected Components (See Note 2)
~~33. Startup and shutdown of the Shutdown Cooling System~~	~~300~~	~~L,C~~
34. Decrease in feedwater temperature	20.	L,C
35. Increase in feedwater flow rate	20.	L,C
36. Increase in steam flow rate	20.	L,C
37. Inadvertent opening of a steam generator safety valve	10.	L,C
38. Loss of load (turbine speed control systems operate properly)	19.	L,C
39. Turbine trip	20.	L,C
40. Loss of condenser vacuum	20.	L,C
41. Loss of non-emergency AC power to the station auxiliaries	10.	L,C,S,B
42. Loss of normal feedwater flow	20.	L,C
43. Loss of forced reactor coolant flow	20.	L,C,~~S~~
44. Uncontrolled CEA withdrawal from subcritical or low power conditions	10.	L,C
45. Uncontrolled CEA withdrawal at power	10.	L,C
46. CEA misoperation, system malfunction, operator error or inadvertent RPCS operation ..	20.	L,C
47. Natural circulation cooldown	30.	L,C,~~S~~
48. Loss of component cooling water to the letdown heat exchanger	10.	L,C
49. Inadvertent boron dilution	10.	L,C
50. CVCS malfunction that increases RCS inventory	20.	L,C
51. CVCS leak test	40.	L,C,S,B
52. CVCS hydrostatic test	40.	L,C,S,B

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Event	Occurrences (See Note 1)	Affected Components (See Note 2)
53. Failure of small lines carrying coolant outside containment (sample line break)	5.	L,C
54. Inadvertent MSIS at zero power	5.	L,C
55. Closure of a single MSIV	5.	L,C
56. Loss of load (turbine speed control system fails)	1.	L,C
57. Inadvertent opening of a SDS Valve	1.	L,C
58. Steam system piping failure (SLB)	1.	L,C,~~S~~
59. Feedwater system pipe break (FWLB)	1.	L,C,~~S~~
60. Reactor coolant pump rotor seizure	1.	L,C,~~S~~
61. Reactor coolant pump shaft break	1.	L,C,~~S~~
62. Steam generator tube rupture (SGTR)	1.	L,C,~~S~~
63. Loss of coolant accident (LOCA)	1.	L,C,~~S~~
64. Inadvertent opening of a pressurizer safety valve	1.	L,C,~~S~~
65. Rod ejection accident	1.	L,C

Notes (Table 9.3.4-2):

(1) This is the total number of occurrences over 60 years for each event. This frequency of occurrence is for design purposes only and does not necessarily reflect the actual expected number of operational occurrences. An asterisk (*) denotes that the frequency is in each direction (e.g., there are 2000 power steps of + 10 % and 2000 power steps of -10 %)

(2) The affected components are defined as follows:
- L ... Letdown line from the RCS nozzle to and including CH-523
- C ... Charging line from and including CH-524 to the RCS charging nozzle
- S ... Seal injection line from and including CH-255 to the reactor coolant pump seals
- B ... Bleedoff line from the RCP seals to and including CH-505

TABLE 9.3.4-3

EXCESS REACTOR COOLANT GENERATED DURING TYPICAL PLANT OPERATIONS

PLANT OPERATION	VOLUME GENERATED
Plant shutdown to refueling at 90% core cycle.	155,250 gallons
Plant startup from refueling at beginning of core cycle.	293,603 gallons
Plant shutdown to cold shutdown and startup at 50% core cycle.	79,538 gallons
Anticipated daily leakage to reactor drain tank and equipment drain tank.	250 gallons/day

E

B

TABLE 9.3.4-4

(Sheet 1 of 11)

PRINCIPAL COMPONENT DATA SUMMARY

Regenerative Heat Exchanger

Quantity	1
Type	Shell and tube, vertical
Code (tube and shell side)	ASME III, Class 2
Tube Side (Letdown)	
Fluid	Reactor coolant, 2.5 wt % boric acid, maximum
Design pressure	2485 psig
Design temperature	650°F
Materials	Austenitic stainless steel
Normal flow	100 gpm
Design flow	200 gpm
Shell Side (Charging)	ASME III, Class 2
Fluid	Reactor coolant, 2.5 wt % boric acid, maximum
Design pressure	3025 psig
Design temperature	550°F
Materials	Austenitic stainless steel
Normal flow	90 gpm
Design flow	200 gpm

Letdown Heat Exchanger

Quantity	1
Type	Shell and tube, horizontal
Code (tube and shell side)	ASME III, Class 2
Tube Side (Letdown)	
Fluid	Reactor coolant, 2.5 wt % boric acid, maximum
Design pressure	2485 psig
Design temperature	550°F
Materials	Austenitic stainless steel
Normal flow	100 gpm
Design flow	200 gpm

TABLE 9.3.4-4 (Cont'd)

(Sheet 2 of 11)

PRINCIPAL COMPONENT DATA SUMMARY

Shell Side (Cooling Water)	ASME III, Class 3
Fluid	Component cooling water
Design pressure	150 psig
Design temperature	250°F
Materials	Carbon steel
Normal flow	950 gpm
Design flow	2400 gpm
Pressure loss	15 psid @ 1500 gpm & 105°F

Seal Injection Heat Exchanger

Quantity	1
Type	Shell and tube (steam heater), vertical
Tube Side (Seal Injection)	
Code	ASME III, Class 3
Fluid	Reactor coolant, 2.5 wt % boric acid, maximum
Design pressure	3025 ~~2735~~ psig
Design temperature	200°F
Materials	Austenitic stainless steel
Pressure loss	10 psi @ 30 gpm & 120°F
Normal flow	26 gpm
Design flow	30 gpm
Shell Side (Steam)	
Code	ASME III, Class 3
Fluid	Steam-saturated
Design pressure	110 psig
Design temperature	360°F
Materials	Carbon steel
Design flow	1740 lbm./hr.

Charging Pumps

Code	ASME III, Class 3
Quantity	2
Type	Centrifugal
Design pressure	3025 ~~2735~~ psig
Design temperature	200°F
Normal flow (151 gpm at charging pump discharge)	90 gpm (to RCS loop)

TABLE 9.3.4-4 (Cont'd)

(Sheet 3 of 11)

PRINCIPAL COMPONENT DATA SUMMARY

Shutoff head	6,300 ft.
Normal suction pressure	38 psig
Normal temperature of pumped fluid	120°F
NPSH required	~~50 ft.~~ 35ft at 130gpm
Materials in contact with pumped fluid	Austenitic stainless steel
Fluid	2.5 wt % boric acid, maximum

Charging Pump Mini-flow Heat Exchanger

Quantity	1
Type	Shell and tube, horizontal
Tube Side (Charging)	
Fluid	Reactor coolant, 2.5 wt % boric acid, maximum
Design Pressure	3025 ~~200~~ psig
Design temperature	200°F
Materials	Austenitic stainless steel
Normal flow	35 gpm
Design Flow	100 gpm
Code	ASME III, Class 3
Shell Side (Cooling Water)	
Fluid	Component cooling water
Design Pressure	150 psig
Design Temperature	250°F
Materials	Carbon Steel
Normal flow	50 ~~2~~ gpm
Design Flow	200 gpm
Code	ASME III, Class 3

Boric Acid Makeup Pumps

Quantity	2
Type	Centrifugal
Design pressure	200 psig
Design temperature	200°F
Rated head	300 ft
Normal flow	180 ~~165~~ gpm
Normal operating temperature	40-120°F
NPSH required	15 ft

TABLE 9.3.4-4 (Cont'd)

(Sheet 4 of 11)

PRINCIPAL COMPONENT DATA SUMMARY

Fluid	2.5 wt % boric acid, maximum
Material in contact with liquid	Austenitic stainless steel
Code	ASME III, Class 3

Reactor Makeup Water Pumps

Quantity	2
Type	Centrifugal
Design pressure	200 psig
Design temperature	200°F
Rated head	300 ft.
Normal flow	180 ~~165~~ gpm
Normal operating temperature	40-120°F
NPSH required	15 ft
Material in contact with pumped fluid	Austenitic stainless steel
Fluid	Demineralized water
Code	None

Holdup Pumps

Quantity	2
Type	Centrifugal
Design pressure	100 psig
Design temperature	200°F
Rated head	145 ft
Normal flow	50 gpm
Normal operating temperature	40-120°F
NPSH required	10 ft
Materials in contact with pumped fluid	Austenitic stainless steel
Fluid	2.5 wt % boric acid, maximum
Code	None

Reactor Drain Pumps

Quantity	2
Type	Centrifugal
Design pressure	200 psig
Design temperature	200°F
Rated head	145 ft
Normal flow	50 gpm
Normal operating temperature	120°F
NPSH required	10 ft

TABLE 9.3.4-4 (Cont'd)

(Sheet 5 of 11)

PRINCIPAL COMPONENT DATA SUMMARY

Materials in contact with pumped fluid	Austenitic stainless steel
Fluid	2.5 wt % boric acid, maximum
Code for fluid end	ASME III, Class 3

Volume Control Tank

Quantity	1
Type	Vertical, cylindrical
Internal volume	5,800 gallons (approx)
Design pressure, internal	75 psig
Design pressure, external	15 psig
Normal operating temperature	120°F
Normal operating pressure	20 psig
Blanket gas (during plant operation)	Hydrogen
Code	ASME III, Class 3
Fluid	2.5 wt % boric acid, maximum
Material	Austenitic stainless steel

Boric Acid Batching Tank

Quantity	1
Internal volume	630 gallons (minimum)
Design pressure	Atmospheric
Design temperature	200°F
Normal operating temperature	155°F
Type heater	Electric immersion
Heater capacity, minimum	45kW
Fluid	12 wt % boric acid, maximum
Material	Austenitic stainless steel

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TABLE 9.3.4-4 (Cont'd)

(Sheet 6 of 11)

PRINCIPAL COMPONENT DATA SUMMARY

Normal operating pressure	Atmospheric
Code	None

put on previous page

Equipment Drain Tank

Quantity	1
Type	Horizontal, cylindrical
Internal volume	10,500 gallons (minimum)
Design pressure	30 psig internal, 15 psig external
Design temperature	300°F
Normal operating pressure	3 psig
Normal operating temperature	120°F
Code	ASME III, Class 3
Fluid	2.5 wt % boric acid, maximum
Material	Austenitic stainless steel

Reactor Drain Tank

Quantity	1
Type	Horizontal, cylindrical
Design pressure (internal)	130 psig
Design pressure (external)	15 psig
Design temperature	350°F psig
Normal operating pressure	3 psig
Normal operating temperature	120°F
Internal volume	2850 gallons (minimum)
Blanket gas	Nitrogen
Material	Austenitic stainless steel
Code	ASME VIII
Fluid	2.5 wt % boric acid, maximum

Holdup Tank

Quantity	1
Type	Vertical (field fabricated)
ital Useful ~~Internal~~ volume	475,000 ~~435,000~~ gallons
Design pressure	1.5 psig
Design temperature	200°F
Operating pressure	Atmospheric
Operating temperature	40-120°F
Material (wetted)	Austenitic stainless steel
Code	API-650
Fluid	2.5 wt % boric acid, maximum

TABLE 9.3.4-4 (Cont'd)

(Sheet 7 of 11)

PRINCIPAL COMPONENT DATA SUMMARY

Quantity	1	

Reactor Makeup Water Tank

Type	Vertical (field fabricated)	
Total useful ~~internal~~ volume	450,000 ~~445,000~~ gallons	
Design pressure	1.5 psig	
Design temperature	200°F	
Operating pressure	Atmospheric	
Operating temperature	40-120°F	
Material (wetted)	Austenitic stainless steel	I
Code	API-650	
Fluid	Demineralized water	

Boric Acid Storage Tank

Quantity	1	
Type	Vertical (field fabricated)	
Total useful ~~internal~~ volume	250,000 ~~180,000~~ gallons	
Design pressure	1.5 psig	
Design temperature	200°F	
Operating pressure	Atmospheric	
Operating temperature	60-120°F	
Material (wetted)	Austenitic stainless steel	
Code	ASME III, Class 3	K
Fluid	2.5 wt % boric acid, maximum	

Purification and Deborating Ion Exchangers

Quantity	3	
Type	Flushable	
Design pressure	200 psig	I
Design temperature	200°F	
Normal operating temperature	120°F	
Resin volume, each (useful)	3? 0 ft^3 (minimum required)	
Normal flow	[illegible] gpm	
Design ~~Maximum~~ flow	150 ~~170~~ gpm	
Code for vessel	ASME III, Class 3	K
Retention screen size	80 U.S. mesh	
Material	Austenitic stainless steel	
Resin	Cation/anion mixed bed for purification; anion bed for deborating	I
Fluid	2.5 wt % boric acid, maximum	

TABLE 9.3.4-4 (Cont'd)

(Sheet 8 of 11)

PRINCIPAL COMPONENT DATA SUMMARY

Pre-holdup Ion Exchanger

Quantity	1
Type	Flushable
Design pressure	200 psig
Design temperature	200°F
Normal operating temperature	120°F
Resin volume, each (useful)	32.0 ft^3 (minimum required)
Normal flow	100 gpm
Design ~~Maximum~~ flow	150 ~~170~~ gpm
Code for vessel	ASME III, Class 3
Retention screen size	80 U.S. mesh
Material	Austenitic stainless steel
Fluid	2.5 wt % boric acid, maximum
Resin	Cation/anion mixed bed

Boric Acid Condensate Ion Exchanger I

Quantity	1
Type	Flushable
Design pressure	200 psig
Design temperature	200°F
Normal operating temperature	120°F
Resin volume, (useful)	32 ft^3 (minimum required)
Normal flow	20 gpm
Design ~~Maximum~~ flow	100 gpm
Code for vessel	ASME VIII
Retention screen size	80 U.S. mesh
Material	Austenitic stainless steel
Fluid	10 ppm boron, maximum
Resin	Anion

Purification Filter

Quantity	2
Type elements	Replaceable cartridge
Retention for 2 micron and larger particles, % by wt	98%
Normal operating temperature	120°F
Design pressure	200 psig

TABLE 9.3.4-4 (Cont'd)

(Sheet 9 of 11)

PRINCIPAL COMPONENT DATA SUMMARY

Purification Filter (Cont'd)

Design temperature	200°F
Design flow	150 ~~200~~ gpm
Normal flow	100 gpm
Code for vessel	ASME III, Class 3
Material	Austenitic stainless steel
Fluid	2.5 wt % boric acid, maximum

Boric Acid Filter

Quantity	1
Type elements	Replaceable cartridge
Retention for 2 micron and larger particles, % by wt	98%
Normal operating temperature	40-120°F
Design temperature	200°F
Design pressure	200 psig
Design flow	200 gpm
Code for vessel	ASME III, Class 3
Materials, wetted	Austenitic stainless steel
Fluid	2.5 wt % boric acid, maximum

Reactor Makeup Water Filter

Quantity	1
Type elements	Replaceable cartridge
Retention for 2 micron and larger particles, % by wt	98%
Normal operating temperature	40-120°F
Design temperature	200°F
Design pressure	200 psig
Design flow	200 gpm
Code for vessel	ASME VIII
Materials, wetted	Austenitic stainless steel
Fluid	Demineralized water

Reactor Drain Filter

Quantity	1
Retention for 2 micron and larger particles, % by wt	98%
Type elements	Replaceable cartridge
Normal operating temperature	120°F

TABLE 9.3.4-4 (Cont'd)

(Sheet 10 of 11)

PRINCIPAL COMPONENT DATA SUMMARY

Design temperature	200°F
Design pressure	200 psig
Design flow	100 gpm
Code for vessel	ASME III, Class 3
Materials, wetted	Austenitic stainless steel
Fluid	2.5 wt % boric acid, maximum

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Seal Injection Filter

Quantity	2
Type elements	Replaceable cartridge
Retention for 5 micron and larger particles, % by wt	95%
Normal operating temperature	125°F
Design pressure	3025 ~~2735~~ psig
Design temperature	200°F
Design flow	30 gpm
Code for vessel	ASME III, Class 3
Materials, wetted	Austenitic stainless steel
Fluid	2.5 wt % boric acid, maximum
Normal flow	26 gpm

K

Boric Acid Concentrator

Quantity	1
~~Design DF (Bottoms to Distillate)~~	~~10^4~~
Maximum distillate effluent concentration	10 ppm boron
Rated ~~Design~~ flow	20 gpm
Cooling water flow	700 gpm (maximum)
Steam required at 50 psig	13,500 lb/hr
Code	ASME VIII

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Gas Stripper

Quantity	1
Design DF	10^3
Rated ~~Design~~ flow (process)	140 ~~200~~ gpm
Cooling water flow	700 gpm (maximum)
Steam required at 50 psig	20,000 lb/hr (maximum)
Code	ASME III, Class 3

TABLE 9.3.4-4 (Cont'd)

(Sheet 11 of 11)

PRINCIPAL COMPONENT DATA SUMMARY

Chemical Addition Package

Chemical Addition Tank:

Quantity	1
Internal volume	8 gallons (minimum)
Design pressure	Atmospheric
Design temperature	150°F
Normal operating temperature	40-90°F
Material	Austenitic stainless steel
Fluid	N_2H_4 or Li^7OH solution
Code	None

I

Chemical Addition Pump:

Quantity	1
Type	Positive displacement, variable capacity
Design pressure	3025 ~~2735~~ psig
Design temperature	150°F
Normal operating temperature	40-90°F
Capacity	0-25 gal/hr
~~Design head~~	~~2735 psig~~
Fluid	N_2H_4 or Li^7OH solution
Material in contact with fluid	Austenitic stainless steel
Code	None

TABLE 9.3.4-5

CHEMICAL AND VOLUME CONTROL SYSTEM PARAMETERS

Normal letdown and purification flow	100 gpm
Normal charging flow (to RCS)	90 gpm
Normal charging mini-recirculation flow	35 gpm
Normal seal injection flow	26 gpm
Reactor coolant pump controlled bleedoff (4 pumps)	16 gpm
Normal letdown temperature at loop	556 ~~558~~°F
Normal charging temperature at loop	445°F
Ion exchanger operating temperature	120°F

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TABLE 9.3.4-6

(Sheet 1 of 8)

CHEMICAL AND VOLUME CONTROL SYSTEM
PROCESS FLOW DATA #

I. CVCS MINIMUM PURIFICATION OPERATION (Minimum Charging and Letdown Flow)

CVCS LOCATION:	1	2	3	4	5	6	7	8
Flow* (gpm)	30	30	30	30	30	30	30	30
Press. (psig)	2235	2235	2235	~~60~~ 460	60	60	60	60
Temp. (°F)	~~558~~ 556	170	120	120	120	120	120	120

CVCS LOCATION:	8b	8c	8d	9	10	11	11b	12
Flow* (gpm)	30	30	46	46	81	46	46	46
Press. (psig)	60	50	50	50	50	~~2525~~ 2630	~~2525~~ 2630	~~2485~~ 2630
Temp. (°F)	120	120	120	120	120	120	120	335

CVCS LOCATION:	12b	12c	12d, e,f,g	12h, i,j,k	12l	12m	13a, b,c,d	13f
Flow* (gpm)	46	0	0	0	46	0	4	16
Press. (psig)	2455	2400	2400	2400	2525	2400	100	100
Temp. (°F)	335	120	120	120	120	120	180	180

CVCS LOCATION:	14
Flow* (gpm)	35
Press. (psig)	100
Temp. (°F)	~~120~~ 100

* @ 120°F

TABLE 9.3.4-6 (Cont'd)

(Sheet 2 of 8)

CHEMICAL AND VOLUME CONTROL SYSTEM
PROCESS FLOW DATA #

II. CVCS NORMAL PURIFICATION OPERATION (Normal Charging and Letdown Flow)

CVCS LOCATION:	1	2	3	4	5	6	7	8
Flow* (gpm)	100	100	100	100	100	100	100	100
Press. (psig)	2235	2235	2235	~~50~~ 460	60	60	60	60
Temp. (°F)	~~558~~ 556	290	120	120	120	120	120	120

CVCS LOCATION:	8b	8c	8d	9	10	11	11b	12
Flow* (gpm)	100	100	116	116	151	116	90	90
Press. (psig)	60	50	50	50	50	~~2500~~ 2630	~~2500~~ 2630	~~2485~~ 2630
Temp. (°F)	120	120	120	120	120	120	120	440

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CVCS LOCATION:	12b	12c	12d, e,f,g	12h,i j,k	12l	12m	13a b,c,d	13f
Flow* (gpm)	90	26	6.5	6.5	90	26	4	16
Press. (psig)	2455	2400	2400	2400	2500	2400	100	100
Temp. (°F)	440	120	125	125	120	125	180	180

CVCS LOCATION:	14
Flow* (gpm)	35
Press. (psig)	100
Temp. (°F)	~~120~~ 100

* at 120°F

TABLE 9.3.4-6 (Cont'd)

(Sheet 3 of 8)

CHEMICAL AND VOLUME CONTROL SYSTEM
PROCESS FLOW DATA #

III. CVCS MAXIMUM PURIFICATION OPERATION (Maximum Charging and Letdown Flow)

CVCS LOCATION:	1	2	3	4	5	6	7	8
Flow * (gpm)	~~170~~ 150	~~170~~ 150	~~170~~ 150	~~170~~ 150	~~170~~ 150	~~170~~ 150	~~170~~ 150	~~170~~ 150
Press. (psig)	2235	2235	2235	~~60~~ 460	60	60	60	60
Temp. (°F)	~~558~~ 556	230	120	120	120	120	120	120

CVCS LOCATION:	8b	8c	8d	9	10	11	11b	12
Flow * (gpm)	~~170~~ 150	~~170~~ 150	~~188~~ 166	~~188~~ 166	~~221~~ 201	~~188~~ 166	~~160~~ 140	~~160~~ 140
Press. (psig)	60	50	50	50	50	~~2450~~ 2630	~~2450~~ 2630	~~2435~~ 2630
Temp. (°F)	120	120	120	120	120	120	120	334

CVCS LOCATION:	12b	12c	12d, e,f,g	12h,i j,k	12l	12m	13a b,c,d	13f
Flow * (gpm)	~~160~~ 140	26	6.5	6.5	~~160~~ 140	26	4	16
Press. (psig)	2400	2400	2400	2400	2450	2400	100	100
Temp. (°F)	265	120	125	125	120	125	180	180

CVCS LOCATION:	14
Flow * (gpm)	35
Press. (psig)	100
Temp. (°F)	~~120~~ 100

* ≯ 120°F

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TABLE 9.3.4-6 (Cont'd)

(Sheet 4 of 8)

CHEMICAL AND VOLUME CONTROL SYSTEM
PROCESS FLOW DATA #

IV. CVCS MAKEUP SYSTEM OPERATION

1) Automatic Mode (Blended Boric Acid Concentration = 900 ppm)

CVCS LOCATION:	15	16	17	18	18b	18c	19
Flow* (gpm)	~~54~~ 57	~~34~~ 37	~~34~~ 37	~~34~~ 37	~~34~~ 37	0	~~34~~ 37
Press. (psig)	41	~~130~~ 155	130	130	130	0	130
Temp. (°F)	120	120	120	120	120	120	120

CVCS LOCATION:	20	21	22	23	23b	24	25
Flow* (gpm)	~~165~~ 180	~~0~~ 163	~~151~~ 143	~~151~~ 143	20	~~131~~ 143	~~165~~ 143
Press. (psig)	130	15	~~130~~ 160	130	15	130	130
Temp. (°F)	120	120	120	120	120	120	120

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* at 120 °F

TABLE 9.3.4-6 (Cont'd)

(Sheet 5 of 8)

CHEMICAL AND VOLUME CONTROL SYSTEM
PROCESS FLOW DATA #

IV. CVCS MAKEUP SYSTEM OPERATION (Cont'd)

2) Dilute Mode

CVCS LOCATION:	15	16	17	18	18b	18c	19
Flow* (gpm)	0	0	0	0	0	0	0
Press. (psig)	41	41	41	41	41	0	41
Temp. (°F)	120	120	120	120	120	120	120

CVCS LOCATION:	20	21	22	23	23b	24	25
Flow* (gpm)	~~145~~ 180	~~0~~ 200	~~145~~ 180	~~145~~ 180	20	~~145~~ 180	~~145~~ 180
Press. (psig)	130	15	~~130~~ 160	130	15	130	130
Temp. (°F)	120	120	120	120	120	120	120

3) Shutdown Boration

CVCS LOCATION:	15	16	17	18	18c
Flow* (gpm)	~~152~~ 170	~~132~~ 150	~~132~~ 150	~~132~~ 150	~~132~~ 150
Press. (psig)	41	~~130~~ 155	130	130	130
Temp. (°F)	120	120	120	120	120

* at 120°F

TABLE 9.3.4-6 (Cont'd)

(Sheet 6 of 8)

CHEMICAL AND VOLUME CONTROL SYSTEM
PROCESS FLOW DATA #

IV. CVCS MAKEUP SYSTEM OPERATION (Cont'd)

4) BAST Boration (Boric Acid Batching Operations)

CVCS LOCATION:	15	16	17	18	18b	18c	18e	18f
Flow* (gpm)	~~165~~ 180	~~145~~ 160	~~145~~ 160	~~145~~ 160	~~145~~ 160	0	~~30~~ 40	~~175~~ 200
Press. (psig)	41	130	130	130	130	0	15	15
Temp. (°F)	120	120	120	120	120	120	155	155

5) Resin Sluicing

CVCS LOCATION:	21	22	23	23b	40	41	42	43
Flow* (gpm)	~~0~~ 40	40	40	20	~~20~~ 40	50scfm	~~20~~ 40gpm +50scfm	~~20~~ 40gpm+ 50scfm
Press. (psig)	15	~~195~~ 160	~~195~~ 130	~~0~~ 15	130	~~90~~ 130	130	130
Temp. (°F)	120	120	120	120	120	120	120	120

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6) Borate Mode

CVCS LOCATION:	15	16	17	18	18c	19	20
Flow* (gpm)	120	100	100	100	0	100	100
Press. (psig)	41	~~130~~ 155	130	130	15	130	130
Temp. (°F)	120	120	120	120	120	120	120

* at 120°F

TABLE 9.3.4-6 (Cont'd)

(Sheet 7 of 8)

CHEMICAL AND VOLUME CONTROL SYSTEM
PROCESS FLOW DATA #

V. BORON RECOVERY SYSTEM

1) Processing Normal VCT Diversion

CVCS LOCATION:	32	33	34
Flow* (gpm)	100	100	100
Press. (psig)	~~60~~ 65	~~60~~ 65	~~60~~ 65
Temp. (°F)	120	120	~~120~~ 130

2) Processing Maximum VCT Diversion

CVCS LOCATION:	32	33	34
Flow* (gpm)	~~170~~ 150	~~170~~ 150	~~170~~ 150
Press. (psig)	~~60~~ 65	~~60~~ 65	~~60~~ 65
Temp. (°F)	120	120	~~120~~ 130

3) Reactor Drain Tank Processing

CVCS LOCATION:	26	27	30	31	32	33	34
Flow* (gpm)	200 gpd	50	50	50	50	50	50
Press. (psig)	3~~4~~	3~~4~~	~~85~~ 75	~~83~~ 75	~~80~~ 70	~~80~~ 70	~~30~~ 65
Temp. (°F)	120	120	120	120	120	120	~~120~~ 130

* at 120 °F

TABLE 9.3.4-6 (Cont'd)

(Sheet 8 of 8)

CHEMICAL AND VOLUME CONTROL SYSTEM
PROCESS FLOW DATA #

V. BORON RECOVERY SYSTEM (Cont'd)

4) Equipment Drain Tank Processing

CVCS LOCATION:	28	28a	29	30	31	32	33	34
Flow* (gpm)	~~150~~ 0	~~100~~ 0	50	50	50	50	50	50
Press. (psig)	3	3 ~~5~~	4	~~83~~ 66	~~83~~ 60	~~80~~ 60	~~88~~ 50	~~30~~ 45
Temp. (°F)	120	120	120	120	120	120	120	~~120~~ 130

5) Holdup Tank Processing

CVCS LOCATION:	35	36	37	38	39
Flow* (gpm)	~~50~~ 52	~~50~~ 52	~~18~~ 20	~~2.5~~ 2	~~50~~ 20
Press. (psig)	20	~~83~~ 82	~~35~~ 50	~~7~~ 40	~~35~~ 50
Temp. (°F)	120	120	120	160	120

NOTE: # Locations correspond to numbers in ellipses on Figure 9.3.4-1 (Sheets 1 through 4)

* at 120 °F

TABLE 9.3.4-7

(Sheet 1 of 2)

CHEMICAL AND VOLUME CONTROL SYSTEM LIST OF ACTIVE VALVES

Valve Number	Flow Diagram Coordinates	Valve Type	Line Size (in.)	Actuator Type
		Reference: Figure 9.3.4-1, Sheet 1 of 4		
~~CH-205~~	~~H-7~~	~~Globe~~	~~2.00~~	~~Solenoid~~
~~CH-208~~	~~G-7~~	~~Globe~~	~~2.50~~	~~Solenoid~~
CH-255	F-3	Globe	1.50	Motor
CH-431	H-6	Check	2.00	None
CH-433	G-6	Check	2.50	None
~~CH-435~~	~~G-7~~	~~Springloaded Check~~	~~2.00~~	~~None~~
CH-447	H-6	Check	2.00	None
CH-448	G-6	Check	2.50	None
CH-515	H-8	Globe	2.00	Pneumatic Diaphragm
CH-516	H-8	Globe	2.00	Pneumatic Diaphragm
CH-523	E-7	Globe	2.00	Pneumatic Diaphragm
~~CH-524~~	~~B-8~~	~~Globe~~	~~2.00~~	~~Motor~~
~~CH-839~~	~~B-7~~	~~Check~~	~~2.00~~	~~None~~
~~CH-747~~	~~F-8~~	~~Check~~	~~2.50~~	~~None~~
CH-787	H-1	Check	1.00	None
CH-802	G-1	Check	1.00	None
CH-807	F-1	Check	1.00	None
CH-812	E-1	Check	1.00	None
CH-835	F-2	Check	1.50	None
CH-866	H-1	Check	1.00	None
CH-867	G-1	Check	1.00	None
CH-868	F-1	Check	1.00	None
CH-869	E-1	Check	1.00	None
		Reference: Figure 9.3.4-1, Sheet 2 of 4		
CH-144	A-8	Hand	3.00	None
CH-154	B-7	Check	3.00	None
CH-155	A-7	Check	3.00	None
CH-164	C-6	Hand	3.00	None
CH-174	C-5	Hand	2.00	None
CH-177	B-4	Check	3.00	None
CH-190		Check	2.00	None

TABLE 9.3.4-7 (Cont'd)

(Sheet 2 of 2)

CHEMICAL AND VOLUME CONTROL SYSTEM LIST OF ACTIVE VALVES

Valve Number	Flow Diagram Coordinates	Valve Type	Line Size (in.)	Actuator Type
	Reference: Figure 9.3.4-1, Sheet 2 of 4 (Cont'd)			
CH-191		Check	2.00	None
CH-501	C-4	Gate	4.00	Motor
CH-504		Gate	4.00	Motor
CH-505	G-7	Globe	1.00	Pneumatic Diaphragm
CH-506	G-7	Globe	1.00	Pneumatic Diaphragm
CH-514	B-5	Globe	3.00	Motor
CH-534		Gate	3.00	Motor
CH-536		Gate	3.00	Motor
CH-590	D-1	Globe	2.00	Motor
CH-591	C-1	Globe	2.00	Motor
CH-705	E-2	Check	2.00	None
CH-719	C-2	Check	2.00	None
CH-750	D-1	Globe	2.00	Motor
CH-753	A-6	Hand	3.00	None
CH-754	D-1	Globe	2.00	Motor
CH-764	C-1	Globe	2.00	Motor
CH-766	C-1	Globe	2.00	Motor
	Reference: Figure 9.3.4-1, Sheet 3 of 4			
CH-494	H-7	Check	1.50	None
CH-560	D-7	Globe	3.00	Pneumatic Diaphragm
CH-561	D-7	Globe	3.00	Pneumatic Diaphragm
CH-580	H-6	Globe	1 50	Pneumatic Diaphragm

Q/R 281.37

Secondary water chemistry is based on the zero solids treatment method. This method employs the use of volatile additives to maintain system pH and to scavenge dissolved oxygen which may be present in the feedwater.

A neutralizing amine is added to establish and maintain alkaline conditions in the feedtrain. Neutralizing amines which can be used for pH control are ammonia, morpholine, and cyclohexylamine. Ammonia should be used in plants employing condensate polishing to avoid resin fouling. Although the amines are volatile and will not concentrate in the steam generator, they will reach an equilibrium level which will establish an alkaline condition in the steam generator.

Hydrazine is added to scavenge dissolved oxygen which may be present in the feedwater. Hydrazine also tends to promote the formation of a protective oxide layer on metal surfaces by keeping these layers in a reduced chemical state.

Both the pH agent and hydrazine can be injected continuously at the discharge headers of the condensate pumps or condensate demineralizer, if installed. These chemicals are added as necessary for chemistry control, and can also be added to the upper steam generator feed line when necessary.

Operating chemistry limits for secondary -side steam generator water, feedwater and condensate (as extracted from PWR Secondary Water Chemistry Guidelines, Revision 2, EPRI report NP-6239, dated December 1988) are given in Tables 10.3.5-1, 10.3.5-2 and 10.3.5-3. E

The limits stated are divided into three groups: normal, abnormal and immediate shutdown. The limits provide high quality chemistry control and yet permit operating flexibility. The normal chemistry conditions can be maintained by any plant operating with little or no condenser leakage. The abnormal steam generator limits are suggested to permit operations with minor system fault conditions until the affected component can be isolated and/or repaired. The immediate shutdown limits represent chemistry conditions at which continued operation could result in severe steam generator corrosion damage. E

The following procedures are recommended for protection against secondary system and steam generator corrosion: A

A. When the normal range is exceeded, immediate investigation of the problem should be initiated, sampling frequency increased to the abnormal level (at least twice per 8 hour shift) and blowdown increased to one percent of the main steaming rate. The problem should be corrected and the parameter(s) returned to the normal range within one week.

Q/R 281.38 & 281.40

INSERT WET LAYUP AND STARTUP DATA FROM NEXT PAGE

TABLE 10.3.5-1

OPERATING CHEMISTRY LIMITS FOR SECONDARY STEAM GENERATOR WATER

Variable	Power Operation Normal Specifications[1]	Abnormal Limits	
pH (mixed system)[2]	8.5 - ~~9.0~~ 9.2[6]		A
(copper free)	≥ 9.0 ~~- 9.5~~[5]		A
Cation Conductivity[3]	≤ 0.8 μmhos/cm	0.8-2.0 μmhos/cm	E
Silica[7]	≤ 300 ppb		E
Chloride	≤ 20 ppb	20-100 ppb	E
Sodium[4]	≤ 20 ppb	20-100 ppb	E
Sulfate	≤ 20 ppb	20-100 ppb	E

NOTES:

(1) Normal specifications are those which should be maintained by continuous steam generator blowdown during proper operation of secondary systems.

(2) A mixed system is any secondary system containing copper alloy components.

(3) If the immediate shutdown limit of 7.0 μmhos/cm is exceeded, the unit should be shut down within four hours.

(4) If the immediate shutdown limit of 500 ppb is exceeded, the unit should be shut down within four hours.

(5) In plants where condensate polishers are in operation, the pH of a copper-free system can be controlled to a value of ≥ 8.8, with action required at < 8.8. A

(6) Action required only if experience shows increased copper transport at pH > 9.2.

(7) This parameter is used for problem diagnosis.

Q/R 281.40

INSERT TO TABLE 10.3.5-1

Variable	Wet Layup	Startup (9)
pH (mixed system) (2)	$\geq$9.8	8.5 - 9.2
(copper free)	$\geq$9.8	$\geq$9.0
Cation Conductivity (3)	----	$\leq$2.0 μmhos/cm
Silica (7)	----	----
Chloride	$\leq$1000 ppb	$\leq$100 ppb
Sodium (4)	$\leq$1000 ppb	$\leq$100 ppb
Sulfate	$\leq$1000 ppb	$\leq$100 ppb
Hydrazine	75 - 200 ppm	----
N_2 (over pressure)	>5 psig	----
Dissolved Oxygen	$\leq$100 ppb (8)	$\leq$5 ppb

NOTES: (1) through (7) see prior page.
(8) Oxygen value applies to steam generator fill source.
(9) Startup values shall be met prior to exceeding 5% reactor power.

Q/R 281.39 & 281.40

TABLE 10.3.5-2

OPERATING CHEMISTRY LIMITS FOR FEEDWATER

Variable	Startup[7] Specifications	Normal[1] Specifications
pH		
a. Mixed system	---	8.8 - 9.2 (5)
b. Copper-free system	---	≥ 9.3 ~~< 9.6~~ (3)
Conductivity (Intensified cation)[4]	---	≤ 0.2 μmhos/cm
Hydrazine (6)	≥ 3 × $[O_2]$	≥ 20 ppb
Dissolved Oxygen	≤ 100 ppb[8]	≤ 5 ppb
Sodium[4]	---	≤ 3 ppb
Iron	---	≤ 20 ppb
Copper[2]	---	≤ 2 ppb

NOTES:

(1) Normal specifications are those which should be maintained during proper operation of secondary systems.

(2) Analysis not required for copper-free systems.

(3) In plants where condensate polishers are in operation, the pH of a copper-free system can be controlled to a value of ≥ 9.0, with action required at < 9.0.

(4) Conductivity and sodium are diagnostic parameters. These values were set as a means of addressing steam purity concerns. It is realized that lower values will be needed to meet blowdown limitations in Table 10.3.5-1. Feedwater sodium values of <<1 ppb are required to meet steam generator water quality. Likewise, cation conductivity values <<0.2 are generally required to meet steam generator water quality.

(5) Action required only if experience shows increased copper transport at pH>9.2.

(6) The hydrazine limit applies to feedwater/condensate downstream of the normal chemical addition point.

(7) Startup values apply when the RCS >210 °F, but reactor power is ≤5%.

(8) It may not be possible to control oxygen at this value before turbine steam seals can be established. This value should be met, however, prior to reaching 5% power.

ATTACHMENT 4

TABLE 18.5.2-3

GENERAL SYSTEM 80+ COMPONENT DATA

Component	Component Design & Operating Data CESSAR-DC (Sys 80+)	CESSAR-F (Sys 80)
Reactor Vessel		
Total core heat output (MWt)	~~3,800~~ 3914	3,800
Design pressure (psia)	2,500	2,500
Primary system pressure (psia)	2,250	2,250
RCS inlet temperature (°F)	~~558~~ 556	568
RCS outlet temperature (°F)	615	621
Design minimum RCS flow rate (gpm)	~~445,600~~ 444,650	445,600
Steam Generator		
Number of units	2	2
Primary Side		
Design pressure (psia)	2,500	2,500
Design temperature (°F)	650	650
Operating pressure (psia)	2,250	2,250
Inlet temperature (°F)	615.8	621
Outlet temperature (°F)	558	564.5
Secondary Side		
Design pressure (psia)	1,200	1,270
Design temperature (°F)	570	575
Full Load Steam Pressure (psia)	1,000	1,070
Full Load Steam Temperature (°F)	545	553
Zero Load Steam Pressure (psia)	1,100	1,170
Total Steam Flow per gen. (lb/h)	~~$8.56x10^6$~~ $8.82x10^6$	$9.59x10^6$
Full load steam quality (%)	99.75	99.75
Feedwater temperature, full power (°F)	450	450
Pressurizer		
Internal free volume (ft^3)	2,400	1,800
Design pressure (psia)	2,500	2,500
Design temperature (°F)	700	700
Operating pressure (psia)	2,250	2,250
Operating temperature (°F)	653	653
Vessel height (ft)	54	42
Volume/Power ratio	0.629	0.472
Pressurizer/RCS volume ratio	0.194	0.147

h. Calc Compensated Level (Calculated, Normally Valid, Compensated, normally PAMI display of density compensated pressurizer level. Normally continuously displayed via a digital, analog and trend display.)

3. RCS T_h (Continuously Displayed)

The left side of Figures 18.7.3-12 and 18.7.3-13 illustrates the displays for RCS T_{hot}, showing the normally displayed trend format and the associated menu page respectively. The following sensor channels and valid parameters are provided on this discrete indicator:

a. T-112HA, T-112HB, T-112HC, T-112HD (525-675°F, Loop 1 T_{hot})

b. T-111HA, T-111HB (50-750°F, PAMI Loop 1 T_{hot}) 10

c. Calc Loop 1 T_{hot} (Calculated, Normally Validated, Normally PAMI display of the average loop 1 T_{hot} in the most accurate range. Used for comparisons between loop 1 and loop 2)

d. T-122HA, T-122HB, T-122HC, T-122HD (525-675°F, Loop 2 T_{hot})

e. T-121HA, T-121HB (50-750°F, PAMI loop 2 T_{hot}) 10

f. Calc Loop 2 T_{hot} (Calculated, Normally Validated, Normally PAMI display of the average loop 2 T_{hot} in the most accurate range. Used for comparisons between loop 1 and loop 2)

g. Calc RCS T_h (Calculated, Normally Validated, Normally PAMI display of the average temperature of loop 1 and loop 2 T_{hot}. Normally continuously displayed via a digital, analog and trend display)

4. RCS T_c (Continuously Displayed)

The right side of Figures 18.7.3-12 and 18.7.3-13 illustrates the displays for RCS T_{cold}, the normally displayed trend format and the associated menu page (loop 1 menu page is shown, loop 2 T_c menu page is similar). The following sensor channels and valid parameters are provided on this discrete indicator:

a. T-112CA, T-112CC (465-615°F, loop 1A T_{cold})

b. T-111CA (50-750°F, PAMI loop 1A T_{cold}) 10

c. Calc Leg 1A T_c (Calculated, Normally Validated, Normally PAMI display of the average loop 1A T_{cold})

d. T-112CB, T-112CD (465-615°F, loop 1B T_{cold})

e. T-111CB (50-750°F, PAMI loop 1B T_{cold}) 10

f. Calc Leg 1B T_c (Calculated, Normally Validated, Normally PAMI display of the average loop 1B T_{cold})

g. T-122 CA, T-122CC (465-615°F, loop 2A T_{cold})

h. T-121CA (50-750°F, PAMI loop 2A T_{cold}) 10

i. Calc Leg 2A T_c (Calculated, Normally Validated, Normally PAMI display of the average loop 2A T_{cold})

j. T-122CB, T-122CD (465-615°F, loop 2B T_{cold})

k. T-121CB (50-750°F, PAMI loop 2B T_{cold}) 10

l. Calc Leg 2B T_c (Calculated, Normally Validated, Normally PAMI display of the average loop 2B T_{cold})

m. Calc Loop 1 T_c (Calculated, Normally Validated, Normally PAMI display of the average leg 1A and leg 1B T_{cold}. Used for comparisons between loop 1 and loop 2 T_{cold})

n. Calc Loop 2 T_c (Calculated, Normally Validated, Normally PAMI display of the average leg 2A and leg 2B T_{cold}. Used for comparisons between loop 1 and loop 2 T_{cold})

o. Calc RCS T_c (Calculated, Normally Validated, Normally PAMI display of the average loop 1 and loop 2 T_{cold}. Normally continuously displayed via a digital, analog and trend display)

NOTE: The "Loop 1" and "Loop 2" touch selections, located beneath the "menu" label, on Figure 18.7.3-13, selects which loop data (1 or 2) is presently being displayed. The figure illustrates the Loop 1 case.

2. Cooling System

Figure 18.7.3-22 illustrates the cooling system menu for RCP 1A (other RCP cooling system menus are similar).

a. HP Cooler Inlet Temperature (T-150, 160, 170, 180)

b. HP Cooler Outlet Temperature (T-151, 161, 171, 181)

c. RCP Essential Cooling Water Flow (F-471, 474, 475, 477)

d. RCP Essential Cooling Water Outlet Temperature (T-471, 472, 473, 474)

3. Pump/Motor

Figure 18.7.3-23 illustrates the Pump/Motor menu page for RCP 1A (other RCP pump/motor menus are similar).

a. Motor Current (RCP-1A, 1B, 2A, 2B)

b. Motor Lower Journal Bearing Temperature (T-116, 126, 136, 146)

c. Motor Lower Thrust Bearing Temperature (T-154, 164, 174, 184)

d. Motor Upper Journal Bearing Temperature (T-194, 195, 196, 197)

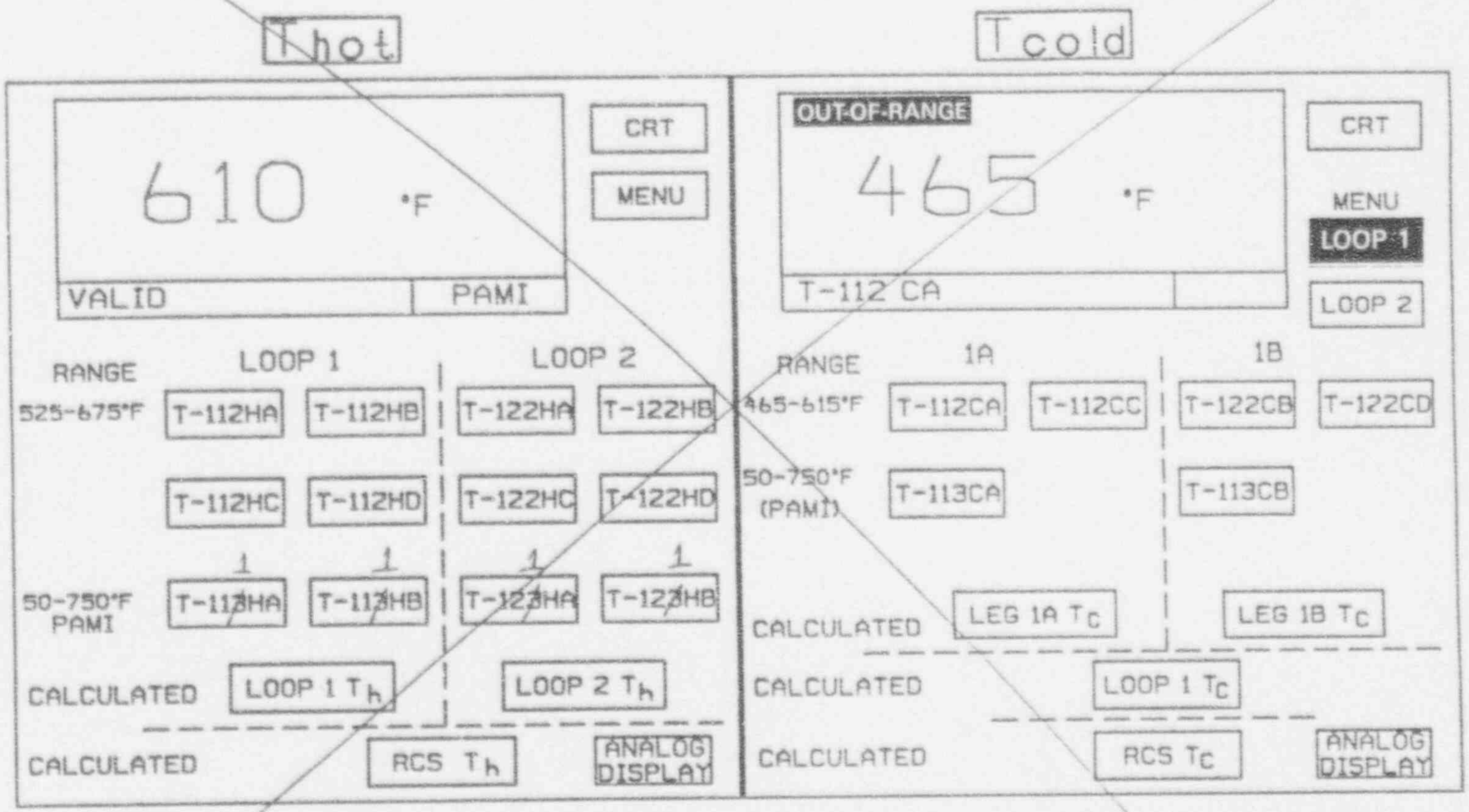
e. Motor Stator Temperature (T-155, 165, 175, 185)

f. Motor ~~Anti-Reverse~~ Forward and Reverse Rotation ~~Device~~ Switch ~~Temperature (T-119, 129, 139, 149)~~ (0-109, 119, 129, 139) 0

g. Pump Lower Journal Bearing Temperature (T-152, 162, 172, 182)

h. Pump Upper Journal Bearing Temperature (T-153, 163, 173, 183)

i. Pump Upper Thrust Bearing Temperatures (T-156, 166, 176, 186)



NPS-PANELFRONTS(SHT12)

Replace w/ Attached Figure

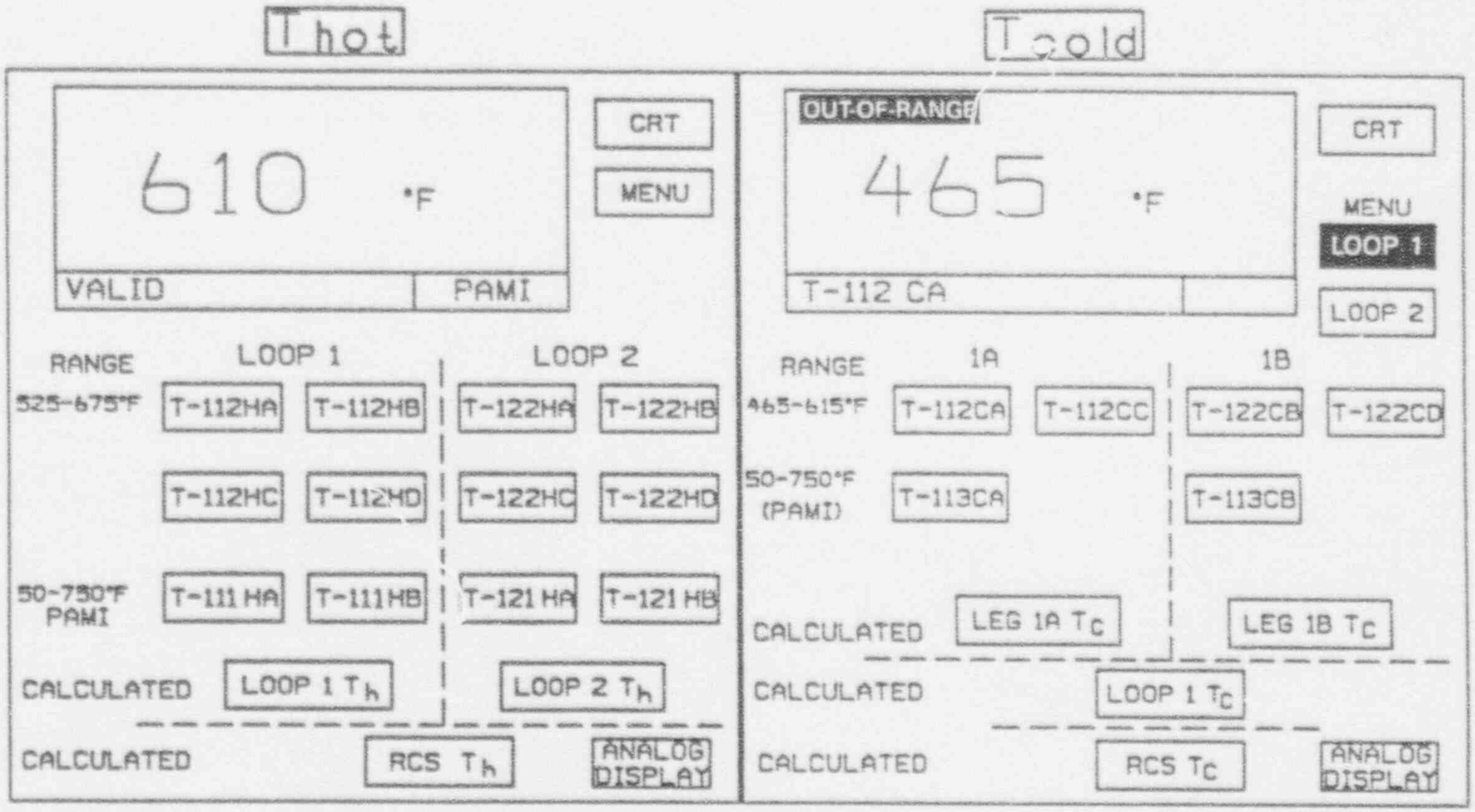
SYSTEM 80+™

T_{HOT} AND T_{COLD} MENU PAGES FOR DIAS

Figure 18.7.3-13

Amendment E
December 30, 1988

SYSTEM 80+™

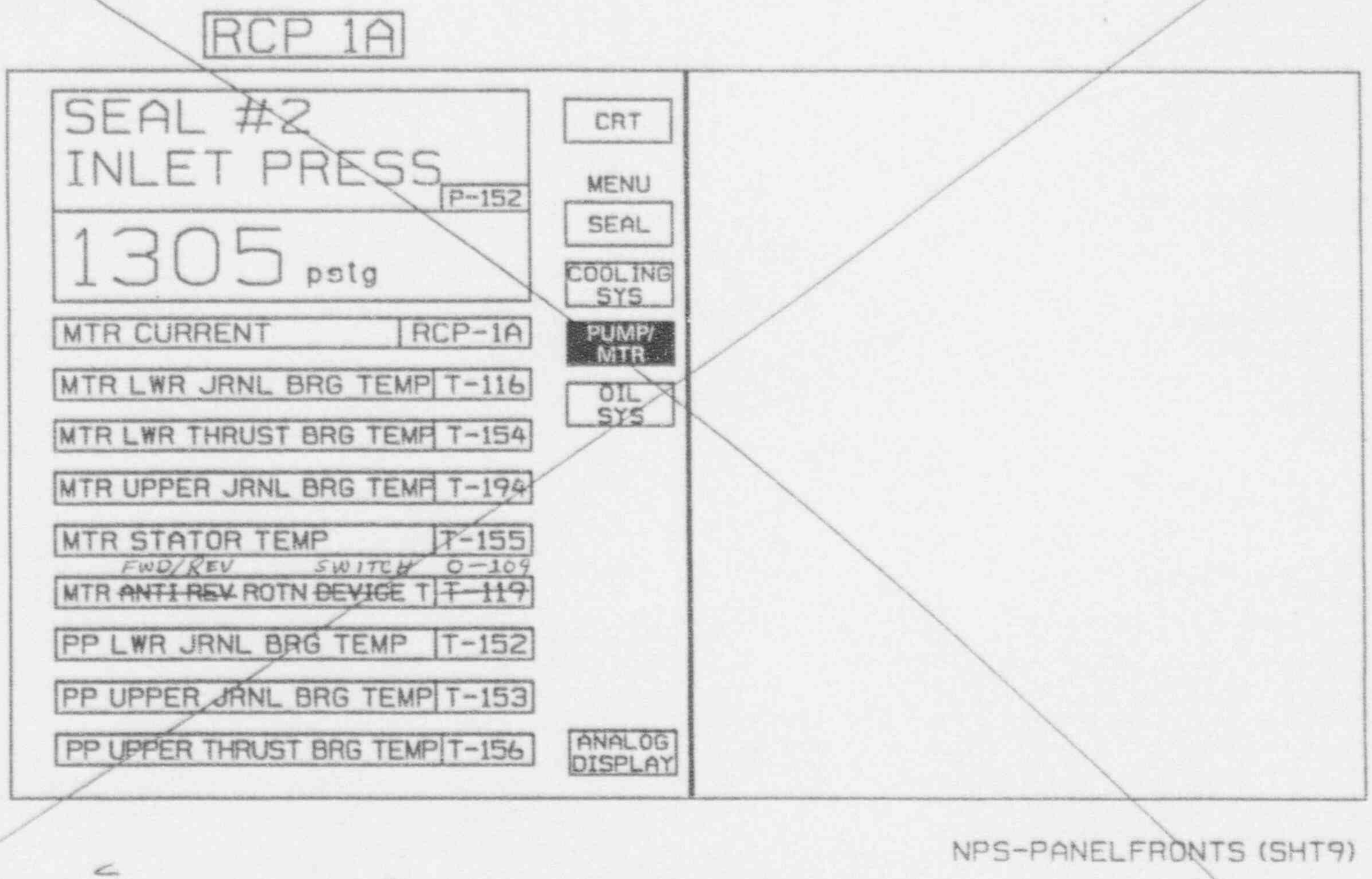


NPS-PANELFRONTS(SHT12)

T_{HOT} AND T_{COLD} MENU PAGES FOR DIAS

Figure 18.7.3-13

Amendment E
December 30, 1988



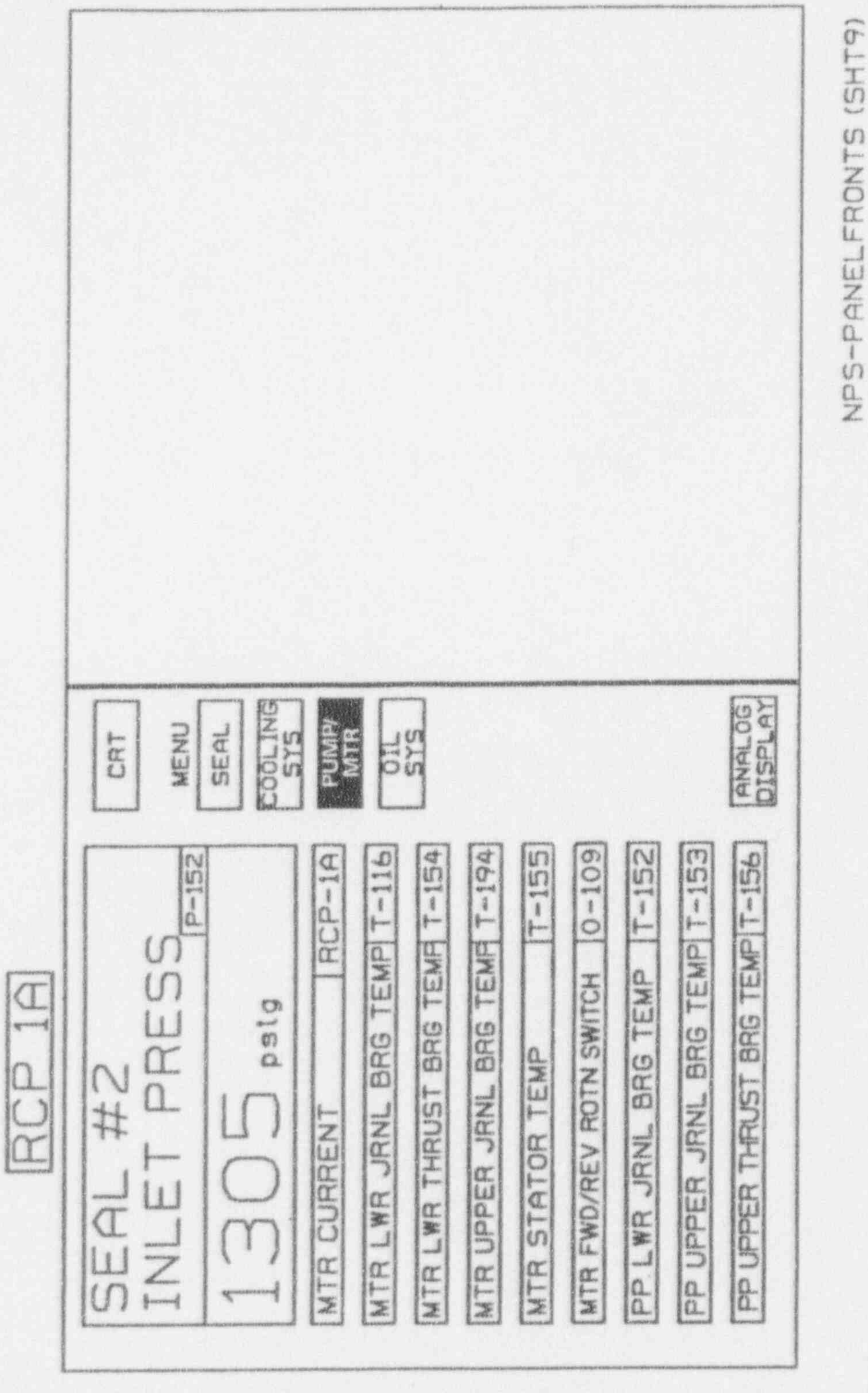
Replace w/ Attached Figure

SYSTEM 80+™

RCP 1A PUMP/MOTOR MENU PAGE

Figure 18.7.3-23

Amendment E
December 30, 1988



Change Date & Amendment # to O

Amendment E
December 30, 1988

SYSTEM 80+™	RCP 1A PUMP/MOTOR MENU PAGE	Figure 18.7.3-23

ATTACHMENT 5

CESSAR-DC, Section 19.10

10.0 SENSITIVITY ANALYSES - LEVEL I

As discussed in section 2.7, PRAs include a number of data analysis and modelling assumptions. A set of eleven sensitivity analyses were run to evaluate the potential impact of modelling and data analysis assumptions for the System 80+ PRA. The initial quantification of core damage frequency was performed in four steps:

1) The core damage cutsets were generated for each core damage sequence.

2) Recovery analysis was performed for each core damage sequence and the cutsets for each core damage sequence were requantified.

3) All core damage sequence cutsets associated with each initiating event were combined into initiating event core damage cutset files. Each initiating event core damage cutset file was quantified to generate an overall core damage frequency for each initiator.

4) The initiating event core damage cutset files were combined to form a Total Core Damage Cutset File. The cutsets in this file were requantified to generate the total core damage frequency.

Several of the sensitivity analyses were performed by requantifying the Total Core Damage Cutset File with temporary data changes in the data base files. The following paragraphs describe the individual sensitivity analyses and their results. Table 10-1 summarizes the results of the sensitivity analyses.

10.1 OPERATOR ERROR RATE - GENERAL

The System 80+ fault trees and event trees include basic events representing failure of the operators to perform specific tasks. The probability that the operator failed to perform the specified tasks was determined using the SHARP[39]

methodology as described in section 2.5 and 5.5. There is uncertainty in the quantification of Human Error Probabilities (HEPs) because of the various assumptions that are made in the calculation process. Previous PRAs have shown that the core damage frequency can be sensitivity to the HEPs used in the analysis. A sensitivity analysis was performed to determine the potential impact on the System 80+ total core damage frequency. All operator error rates presented in table 5-5 and the non-recovery probability for recovery action, RCVRMOV, in table 5-7 were increased by one order of magnitude and the Total Core Damage Cutset File was requantified. The total core damage frequency increased from 1.66E-06/yr to 9.82E-06/yr, a factor of 5.92 increase in total core damage frequency. This indicates that the System 80+ PRA results are somewhat sensitive to human error probabilities. Table 10-2 presents the dominant cutsets (all cutsets above 1.0E-06) for this sensitivity analysis. As can be seen from these cutsets, nine of the fourteen cutsets involve failure of long-term cooling and include two or three operator errors.

10.2 OPERATOR ERROR RATE - CONTROL ROOM RESPONSE

In general, the operator errors included directly in the fault tree models or added as non-recovery actions can be classified as errors committed before the initiating event (pre-existing maintenance errors), failure to perform actions in the control room, and failure to perform local actions (actions outside of the control room. A question has arisen as to the impact on core damage frequency if only operator actions that can be performed from the control room are credited. A sensitivity analysis was performed to evaluate this issue. The failure probability for all @HFD@@@@@@@@@@@ (post-trip local action) components in table 5-5 were set to 1.0 with an error factor of 1.0 and the non-recovery probability for RCVRMOV in table 5-7 was set to 1.0 with an error factor of 1.0. The Total Core Damage Cutset File was then requantified. The core damage frequency changed from 1.66E-06/yr to 1.8E-04/yr, a factor of 108 increase in core damage. This indicates that the System 80+ PRA results are sensitive to the assumption that the operators can and will perform local actions. Table 10-3 presents the dominant cutsets (all cutsets above 1.0E-06) for this sensitivity analysis. As can be seen from this table, 27 of the 29 cutsets involve failure

of long-term cooling. For most of these cutsets, the main operator error involving failure to perform a local action is AHFDCST, operator fails to align CST to EFW storage tanks. In these scenarios, there is more than adequate time for the control room operators to dispatch equipment operators to align the proper valves. Therefore, it is considered realistic to assume that this type of local action would be performed.

10.3 MOTOR-OPERATED VALVE FAILURE RATE

A large number of motor-operated valves are used in the safety related systems in a nuclear power plant. In general, these valves must change position to perform their safety related function. In the past, there has been some concern that the failure rates for motor-operated valves have been under-estimated. A sensitivity analysis was performed to evaluate the impact of increasing all motor-operated valve (MOV) failure rates by one order of magnitude. The base MOV failure rates in the .TC file (see table 5-1) were increased by one order of magnitude. This included the base failure rates for VMA, VMB, VMD, VMF, VMQ, and VMT. In addition, the common cause failure rate for all MOV related common cause failures as presented in table 5-3 were increased by one order of magnitude. The Total Core Damage Cutset File was then requantified. The core damage frequency increased from 1.66E-06 to 5.72E-06, a factor of 3.4 increase in core damage frequency. This indicates that the System 80+ core damage frequency is not highly sensitive to the MOV failure rates. Table 10-4 presents the dominant cutsets (all cutsets above 1.0E-7) for this sensitivity analysis. Eleven of the thirty cutsets involve common cause MOV failures and twelve of the cutsets involve multiple independent MOV failures. The remaining 7 cutsets did not involve MOV failures.

10.4 SITS FOR MEDIUM LOCAS

For the System 80+ PRA, a best-estimate thermo-dynamic analysis was performed to confirm the thermo-dynamic analyst's belief that Safety Injection Tank (SIT) injection was not needed to prevent core damage for medium LOCAs, LOCAs with an effective break size between 0.50 ft^2 and 0.03 ft^2. In previous PRAs, it was assumed that three of the four SITs had to inject to prevent core damage for both large and medium LOCAs and the PRA models were constructed accordingly. In the System 80+ PRA, SIT failure was included in the large LOCA models (see section 4.1) but was not included in the medium LOCA models (see section 4.2). A sensitivity analysis was performed to determine the impact of assuming SIT injection was required to prevent core damage for medium LOCA. This involved effectively incorporating the large LOCA SIT failure model in the medium LOCA models. This was accomplished by taking the cutsets for the Large LOCA core damage sequence LL-4 (see table 9.2.1-4), changing the initiator from large LOCA to the two medium LOCA initiators and inserting the new cutsets into the Total Core Damage Cutset File. Four cutsets were added to the file. They are:

1) MLOCA1 * HVCXS-SET1
2) MLOCA1 * LVCXD-SET1
3) MLOCA2 * HVCXD-SET1
4) MLOCA2 * LVCDX-SET1

The Total Core Damage Frequency Cutset File was then requantified. The total core damage frequency changed from 1.66E-06 to 1.67E-06, a nominal change. This indicates that the System 80+ core damage frequency is not sensitive to the assumption that SIT injection is not needed to prevent core damage for medium LOCAs.

10-5 AGGRESSIVE SECONDARY COOLDOWN NOT FEASIBLE

For small LOCAs and Steam Generator Tube Ruptures (SGTR), it was assumed that if safety injection was not available for inventory control, the RCS could be

depressurized via a rapid cooldown using the secondary side and the Shutdown Cooling Pumps aligned to provide injection for RCS inventory control. This assumption was based on analyses for System 80 plants documented in CEN-239[9,10,11]. A confirmatory analysis was performed to demonstrate that these analyses are valid for System 80+. The question still remains of the potential impact on the core damage frequency if aggressive secondary cooldown for SCS injection were considered not feasible. A sensitivity analysis was performed to evaluate the impact on core damage frequency of assuming that Aggressive Secondary Cooldown for SCS injection was not feasible. This was accomplished by deleting the cutsets for Small LOCA sequences SL-9 and SL-10 (see tables 9.2.3-5 and 9.2.3-6) and SGTR sequences SGTR-15 and SGTR-16 (see tables 9.2.4-9 and 9.2.4-10) from the Total Core Damage Cutset File, setting the probability that the operator fails to perform the aggressive secondary cooldown to 1.0G (AHFFASCSLOCA = 1.0 and AHHFASCSGTR = 1.0) and requantifying the Total Core Damage Cutset File. The total core damage frequency changed from 1.66E-06 to 7.7E-06, a factor of 4.6 increase in the core damage frequency. This indicates that the System 80+ PRA results are not highly sensitive to the assumption that aggressive secondary cooldown for SCS injection is feasible for inventory control for small LOCAs and SGTRs. Table 10-5 presents all cutsets with a frequency of greater than 1.0E-8 for this sensitivity analysis. The top four cutsets in this table involve failure of safety injection following a small LOCA or SGTR and no aggressive secondary cooldown for SCS injection. The safety injection system failures in all four cases are due to common cause failure of various valves.

10.6 RCP SEAL FAILURE ON STATION BLACKOUT

With loss of all station AC power (Station Blackout), RCP seal cooling water will be lost. The NRC has postulated in their evaluation of Station Blackout[57] that under these conditions, the seals will begin to degrade and gross seal leakage on the order of several hundred gpm may occur. The CEOG contends that this is not credible for pumps used in C-E plants[58]. System 80+ uses CE-KSB pumps. The CE-KSB pumps use a system of 3 hydrodynamic seals to seal the pump shaft. (These seals are similar to the Byron Jackson RCP seals except that the Byron Jackson

seals use four stages.) Hydrodynamic seals are less subject to leakage than are the hydrostatic seals used by other RCP manufacturers. The first two seal stages derease the pressure by about 84%, and the third stage decreases it by 16%. Each of the three seal stages is capable of operating at full system pressure.

Several tests have been performed to address the capability of Byron Jackson seal assemblies to maintain integrity and limit seal leakage under loss of seal cooling/station blackout conditions. A simulated station blackout test was run on a prototype seal cartridge for one utility. This test was run for more than 50 hours at steady state SBO conditions: no shaft rotation, no cooling, and plant operating temperature and pressure. Seal controlled leakage remained within normal limits (approximately 1 GPM) for the entire period. Another significant test was a thirty minute loss of cooling water test. In this case, the RCP shaft was rotating, a more severe condition than for SBO. During this test, the seal maintained its function, and the maximum controlled leakage was 2 GPM.

Thus, based on the robust seal design, and the results of the tests and operating experience, CE has asserted that the System 80+ RCP seals can cope with SBO Conditions without developing significant seal leakage. However, a sensitivity analysis was performed to evaluate the potential impact on core damage frequency if there is a finite probability that the System 80+ RCP seals will fail following a Station BlackOut (SBO) or Loss of Component Cooling Water initiating events.

The impact on core damage frequency of RCP seal failure following an SBO can be estimated by creating a new core damage sequence, SBORCP, generating the cutsets, and quantifying the cutsets to get an incremental core damage frequency that can be added to the overall core damage frequency. The new sequence can be expressed as:

SBORCP = (Station Blackout Occurs) AND [10-1]
(RCP seals fail given that the SBO has occurred) AND
(Offsite power not recovered in time to prevent core uncovery and damage)

An SBO is defined to be a Loss Of Offsite Power (LOOP) with demand failure of both diesel generators and failure of the Standby AC Source. If, following an SBO, RCP failures occur, core damage can be prevented if offsite power is restored and the injection pumps started before core is uncovered. The available time to recover offsite power is a function of the RCP seal leak rate. For this sensitivity analysis, it was assumed that the RCP seal leak rate was such that the time available to recover offsite power before the core would become uncovered was one hour. The fault tree presented in figure 10-1 was used to model the new sequence, SBORCP.

The initiating event, Loss of Component Cooling Water, will result in the loss of cooling water flow to the seals of two of the four RCPs. At the same time, cooling water flow to the charging pumps may also be lost. This would result in loss of seal injection flow to the RCPs also. In January, 1992, ABB-CE committed to install a backup seal injection system to provide additional protection for the RCP seals given loss of seal cooling and seal injection. This backup system will consist of a positive displacement pump that will provide seal inject flow via the existing seal injection lines in the CVCS system. In order to provide seal protection for loss of CCW, this pump will be self cooled. The pump driver will be powered from one of the four 4.16 Kv vital buses which can be powered by offsite power, the emergency diesel generators or the standby AC source. This pump will be manually loaded to an available 4.16 Kv bus when needed.

The potential impact of an RCP seal failure following loss of component cooling water on core damage frequency can be estimated by creating a new core damage sequence, LCCWRCP, generating the cutsets, and quantifying the cutsets to get an incremental core damage frequency that can be added to the overall core damage frequency. The new sequence can be expressed as:

LCCWRCP = (Loss of Component Cooling Water Occurs) AND (Backup Seal Injection System Fails) AND (RCP seals fail given that the loss of component cooling water has occurred) AND (Safety Injection System fails to provide inventory control)

The fault tree presented in figure 10-2 was used to represent this sequence.

The data for all basic events In the above models, with the exception of RCPSEAL4, RCPSEAL2, and RCPSIBU, is presented in the data tables in chapter 5. The calculations of the values for RCPSEAL4, RCPSEAL2, and RCPSIBU are presented in the following paragraphs.

ABB-CE plants are designed with Reactor Coolant Pumps (RCPs) with multiple stages, in series, each able to hold the full primary pressure. The earlier plants have Byron Jackson pumps and seals with 4 stages in a replaceable seal cartridge. The three newer plants have KSB pumps using seals with 3 stages. Four plants are using Bingham seals (4 stages) on BJ pumps. All of the seal designs have the characteristic that each stage is able to hold the full primary pressure. Reference 73 summarizes the pumps, seals and operating experience for ABB-CE reactors. In the analysis presented here, the full assembly of the three or four stages will be referred to as the seal and the individual sections are called the stages.

ABB-CE reactors have accumulated 190 reactor years of operating experience to date. There has never been a seal induced LOCA at any of these plants for any reason, including loss of seal cooling.

In the April 19, 1991 Federal Register Notice, the NRC announced the release of their proposed resolution to Generic Issue 23 in Draft Regulatory Guide DG-1008 and supporting documents. In response to that draft, the CEOG had ABB-CE survey the CE reactor owners and document their RCP seal performance in CEN-408[73]. In addition, earlier seal events were identified from references 74 through 87. These events are both from reactor operations and planned tests. Table 10-6 summarizes the events. The first two columns give the dates and plant. Tests are noted with a "T". The number of pumps or seals are noted in the next column. The fourth column gives the number of stages that were exposed to loss of cooling. The fifth column gives the number of leaks reported. All but one leak with reported leakage rates were 3 gpm or less. The only exception was the 8/1/88 event where a 20 gpm leak was observed and two stages were reported

damaged in a single seal. A leakage of 20 gpm is still well below the maximum capacity of the charging pumps for all the ABB-CE plants and therefore not a LOCA. Even under SBO conditions, this rate of coolant loss would not lead to core uncovery until well over 24 hours which is well over the NRC mandated "coping time" of 4 hours. This one event was considered as a common cause failure and used in calculating β. In some cases, the pumps were kept running for a period after cooling loss. These conditions are considered more severe than SBO conditions because more heat is imparted into the seal. No events (with or without seal cooling) at ABB-CE reactors have been seal LOCAs.

Details of the exact type of damage to the stages was lacking in most event descriptions. In normal operation the seals leak slightly by design (about 1 gpm bleed-off). This bleed-off is necessary for lubrication. In most events, no increase in bleed-off was observed. Where there was an increase in leakage, most reported leakages were 3 gpm or less. In a few events it was reported that a stage was damaged.

The approach taken in this analysis was to first calculate the conditional probability of a stage failure given loss of cooling and then calculate the probability of having all three or four stages in a single seal fail by using the Multiple Greek Letter Method[68]. First, optimistic and pessimistic stage failure rates were estimated. The optimistic failure rate was based on the assumption that the only stage failures were the events where a stage failure was reported. The pessimistic failure rate was estimated assuming that any increase in leakage was caused by a stage failure. The median value was then calculated by assuming a log-normal failure probability distribution and the optimistic failure probability represented the 5% fractile and the pessimistic estimate represented the 95% value.

Having calculated the probability that a single stage would fail, it is now possible by using the multiple greek letter method of common cause failure analysis, to calculate the probability that 3 or 4 consecutive stages in a complete seal would fail simultaneously. The equation for the probability of a complete seal assembly failing is:

$$Q = \beta * \gamma * \delta * P_{stage} \qquad [10\text{-}2]$$

where: Q = probability per demand of total seal failure given loss of cooling.

β = conditional probability that the common cause of a component failure will be shared by one or more additional components.

γ = conditional probability that the common cause of a component failure that is shared by one or more components will be shared by two or more components additional to the first.

δ = conditional probability that the common cause of a component failure that is shared by two or more components will be shared by three or more components additional to the first.

and P_{stage} = total failure probability of a stage due to all independent and common cause events.

There was one event where two stages were reported as failed. This permits the calculation of β. Since no events reported the loss of three or more stages, the values used for γ and δ will be recommended values[7] of 0.5 and 0.9 respectively. For a three stage seal, δ = 1.0.

The optimistic estimate of the stage failure probability is based on reported stage failures and assigned a 5% fractile in a probability distribution. From column 6 of Table 10-6 there were 6 reported stage failures in 245 stage exposures or 2.45E-2/d. The 95% fractile was based on the assumption that any leakage increase was a stage failure (column 4 of Table 1) and is 15 stages in 245 stage exposures or 6.12E-2/d. Assuming that the failure probability distribution is log normal, the median value is the square root of the product

of these two extremes and is equal to P_{stage} = 3.87E-2/d.

β is the ratio of stages that failed in a common mode to the total number of failed stages. With the 8/1/88 event being the only common mode failure, β = 2/(3.87E-2 * 245) = 0.211.

The equation for the probability , Q, of a complete seal assembly to fail given loss of cooling is:

$$Q = \beta * \gamma * \delta * P_{stage} \qquad [10\text{-}3]$$

where: Q = failure probability per demand of RCP seal given loss of cooling

β = 0.211

γ = 0.5

δ = 0.9 for four stages
1.0 for three stages

and P_{stage} = 3.87E-3/d

This results in:

$Q_{4\ stages}$ = 3.7E-3/d

$Q_{3\ stages}$ = 4.1E-3/d

For a Station Blackout event, all four RCP seals are exposed. Therefore the probability of a seal LOCA is four times that for a single RCP. Thus:

RCPSEAL4 = 4 * 4.1E-03 = 1.64E-02/demand.

An error factor of 3.0 was assigned to this value.

For a Loss of One Division of Component Cooling Water, only two RCP seals are exposed. Therefore, the probability of an RCP seal LOCA is two times the base RCP seal failure probability. Thus:

$$RCPSEAL2 = 2 * 4.1E\text{-}03 = 8.2E\text{-}03/\text{demand}.$$

An error factor of 3.0 was assigned to this value.

As stated above, this backup system will consist of a positive displacement pump that will provide seal inject flow via the existing seal injection lines in the CVCS system. In order to provide seal protection for loss of CCW, this pump will be self cooled. The pump driver will be powered from one of the four 4.16 Kv vital buses which can be powered by offsite power, the emergency diesel generators or the standby AC source.

The detailed design for the standby seal injection system has not yet been developed. However, a system unavailability can be estimated based on the above information. The system will have one positive displacement pump. For the purposes of this analysis, it is assumed that a motor driven pump will be used. The following information was extracted from the KAG[7]:

P(f) = 2.5E-05/hour for a motor driven pump.
Q = 2.0E-03/demand

Assuming a mission time of 24 hours, the system unavailability contribution due to the pump is:

$$U_p = 2.0E\text{-}3 + 24 * 2.5E\text{-}05 = 2.6E\text{-}03.$$

It is also assumed that the system will have four motor-operated valves that will have to open. It is assumed that these valves will be powered from the 125 VDC vital buses. Based on data in the KAG[7], the unavailability contribution attributable to the valves not opening would be:

$$Q_v = 4 * 4.0E\text{-}03/\text{demand} = 1.6E\text{-}02$$

Thus the total estimated unavailability of the system is:

$$U_s = U_p + Q_v = 2.6E\text{-}03 + 1.6E\text{-}02 = 1.9E\text{-}02$$

For the purposes of this analysis, this value was rounded up to 5.0E-02 to account for operator errors. Thus:

RCPSIBU = 5.0E-02

An error factor of 5.0 was assigned to this value.

The fault trees for SBORCP and LCCWRCP were quantified using the data above and the base failure data from section 5. The resultant core damage frequency for SBORCP was 3.59E-09/year. The resultant core damage frequency for LCCWRCP was calculated to be 1.04E-09. Thus, the calculated increase in core damage frequency attributable to RCP Seal LOCAs would be 4.63E-09. This would be an increase of less than 1%. Thus, the System 80+ PRA core damage frequency results are not sensitive to the assertion that the RCP seals will not fail following an SBO. Additional analyses were performed to further evaluate the impact on core damage frequency of variations in the assumed conditional probability of RCP seal failure on SBO. The values used for the conditional RCP seal failure rate in these analyses included the 5th percentile values for RCPSEAL4 and RCPSEAL2, the 95th percentile values and an order of magnitude increase for the mean values. In addition, the impact of the conditional RCP seal failure rate value of 0.6 used in NUREG-1150 was evaluated. The 0.6 value is applicable only to the Westinghouse seal design which is known to have RCP seal leak problems on station blackout. The ~~Westinghouse~~ RCP seal design evaluated in NUREG-1150 is significantly different from the CE-KSB seal design

As shown on figures 10-3 and 10-4, Station Blackouts with an assumed RCP Seal LOCA do not cause a significant increase in the total System 80+ core damage frequency over the entire range of evaluated conditional seal failure

probabilities. Even using the conditional seal failure probability for Westinghouse seals would result in only a 10% increase in total core damage frequency. Thus, the System 80+ core damage frequency is not sensitive to the assertion that the RCP seals will not fail as a result of a station blackout. These sequences would also be expected to have little impact on the containment performance analyses because they are similar in nature to other sequences that have been evaluated and SBORCP and LCCWRCP have low frequencies (less than 1% of the total). For example, LCCWRCP would map into PDS 167 which has a total frequency of 5.2E-08 (see table 12.1-7).

10.7 COMPONENTS UNAVAILABLE DUE TO MAINTENANCE

Components in safety related systems are periodically tested per technical specification requirements. In some cases, the components may be unable to perform their safety related function during the test. In addition, if the component is found to be failed during the test, it is taken out of service for maintenance. While the component is out of service for maintenance, it is unable to perform its safety related function. Component unavailability due to test and maintenance was included in the System 80+ PRA models. The basic events for component unavailability due to test and maintenance are presented in table 5-4. A sensitivity analysis was performed to evaluate the impact on core damage frequency if it was assumed that all components were able to perform their safety related function while in test and that no maintenance was performed on safety related equipment while the plant was at power. This was done by setting the failure rate for all basic events in table 5-4 to 0.0 and re-quantifying the Total Core Damage Cutset File. The total core damage frequency did not change from the base value of 1.7E-06/yr. This indicates that the results of the System 80+ PRA would not improve if there was no testing or maintenance unavailability for safety related components.

10.8 ADVERSE MTC

An ATWS is an event in which an anticipated transient occurs but the reactor is not shutdown by automatic insertion of the control rods. One factor that

influences the progression of an ATWS event is the Moderator Temperature Coefficient (MTC). If the MTC is more positive than a calculated critical value, the peak RCS pressure will exceed the level C stress limit pressure and a non-mitigatable LOCA is assumed to occur (see section 4.13.1). For System 80+, the critical MTC was calculated to be -0.30E-4 $\Delta\rho$/°F. For MTC values more positive than -0.30E-4 $\Delta\rho$/°F, the peak RCS pressure will exceed the level C stress limit pressure if less than tree PSVs open. For System 80+, it has been determined that the MTC value should be less than -0.30E-4 $\Delta\rho$/°F for 99% of the core life (see figure 5-2, page 5-68). The dominant ATWS core damage sequence is (ATWS occurs) AND (MTC is adverse). A sensitivity analysis was run to evaluate the impact on core damage frequency if the MTC was found to be adverse over a larger fraction of the core life. For this sensitivity analysis, the value for SE-MTC, adverse MTC, in the CAFTA data base was increased from 0.01 to 0.1, and the Total Core Damage Cutset File was requantified. The total core damage frequency increased from 1.66E-6/year to 2.09E-6 per year, a factor of 1.3 increase in core damage frequency. This indicates that the overall System 80+ core damage frequency is not very sensitive to the adverse MTC probability.

10.9 LOSS OF OFFSITE POWER FREQUENCY

For the System 80+ PRA, the core damage frequency contribution attributable to events initiated by a Loss Of Offsite Power (LOOP) was calculated to be 2.0E-08/year. This represents only 1.3% of the total core damage frequency for internal events. In past PRAs, the core damage frequency attributable to LOOP has been greater in both absolute value and relative contribution to the total. There are a number of reasons for the reduction in the core damage frequency contribution for LOOP for System 80+. These include the capability of the main turbine/generator to runback and pickup hotel load on loss of offsite power, two separate switchyards for incoming power, a four train EFW system with two 100% capacity turbine driven pumps, 6 vital batteries which provide an 8 hour coping capability and a standby combustion turbine which can backup the diesels. Based on the first three features described in the preceding sentence, the Loss Of Offsite Power (LOOP) initiating event was defined as a loss of site power which required the startup and loading of the emergency diesel generators. The LOOP

frequency calculated for System 80+ (see section 3.3.7, page 3-43) is 5.0E-3 per year. This is almost an order of magnitude lower than a LOOP frequency based purely on the failure of the grid. This value, as presented in the KAG[7], is 3.5E-2/year. A sensitivity analysis was performed to determine the impact on the overall core damage frequency if the LOOP frequency was increased by an order of magnitude. The value for the event, LOOP, was increased from 5.0E-3/year to 5.0E-2/year in the CAFTA database. The Total Core Damage Cutset File was then requantified. The total core damage frequency increased from 1.66E-6 to 1.96E-6, an 18% increase in the total core damage frequency. This indicates that the System 80+ overall core damage frequency is not highly sensitive to the LOOP frequency. A hand calculation indicated that the LOOP core damage frequency would increase from 2.0E-8 to 3.2E-7. A core damage frequency contribution of 3.7E-7 is 11% of the new total core damage frequency.

An additional LOOP sensitivity analysis was performed to evaluate the effect of changing the base Loss of Grid frequency from 0.035/year to 0.15/year. The effective System 80+ LOOP frequency for this case was 2.32E-2/year as opposed to the base case frequency of 5.0E-3/year. For this case, the total core damage frequency increased from 1.66E-06/year to 1.78E-06/year, a change of only 7%. This is not a significant increase.

10.10 OTHER SENSITIVITY ANALYSES.

10.10.1 Vessel Rupture

Vessel rupture was originally evaluated in WASH 1400. It is typically defined as a rupture of the vessel or a large LOCA in excess of the ECCS capabilities. Vessel rupture is assumed to directly lead to core damage. This event and its initiating frequency have essentially been accepted as is since WASH 1400 because it has little impact on the overall core damage frequency for existing plants. However, it contributes approximately 6% of the total core damage frequency. With current materials and current manufacturing methods, it has been questioned as to whether or not vessel rupture is a credible event for an ALWR. A

sensitivity analysis was performed to evaluate the impact on plant core damage frequency if vessel rupture was assumed not to be credible. This was accomplished by subtracting the Vessel rupture core damage frequency contribution from the total plant core damage frequency. As expected, the plant core damage frequency decreased from 1.66E-06/year to 1.56E-06/year, a decrease of about 6%.

10.10.2 Common Cause Failures

As discussed in section 9.4, the System 80+ plant core damage frequency is dominated by common cause failures. It has been contended that with complete divisional separation, improved staff training, improved maintenance techniques and proper selection of components, the potential for common mode failure can be essentially eliminated. A sensitivity analysis was performed to evaluate the impact on plant core damage frequency of the assumption that all common mode failures except for diesels and batteries were eliminated. This analysis was performed by setting all common mode failure rates except for those for diesel generators and batteries to 0.0 and requantifying the Total Core Damage Cutset File. The plant core damage frequency decreased by essentially an order of magnitude, from 1.66E-06/year to 2.15E-07/year. This is a significant decrease. Combining the assumption that vessel rupture is not credible and that common mode failure of equipment other than the diesel generators and batteries are not credible results in a total core damage frequency of 1.15E-7/year. This is a factor of 14.4 decrease from the base core damage frequency. The combined impact of these two assumptions on plant core damage frequency is significant.

Table 10-1
SUMMARY OF SYSTEM 80+ PRA SENSITIVITY ANALYSIS RESULTS

CASE	DESCRIPTION	MODELED AS	OLD CDF	NEW CDF	CHANGE FACTOR
1	Operator Error Rate - General	Increase all operator error rates by factor of 10	1.66E-06	9.82E-06	5.9
2	Operator Error Rate - Control Room Response	Set operator error rate for all actions performed outside of the control room to 1.0	1.66E-06	1.8E-04	108.4
3	Motor-Operated Valve Failure Rate	Increase generic failure rates for all MOV failures by factor of 10. Increase MOV common cause failure rates by factor of 10.	1.66E-06	5.72E-05	3.4
4	SIT Injection for Medium LOCA	Add SIT injection failure cutsets for MLOCA1 and MLOCA2 based on large LOCA SIT failure	1.66E-06	1.67E-06	-
5	Aggressive Secondary Cooldown not Feasible	Delete Small LOCA sequences 9 and 10 and SGTR sequences 15 and 16. Set operator error rate for failure to initiate aggressive secondary cooldown to 1.0	1.66E-06	7.7E-06	4.6
6	RCP Seal Failure on Station Blackout or Loss of Seal Injection or Seal Cooling	Create models for core damage due to RCP seal failure on SBO and Loss of Component Cooling Water	1.66E-06	1.66E-06	-
7	Components Unavailable Due to Maintenance	Set unavailability due to test and maintenance to 0.0 for all components	1.66E-06	1.65E-06	-

Table 10-1 SUMMARY OF SYSTEM 80+ PRA SENSITIVITY ANALYSIS RESULTS					
CASE	DESCRIPTION	MODELED AS	OLD CDF	NEW CDF	CHANGE FACTOR
8	Adverse MTC More Probable	Increase probability of having an adverse MTC when an ATWS occurs by a factor of 10	1.66E-06	2.09E-06	1.3
9	LOOP Frequency higher	Increase LOOP frequency by factor of 10	1.66E-06	1.96E-06	1.18
9A	LOOP Frequency Higher	Increase Loss of Grid Frequency to 0.15/year	1.66E-06	1.78e-06	1.07
10A	Vessel Rupture Not Credible	Set Vessel Rupture Rate to 0.0	1.66E-06	1.56E-06	-1.07
10B	Common Mode Failure Not Credible	Set all Common Mode Failure Ratess except for Diesels and Batteries to 0.0	1.66E-06	2.15E-07	-7.7
10C	Vessel Rupture and Common Mode Not Credible	Set Vessel Rupture and all Common Mode Failure Rates Except for Diesels and Batteries to 0.0	1.66E-06	1.15E-07	-14.4

Table 10-2
DOMINANT CUTSETS FOR SENSITIVITY CASE 1
OPERATOR ERROR RATE - General

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
1) SYSBOP					*9.82E-06
1) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	7.10E-01<	1.80E-06
HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
2) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	6.40E-01<	1.08E-06
HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
3) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	1.05E-06
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
4) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	1.05E-06
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
5) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	7.10E-01<	4.41E-07
HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
6) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	6.40E-01<	2.67E-07
FHFFSIAS	OPERATOR FAILS TO GENERATE SAFETY INJECTION ACTUATION SIGNAL		4.6E-03	4.60E-02<	
FSSXSIAS	COMMON CAUSE FAILURE OF SAFETY INJECTION ACTUATION SIGNALS		3.02E-03	3.02E-03	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
7) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	6.40E-01<	2.65E-07
HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
8) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	7.10E-01<	1.85E-07
HPSXR-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO RUN		5.80E-05	5.80E-05	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
9) ATWS	ANTICIPATED TRANSIENT WITHOUT SCRAM		4.75E-06	4.75E-06	1.41E-07
UHFFBORONRCS	OPERATOR FAILS TO INITIATE BORON DELIVERY TO RCS VIA CHARGING PUMP		3.25E-02	3.25E-01<	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
10) FHFFEFWS	OPERATOR FAILS TO ACTUATE EFWS COMPONENTS		4.6E-03	4.60E-02<	1.25E-07
FSERAPS	NO EFAS ACTUATION SIGNAL FROM ALTERNATE PROTECTION SYSTEM		2.60E-02	2.60E-02	
FSEX-EFAS	COMMON CAUSE FAILURE OF EMERGENCY FEEDWATER ACTUATION SIGNAL		2.79E-03	2.79E-03	
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
11) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	7.10E-01<	1.12E-07
HPSXD-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO START		3.49E-05	3.49E-05	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
12) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	6.40E-01<	1.11E-07
HPSXR-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO RUN		5.80E-05	5.80E-05	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
13) HHFFHOTLEG	OPERATOR FAILS TO INITIATE HOT LEG INJECTION		1.38E-04	1.38E-03<	1.09E-07
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
14) HVMXD-SET2	COMMON CAUSE FAILURE OF HOT LEG M-O ISO VALVES		8.00E-04	8.00E-04	1.00E-07
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	

Table 10-2 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 1
OPERATOR ERROR RATE - General

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
15) VR	VESSEL RUPTURE		1.00E-07	1.00E-07	1.00E-07
16) HHFFHOTLEG	OPERATOR FAILS TO INITIATE HOT LEG INJECTION		1.38E-04	1.38E-03<	9.62E-08
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
17) HHFFHOTLEG	OPERATOR FAILS TO INITIATE HOT LEG INJECTION		1.38E-04	1.38E-03<	9.62E-08
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
18) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	9.35E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-03<	
19) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	7.10E-01<	8.95E-08
HVCXD-SET5	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-404/5, SI-434, SI		2.80E-05	2.80E-05	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
20) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	7.10E-01<	8.95E-08
HVCXD-SET7	COMMON CAUSE FAILURE OF ALL 4 DVI CHECK VALVES (SI-217/SI-247) TO		2.80E-05	2.80E-05	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
21) HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	8.35E-08
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	3.30E-02<	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
22) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	6.40E-01<	6.70E-08
HPSXD-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO START		3.49E-05	3.49E-05	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
23) JHFDSCSLTC	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR LONG-TERM COOL		1.10E-04	1.10E-03<	6.53E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-03<	
24) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	6.49E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-03<	
25) HVMXD-SET2	COMMON CAUSE FAILURE OF HOT LEG M-O ISO VALVES		8.00E-04	8.00E-04	6.31E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
26) HVMXD-SET2	COMMON CAUSE FAILURE OF HOT LEG M-O ISO VALVES		8.00E-04	8.00E-04	5.58E-08
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
27) HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	5.56E-08
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	3.30E-02<	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
28) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	5.53E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-03	
29) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	5.53E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-03	
30) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	6.40E-01<	5.38E-08
HVCXD-SET5	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-404/5, SI-434, SI		2.80E-05	2.80E-05	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	

Table 10-2 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 1
OPERATOR ERROR RATE - General

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
31) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	6.40E-01<	5.38E-08
HVCXD-SET7	COMMON CAUSE FAILURE OF ALL 4 DVI CHECK VALVES (SI-217/SI-247) TO		2.80E-05	2.80E-05	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
32) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-04	4.75E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-03<	
33) ATWS	ANTICIPATED TRANSIENT WITHOUT SCRAM		4.75E-06	4.75E-06	4.75E-08
SE-MTC	ADVERSE MTC (> -0.3)		0.01	1.00E-02	
34) JHFDSCSLTC	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR LONG-TERM COOL		1.10E-04	1.10E-03<	4.54E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-03<	
35) HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	4.43E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
36) ELCX125C1E	COMMON CAUSE FAILURE OF 125 VDC CLASS 1E BUS		1.08E-07	1.08E-07	4.43E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
37) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	7.10E-01<	4.31E-08
HBDXD-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMP BREAKERS TO CLOSE		1.35E-05	1.35E-05	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
38) HVCXD-SET3	COMMON CAUSE FAILURE OF 3 OR MORE SI LINE CHECK VALVES TO OPEN		6.10E-04	6.10E-04	4.25E-08
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
39) HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	3.92E-08
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
40) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-04	3.30E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-03<	
41) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	3.30E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
42) SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	3.23E-08
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
AVMAEF-102	MOV EF-102 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AVSDEF-104	DC MOTOR VALVE EF-104 TRANSFERS CLOSED	1.68E-06	18	1.09E-02	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	
43) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	2.96E-08
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
JVMASI-310	SCS HX1 FLOW CONTROL VALVE SI-310 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-03<	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	

Table 10-2 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 1
OPERATOR ERROR RATE - General

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
44) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	2.96E-08
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
JVMASI-601	SCS TRAIN 1 DISCHARGE ISO VALVE SI-601 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-03<	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	
45) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	2.96E-08
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
JVMASI-655	SCS SUCTION M-O ISO VALVE SI-655 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-03<	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	
46) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	2.96E-08
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
JVMACC-111	CCW/SCSHX1 M-O VALVE CC-111 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-03<	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	
47) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	6.40E-01<	2.59E-08
HBDXD-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMP BREAKERS TO CLOSE		1.35E-05	1.35E-05	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
48) JHFDSCSLTC	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR LONG-TERM COOL		1.10E-04	1.10E-03<	2.30E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
49) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	2.29E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
50) JVMXCC-111/211	COMMON CAUSE FAILURE OF SCS/CCW VALVES CC-111/CC-211 TO OPEN		2.00E-04	2.00E-04	2.14E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-03<	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	
51) JVMXD-SET1	COMMON CAUSE FAILURE OF SCS HX FLOW CONTROL VALVES TO (SI-310/SI-3		2.00E-04	2.00E-04	2.14E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-03<	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	
52) JVMXD-SET2	COMMON CAUSE FAILURE OF SCS DISCHARGE ISO VALVES TO (SI-601/SI-600		2.00E-04	2.00E-04	2.14E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-03<	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	
53) JVMXSI-655/656	COMMON CAUSE FAILURE OF SI-655/SI-656 MOVs TO OPEN		2.00E-04	2.00E-04	2.14E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-03<	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	

Table 10-2 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 1
OPERATOR ERROR RATE - General

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
54) FHFFSIAS	OPERATOR FAILS TO GENERATE SAFETY INJECTION ACTUATION SIGNAL		4.6E-03	4.60E-02<	2.06E-08
FSSXSIAS	COMMON CAUSE FAILURE OF SAFETY INJECTION ACTUATION SIGNALS		3.02E-03	3.02E-03	
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	3.30E-02<	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
55) HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	2.05E-08
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	3.30E-02<	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
56) DVRBMSSVS	MAIN STEAM SAFETY VALVES (MSSVs) FAIL TO RESEAT		5.60E-02	5.60E-02	1.86E-08
PHFFSIPUMP	OPERATOR FAILS TO THROTTLE SAFETY INJECTION PUMP IN TIME		2.00E-04	2.00E-03<	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
UHFDRFIRWSTSGTR	OPERATOR FAILS TO ALIGN CVCS TO FILL IRWST FOLLOWING SGTR		3.7E-03	3.70E-02<	
57) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	1.70E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
AVCAEF-214	CHECK VALVE EF-214 FAILS TO OPEN	2.00E-04	1	2.00E-04	
58) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-04	1.68E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
59) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	1.64E-08
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
JVMASI-653	SCS SUCTION M-O ISO VALVE SI-653 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-03<	
60) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	1.64E-08
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
JVMASI-651	SCS SUCTION M-O ISO VALVE SI-651 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-03<	
61) JHFDSCSLTC	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR LONG-TERM COOL		1.10E-04	1.10E-03<	1.60E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
62) HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	1.37E-08
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	3.30E-02<	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
63) HVMXD-SET1	COMMON CAUSE FAILURE OF 3 OR MORE SI M-O VALVES TO OPEN		1.89E-04	1.89E-04	1.32E-08
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
64) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	1.29E-08
LOOP	LOSS OF OFFSITE POWER		5.00E-03	5.00E-03	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
65) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	1.29E-08
LOOP	LOSS OF OFFSITE POWER		5.00E-03	5.00E-03	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
66) DHFFRECLOSEADV	OPERATOR FAILS TO RECLOSE ADVs ON THE RUPTURED SG-2		3.75E-03	3.75E-02<	1.25E-08
PHFFSIPUMP	OPERATOR FAILS TO THROTTLE SAFETY INJECTION PUMP IN TIME		2.00E-04	2.00E-03<	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
UHFDRFIRWSTSGTR	OPERATOR FAILS TO ALIGN CVCS TO FILL IRWST FOLLOWING SGTR		3.7E-03	3.70E-02<	

Table 10-2 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 1
OPERATOR ERROR RATE - General

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
67) JHFDSCSLTC	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR LONG-TERM COOL		1.10E-04	1.10E-03<	1.19E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
AVCAEF-214	CHECK VALVE EF-214 FAILS TO OPEN	2.00E-04	1	2.00E-04	
68) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	1.18E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
AVCAEF-214	CHECK VALVE EF-214 FAILS TO OPEN	2.00E-04	1	2.00E-04	
69) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-04	1.16E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
70) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	1.16E-08
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
71) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	1.16E-08
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
72) HVCXD-SET4	COMMON CAUSE FAILURE OF HOT LEG CHECK VALVES		1.44E-04	1.44E-04	1.14E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
73) FHFFSIAS	OPERATOR FAILS TO GENERATE SAFETY INJECTION ACTUATION SIGNAL		4.6E-03	4.60E-02<	1.10E-08
FSSXSIAS	COMMON CAUSE FAILURE OF SAFETY INJECTION ACTUATION SIGNALS		3.02E-03	3.02E-03	
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
74) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	1.09E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
MVMASF-002	MOV SF-002 FAILS TO OPEN	4.00E-03	1	4.00E-03	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	
75) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	1.09E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
MVMASF-005	MOV SF-005 FAILS TO OPEN	4.00E-03	1	4.00E-03	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	
76) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	1.09E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
MVMASF-005	MOV SF-005 FAILS TO OPEN	4.00E-03	1	4.00E-03	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	
77) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	1.09E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
MVMASF-002	MOV SF-002 FAILS TO OPEN	4.00E-03	1	4.00E-03	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	
78) HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	1.09E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	

Table 10-2 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 1
OPERATOR ERROR RATE - General

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
79) LSSB	LARGE SECONDARY SIDE BREAK		1.50E-03	1.50E-03	1.08E-08
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
AVMAEF-102	MOV EF-102 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AVSDEF-104	DC MOTOR VALVE EF-104 TRANSFERS CLOSED	1.68E-06	18	1.09E-02	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	
80) APAXDEFWP102-104	COMMON CAUSE DEMAND FAILURE OF EFW MOTOR PUMPS		2.37E-04	2.37E-04	1.06E-08
APTXDP101-103	COMMON CAUSE DEMAND FAILURE OF EFW TURBINE PUMPS		1.19E-03	1.19E-03	
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
81) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	1.04E-08
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
JVMASI-310	SCS HX1 FLOW CONTROL VALVE SI-310 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	
82) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	1.04E-08
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
JVMASI-601	SCS TRAIN 1 DISCHARGE ISO VALVE SI-601 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	
83) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	1.04E-08
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
JVMASI-655	SCS SUCTION M-O ISO VALVE SI-655 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	
84) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	1.04E-08
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-02<	
JVMACC-111	CCW/SCSHX1 M-O VALVE CC-111 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E+00<	
85) HVCXD-SET4	COMMON CAUSE FAILURE OF HOT LEG CHECK VALVES		1.44E-04	1.44E-04	1.00E-08
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
86) HVCXD-SET4	COMMON CAUSE FAILURE OF HOT LEG CHECK VALVES		1.44E-04	1.44E-04	1.00E-08
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	

TABLE 10-3
DOMINANT CUTSETS FOR SENSITIVITY CASE 2
OPERATOR ERROR RATE - Control Room Response

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
1) SYSBOP					*1.80E-04
1) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	8.50E-05
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
2) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	5.90E-05
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
3) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-04	4.32E-06
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
4) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-04	3.00E-06
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
5) HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	2.53E-06
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	1.00E+00<	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
6) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-04	2.27E-06
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
7) HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	1.69E-06
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	1.00E+00<	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
8) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-04	1.57E-06
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
9) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	1.49E-06
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
JVMASI-651	SCS SUCTION M-O ISO VALVE SI-651 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
10) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	1.49E-06
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
JVMASI-310	SCS HX1 FLOW CONTROL VALVE SI-310 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.00E+00<	
11) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	1.49E-06
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
JVMASI-601	SCS TRAIN 1 DISCHARGE ISO VALVE SI-601 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.00E+00<	

TABLE 10-3 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 2
OPERATOR ERROR RATE - Control Room Response

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
12) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	1.49E-06
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
JVMASI-655	SCS SUCTION M-O ISO VALVE SI-655 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.00E+00<	
13) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	1.49E-06
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
JVMACC-111	CCW/SCSHX1 M-O VALVE CC-111 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.00E+00<	
14) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	1.49E-06
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
JVMASI-653	SCS SUCTION M-O ISO VALVE SI-653 FAILS TO OPEN	4.00E-03	1	4.00E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
15) JVMXCC-111/211	COMMON CAUSE FAILURE OF SCS/CCW VALVES CC-111/CC-211 TO OPEN		2.00E-04	2.00E-04	1.08E-06
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.00E+00<	
16) JVMXD-SET1	COMMON CAUSE FAILURE OF SCS HX FLOW CONTROL VALVES TO (SI-310/SI-3		2.00E-04	2.00E-04	1.08E-06
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.00E+00<	
17) JVMXD-SET2	COMMON CAUSE FAILURE OF SCS DISCHARGE ISO VALVES TO (SI-601/SI-600		2.00E-04	2.00E-04	1.08E-06
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.00E+00<	
18) JVMXSI-655/656	COMMON CAUSE FAILURE OF SI-655/SI-656 MOVs TO OPEN		2.00E-04	2.00E-04	1.08E-06
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.00E+00<	
19) AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	9.00E-07
PHFFSIPUMP	OPERATOR FAILS TO THROTTLE SAFETY INJECTION PUMP IN TIME		2.00E-04	2.00E-04	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
20) HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	6.21E-07
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	1.00E+00<	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
21) JHFDSCSLTC	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR LONG-TERM COOL		1.10E-04	1.10E-04	5.94E-07
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	

TABLE 10-3 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 2
OPERATOR ERROR RATE - Control Room Response

MODULE/EVENT NAME	DESCRIPTION	RATE	B.E. EXPOSURE	PROB.	MOD./CS. PROB.
22) HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	4.14E-07
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	1.00E+00<	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
23) JHFDSCSLTC	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR LONG-TERM COOL		1.10E-04	1.10E-04	4.13E-07
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
24) HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	2.65E-07
JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-04	
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
25) HPSXR-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO RUN		5.80E-05	5.80E-05	2.61E-07
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	1.00E+00<	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
26) JVCXD-SET1	COMMON CAUSE FAILURE OF SCS DISCHARGE CHECK VALVES (SI-178/SI-168)		3.60E-05	3.60E-05	1.94E-07
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
27) JVCXD-SET3	COMMON CAUSE FAILURE OF 2 DVI CHECK VALVES (SI-247/SI-227) TO OPEN		3.60E-05	3.60E-05	1.94E-07
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
28) HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	1.84E-07
JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-04	
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
29) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	7.10E-02	1.80E-07
HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
30) HPSXR-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO RUN		5.80E-05	5.80E-05	1.74E-07
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	1.00E+00<	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
31) HPSXD-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO START		3.49E-05	3.49E-05	1.57E-07
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	1.00E+00<	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
32) DVRBMSSVS	MAIN STEAM SAFETY VALVES (MSSVs) FAIL TO RESEAT		5.60E-02	5.60E-02	1.42E-07
HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
UHFDRFIRWSTSGTR	OPERATOR FAILS TO ALIGN CVCS TO FILL IRWST FOLLOWING SGTR		3.7E-03	1.00E+00<	
33) HVCXD-SET5	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-404/5, SI-434, SI		2.80E-05	2.80E-05	1.26E-07
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	1.00E+00<	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
34) HVCXD-SET7	COMMON CAUSE FAILURE OF ALL 4 DVI CHECK VALVES (SI-217/SI-247) TO		2.80E-05	2.80E-05	1.26E-07
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	1.00E+00<	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	

TABLE 10-3 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 2
OPERATOR ERROR RATE - Control Room Response

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
35) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	6.40E-02	1.08E-07
HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
36) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-[illegible]	1.05E-07
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
37) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	1.05E-07
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
38) HPSXD-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO START		3.49E-05	3.49E-05	1.05E-07
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	1.00E+00<	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
39) VR	VESSEL RUPTURE		1.00E-07	1.00E-07	1.00E-07
40) HVCXD-SET5	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-404/5, SI-434, SI		2.80E-05	2.80E-05	8.40E-08
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	1.00E+00<	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
41) HVCXD-SET7	COMMON CAUSE FAILURE OF ALL 4 DVI CHECK VALVES (SI-217/SI-247) TO		2.80E-05	2.80E-[illegible]	8.4[illegible]E-08
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-[illegible]	[illegible].00E+00<	
SLOCA	SMALL LOCA		3.00E-[illegible]	[illegible].00E-03	
42) HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	6.51E-08
JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-04	
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.00E+00<	
43) HVMXD-SET2	COMMON CAUSE FAILURE OF HOT LEG M-O ISO VALVES		8.00E-04	8.00E-04	6.31E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
44) FHFFSIAS	OPERATOR FAILS TO GENERATE SAFETY INJECTION ACTUATION SIGNAL		4.6E-03	4.60E-03	6.25E-08
FSSXSIAS	COMMON CAUSE FAILURE OF SAFETY INJECTION ACTUATION SIGNALS		3.02E-03	3.02E-03	
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	1.00E+00<	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
45) HBDXD-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMP BREAKERS TO CLOSE		1.35E-05	1.35E-05	6.07E-08
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	1.00E+00<	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
46) HVMXD-SE[illegible]	COMMON CAUSE FAILURE OF HOT LEG M-O ISO VALVES		8.00E-04	8.00E-04	5.58E-08
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
47) HVMXD-SET2	COMMON CAUSE FAILURE OF HOT LEG M-O ISO VALVES		8.00E-04	8.00E-04	5.58E-08
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.00E+00<	
48) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	5.53E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-03	
49) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	5.53E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMXB[illegible]	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-03	

TABLE 10-3 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 2
OPERATOR ERROR RATE - Control Room Response

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
50) DVRBMSSVS	MAIN STEAM SAFETY VALVES (MSSVs) FAIL TO RESEAT		5.60E-02	5.60E-02	5.04E-08
PHFFSIPUMP	OPERATOR FAILS TO THROTTLE SAFETY INJECTION PUMP IN TIME		2.00E-04	2.00E-04	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
UHFDRFIRWSTSGTR	OPERATOR FAILS TO ALIGN CVCS TO FILL IRWST FOLLOWING SGTR		3.7E-03	1.00E+00<	
51) ATWS	ANTICIPATED TRANSIENT WITHOUT SCRAM		4.75E-06	4.75E-06	4.75E-08
SE-MTC	ADVERSE MTC (> -0.3)		0.01	1.00E-02	
52) HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	4.43E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
53) ELCX125C1E	COMMON CAUSE FAILURE OF 125 VDC CLASS 1E BUS		1.08E-07	1.08E-07	4.43E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
54) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	7.10E-02	4.41E-08
HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
55) HVCXD-SET3	COMMON CAUSE FAILURE OF 3 OR MORE SI LINE CHECK VALVES TO OPEN		6.10E-04	6.10E-04	4.25E-08
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
56) HBDXD-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMP BREAKERS TO CLOSE		1.35E-05	1.35E-05	4.05E-08
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	1.00E+00<	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
57) HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	3.92E-08
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
58) DVRBMSSVS	MAIN STEAM SAFETY VALVES (MSSVs) FAIL TO RESEAT		5.60E-02	5.60E-02	3.48E-08
HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
UHFDRFIRWSTSGTR	OPERATOR FAILS TO ALIGN CVCS TO FILL IRWST FOLLOWING SGTR		3.7E-03	1.00E+00<	
59) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	3.30E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
60) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	6.40E-02	2.65E-08
HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
61) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	2.29E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
62) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	7.10E-02	1.85E-08
HPSXR-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO RUN		5.80E-05	5.80E-05	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
63) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	1.70E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
AVCAEF-214	CHECK VALVE EF-214 FAILS TO OPEN	2.00E-04	1	2.00E-04	
64) DVRBMSSVS	MAIN STEAM SAFETY VALVES (MSSVs) FAIL TO RESEAT		5.60E-02	5.60E-02	1.46E-08
HPSXR-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO RUN		5.80E-05	5.80E-05	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
UHFDRFIRWSTSGTR	OPERATOR FAILS TO ALIGN CVCS TO FILL IRWST FOLLOWING SGTR		3.7E-03	1.00E+00<	
65) HVMXD-SET1	COMMON CAUSE FAILURE OF 3 OR MORE SI M-O VALVES TO OPEN		1.89E-04	1.89E-04	1.32E-08
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	

TABLE 10-3 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 2
OPERATOR ERROR RATE - Control Room Response

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
66) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	1.18E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
AVCAEF-214	CHECK VALVE EF-214 FAILS TO OPEN	2.00E-04	1	2.00E-04	
67) HVCXD-SET4	COMMON CAUSF FAILURE OF HOT LEG CHECK VALVES		1.44E-04	1.44E-04	1.14E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
68) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	7.10E-02	1.12E-08
HPSXD-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO START		3.49E-05	3.49E-05	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
69) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	6.40E-02	[illegible]
HPSXR-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO RUN		5.80E-05	5.80E-05	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
70) HHFFHOTLEG	OPERATOR FAILS TO INITIATE HOT LEG INJECTION		1.38E-04	1.38E-04	1.09E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
71) HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	1.09E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
72) HVCXD-SET4	COMMON CAUSE FAILURE OF HOT LEG CHECK VALVES		1.44E-04	1.44E-04	1.00E-08
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
73) HVCXD-SET4	COMMON CAUSE FAILURE OF HOT LEG CHECK VALVES		1.44E-04	1.44E-04	1.00E-08
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	

TABLE 10-4
DOMINANT CUTSETS FOR SENSITIVITY CASE 3
MOTOR-OPERATED VALVE FAILURE RATE

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
1) SYS80P					*5.72E-06
1) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	5.53E-07
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-02<	
2) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	5.53E-07
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-02<	
3) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	7.10E-02	4.41E-07
HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-03<	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
4) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	6.40E-02	2.65E-07
HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-03<	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
5) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	7.10E-02	1.80E-07
HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
6) MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	1.26E-07
HVMASI-331	HOT LEG 2 M-O ISO VALVE SI-331 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
HVMASI-604	HOT LEG 1 M-O ISO VALVE SI-604 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
7) MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	1.26E-07
HVMASI-609	HOT LEG 2 M-O ISO VALVE SI-609 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
HVMASI-321	HOT LEG 1 M-O VALVE SI-321 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
8) MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	1.26E-07
HVMASI-609	HOT LEG 2 M-O ISO VALVE SI-609 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
HVMASI-604	HOT LEG 1 M-O ISO VALVE SI-604 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
9) MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	1.26E-07
HVMASI-331	HOT LEG 2 M-O ISO VALVE SI-331 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
HVMASI-321	HOT LEG 1 M-O VALVE SI-321 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
10) LLOCA	LARGE LOCA		6.97E-05	6.97E-05	1.12E-07
HVMASI-331	HOT LEG 2 M-O ISO VALVE SI-331 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
HVMASI-321	HOT LEG 1 M-O VALVE SI-321 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
11) LLOCA	LARGE LOCA		6.97E-05	6.97E-05	1.12E-07
HVMASI-331	HOT LEG 2 M-O ISO VALVE SI-331 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
HVMASI-604	HOT LEG 1 M-O ISO VALVE SI-604 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
12) LLOCA	LARGE LOCA		6.97E-05	6.97E-05	1.12E-07
HVMASI-609	HOT LEG 2 M-O ISO VALVE SI-609 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
HVMASI-321	HOT LEG 1 M-O VALVE SI-321 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
13) LLOCA	LARGE LOCA		6.97E-05	6.97E-05	1.12E-07
HVMASI-609	HOT LEG 2 M-O ISO VALVE SI-609 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
HVMASI-604	HOT LEG 1 M-O ISO VALVE SI-604 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
14) MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	1.12E-07
HVMASI-609	HOT LEG 2 M-O ISO VALVE SI-609 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
HVMASI-321	HOT LEG 1 M-O VALVE SI-321 FAILS TO OPEN	4.00E-03	1	4.00E-02<	

TABLE 10-4 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 3
MOTOR-OPERATED VALVE FAILURE RATE

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
15) MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	1.12E-07
HVMASI-331	HOT LEG 2 M-O ISO VALVE SI-331 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
HVMASI-604	HOT LEG 1 M-O ISO VALVE SI-604 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
16) MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	1.12E-07
HVMASI-609	HOT LEG 2 M-O ISO VALVE SI-609 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
HVMASI-604	HOT LEG 1 M-O ISO VALVE SI-604 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
17) MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	1.12E-07
HVMASI-331	HOT LEG 2 M-O ISO VALVE SI-331 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
HVMASI-321	HOT LEG 1 M-O VALVE SI-321 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
18) HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-03<	1.09E-07
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
19) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	6.40E-02	1.08E-07
HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
20) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	1.05E-07
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
21) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	1.05E-07
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
22) VR	VESSEL RUPTURE		1.00E-07	1.00E-07	1.00E-07
23) HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-03<	9.62E-08
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
24) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-03<	8.79E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-02<	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
25) HVMXD-SET2	COMMON CAUSE FAILURE OF HOT LEG M-O ISO VALVES		8.00E-04	8.00E-04	6.31E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
26) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-03<	6.11E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-02<	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
27) HVMXD-SET2	COMMON CAUSE FAILURE OF HOT LEG M-O ISO VALVES		8.00E-04	8.00E-04	5.58E-08
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
28) ATWS	ANTICIPATED TRANSIENT WITHOUT SCRAM		4.75E-06	4.75E-06	4.75E-08
SE-MTC	ADVERSE MTC (> -0.3)		0.01	1.00E-02	
29) APTXDP101-103	COMMON CAUSE DEMAND FAILURE OF EFW TURBINE PUMPS		1.19E-03	1.19E-03	4.68E-08
AVMXEF102-103	COMMON CAUSE FAILURE OF EFW DIST. AC VALVES EF-102 & EF-103		2.00E-04	2.00E-03<	
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-02<	
30) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-03<	4.53E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-02<	
AVCAEF-214	CHECK VALVE EF-214 FAILS TO OPEN	2.00E-04	1	2.00E-04	

TABLE 10-4 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 3
MOTOR-OPERATED VALVE FAILURE RATE

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
31) HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	4.43E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
32) ELCX125C1E	COMMON CAUSE FAILURE OF 125 VDC CLASS 1E BUS		1.08E-07	1.08E-07	4.43E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
33) HVCXD-SET3	COMMON CAUSE FAILURE OF 3 OR MORE SI LINE CHECK VALVES TO OPEN		6.10E-04	6.10E-04	4.25E-08
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
34) HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	3.92E-08
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
35) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	3.30E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
36) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-03<	3.15E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-02<	
AVCAEF-214	CHECK VALVE EF-214 FAILS TO OPEN	2.00E-04	1	2.00E-04	
37) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	3.04E-08
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-02<	
JVMASI-651	SCS SUCTION M-O ISO VALVE SI-651 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
38) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	3.04E-08
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-02<	
JVMASI-653	SCS SUCTION M-O ISO VALVE SI-653 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
39) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-03<	2.49E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-02<	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-04	
40) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	2.29E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
41) SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	2.13E-08
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-02<	
AVMDEF-106	MOV EF-106 FAILS TO REMAIN OPEN	1.40E-07	18	9.07E-03<	
AVSDEF-104	DC MOTOR VALVE EF-104 TRANSFERS CLOSED	1.68E-06	18	1.09E-02	
42) HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-03<	2.05E-08
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	3.30E-03	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
43) HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-03<	1.93E-08
LOOP	LOSS OF OFFSITE POWER		5.00E-03	5.00E-03	
SE-PSV	FAILURE OF PRIMARY SAFTEY VALVE TO RESEAT		2.8E-03	2.80E-03	
44) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	7.10E-02	1.85E-08
HPSXR-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO RUN		5.80E-05	5.80E-05	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	

TABLE 10-4 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 3
MOTOR-OPERATED VALVE FAILURE RATE

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
45) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-03<	1.73E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-02<	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-04	
46) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	1.70E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
AVCAEF-214	CHECK VALVE EF-214 FAILS TO OPEN	2.00E-04	1	2.00E-04	
47) SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	1.69E-08
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-02<	
AVMAEF-102	MOV EF-102 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
AVSDEF-104	DC MOTOR VALVE EF-104 TRANSFERS CLOSED	1.68E-06	18	1.09E-02	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E-01	
48) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-03<	1.68E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
49) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	1.59E-08
HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-03<	
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
50) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	1.59E-08
HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-03<	
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
51) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	1.57E-08
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-02<	
JVMASI-653	SCS SUCTION M-O ISO VALVE SI-653 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
AVCAEF-214	CHECK VALVE EF-214 FAILS TO OPEN	2.00E-04	1	2.00E-04	
52) LHVAC	LOSS OF ONE DIVISION OF HVAC		4.08E-02	4.08E-02	1.57E-08
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-02<	
JVMASI-651	SCS SUCTION M-O ISO VALVE SI-651 FAILS TO OPEN	4.00E-03	1	4.00E-02<	
AVCAEF-214	CHECK VALVE EF-214 FAILS TO OPEN	2.00E-04	1	2.00E-04	
53) HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-03<	1.37E-08
JHFDRHRI	OPERATOR FAILS TO ALIGN SHUTDOWN COOLING SYSTEM FOR INJECTION OPER		3.30E-03	3.30E-03	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
54) HVMXD-SET1	COMMON CAUSE FAILURE OF 3 OR MORE SI M-O VALVES TO OPEN		1.89E-04	1.89E-04	1.32E-08
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
55) DVMXD-ADVSG1	COMMON CAUSE FAILURE OF 2 ADVs ON STEAM GENERATOR 1 TO OPEN		2.00E-04	2.00E-03<	1.24E-08
HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-03<	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
56) DVMXD-ADVSG2	COMMON CAUSE FAILURE OF 2 ADVs ON STEAM GENERATOR 2 TO OPEN		2.00E-04	2.00E-03<	1.24E-08
HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-03<	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
57) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	1.18E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
AVCAEF-214	CHECK VALVE EF-214 FAILS TO OPEN	2.00E-04	1	2.00E-04	

TABLE 10-4 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 3
MOTOR-OPERATED VALVE FAILURE RATE

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
58) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-03<	1.16E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
59) HVCXD-SET4	COMMON CAUSE FAILURE OF HOT LEG CHECK VALVES		1.44E-04	1.44E-04	1.14E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
60) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	7.10E-02	1.12E-08
HPSXD-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO START		3.49E-05	3.49E-05	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
61) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	6.40E-02	1.11E-08
HPSXR-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO RUN		5.80E-05	5.80E-05	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
62) KHFFHOTLEG	OPERATOR FAILS TO INITIATE HOT LEG INJECTION		1.38E-04	1.38E-04	1.09E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
63) HVCXD-SET4	COMMON CAUSE FAILURE OF HOT LEG CHECK VALVES		1.44E-04	1.44E-04	1.00E-08
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
64) HVCXD-SET4	COMMON CAUSE FAILURE OF HOT LEG CHECK VALVES		1.44E-04	1.44E-04	1.00E-08
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
65) HVMXD-SET2	COMMON CAUSE FAILURE OF HOT LEG M-O ISO VALVES		8.00E-04	8.00E-04	1.00E-08
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E-01	

TABLE 10-5
DOMINANT CUTSETS FOR SENSITIVITY CASE 4
AGGRESSIVE SECONDARY COOLDOWN NOT FEASIBLE

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
1) SYS80P					*7.70E-06
1) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	1.00E+00<	2.53E-06
HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
2) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	1.00E+00<	1.69E-06
HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
3) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	1.00E+00<	6.21E-07
HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
4) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	1.00E+00<	4.14E-07
HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
5) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	1.00E+00<	2.61E-07
HPSXR-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO RUN		5.80E-05	5.80E-05	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
6) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	1.00E+00<	1.74E-07
HPSXR-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO RUN		5.80E-05	5.80E-05	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
7) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	1.00E+00<	1.57E-07
HPSXD-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO START		3.49E-05	3.49E-05	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
8) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	1.00E+00<	1.26E-07
HVCXD-SET5	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-404/5, SI-434, SI		2.80E-05	2.80E-05	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
9) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	1.00E+00<	1.26E-07
HVCXD-SET7	COMMON CAUSE FAILURE OF ALL 4 DVI CHECK VALVES (SI-217/SI-247) TO		2.80E-05	2.80E-05	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
10) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	1.05E-07
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
11) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	1.05E-07
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
12) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	1.00E+00<	1.05E-07
HPSXD-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO START		3.49E-05	3.49E-05	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
13) VR	VESSEL RUPTURE		1.00E-07	1.00E-07	1.00E-07
14) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	1.00E+00<	8.40E-08
HVCXD-SET5	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-404/5, SI-434, SI		2.80E-05	2.80E-05	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
15) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	1.00E+00<	8.40E-08
HVCXD-SET7	COMMON CAUSE FAILURE OF ALL 4 DVI CHECK VALVES (SI-217/SI-247) TO		2.80E-05	2.80E-05	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	

TABLE 10-5 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 4
AGGRESSIVE SECONDARY COOLDOWN NOT FEASIBLE

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
16) HVMXD-SET2	COMMON CAUSE FAILURE OF HOT LEG M-O ISO VALVES		8.00E-04	8.00E-04	6.31E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
17) AHFFASCSGTR	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SGTR		7.10E-02	1.00E+00<	6.07E-08
HBDXD-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMP BREAKERS TO CLOSE		1.35E-05	1.35E-05	
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
18) HVMXD-SET2	COMMON CAUSE FAILURE OF HOT LEG M-O ISO VALVES		8.00E-04	8.00E-04	5.58E-08
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
19) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	5.53E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-03	
20) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	5.53E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMXBLDV	COMMON CAUSE FAILURE OF BLEED VALVES		4.80E-03	4.80E-03	
21) ATWS	ANTICIPATED TRANSIENT WITHOUT SCRAM		4.75E-06	4.75E-06	4.75E-08
SE-MTC	ADVERSE MTC (> -0.3)		0.01	1.00E-02	
22) HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	4.43E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
23) ELCX125C1E	COMMON CAUSE FAILURE OF 125 VDC CLASS 1E BUS		1.08E-07	1.08E-07	4.43E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
24) HVCXD-SET3	COMMON CAUSE FAILURE OF 3 OR MORE SI LINE CHECK VALVES TO OPEN		6.10E-04	6.10E-04	4.25E-08
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
25) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	1.00E+00<	4.17E-08
FHFFSIAS	OPERATOR FAILS TO GENERATE SAFETY INJECTION ACTUATION SIGNAL		4.6E-03	4.60E-03	
FSSXSIAS	COMMON CAUSE FAILURE OF SAFETY INJECTION ACTUATION SIGNALS		3.02E-03	3.02E-03	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
26) AHFFASCSLOCA	OPERATOR FAILS TO PERFORM AGGRESSIVE SECONDARY COOLDOWN FOR SMALL		6.4E-02	1.00E+00<	4.05E-08
HBDXD-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMP BREAKERS TO CLOSE		1.35E-05	1.35E-05	
SLOCA	SMALL LOCA		3.00E-03	3.00E-03	
27) HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	3.92E-08
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
28) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	3.30E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
29) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	2.29E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
30) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	1.70E-08
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
AVCAEF-214	CHECK VALVE EF-214 FAILS TO OPEN	2.00E-04	1	2.00E-04	
31) HVMXD-SET1	COMMON CAUSE FAILURE OF 3 OR MORE SI M-O VALVES TO OPEN		1.89E-04	1.89E-04	1.32E-08
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
32) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	1.18E-08
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
AVCAEF-214	CHECK VALVE EF-214 FAILS TO OPEN	2.00E-04	1	2.00E-04	

TABLE 10-5 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 4
AGGRESSIVE SECONDARY COOLDOWN NOT FEASIBLE

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
33) HVCXD-SET4	COMMON CAUSE FAILURE OF HOT LEG CHECK VALVES		1.44E-04	1.44E-04	1.14E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
34) HHFFHOTLEG	OPERATOR FAILS TO INITIATE HOT LEG INJECTION		1.38E-04	1.38E-04	1.09E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
35) HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	1.09E-08
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
36) HVCXD-SET4	COMMON CAUSE FAILURE OF HOT LEG CHECK VALVES		1.44E-04	1.44E-04	1.00E-08
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
37) HVCXD-SET4	COMMON CAUSE FAILURE OF HOT LEG CHECK VALVES		1.44E-04	1.44E-04	1.00E-08
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
38) HVMXD-SET2	COMMON CAUSE FAILURE OF HOT LEG M-O ISO VALVES		8.00E-04	8.00E-04	1.00E-08
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
RCVRMOV	FAILURE TO MANUALLY OPEN MOTOR OPERATED VALVE		1.80E-01	1.80E-01	
39) HHFFHOTLEG	OPERATOR FAILS TO INITIATE HOT LEG INJECTION		1.38E-04	1.38E-04	9.62E-09
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
40) HHFFHOTLEG	OPERATOR FAILS TO INITIATE HOT LEG INJECTION		1.38E-04	1.38E-04	9.62E-09
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
41) HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	9.62E-09
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
42) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	9.35E-09
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-04	
43) HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	7.87E-09
LOOP	LOSS OF OFFSITE POWER		5.00E-03	5.00E-03	
SE-PSV	FAILURE OF PRIMARY SAFTEY VALVE TO RESEAT		2.8E-03	2.80E-03	
44) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	6.64E-09
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMARC-406	MOV RC-406 FAILS TO OPEN		2.40E-02	2.40E-02	
VVMARC-407	MOV RC-407 FAILS TO OPEN		2.40E-02	2.40E-02	
45) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	6.64E-09
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMARC-408	MOV RC-408 FAILS TO OPEN		2.40E-02	2.40E-02	
VVMARC-409	MOV RC-409 FAILS TO OPEN		2.40E-02	2.40E-02	
46) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	6.64E-09
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMARC-406	MOV RC-406 FAILS TO OPEN		2.40E-02	2.40E-02	
VVMARC-409	MOV RC-409 FAILS TO OPEN		2.40E-02	2.40E-02	
47) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	6.64E-09
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMARC-408	MOV RC-408 FAILS TO OPEN		2.40E-02	2.40E-02	
VVMARC-407	MOV RC-407 FAILS TO OPEN		2.40E-02	2.40E-02	
48) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	6.64E-09
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMARC-408	MOV RC-408 FAILS TO OPEN		2.40E-02	2.40E-02	
VVMARC-407	MOV RC-407 FAILS TO OPEN		2.40E-02	2.40E-02	

TABLE 10-5 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 4
AGGRESSIVE SECONDARY COOLDOWN NOT FEASIBLE

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
49) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	6.64E-09
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMARC-406	MOV RC-406 FAILS TO OPEN		2.40E-02	2.40E-02	
VVMARC-409	MOV RC-409 FAILS TO OPEN		2.40E-02	2.40E-02	
50) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	6.64E-09
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMARC-406	MOV RC-406 FAILS TO OPEN		2.40E-02	2.40E-02	
VVMARC-407	MOV RC-407 FAILS TO OPEN		2.40E-02	2.40E-02	
51) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	6.64E-09
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
VVMARC-408	MOV RC-408 FAILS TO OPEN		2.40E-02	2.40E-02	
VVMARC-409	MOV RC-409 FAILS TO OPEN		2.40E-02	2.40E-02	
52) JVCXD-SET2	COMMON CAUSE FAILURE OF 2 SI CHECK VALVES (SI-143,SI-543,SI-123,SI		1.44E-04	1.44E-04	6.49E-09
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
AHFDCST	OPERATOR FAILS TO ALIGN CST TO EFW STORAGE TANKS		1.10E-04	1.10E-04	
53) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	6.47E-09
HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
54) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	6.47E-09
HVCXD-SET6	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-113/143, SI-540/5		5.62E-04	5.62E-04	
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
55) HPSXR-SET1	COMMON CAUSE FAILURE OF 3 OR MORE SI PUMPS TO RUN		7.31E-05	7.31E-05	5.10E-09
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
56) HPSXR-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO RUN		5.80E-05	5.80E-05	4.58E-09
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
57) HPSXD-SET1	COMMON CAUSE FAILURE OF 3 OR MORE SI PUMPS TO START		6.46E-05	6.46E-05	4.50E-09
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
58) HPSXR-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO RUN		5.80E-05	5.80E-05	4.04E-09
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
59) EDDXDDGA-B	COMMON CAUSE DEMAND FAILURE OF DGs		2.80E-04	2.80E-04	3.43E-09
LOOP	LOSS OF OFFSITE POWER		5.00E-03	5.00E-03	
RCVRPWR10	FAILURE TO RECOVER OFFSITE POWER IN 10 HOURS		4.90E-02	4.90E-02	
RCVRSBAC	FAILURE TO START AND LOAD STANDBY AC POWER		5.00E-02	5.00E-02	
60) ESXXSEQ	COMMON CAUSE FAILURE OF DG LOAD SEQUENCERS		2.25E-04	2.25E-04	2.76E-09
LOOP	LOSS OF OFFSITE POWER		5.00E-03	5.00E-03	
RCVRPWR10	FAILURE TO RECOVER OFFSITE POWER IN 10 HOURS		4.90E-02	4.90E-02	
RCVRSBAC	FAILURE TO START AND LOAD STANDBY AC POWER		5.00E-02	5.00E-02	
61) HPSXD-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO START		3.49E-05	3.49E-05	2.75E-09
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
62) HPSXD-SET2	COMMON CAUSE FAILURE OF ALL 4 SI PUMPS TO START		3.49E-05	3.49E-05	2.43E-09
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	

TABLE 10-5 (cont'd)
DOMINANT CUTSETS FOR SENSITIVITY CASE 4
AGGRESSIVE SECONDARY COOLDOWN NOT FEASIBLE

MODULE/EVENT NAME	DESCRIPTION	RATE	EXPOSURE	B.E. PROB.	MOD./CS. PROB.
63) EDDJDGA	EMERGENCY DIESEL GENERATOR DG A FAILS TO START & LOAD	1.40E-02	1	1.40E-02	2.40E-09
EDDJDGB	EMERGENCY DIESEL GENERATOR DG B FAILS TO START & LOAD	1.40E-02	1	1.40E-02	
LOOP	LOSS OF OFFSITE POWER		5.00E-03	5.00E-03	
RCVRPWR10	FAILURE TO RECOVER OFFSITE POWER IN 10 HOURS		4.90E-02	4.90E-02	
RCVRSBAC	FAILURE TO START AND LOAD STANDBY AC POWER		5.00E-02	5.00E-02	
64) HVCXD-SET1	COMMON CAUSE FAILURE OF 3 OR MORE SI CHECK VALVES TO OPEN		3.42E-05	3.42E-05	2.38E-09
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
65) HVCXD-SET2	COMMON CAUSE FAILURE OF 3 OR MORE SI PUMP DISCHARGE LEG CHECK VALV		3.42E-05	3.42E-05	2.38E-09
LLOCA	LARGE LOCA		6.97E-05	6.97E-05	
66) LLOCA	LARGE LOCA		6.97E-05	6.97E-05	2.38E-09
LVCXD-SET1	COMMON CAUSE FAILURE OF 3 OR MORE SIT DISCHARGE CHECK VALVES TO OP		3.42E-05	3.42E-05	
67) HVCXD-SET5	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-404/5, SI-434, SI		2.80E-05	2.80E-05	2.21E-09
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
68) HVCXD-SET7	COMMON CAUSE FAILURE OF ALL 4 DVI CHECK VALVES (SI-217/SI-247) TO		2.80E-05	2.80E-05	2.21E-09
MLOCA2	MEDIUM LOCA 2		7.89E-05	7.89E-05	
69) HVCXD-SET5	COMMON CAUSE FAILURE OF SI LINE CHECK VALVES (SI-404/5, SI-434, SI		2.80E-05	2.80E-05	1.95E-09
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
70) HVCXD-SET7	COMMON CAUSE FAILURE OF ALL 4 DVI CHECK VALVES (SI-217/SI-247) TO		2.80E-05	2.80E-05	1.95E-09
MLOCA1	MEDIUM LOCA 1		6.97E-05	6.97E-05	
71) HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	1.93E-09
LOOP	LOSS OF OFFSITE POWER		5.00E-03	5.00E-03	
SE-PSV	FAILURE OF PRIMARY SAFTEY VALVE TO RESEAT		2.8E-03	2.80E-03	
72) EXLXESF	COMMON CAUSE FAILURE OF 480 V LC ESF TRANSFORMERS		3.78E-07	3.78E-07	1.70E-09
SGTR	STEAM GENERATOR TUBE RUPTURE		4.5E-03	4.50E-03	
73) EJEXDGRM	COMMON CAUSE FAILURE OF DG ROOM DAMPERS		1.37E-04	1.37E-04	1.68E-09
LOOP	LOSS OF OFFSITE POWER		5.00E-03	5.00E-03	
RCVRPWR10	FAILURE TO RECOVER OFFSITE POWER IN 10 HOURS		4.90E-02	4.90E-02	
RCVRSBAC	FAILURE TO START AND LOAD STANDBY AC POWER		5.00E-02	5.00E-02	
74) JVMXSI-651/654	COMMON CAUSE FAILURE OF ALL 4 SUCTION VALVES FROM RCS (SI-651/SI-6		8.00E-04	8.00E-04	1.68E-09
TOTH	TRANSIENTS-OTHER		5.9E-01	5.90E-01	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	
AVNAEF-215	MANUAL VALVE EF-215 FAILS TO OPEN	3.88E-04	1	3.88E-04	
75) AVCXDIST	COMMON CAUSE FAILURE OF DISTRIBUTION LINE CHECK VALVES		2.81E-05	2.81E-05	1.59E-09
HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
76) AVCXEFWP	COMMON CAUSE FAILURE OF EFW PUMP DISCHARGE CHECK VALVES		2.81E-05	2.81E-05	1.59E-09
HVMXD-SET3	COMMON CAUSE FAILURE OF SI LINE M-O ISO VALVES (SI-616/SI-646) TO		1.38E-04	1.38E-04	
LOFW	LOSS OF FEEDWATER		4.1E-01	4.10E-01	
77) ATWS	ANTICIPATED TRANSIENT WITHOUT SCRAM		4.75E-06	4.75E-06	1.41E-09
UHFFBORONRCS	OPERATOR FAILS TO INITIATE BORON DELIVERY TO RCS VIA CHARGING PUMP		3.25E-02	3.25E-02	
VHFFFEEDBLEED	OPERATOR FAILS TO INITIATE FEED & BLEED SYSTEM		9.15E-03	9.15E-03	

TABLE 10-6
EVENTS WITH LOSS OF COOLING TO RCP PUMPS

DATE	PLANT	PUMPS	STAGES	LEAKS	FAILED STAGES	DESCRIPTION
4/17/74	FT CLH	4	16	2	0	ISOLATED CCW TO RCPS ON ESFAS SIGNAL 4 RCPs RUN FOR 45 MIN WITHOUT COOLING
9/20/75	FT CLH	4	16	1	1	ALL 4 SEALS CHANGED AFTER LOSS OF CCW
4/15/77	SL-1	4	16	0	0	LOSS OF CONTAINMENT INSTRU. AIR CAUSED LOSS OF CCW
12/19/78	SONG BJ T	1	4	0	0	BJ PUMP TEST, 30 MIN. PUMP RUNNING, PUMP RUN FOR 2.5 HRS WITH COOLING
6/11/80	SL-1	4	16	0	0	CCW TO RCPs LOST (8+ HRS)
6/24/80	ANO-2	4	16	1	0	PARTIAL LOSS OF AC, LEAK 1.5 - 2 GPM
8/26/80	SL2 T*	1	4	0	0	BJ PUMP TEST, 50 HRS, SBO TEST, NORMAL BLEEDOFF 0.6 GPM
XX/XX/81	FT CLH	4	16	0	0	CCW LOST 1 HR IN HOT STANDBY, PUMPS RESTARTED O.K.
3/XX/83	SONG-2	4	16	0	0	
11/21/83	PV-KSB T	1	3	0	0	TEST OF KSB PUMP FOR PV-1, 36 MIN., PUMP STOPPED
11/15/84	MLS-2	1	4	0	0	CCW LOST FOR 9 HRS
11/16/84	MLS-2	1	4	1	1	CCW LOST FOR 6 HRS, I STAGE REPORTED FAILED, NO LEAKAGE
12/19/84	SL-2	2	8	2	0	LOSS OF CCW TO 2 PUMPS FOR 30 MIN
2/20/85	WTF-3	4	16	1	0	LOSS CCW, 3 PMPS NOT RESTORED, 3 GPM FROM 1 PUMP 4.5 HR
8/8/85	SL-2	4	16	2	0	LOSS OF CCW, 2 SEALS LOST COOLING FOR 4.5 HRS
4/4/86	PV-2	1	4	0	0	COOLING LOST FOR 3 HRS, PUMP RAN FOR 10 MIN.
7/1/86	PV-2	4	12	2	1	ALL COOLING LOST FOR 3 HRS, PUMPS OPERATED FOR 10 MIN., 3RD STAGE REPORTED DISASSEMBLED
12/86	AECL-BG T	9	18	0	0	9 TESTS, 2 SEALS, BINGHAM-WILLAMETTE,NO FAILURE, 24 HR TEST
12/86	AECL-BJ T	7	14	0	0	7 TESTS, 2 SEALS, BJ, NO FAILURES
12/87	N9000 T	1	3	0	0	BJ MODEL TEST FOR STATION BLACKOUT
7/6/88	PV-1	1	3	0	0	CCW LOST TO 1 PUMP, 8 HRS
8/1/88	ANO-2	1	4	2CC**	2	SENSING LINE BROKE, 2 FACES BROKEN,LEAK 20 GPM
3/3/89	PV-3	4	12	1	1	LOSS SEAL INJECTION, CCW TO ALL PUMPS FOR 73 MIN., 1ST STAGE REPORTED FAILED
???	BINGM T	1	4	0	0	SONG TEST ON 4.5" SHAFT, 30 MIN.
TOTALS: 24 EVENTS		72	245	15	6	

* T = TEST(S)
** CC = COMMON CAUSE FAILURE

Figure 10-1
Fault Tree for Core Damage Due to RCP Seal LOCA on SBO

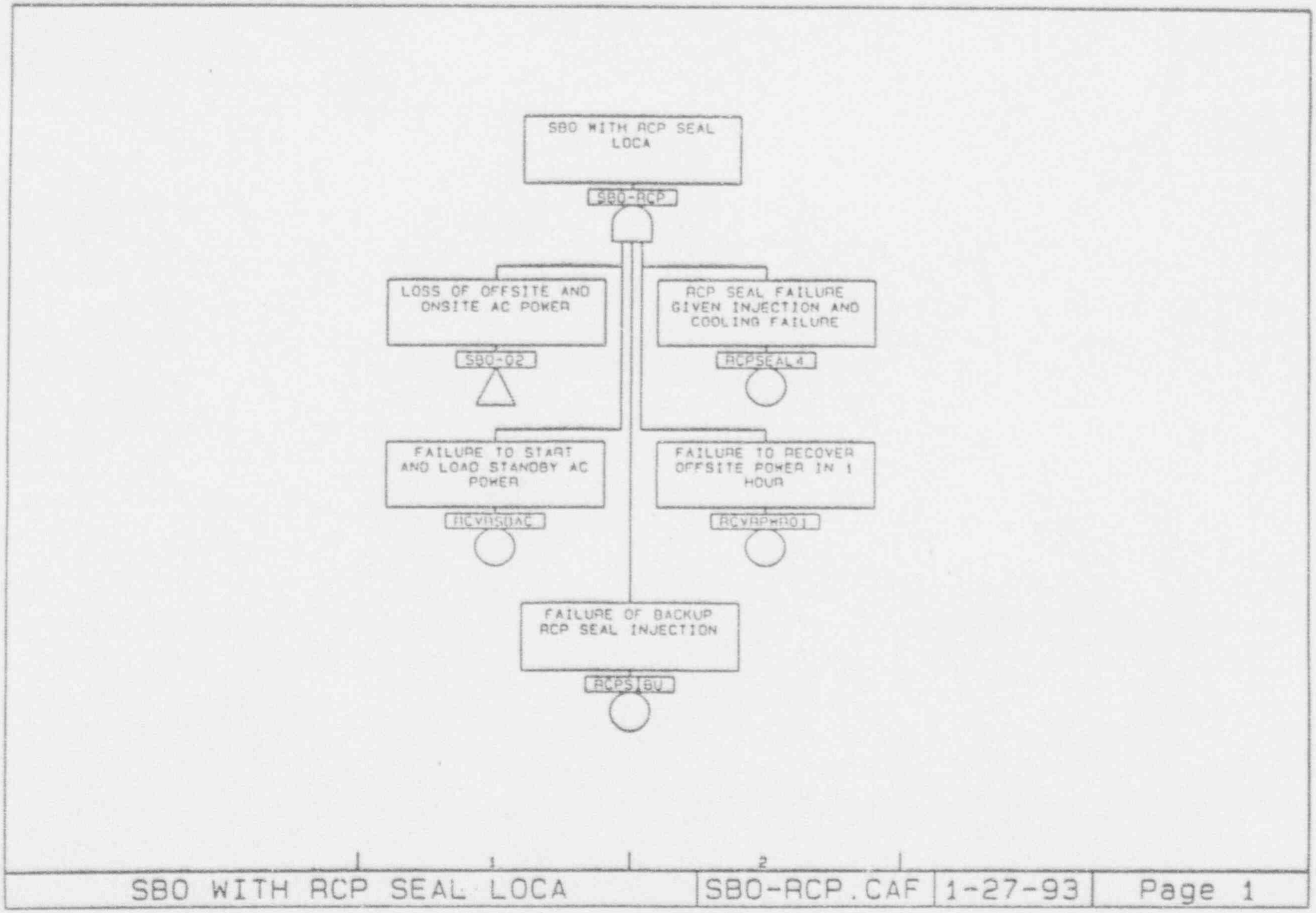


Figure 10-2
Fault Tree for Core Damage Due to RCP Seal LOCA Following Loss of CCW

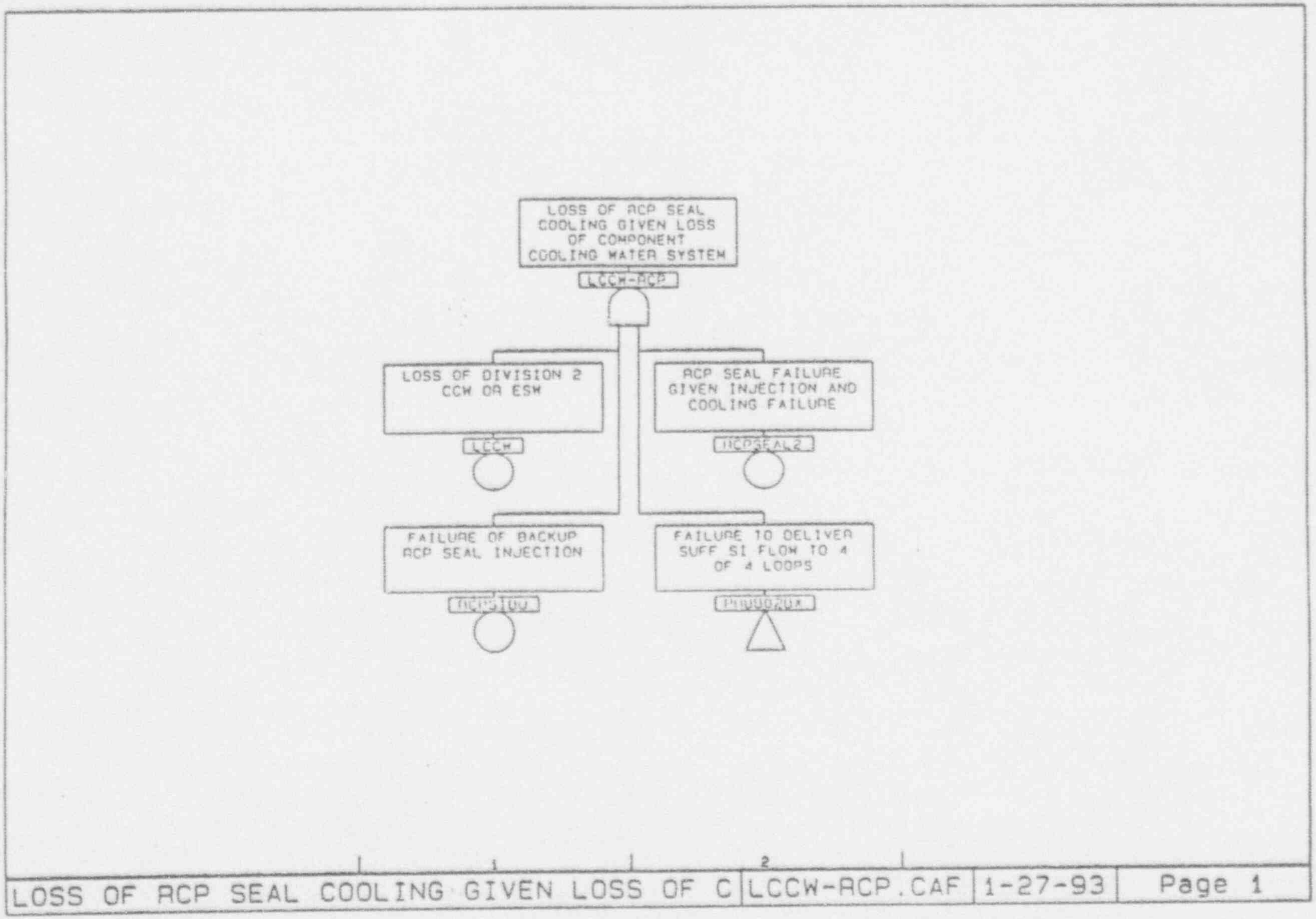


Figure 10-3
Change in Core Damage Frequency Vs Assumed Conditional RCP Seal Failure Rate

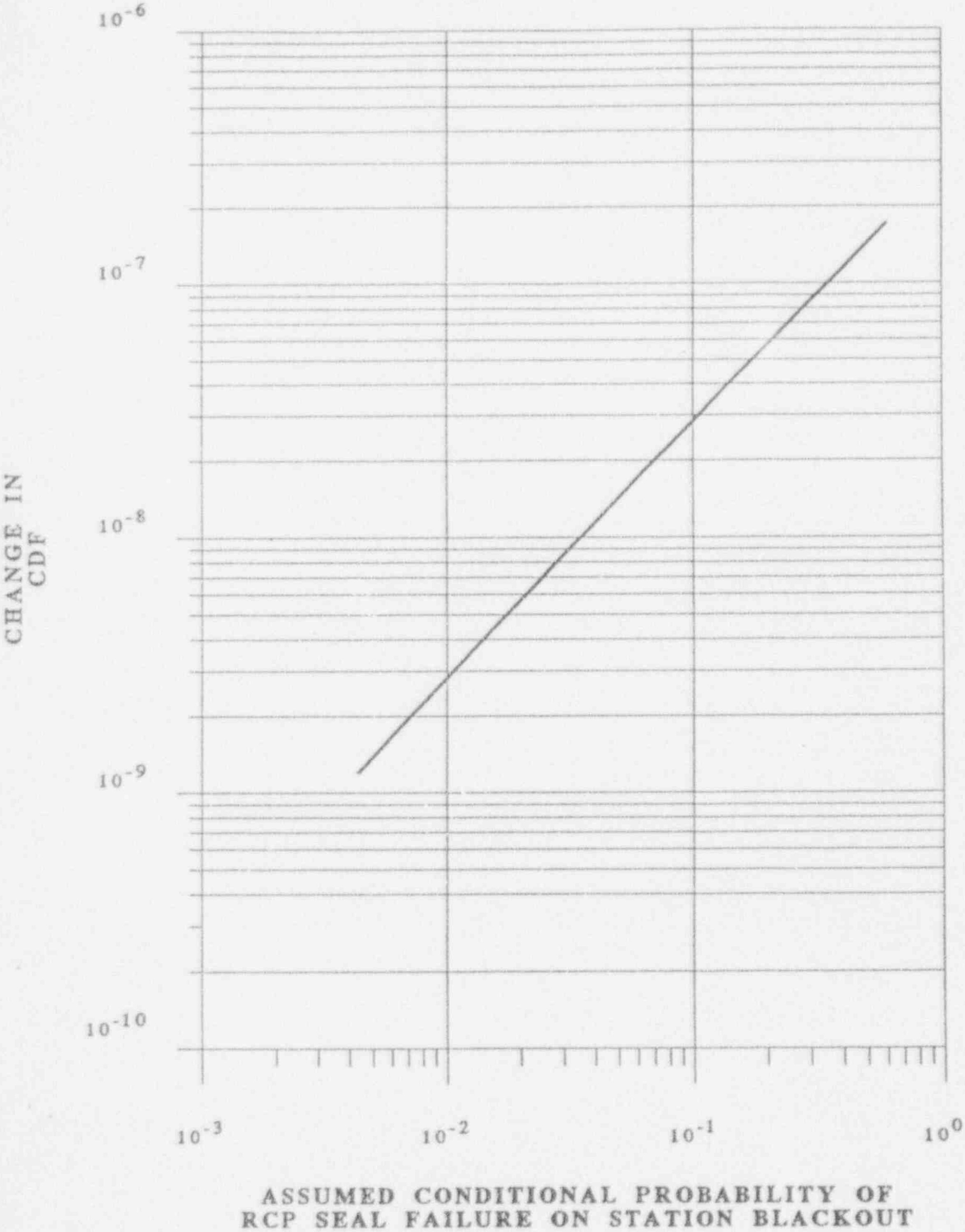
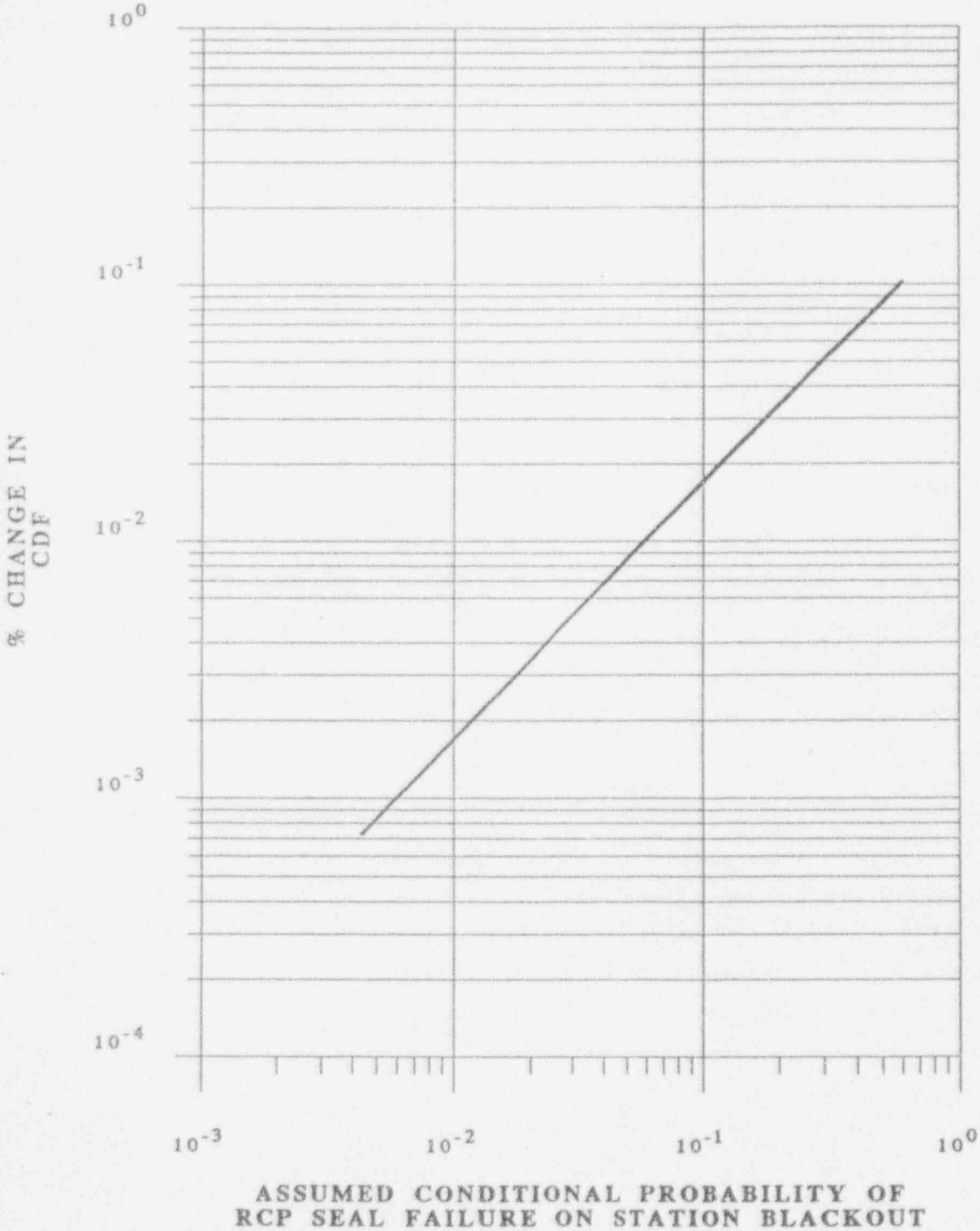


Figure 10-4
Percent Change in Core Damage Frequency Vs
Assumed Conditional RCP Seal Failure Rate



19.14

~~14.0~~ LEVEL II AND LEVEL III SENSITIVITY ANALYSES

In performing the containment response analysis and the consequence analysis, several assumptions were made regarding the progression of severe accident phenomena. Sensitivity analyses were performed to assess the potential impact on the System 80+ design due to certain assumptions regarding the physical processes and phenomena involved in the release of radioactive materials to the environment and the consequences of the release. The sensitivity analyses associated with containment response are presented in Section 14.1 and those associated with the release consequences are presented in Section 14.2.

14.1 CONTAINMENT RESPONSE SENSITIVITY ANALYSES

The results of the "present" or base case analysis of containment response show that the conditional probability for intact containment releases, RC1, is 0.902. Likewise, the conditional probabilities for (1) late containment failure releases, RC2, (2) early containment failure releases, RC3, (3) containment isolation failure releases, RC4, and (4) containment bypass releases, RC5, are 0.076, 0.011, 0.011 and 0.0 respectively. To assess the effects of certain assumptions used in the base case analysis, ten containment response sensitivity analyses were performed. These analyses assessed the impact of specified assumptions on the containment failure modes and the overall conditional containment failure probability. These analyses involved changing certain conditions or assumptions that are modeled in the containment event trees (CETs) and then re-quantifying the CETs to ascertain the impact. Table 14.1-1 summarizes the results of the sensitivity analyses for containment response. The base case results are also included in Table 14.1-1 for comparison purposes. The following sub-sections describe the individual sensitivity analysis and present their results.

14.1.1 Availability of Hydrogen Ignitors

The hydrogen ignitors are provided to prevent the build-up of hydrogen inside the containment following a severe accident. With the ignitors operating, hydrogen is burned at a low enough concentration to prevent a threat to containment

integrity. Because multiple failures of the ignitors in each division must occur to prevent the burning of hydrogen at a low enough concentration, a low probability was assigned to the unavailability of the ignitors. A sensitivity analysis was performed to determine the effects of the hydrogen ignitors being unavailable. For this case the ignitors were assumed to always fail (i.e., a failure probability of 1 was assigned to "IGFAIL"). The CETs were then re-quantified for all PDSs.

The results for this sensitivity case are presented in Table 14.1-1 as case number 1. The results for this case show that the conditional probabilities for the various release classes are unchanged, when compared to the base case. containment would remain intact 90.2% of the time following a severe accident. This implies that the containment response analysis for the System 80+ design is not sensitive to the availability of the hydrogen ignitors.

14.1.2 Containment Characteristics More Favorable to DDT

The System 80+ containment characteristics are considered unfavorable for a deflagration to transition into a detonation. Therefore, for the PRA a low probability value is assigned to the event (DDTOK) in the supporting logic modules that represents deflagration to transition into detonation. Although this type of event is considered to be highly unlikely to impossible for the System 80+ design, a sensitivity analysis was performed to determine the potential impact on the various release classes. To model "DDTOK" as more likely than assumed, the probability of this event was changed from 0.0 or 0.01 to 0.05 for all PDSs except those PDSs where the release point is in the In-containment Refueling Water Storage Tank (IRWST), refer to Table 12.1-1.

The results for this sensitivity case are similar to those for the hydrogen ignitors and are presented in Table 14.1-1 as case number 2. The results show no change in the conditional probabilities for the various release classes. This implies that the System 80+ containment characteristics are not sensitive to deflagration that leads to detonation transition, based on the assumption made in this sensitivity analysis that the deflagration to detonation characteristics are much more likely (by a factor of 5) to occur.

14.1.3 Low Heat Transfer Rate for "WET" Cavity

If the cavity is filled with water, some of the energy from the corium will be transferred to the water and the rest will cause concrete ablation. A high heat transfer rate between the corium and the water enables the corium to be cooled which in turn limits the amount of concrete ablation. However, if the heat transfer rate is low, the containment shell may be penetrated before concrete ablation is terminated. To assess the effects of low heat transfer rate from the corium to the cavity water, a sensitivity analysis was performed. For this case, it was assumed that the probability of the heat transfer rate is twice as low as the value assumed for the base case analysis of the PRA. To model this decrease in the heat transfer rate, the probability that the heat transfer rate is low enough for the corium to penetrate the containment shell was increased from 0.01 to 0.02 for all PDSs associated with "WET" cavity.

The sensitivity results for this case are presented in Table 14.1-1 as case number 3. After re-quantifying the CETs for this sensitivity case, the results show that the conditional probability for intact containment releases, RC1, decreased slightly from 0.902 to 0.890. Conversely, the conditional probability for late containment failure releases, RC2, increased from 0.076 to 0.088. This corresponds to an increase in the conditional probability for release class RC2 of approximately 15%. The conditional probabilities for release classes RC3, RC4, and RC5 are shown to remain unchanged. With such an increase in the conditional probability for release class RC2, late containment failure of the System 80+ design is somewhat sensitive to the ability to transfer heat from the corium to the cavity water.

14.1.4 Cavity Not Always Fill with Water

As discussed in the previous sensitivity case in Section 14.1.3, a high heat transfer rate from the corium to the water covering the debris is beneficial. This allows cooling of the corium and the eventual termination of concrete ablation prior to the containment shell being penetrated. The base case of the PRA assumes that the cavity is always full for PDSs associated with "WET" cavity. This sensitivity case assess the impact on the conditional probabilities for the

release classes if the cavity had a 50% chance of being full, instead of always being full as assumed in the base case analysis. To determine the impact, the probability of "WETCAVITY" was changed from 1.0 (cavity is always full) to 0.5 and then the CETs were re-quantified.

The sensitivity results for this case are presented in Table 14.1-1 as case number 4. The results show that the conditional probability for intact containment releases, RC1, decreased from 0.902 to 0.450 and conversely, the conditional probability for late containment failure releases, RC2, increased from 0.076 to 0.533. The conditional probabilities for the other release classes (RC3, RC4 and RC5) remained unchanged. The results of this case implies that the late containment failure probability would increase by a factor of approximately 7 if there is a 50% chance that the cavity is not filled with water. This implies that late containment failure is very sensitive to the amount of water discharged to the cavity by the Cavity Flooding System following a severe accident.

14.1.5 Recovery of Containment Heat Removal Function

The containment spray system is the preferred system used to remove heat from the containment during an accident. MAAP analyses have shown that it would take approximately 48 hours to over-pressurize the containment for those accident sequences which include failure of containment heat removal. Based on the time it takes to over-pressurized the containment and the events that cause the loss of containment sprays, the PRA assumed that it is possible to recover the containment sprays before the containment fails. Failures of containment spray equipment inside the containment are assumed to be non-recoverable during a severe accident. The System 80+ design includes provisions for connecting an external source of water to one containment spray header via a standpipe and flanged connection to the containment spray line near the containment penetration. Flow would be provided by a skid mounted pumping device which is independent of station power. This backup system would be used to provide spray flow given any failure of containment spray equipment outside the containment. The assigned conditional non-recovery probability of the containment spray backup system is 0.1 for the base case analysis. This implies that 90% of the time the

containment spray backup system would perform its function when required. To assess the effects of containment spray being less likely to be recovered, a sensitivity analysis was performed. For this sensitivity analysis, the conditional probability of not recovering containment spray was doubled (i.e., the probability increased for 0.1 to 0.2) to represent the containment spray less likely to be recovered when required.

The sensitivity results for this case are presented in Table 14.1-1 as case number 5. After re-quantifying the CETs for this case, the results show that the conditional probability for intact containment releases, RC1, would decrease form 0.902 to 0.895. Conversely, the conditional probability for late containment failure releases, RC2, would increase from 0.076 to 0.083. The conditional probability for the other types of releases (RC3, RC4, and RC5) would remain unchanged. The change in the conditional probability for release class RC2 represents a increase of approximately 9%. This implies that late containment failure releases are somewhat sensitive to the reliability of the emergency containment heat removal system and the recovery of containment heat removal following a severe accident.

14.1.6 Induced Failure of P S Piping

High temperature steam and high pressure steam circulates through the RCS following a severe accident with high pressure core damage sequences. Research has indicated that this condition may induce a temperature related creep rupture of the RCS pressure boundary prior to vessel failure. The likely locations of such breach include the hot leg and the pressurizer surge line. The PRA assumes that temperature induced failure of the RCS hot leg or surge line is likely to occur in the System 80+ design because of the high level of zirconium per thermal megawatt of power and the type of material (carbon steel) used to construct the hot leg. Because of the likelihood of failure, a probability of 0.65 was assigned to "HSINTACT" which represents temperature induced failure of the RCS piping. To determine the potential impact of this assumption on the conditional probabilities of the release classes, two sensitivity cases were performed. The first case assumes that a temperature induced creep failure of the RCS piping would always occur and is modeled by changing the value of "HSINTACT" from 0.65

to 1.0. The second case is the opposite, and assumes that a temperature induced creep failure of the RCS piping would never occur. For the second case, the value of "HSINTACT" is changed from 0.65 to 0.0. Note that because temperature induced creep failure of the RCS piping involves high pressure core damage sequences, the value of "HSINTACT" is changed for those PDSs with RCS leak rate occurring due to cycling of the relief valves and RCS pressure being high.

The sensitivity results for the temperature induced creep failure of the RCS piping are presented in Table 14.1-1 as cases 6A and 6B, respectively. The results for both these cases show that conditional probabilities for all the release classes remained unchanged and the System 80+ design is insensitive to temperature induced creep failure of RCS piping.

14.1.7 Depressurization of RCS by SDS

It is advantageous to depressurize a high or medium pressure accident sequence to below the 250 psig target threshold prior to vessel breach. The primary purpose of this action is to reduce the potential threat of direct containment heating. In addition, this action would also reduce the potential for rocket induced failure of the containment and reduce the potential for containment failure induced by cavity collapse resulting from high pressure melt eject loadings. The Safety Depressurization System (SDS) is modeled as the primary means of depressurizing the RCS. In the PRA Level I analysis it was shown that failure of SDS is dominated by operator error. Although the onset of core damage would occur if the SDS was not initiated, there was still sufficient time to depressurize the RCS prior to vessel breach. Consequently, it was assumed that there is an 80% chance the operator would open the SDS valves in time to depressure RCS during core damage. A value of 0.2 was therefore assigned to "NOSDSDP" which represents failure of the SDS to depressurize the RCS following a severe accident. To asses what effects, if any, this assumption may have on the release classes, two sensitivity cases were performed. The first case assumed that the SDS would always depressurize the RCS for PDSs with high RCS pressure and a leak rate via the cycling of the relief valves. For this case, the value of "NOSDSDP" was changed from 0.2 to 0.0. The second case assumed that the SDS would never depressurize the RCS and consequently the value of "NOSDSDP"

was changed from 0.2 to 1.0.

The sensitivity results for depressurizing the RCS are presented in Table 14.1-1 as cases 7A and 7B, respectively. The results for both these cases show that conditional probabilities for all the release classes remained unchanged and the System 80+ design is insensitive to RCS depressurization using the SDS for high pressure core damage accident sequences.

14.1.8 Containment Isolation

The detailed containment penetration designs for the electrical cables, personnel airlocks, equipment hatch, and fuel transfer tube were not available for evaluation during the period when the PRA was being performed. An estimate probability of 2.1E-03 was therefore used for failure to isolate the containment. This estimate probability was based on prior PRAs and historical data. To assess the impact of containment isolation reliability on the various release classes for the System 80+ design, a sensitivity analysis that assumed a less reliable containment isolation system was performed. For this case, the containment isolation system was assumed to be less reliable than the base case by a factor of 5. The more unreliable containment isolation system was reflected in the model by changing the value of "ISOL" from 2.1E-03 to 1.0E-02.

The sensitivity results for this case are presented in Table 14.1-1 as case number 8. The results show that the conditional probability for intact releases, RC1, decreased from 0.902 to 0.893 which is approximately 1%. For the late containment failure releases, RC2, the conditional probability increased from 0.076 to 0.078. This increase is approximately 3%. Release classes RC3 and RC5 remained unchanged. Of all the release classes, containment isolation failure releases, RC4, was affected the most. The conditional probability for this release class changed from 0.011 to 0.017 or an increase of approximately 55%. This implies that the containment isolation failure releases are extremely sensitive to the reliability of the containment isolation system for the System 80+ design.

TABLE 14.1-1

SUMMARY OF CONTAINMENT RESPONSE SENSITIVITY ANALYSIS
RESULTS FOR SYSTEM 80+

CASE No.	DESCRIPTION	MODELED AS	CONDITIONAL PROBABILITY OF RELEASE CLASS				
			RC1	RC2	RC3	RC4	RC5
BASE	AS DESCRIBED IN THE PRA	AS MODELED IN THE PRA	0.902	0.076	0.011	0.011	0.0
1	H_2 IGNITORS UNAVAILABLE	CHANGE PROBABILITY OF "IGFAIL" FROM CURRENT VALUE TO 1.0 OF ALL PDSs.	0.902	0.076	0.011	0.011	0.0
2	CONTAINMENT CHARACTERISTICS FAVOR DEFLAGRATION TO DETONATION TRANSITION	INCREASE THE PROBABILITY OF "DDTOK" FROM 0.0 OR 0.01 TO 0.05 FOR ALL PDSs.	0.902	0.076	0.011	0.011	0.0
3	LOW HEAT TRANSFER FROM CORIUM TO CAVITY WATER	INCREASE THE PROBABILITY OF "LOWHTXFER" FROM 0.01 TO 0.02 FOR ALL PDSs ASSOCIATED WITH "WET" CAVITY.	0.890	0.088	0.011	0.011	0.0
4	INSUFFICIENT HEAT TRANSFER FROM CORIUM TO CAVITY WATER	CHANGE THE PROBABILITY OF "WETCAVITY" FROM 1.0 TO 0.5 FOR ALL PDSs ASSOCIATED WITH "WET" CAVITY.	0.445	0.538	0.006	0.011	0.0
5	CONTAINMENT HEAT REMOVAL LESS LIKELY TO BE RECOVERED	INCREASE THE PROBABILITY OF "NCHRECOV" FROM 0.1 TO 0.2.	0.895	0.083	0.011	0.011	0.0
6A	INDUCED FAILURE OF RCS PIPING ALWAYS OCCUR	CHANGE THE PROBABILITY OF "HSINTACT" FROM 0.65 TO 1.0 FOR PDSs WITH RCS LEAK RATE = "CRV" AND RCS PRESSURE = "HIGH"	0.902	0.076	0.011	0.011	0.0
6B	INDUCED FAILURE OF RCS PIPING NEVER OCCURS	CHANGE THE PROBABILITY OF "HSINTACT" FROM 0.65 TO 0.0 FOR PDSs WITH RCS LEAK RATE = "CRV" AND RCS PRESSURE = "HIGH"	0.902	0.076	0.011	0.011	0.0

TABLE 14.1-1 (Cont'd)

SUMMARY OF CONTAINMENT RESPONSE SENSITIVITY ANALYSIS
RESULTS FOR SYSTEM 80+

CASE No.	DESCRIPTION	MODELED AS	CONDITIONAL PROBABILITY OF RELEASE CLASS				
			RC1	RC2	RC3	RC4	RC5
7A	THE RCS IS NOT DEPRESSURIZED BY THE SDS	CHANGE PROBABILITY OF "NOSDSDP" FROM 0.2 TO 1.0 FOR PDSs WITH RCS LEAK RATE = "CRV" AND RCS PRESSURE = "HIGH".	0.902	0.076	0.011	0.011	0.0
7B	THE RCS IS DEPRESSURIZED BY THE SDS	CHANGE PROBABILITY OF "NOSDSDP" FROM 0.2 TO 0.0 FOR PDSs WITH RCS LEAK RATE = "CRV" AND RCS PRESSURE = "HIGH".	0.902	0.076	0.011	0.011	0.0
8	CONTAINMENT IS LESS LIKELY TO BE ISOLATED	CHANGE PROBABILITY OF "ISOL" FROM 2.1E-03 TO 1.0E-02.	0.893	0.078	0.011	0.017	0.0

14.2 SENSITIVITY ANALYSES OF RELEASE CONSEQUENCES

The base case results for the consequence analysis show that the NRC's goals for large releases are met with substantial margins. The risks associated with large releases are measured by dose at distance, early fatalities, and late fatalities. The base case results show that the probabilities of exceeding a whole-body dose of 25 Rems at 300 meters and one-half mile from the reactor site boundary are 2.3E-08 and 1.9E-08 per year, respectively. In both cases, the probability of exceeding these risk measures is much less than the NRC's goal of 1.0E-06 per year. The base case results also show that the probabilities of exceeding one prompt and one late fatality are 2.1E-09 and 2.0E-07 per year, respectively. These risk measures for the System 80+ design are also within the NRC's goal of 1.0E-06 per year. To assess the effects of certain assumptions on the base case results for the risk measures, several sensitivity analyses were performed. Table 14.2-1 summarizes the results of the sensitivity analyses for the risk consequences, and the associated Complementary Cumulative Distribution Functions (CCDFs) for the sensitivity cases are presented in Figures 14.2-1 through 14.2-28. It should be noted that although certain sensitivity cases show increase in the probability of exceeding the risk measures, the System 80+ design would still meet the NRC's goals for large releases. The following sub-sections describe the individual sensitivity analysis and present their results.

14.2.1 Location of Release Point

The location of the releases is one of the parameters used to characterize a release classes. In the base case analysis, it is assumed that releases to the environment caused by containment over-pressure sequences occur at 54 feet above grade level (i.e., at the equipment and personnel hatches at elevation 146 ft.). For containment bypass failures and failures due to melt-through into the subsphere region, the releases pass through the auxiliary building and eventually to the environment at grade level. Releases caused by a steam generator tube rupture and a stuck open main steam safety valve is assumed to occur at 64 ft. above grade. The other isolation failure releases are assume to occur at 54 ft. above grade. As indicated above, the release point varies for the different release classes addressed in the base case analysis. To assess the impact of the

release point on the risk measures, two sensitivity cases were performed. The first case assumed that all releases to the environment occurred at the top of the containment building which is approximately 173 ft. or 52.8 meters. The results for this case is presented in Table 14.2-1 as case No. 1A. The second case that assessed the impact of the release point assumed that all releases to the environment occurred at grade level. The results for this case is presented in Table 14.2-1 as case No. 1B.

The CCDFs for the four risk measures of case 1A are presented in Figures 14.2-1 through 14.2-4. Similarly, the CCDFs for case 1B are presented in Figures 14.2-5 through 14.2-8.

The results shown in Table 14.2-1 for cases 1A and 1B indicate that if all the releases occurred at the top of the containment building or at grade level it would have no impact on the probability of exceeding the dose at one-half mile, or on the probability of exceeding one latent fatality. For these risk measures, the probabilities of exceedance would be the same as the base case, whether the releases occurred at the top of the containment or at grade level.

The results show that the whole-body dose at 300 meters is less likely to be exceeded, when compared to the base case results. The probability of exceeding the dose at 300 meters decreased for releases that occurred at the top of the containment building and at grade level. If the releases occurred at the top of the containment building, the probability of exceeding the 25 Rem dose at 300 meters decreased from 2.3E-08 to 2.0E-08 (approximately 13%). Likewise, if the releases occurred at grade level, the probability of exceeding the 25 Rem dose at 300 meters also decreased from 2.3E-08 to 1.9E-08 (approximately 17%). Based on these percentages, the risk measure for dose at 300 meters is relatively insensitive to the location of the releases when compared with the base case.

The probability of exceeding one early fatality decreased for releases that occurred at the top of the containment and for releases that occurred at grade level, as shown in Table 14.2-1. If the releases occurred at the top of the containment, the probability of exceeding one early fatality would decrease from 1.6E-09 to 7.0E-10 (approximately 56%). If the releases occurred at grade level,

the probability of exceeding one early fatality would actually increase slightly from 1.6E-09 to 1.7E-09 (approximately 6%). This implies that early fatalities would be sensitive to all releases which occurred at the top of the containment and relatively insensitive to all releases which occurred at grade level.

14.2.2 Iodine and Cesium Release Fractions

The isotopic content of the release for each release class was calculated using S80SOR. This code, a modified version of the ZISOR code, reflects the System 80+ design features. One or more PDSs were selected for each release class, and a S80SOR run was made to calculate the source term isotopic content for the specified release class. The source term isotopic content is the weighted average of the source terms of the PDSs selected to characterize the release class. This sensitivity case was performed to assess the impact on the overall plant risk due to an increase in the release fractions. For this case, the release fractions were increased by an order of magnitude except those release classes in which the total fraction would exceed 100%. For the exceptions, the release fractions were increased by a factor of 2. The results for this case are presented in Table 14.2-1 as case No. 2. The CCDFs for the four risk measures are presented in Figures 14.2-9 through 14.2-12.

The results for this case, as shown in Table 14.2-1, indicate that the probabilities of exceeding all four risk measures would increase, with the probability of exceeding one early fatality most affected. Although the risk measures increased when compared with the base case, the NRC's goals for large releases would still be met. The probability of exceeding the 25 Rem whole-body dose at 300 meters increased by a factor of 1.6 (from 2.3E-08 to 3.6E-08). Likewise, the probability of exceeding the 25 Rem whole-body dose at one half mile increased by a factor of 1.3 (from 1.8E-08 to 2.3E-08). The probability of exceeding one early fatality and one latent fatality increased by a factor of 2.7 and 1.1, respectively. The probability increased from 1.6E-09 to 4.3E-09 for one early fatality. For one latent fatality, the probability increased from 1.9E-07 to 2.1E-07. The results for this case imply that all four risk measure are sensitive to the isotopic content or release fraction that is used to characterized the release class.

14.2.3 Containment Bypass Releases Unscrubbed

Containment bypass releases are characterized by an interfacing system LOCA via the suction line of the Shutdown Cooling System (SCS). The release path is through the broken SCS line into the subsphere region of the auxiliary building. This region will be flooded to a depth of 4 to 8 ft of water and the releases from the RCS are therefore subject to scrubbing. The amount of scrubbing depends on the depth of water in the flooded region. This case was performed to determine the effects on the risk measures if the containment bypass releases were not scrubbed. The results for this case are presented in Table 14.2-1 as case No. 3. The CCDFs for the four risk measures are presented in Figures 14.2-13 through 14.2-16.

The results for this case, as shown in Table 14.2-1, indicate that the probability of exceeding the 25 Rem whole-body dose at one-half mile and the probability of exceeding one latent fatality would remain unchanged. The table also shows that the probability for the whole-body dose at 300 meters would decreased from 2.3E-08 to 2.2E-08 (approximately 4%). The probability of exceeding one latent fatality would increase slightly from 1.6E-09 to 1.8E-09 (approximately 13%). These results imply that there would be an insignificant increase in latent fatalities if containment bypass releases were unscrubbed prior to release to the environment.

14.2.4 Reliability of Containment Isolation Systems

Containment isolation failure is characterized by two types of scenarios. The first type of scenarios involves a SGTR with a stuck open secondary side valve, and the second type of scenarios involves core damage accident sequences in combination with direct loss of the containment isolation function. Releases due to containment isolation failure are influenced by the availability of liquid on the secondary side of the steam generator and the availability of containment spray. This case was performed to assess the impact of the reliability of containment isolation systems on risk. In performing the sensitivity analysis for this case, the probability of containment isolation failure was increased by an order of magnitude. The results for this case are presented in Table 14.2-1

as case No. 4. The CCDFs for the associated risk measures are presented in Figures 14.2-17 through 14.2-20.

The results for this case, as shown in Table 14.2-1, indicate that the probabilities for all of the four risk measures would increase if the failure probability of the containment systems increased. The probability for whole-body dose at 300 meters would increase by a factor of 8 approximately, from 2.3E-08 to 1.9E-07. For whole-body dose at one-half mile, the probability of exceeding 25 Rem would increase by 9 approximately. Similarly, the probabilities of exceeding one early and one latent fatality would increase by a factor of 10 and 2, respectively. This implies that a less reliable containment isolation function would increase the overall risk of the plant.

14.2.5 Basemat Melt-through

Basemat melt-through refers to the process of concrete decomposition and destruction associated with corium interacting with the reactor cavity basemat. This type of accident progression is slow and provided the corium melts through to the containment subsoil, the corium releases to the environment is expected to be negligible for the System 80+ design. Because of the low credibility of this scenario, a low probability of occurrence is assigned to basemat melt-through. This sensitivity case was performed to determine the impact on risk if a basemat melt-through event were to occur more frequently than what is currently assumed in the base case analysis. For this case, the probability of basemat melt-through was increased by 10%. The results for this case are presented in Table 14.2-1 as case No. 5. The corresponding CCDFs for the four risk measures are presented in Figures 14.2-20 through 14.2-24.

The results for this case, as shown in Table 14.2-1, indicate that the probabilities of exceeding the dose at distance and the early fatalities remained unchanged. The probability of exceeding one latent fatality would increase by a factor of 2 approximately. This implies that if the likelihood of basemat melt-through increased then the risk associated with latent fatalities for the System 80+ design would be affected the most.

14.2.6 Interfacing System LOCA

An interfacing system LOCA is the loss of RCS inventory outside the containment via a low pressure system which interfaces with the RCS. The releases associated with this type of event include vaporization releases with no scrubbing inside the containment atmosphere prior to the release but with the source term attenuated due to deposition in the auxiliary building. The frequency for this release class was estimated to be 5.1E-10 per year. This low frequency is attributed to the various design enhancements that are incorporated in the Safety Injection System and the Shutdown Cooling System. In the base case analysis, releases associated with interfacing system LOCA (i.e., containment bypass releases) were identified as insignificant contributors to the overall risk of the plant. This sensitivity case was performed to assess the impact on the overall risk of the plant if interfacing system LOCA occurred more frequently than what is calculated in the base case analysis. For this case, the frequency of interfacing system LOCA was increased by two orders of magnitude. The results for this case are presented in Table 14.2-1 as case No. 6. The corresponding CCDFs for the four risk measures are presented in Figures 14.2-25 through 14.2-30.

The results for this case, as shown in Table 14.2-1, indicate that the overall risk of the plant would increase. The probabilities of exceeding the dose at distance risk measures increased by a factor of 3 for both the 25 Rem whole-body dose at 300 meters and the 25 Rem whole-body dose at one-half mile from the reactor site boundary. The increase in probabilities for the early and latent fatalities was not as large as the increase in probabilities for the dose at distance. The probabilities of exceeding one early fatality and one latent fatality both increased by a factor of approximately 1.3.

TABLE 14.2-1

SUMMARY OF SENSITIVITY RESULTS OF RISK CONSEQUENCES FOR SYSTEM 80+

CASE No.	DESCRIPTION	PROBABILITY OF EXCEEDING			
		25 Rems @ 300 M	25 Rems @ ½ Mi	1 EARLY FATALITY	1 LATENT FATALITY
BASE	AS DESCRIBED IN SECTION 13	2.3E-08	1.8E-08	1.6E-09	1.9E-07
1A	RELEASES OCCUR AT THE TOP OF THE CONTAINMENT (173.2 ft OR 52.8 M)	2.0E-08	1.8E-08	7.0E-10	1.9E-07
1B	RELEASES OCCUR AT GRADE LEVEL	1.9E-08	1.9E-08	1.7E-09	1.9E-07
2	INCREASED IODINE AND CESIUM RELEASE FRACTIONS	3.6E-08	2.3E-08	4.3E-09	2.1E-07
3	CONTAINMENT BYPASS RELEASES UNSCRUBBED	2.2E-08	1.8E-08	1.8E-09	1.9E-07
4	CONTAINMENT ISOLATION SYSTEM IS LESS RELIABLE	1.9E-07	1.7E-07	1.6E-08	3.7E-07
5	BASEMAT MELT-THROUGH OCCURS MORE FREQUENTLY	2.3E-08	1.8E-08	1.6E-09	3.2E-07
6	ISLOCA OCCURS MORE FREQUENTLY	7.2E-08	6.2E-08	2.0E-09	2.4E-07

FIGURE 14.2-1

CCDF FOR DOSE @ 300 METERS FOR CASE 1A

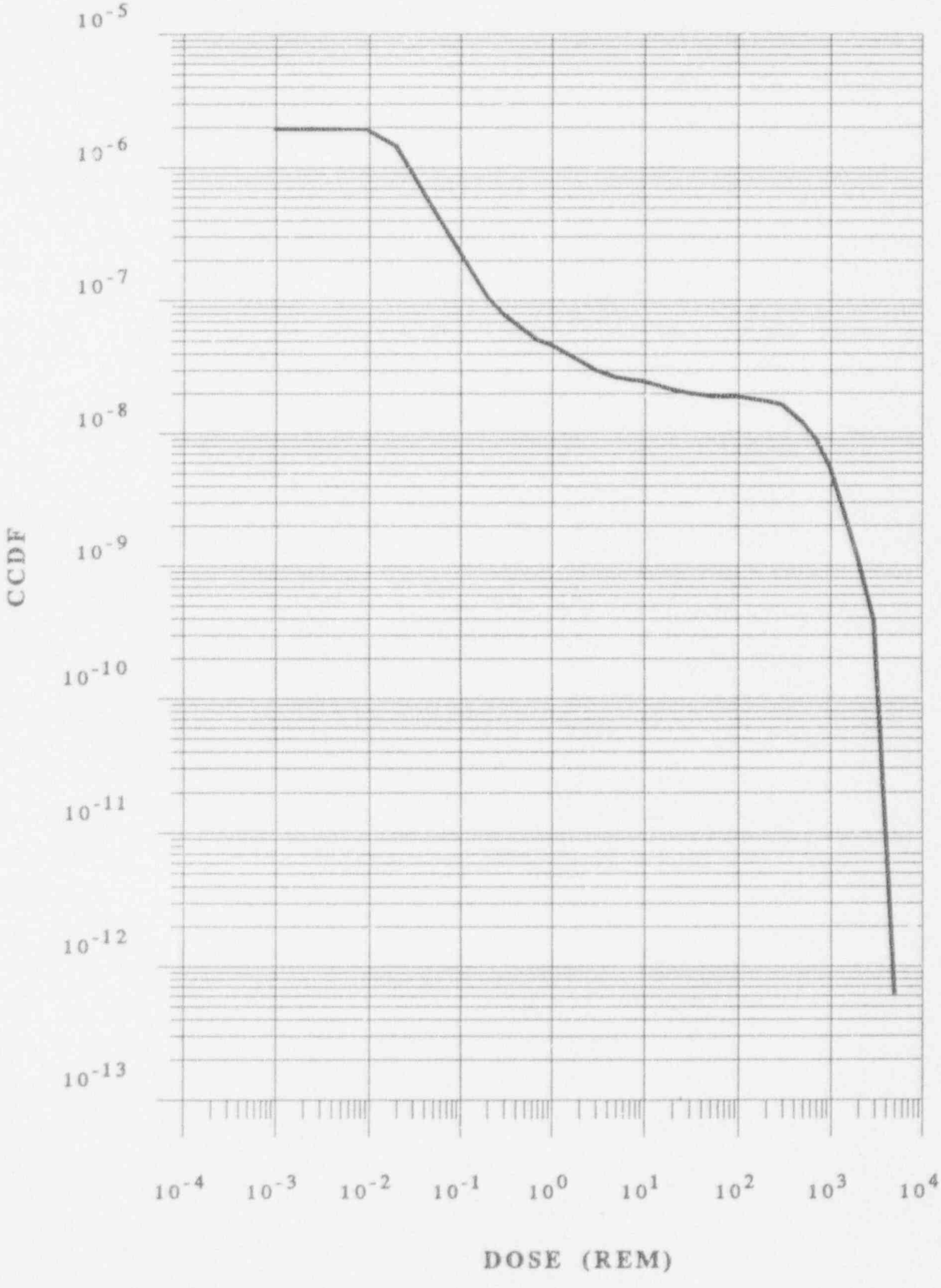


FIGURE 14.2-2

CCDF FOR DOSE @ 1/2 MILE FOR CASE 1A

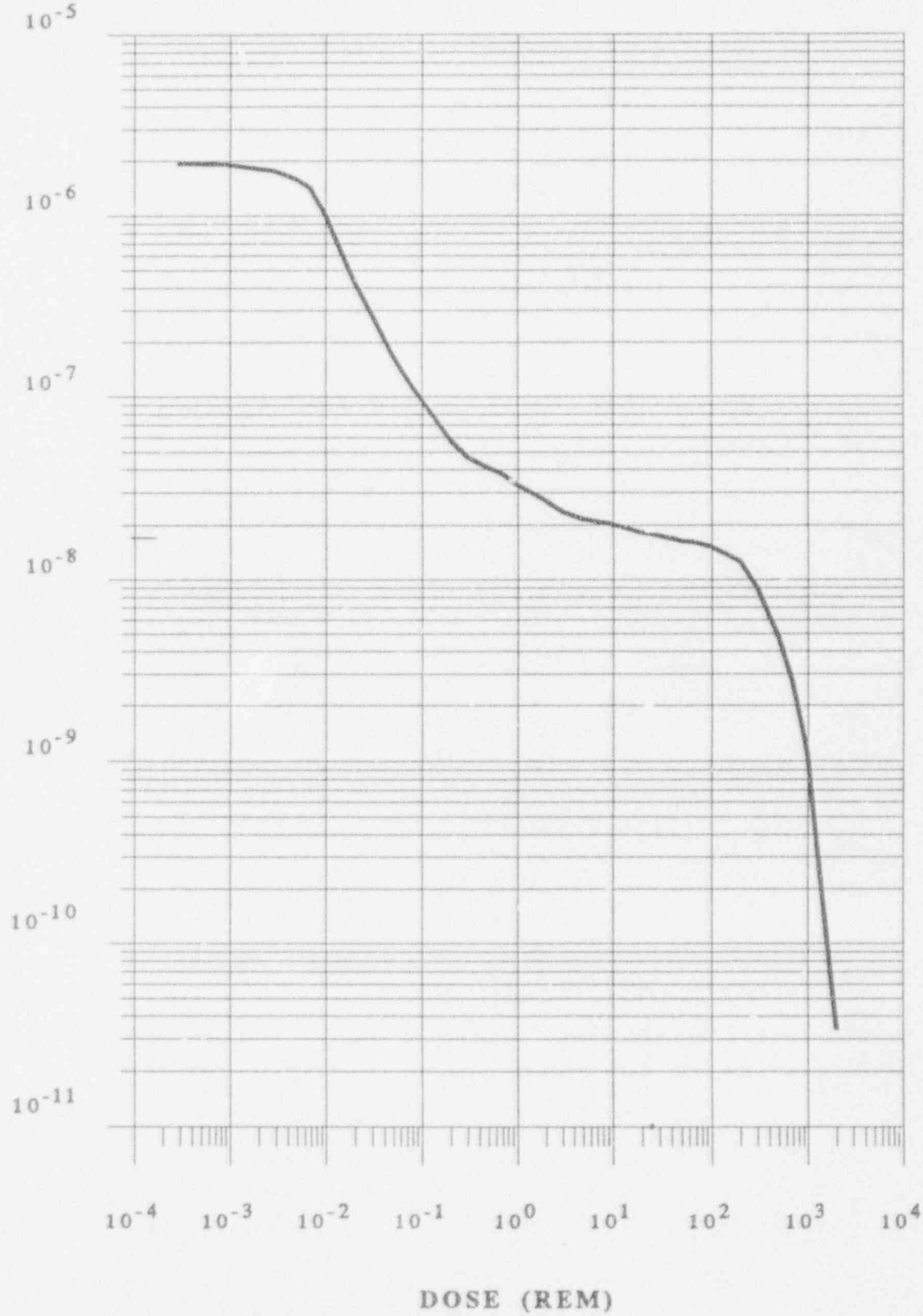


FIGURE 14.2-3

PROBABILITY OF EXCEEDANCE FOR EARLY FATALITY FOR CASE 1A

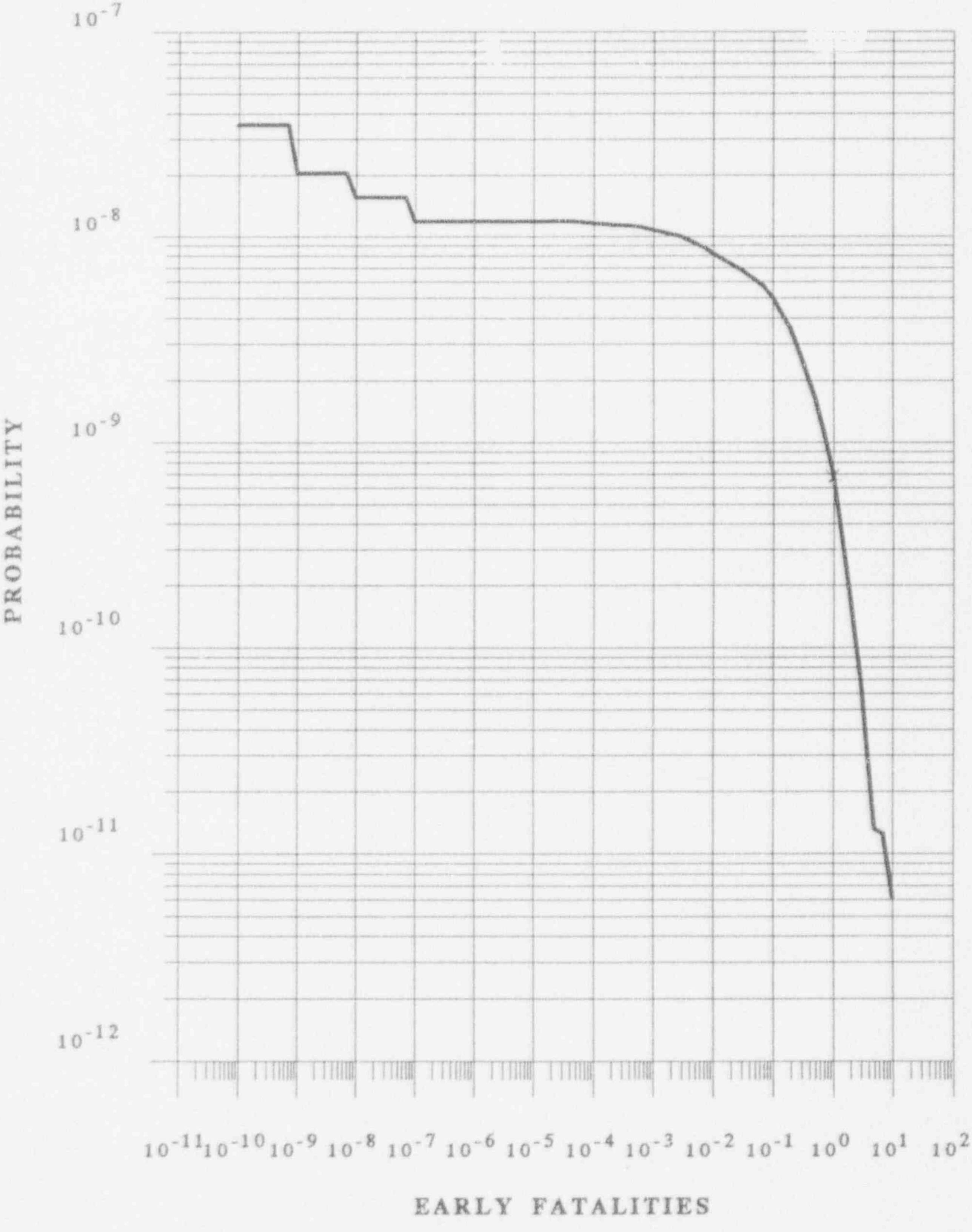


FIGURE 14.2-4

PROBABILITY OF EXCEEDANCE FOR LATENT FATALITY FOR CASE 1A

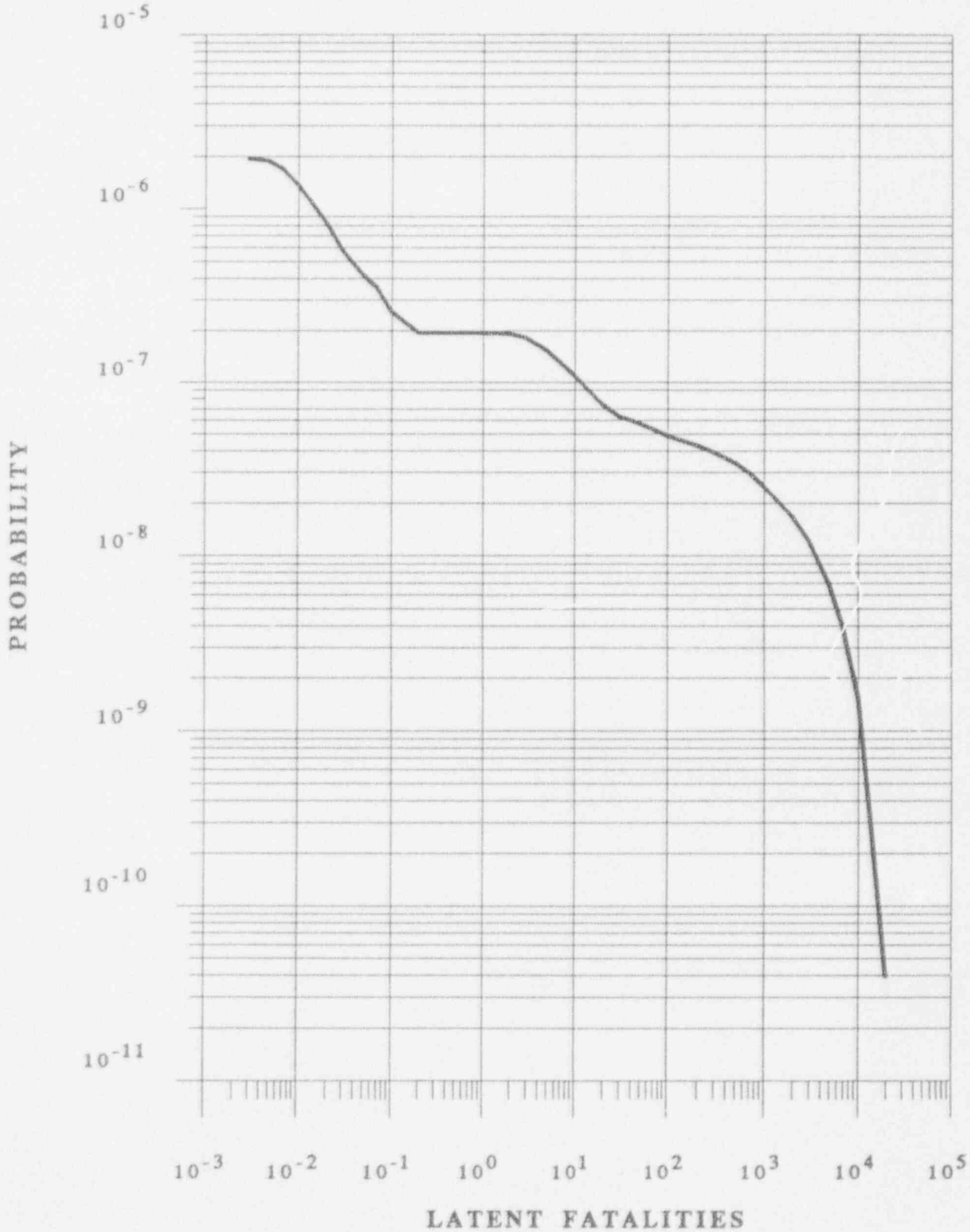
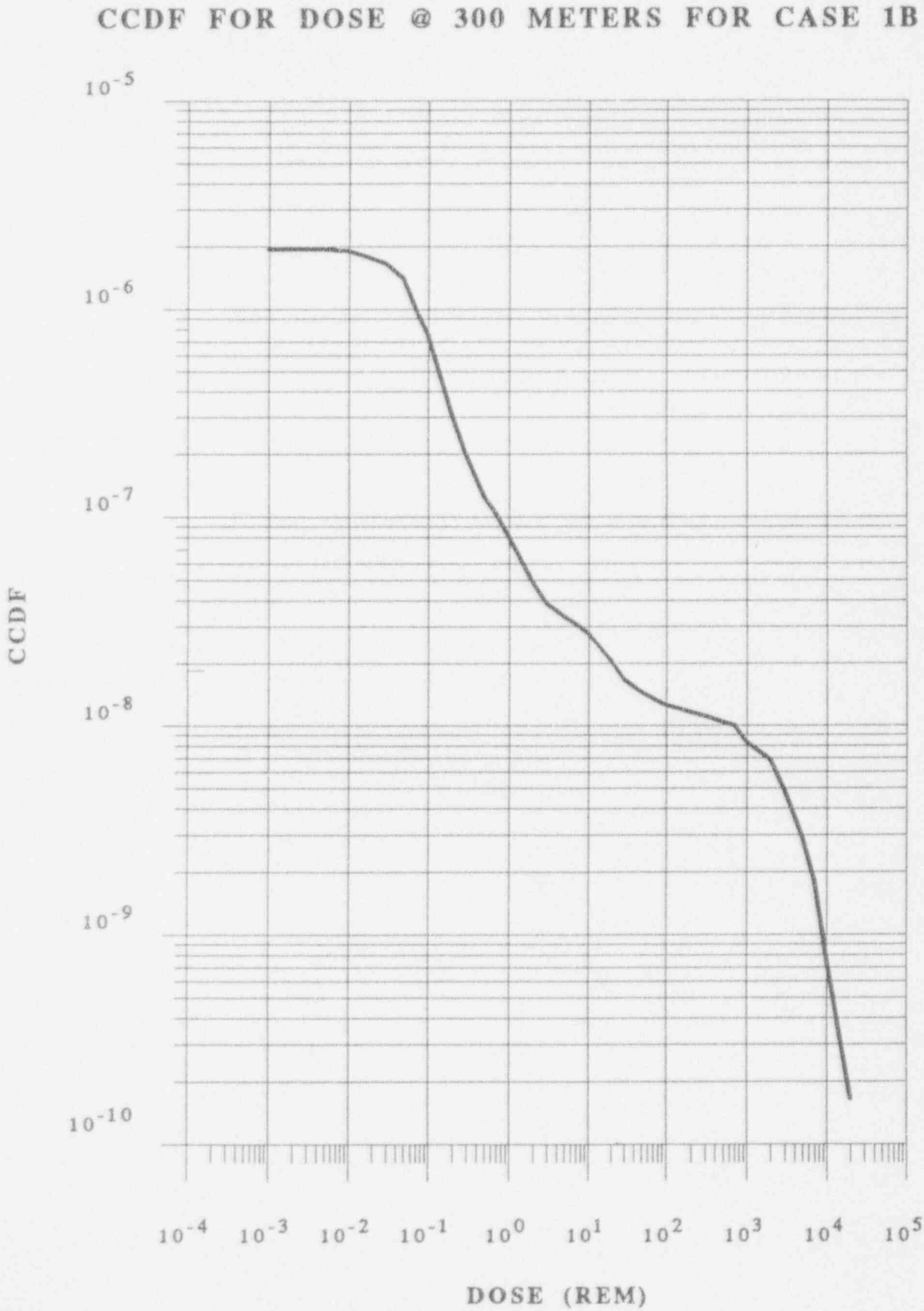


FIGURE 14.2-5



CCDF FOR DOSE @ 300 METERS FOR CASE 1B
10^{-5}
10^{-6}
10^{-7}
10^{-8}
10^{-9}
10^{-10}
CCDF
10^{-4} 10^{-3} 10^{-2} 10^{-1} 10^{0} 10^{1} 10^{2} 10^{3} 10^{4} 10^{5}
DOSE (REM)

FIGURE 14.2-6

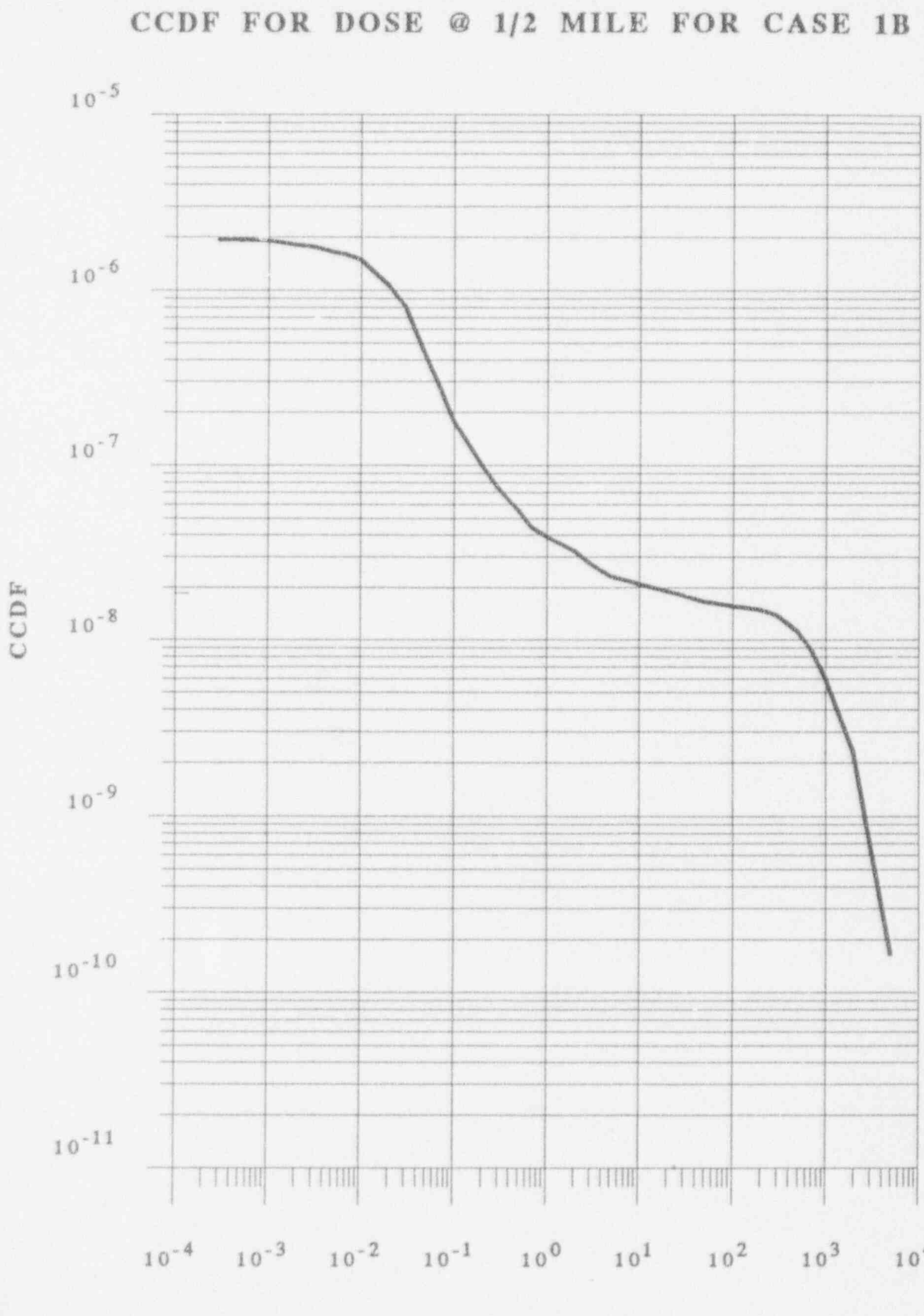

CCDF FOR DOSE @ 1/2 MILE FOR CASE 1B
10^{-5}
10^{-6}
10^{-7}
10^{-8}
10^{-9}
10^{-10}
10^{-11}
CCDF
10^{-4} 10^{-3} 10^{-2} 10^{-1} 10^{0} 10^{1} 10^{2} 10^{3} 10^{4}
DOSE (REM)

FIGURE 14.2-7

PROBABILITY OF EXCEEDANCE FOR EARLY FATALITY FOR CASE 1B

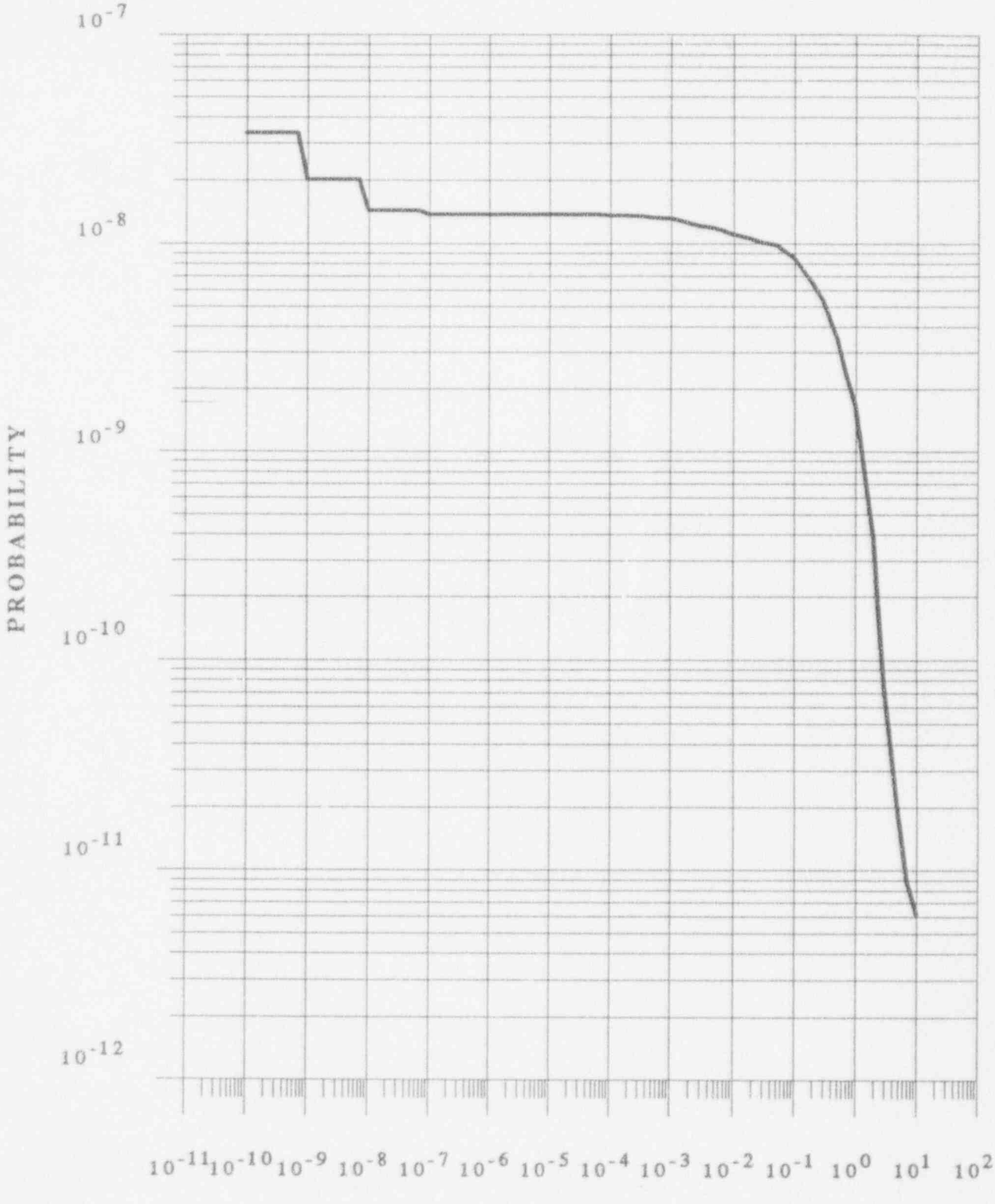


FIGURE 14.2-8

PROBABILITY OF EXCEEDANCE FOR LATENT FATALITY FOR CASE 1B

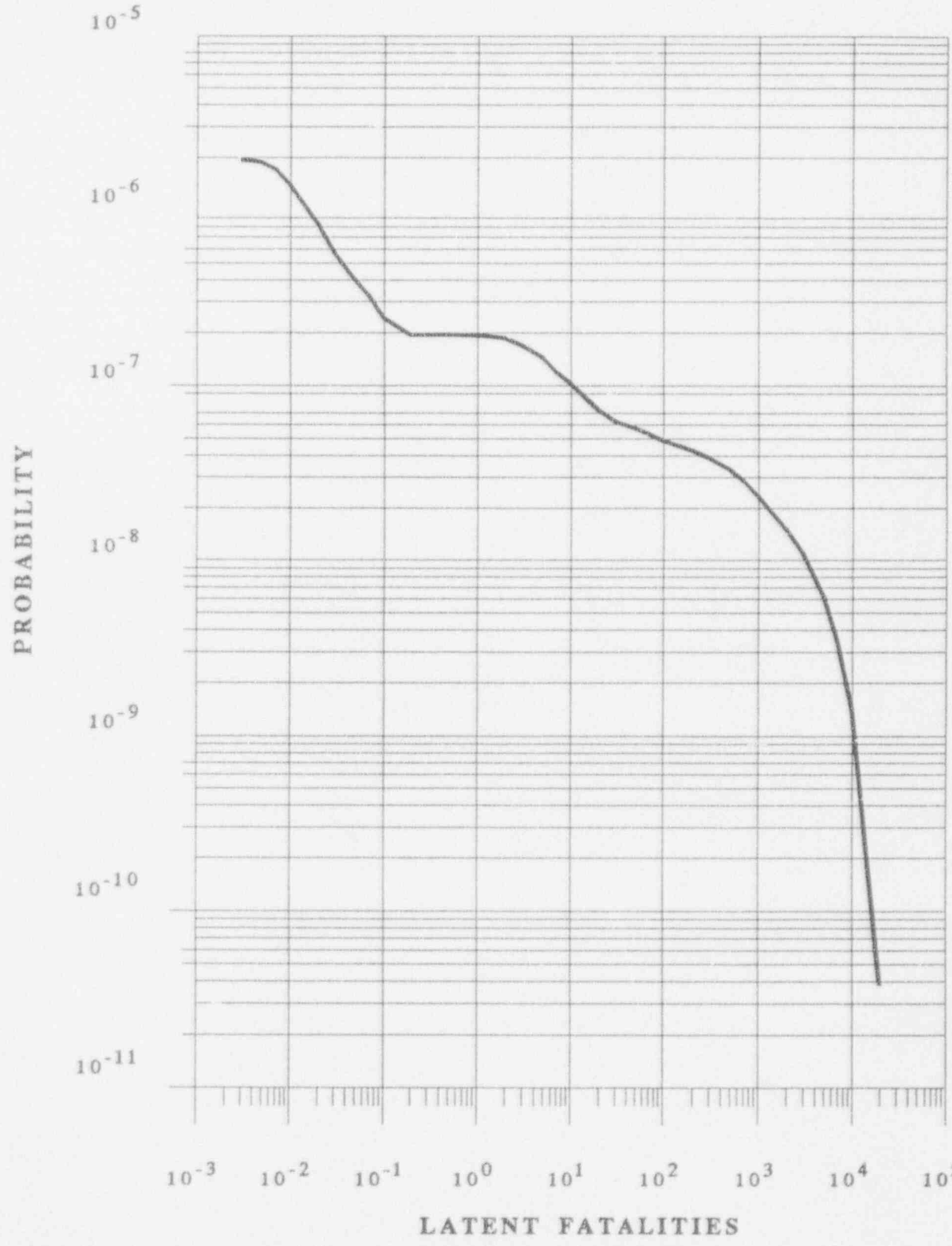


FIGURE 14.2-9

CCDF FOR DOSE @ 300 METERS FOR CASE 2

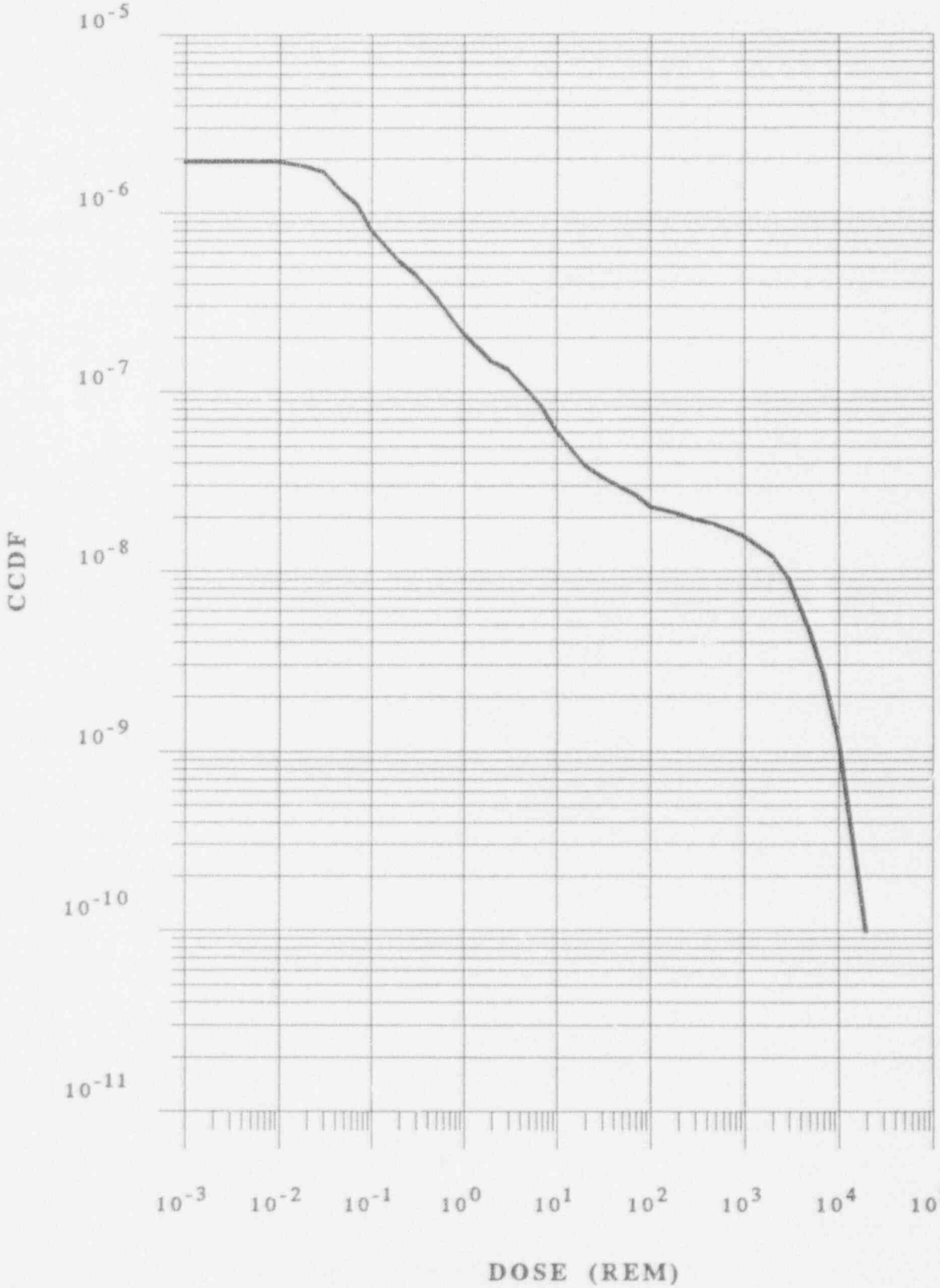


FIGURE 14.2-10

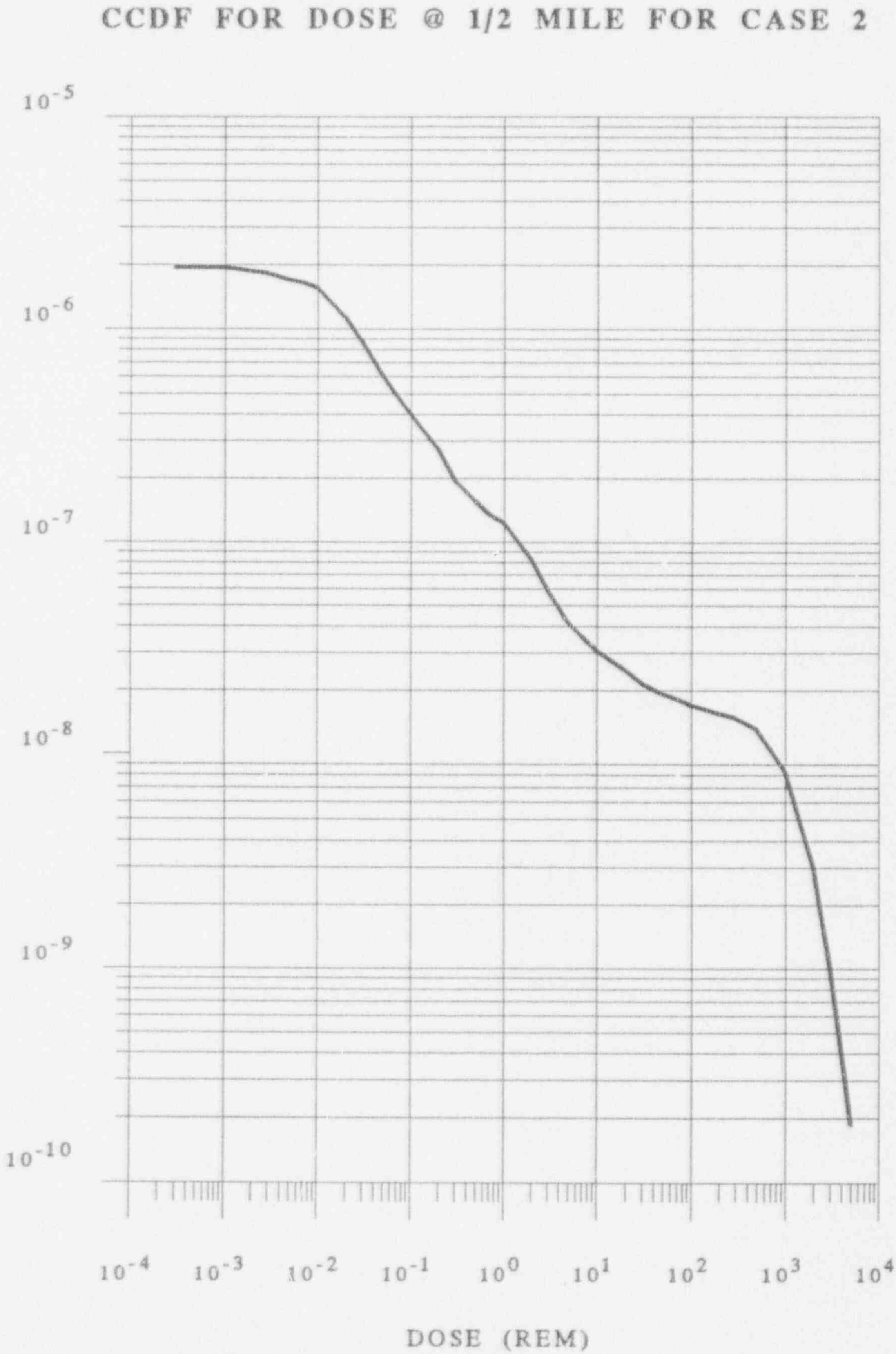


FIGURE 14.2-11

PROBABILITY OF EXCEEDANCE FOR EARLY FATALITY FOR CASE 2

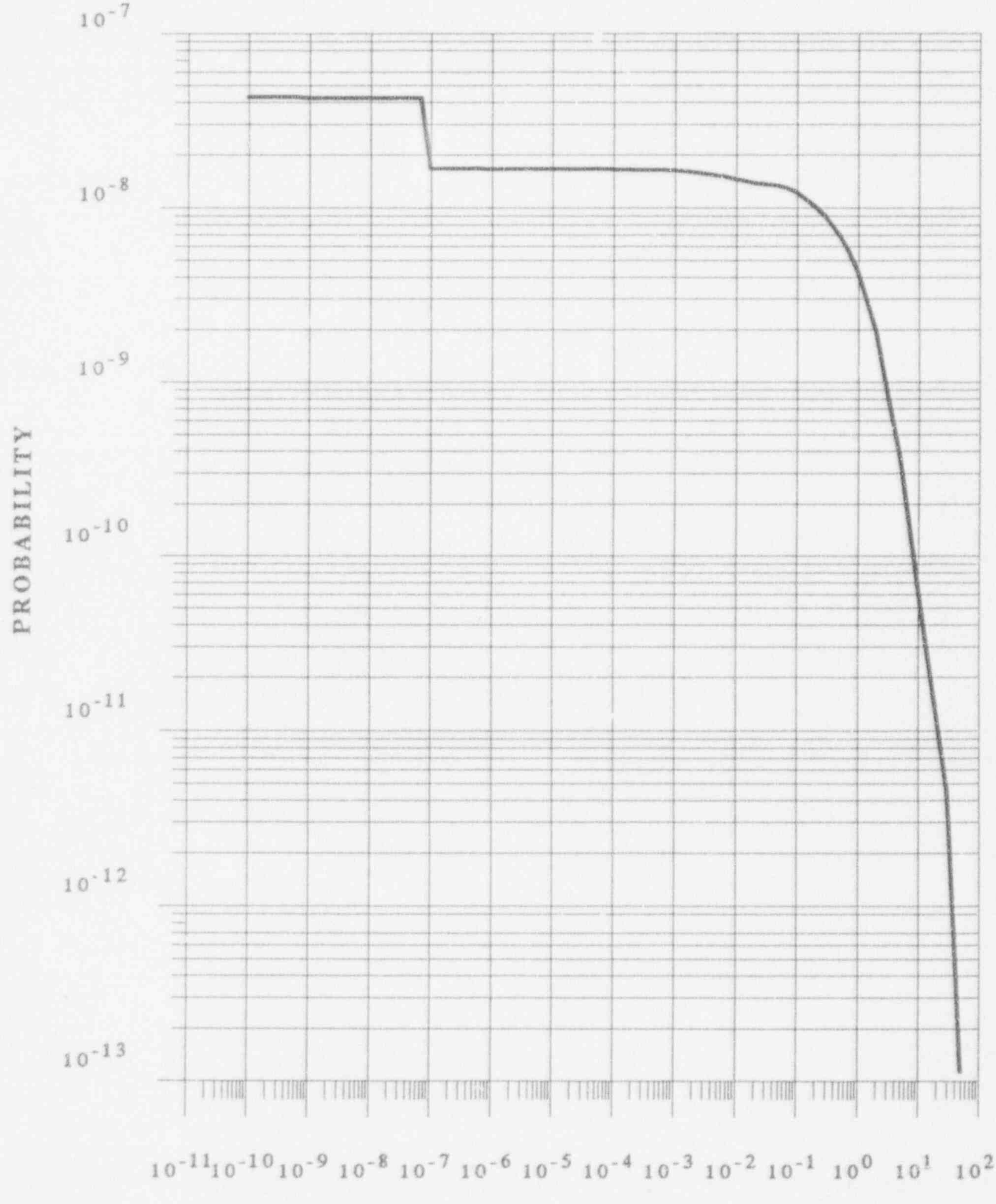


FIGURE 14.2-12

PROBABILITY OF EXCEEDANCE FOR LATENT FATALITY FOR CASE 2

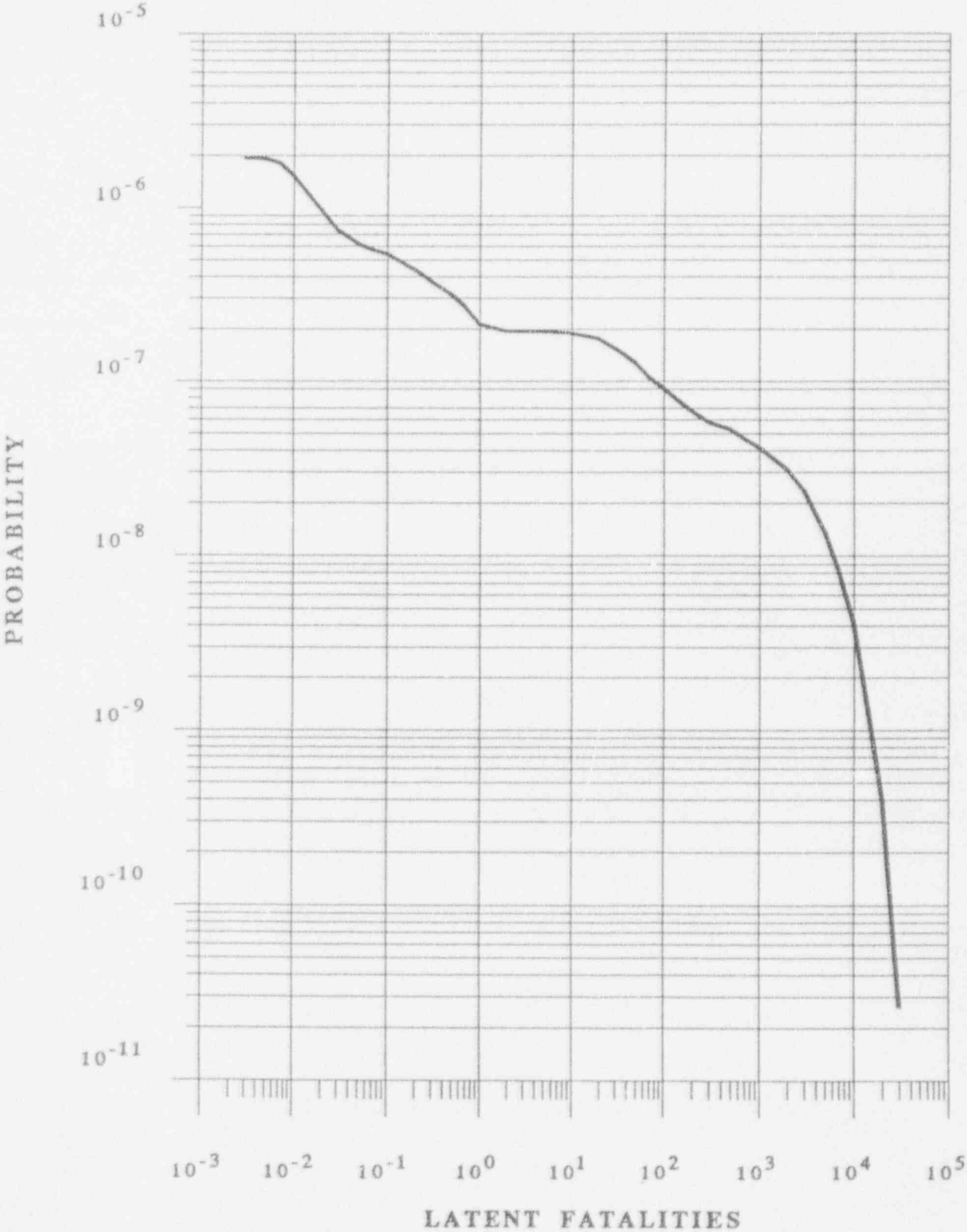


FIGURE 14.2-13

CCDF FOR DOSE @ 300 METERS FOR CASE 3

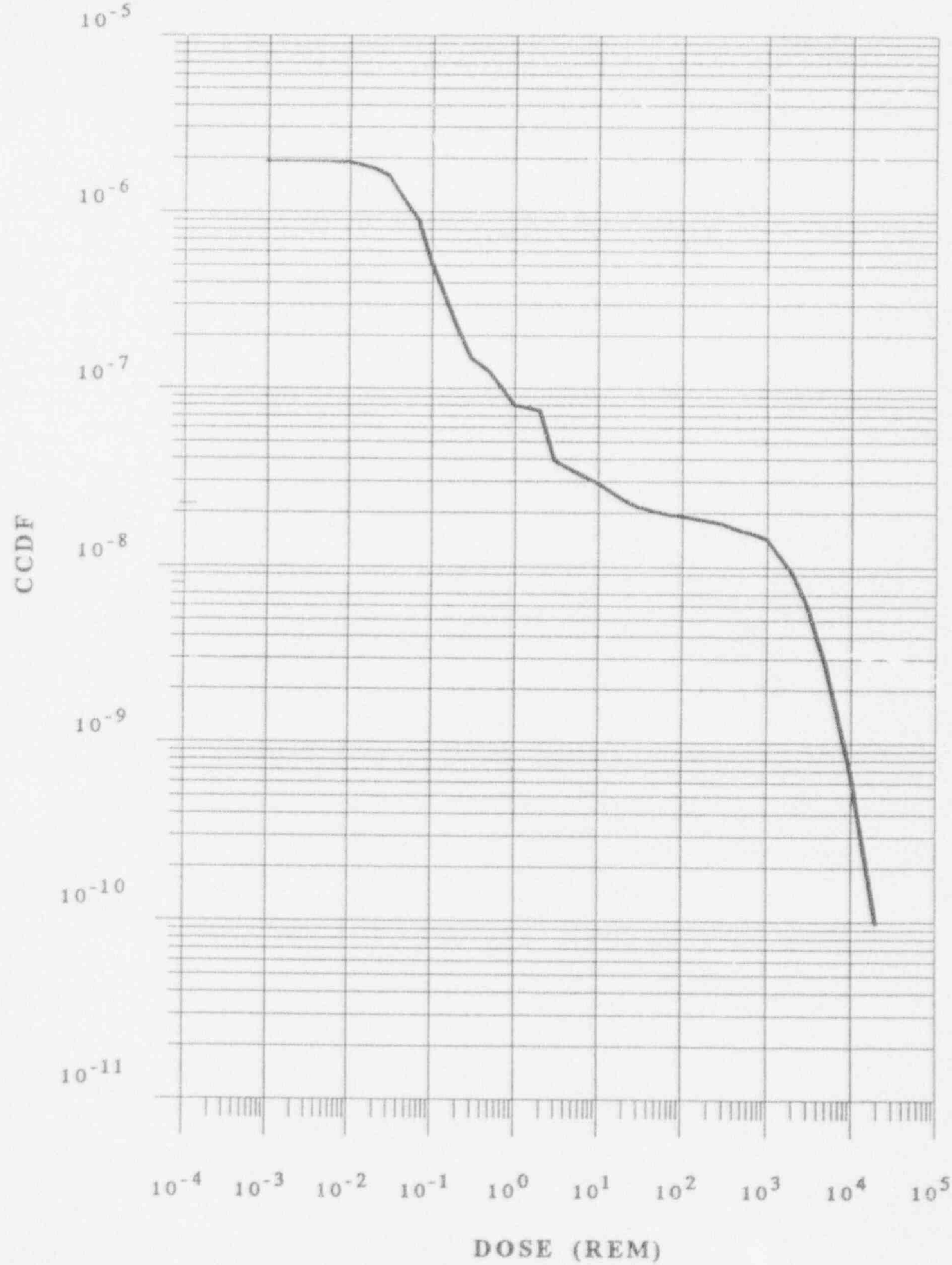


FIGURE 14.2-14

CCDF FOR DOSE @ 1/2 MILE FOR CASE 3

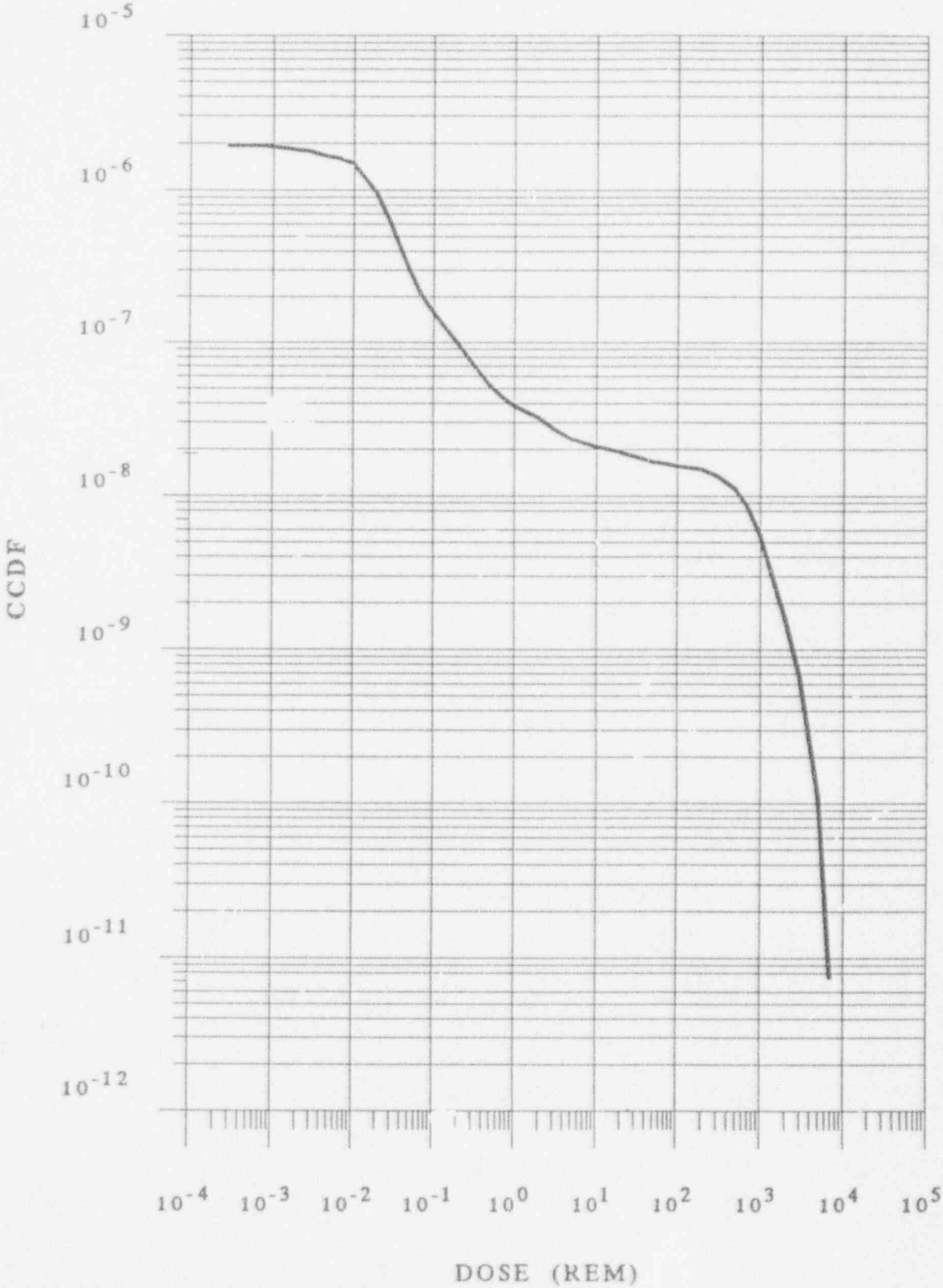


FIGURE 14.2-15

PROBABILITY OF EXCEEDANCE FOR EARLY FATALITY FOR CASE 3

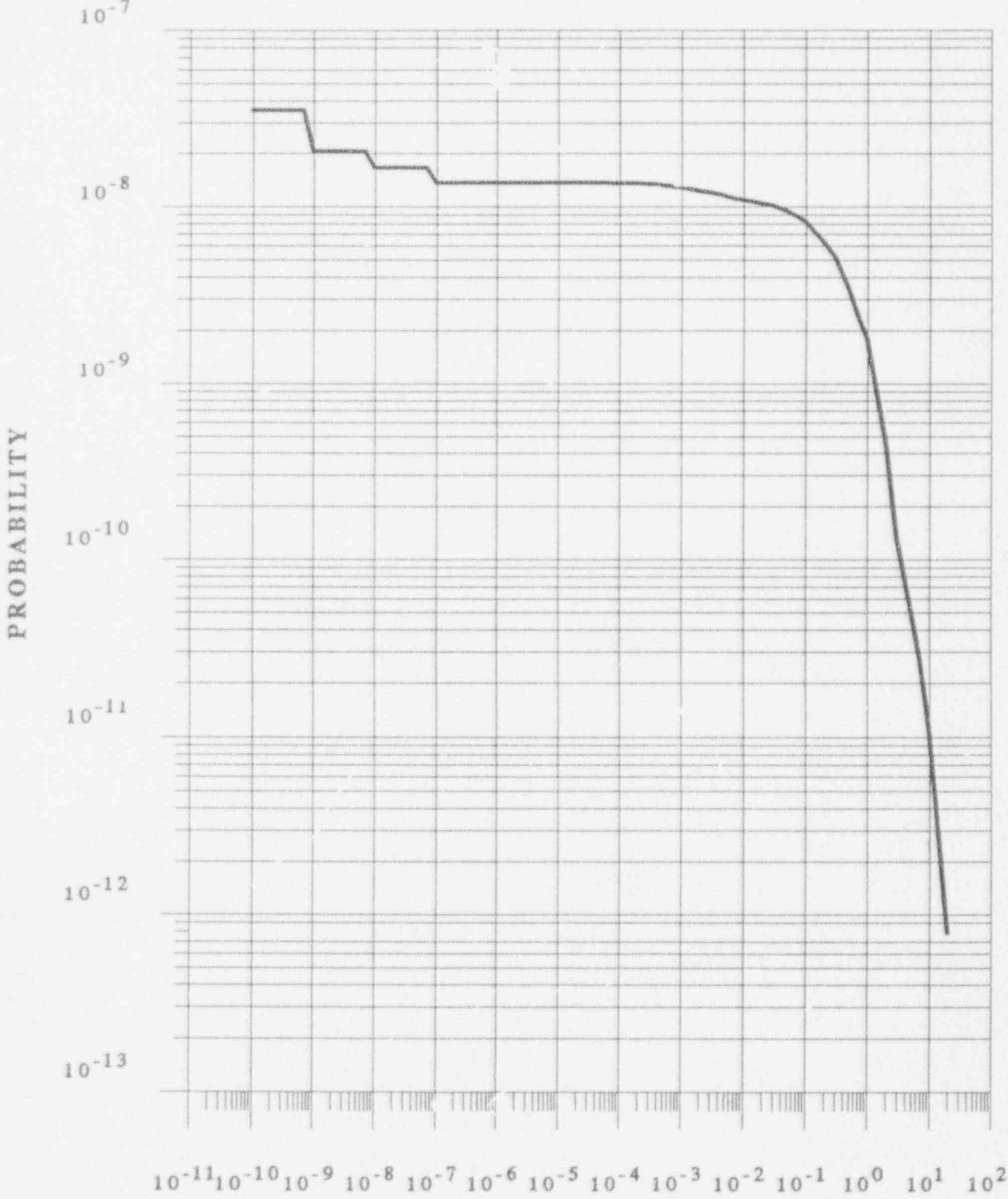


FIGURE 14.2-16

PROBABILITY OF EXCEEDANCE FOR LATENT FATALITY FOR CASE 3

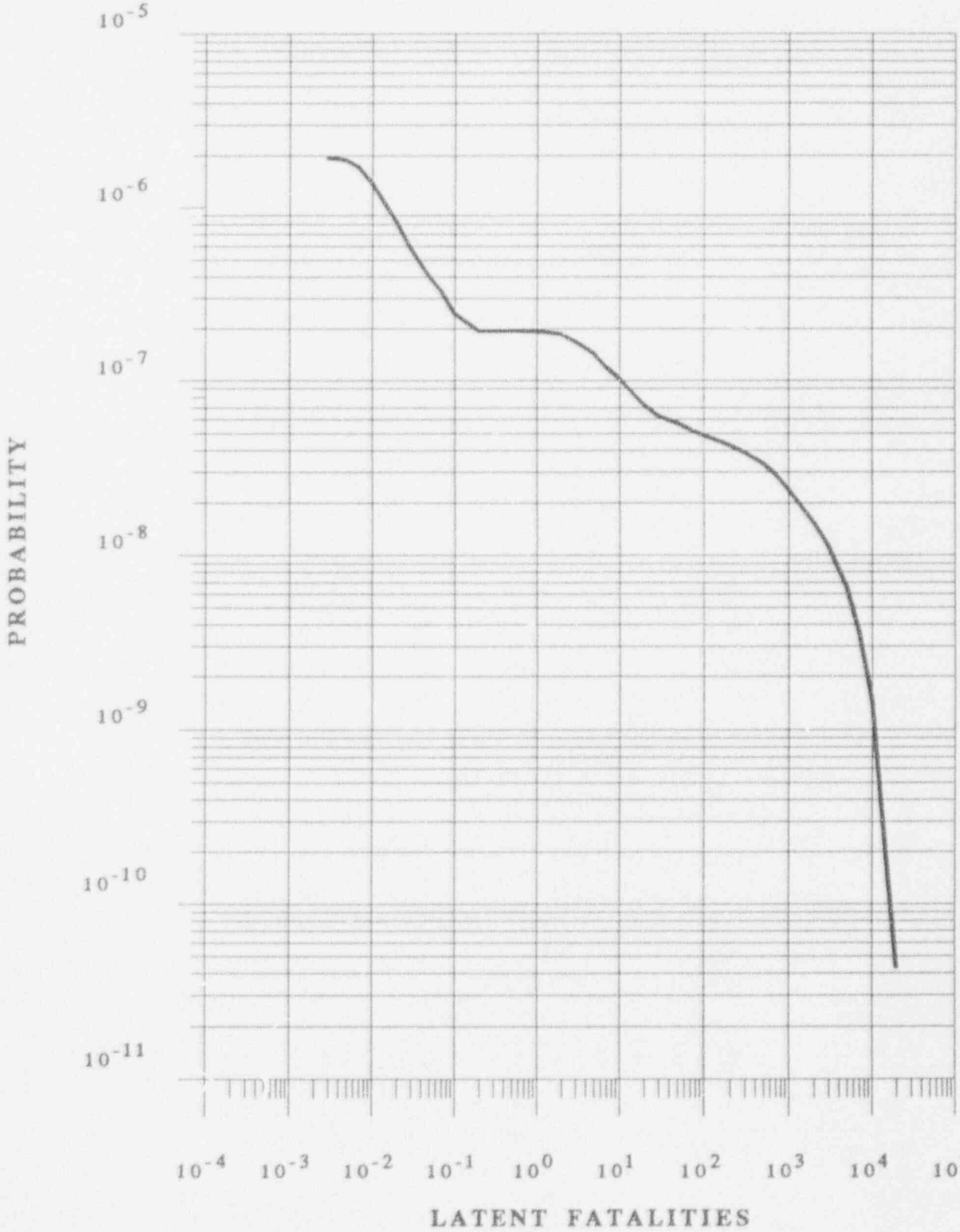


FIGURE 14.2-17

CCDF FOR DOSE @ 300 METERS FOR CASE 4

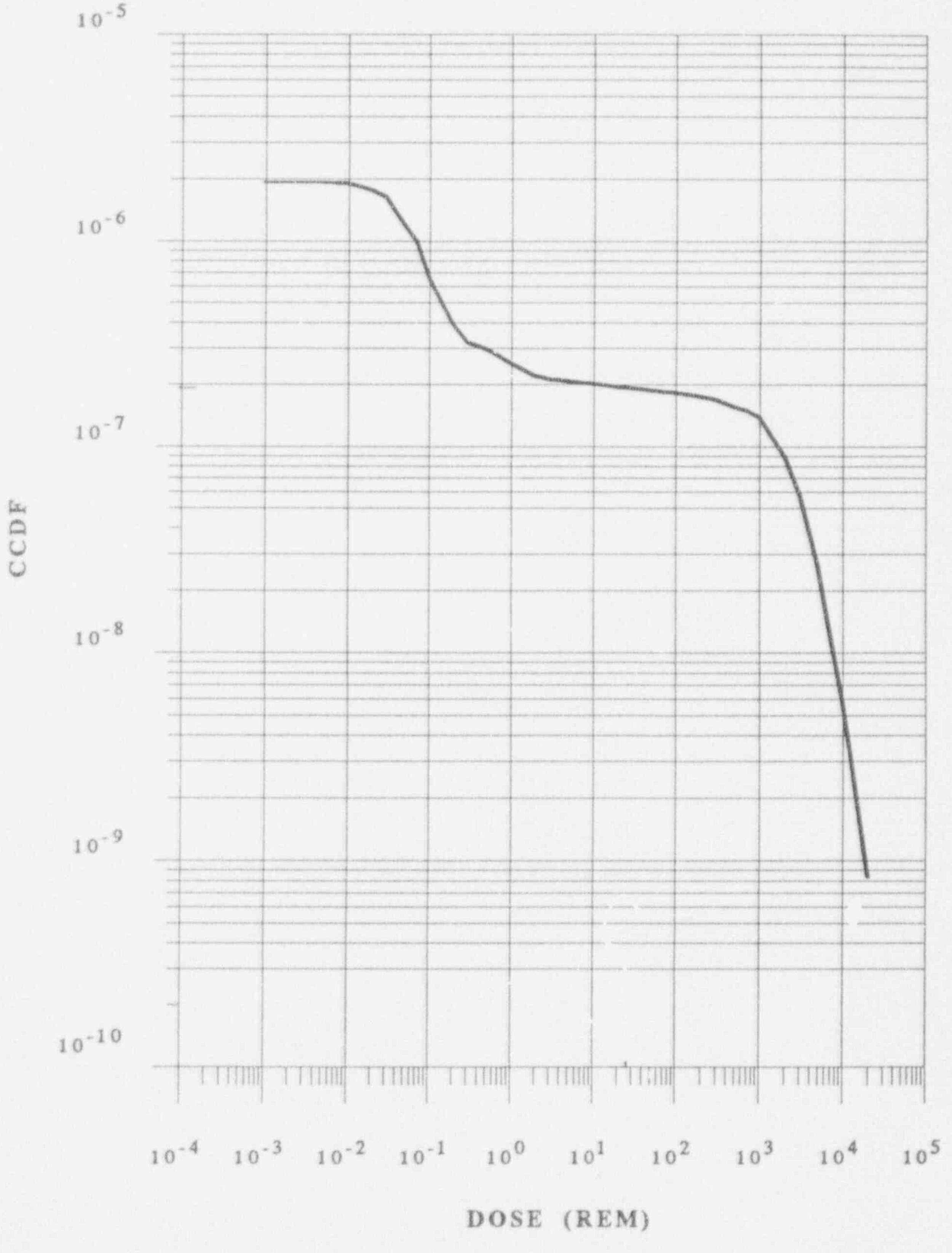


FIGURE 14.2-18

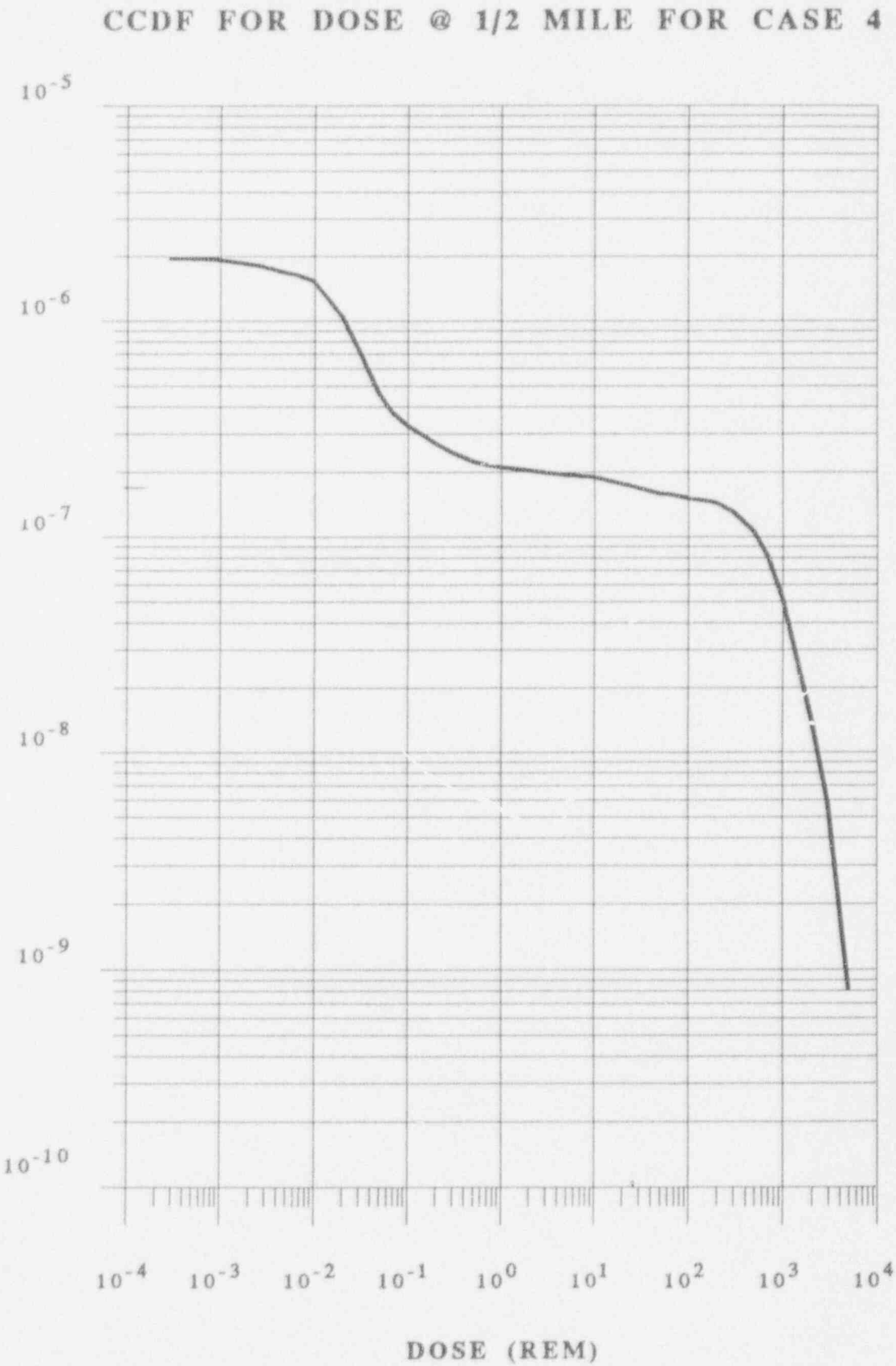


FIGURE 14.2-19

PROBABILITY OF EXCEEDANCE FOR EARLY FATALITY FOR CASE 4

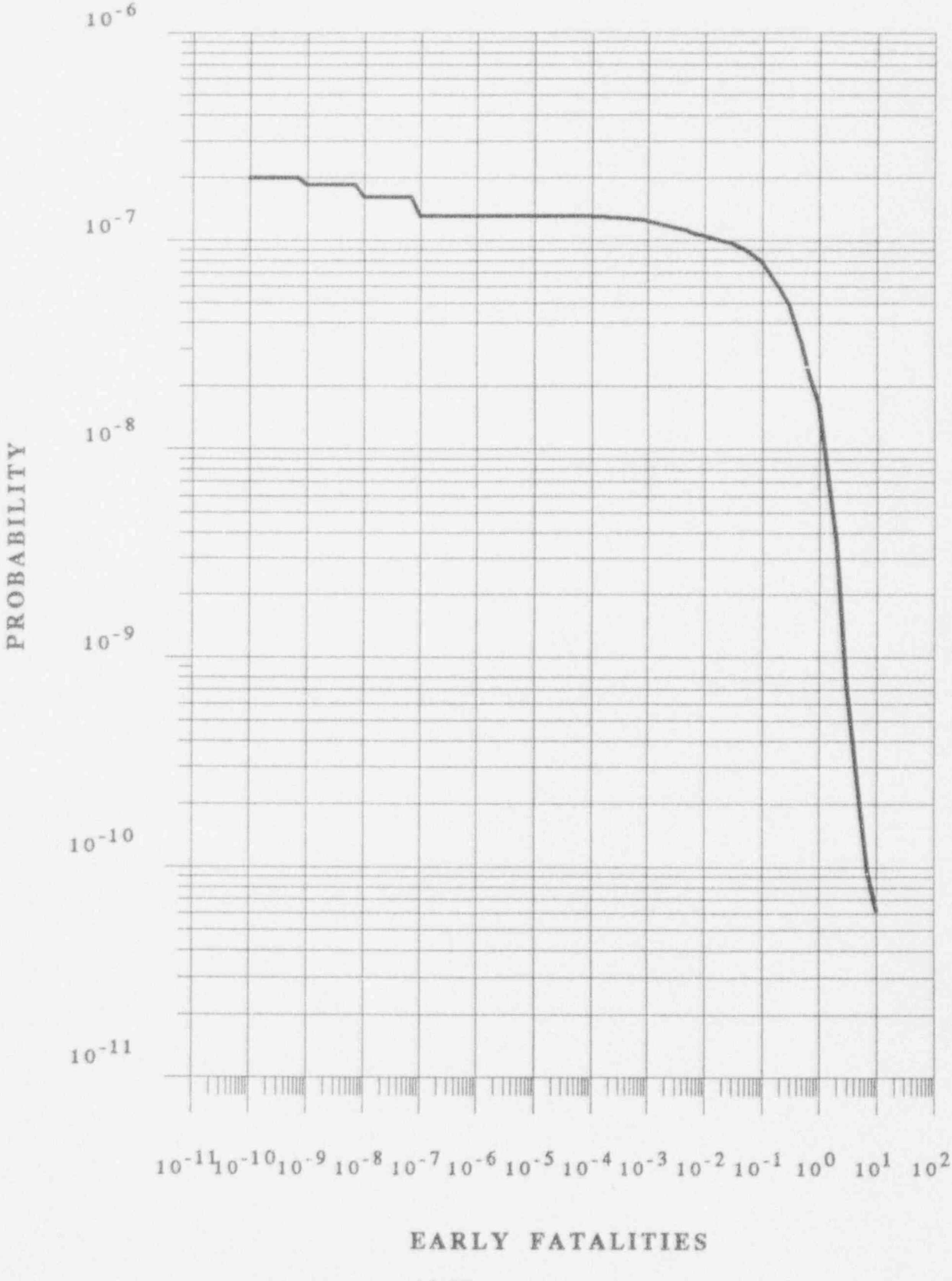


FIGURE 14.2-20

PROBABILITY OF EXCEEDANCE FOR LATENT FATALITY FOR CASE 4

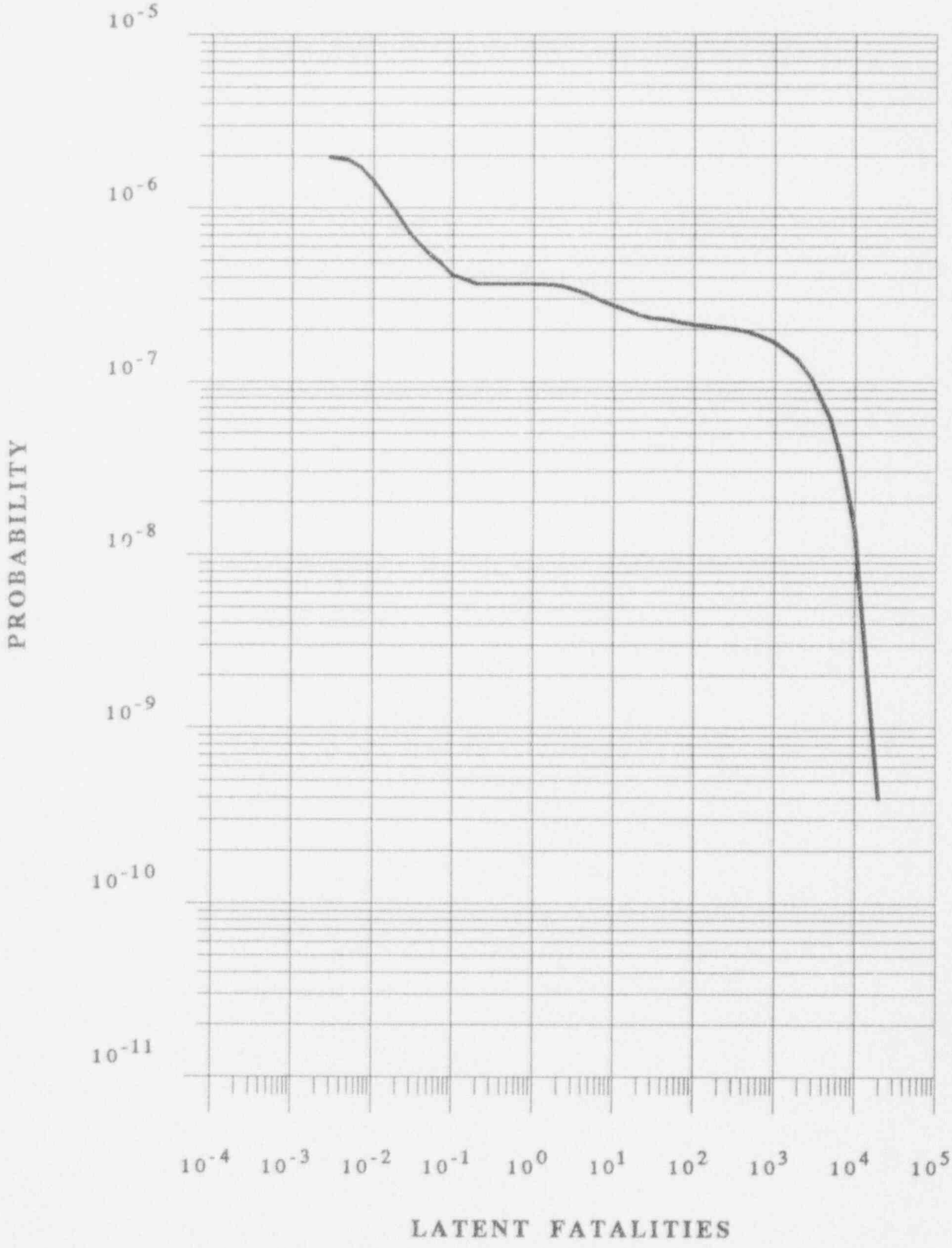


FIGURE 14.2-21

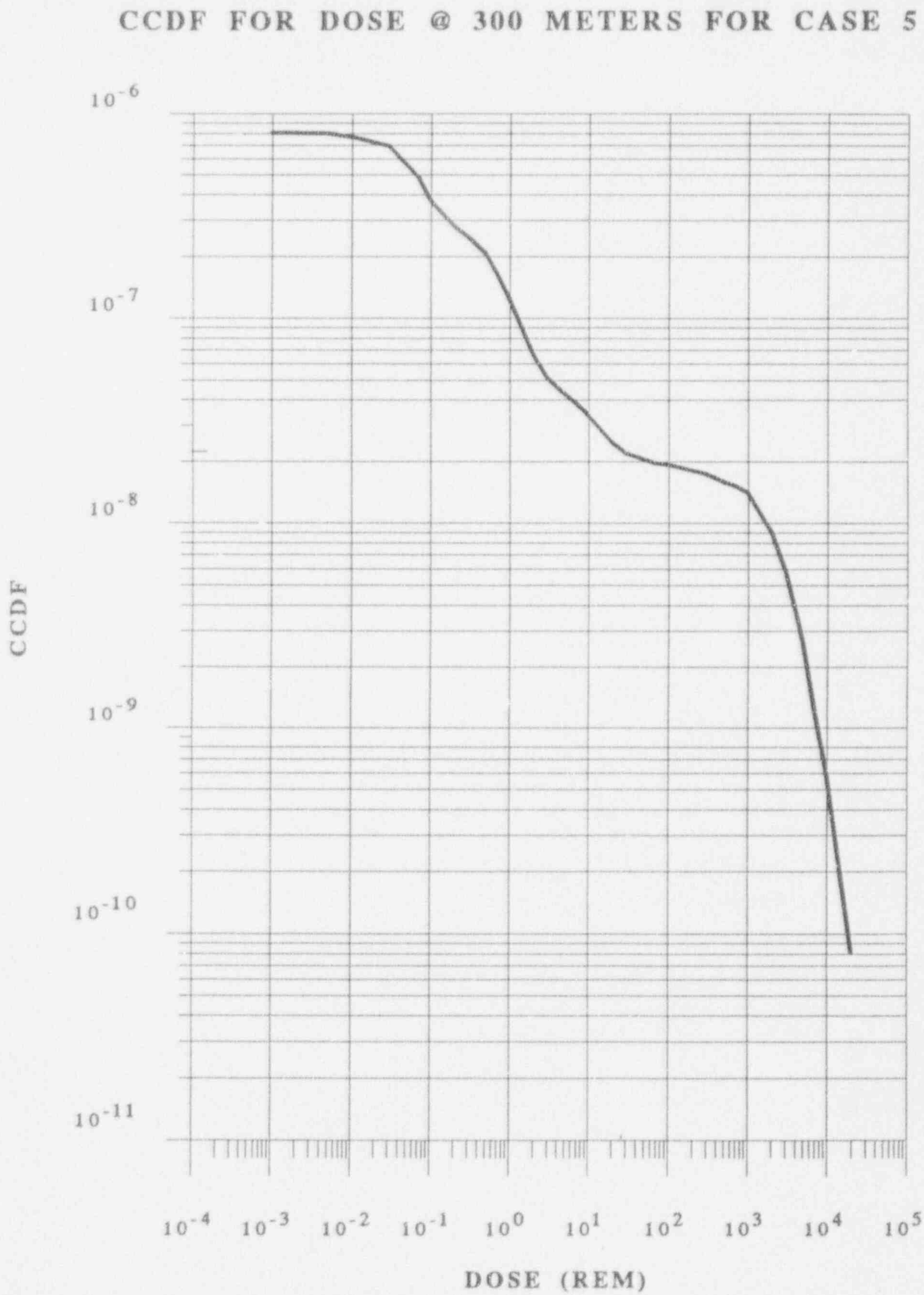


FIGURE 14.2-22

CCDF FOR DOSE @ 1/2 MILE FOR CASE 5

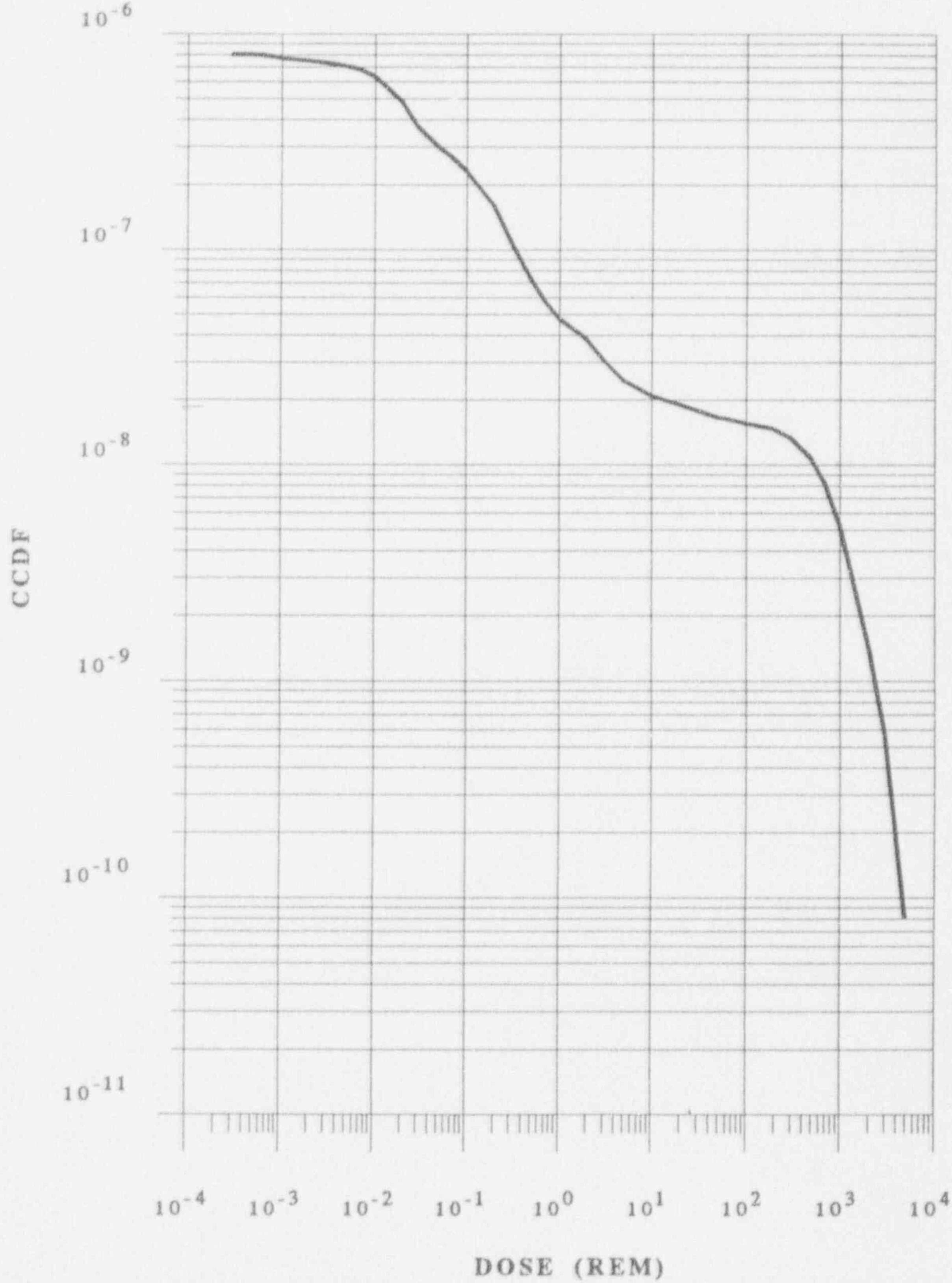


FIGURE 14.2-23

PROBABILITY OF EXCEEDANCE FOR EARLY FATALITY FOR CASE 5

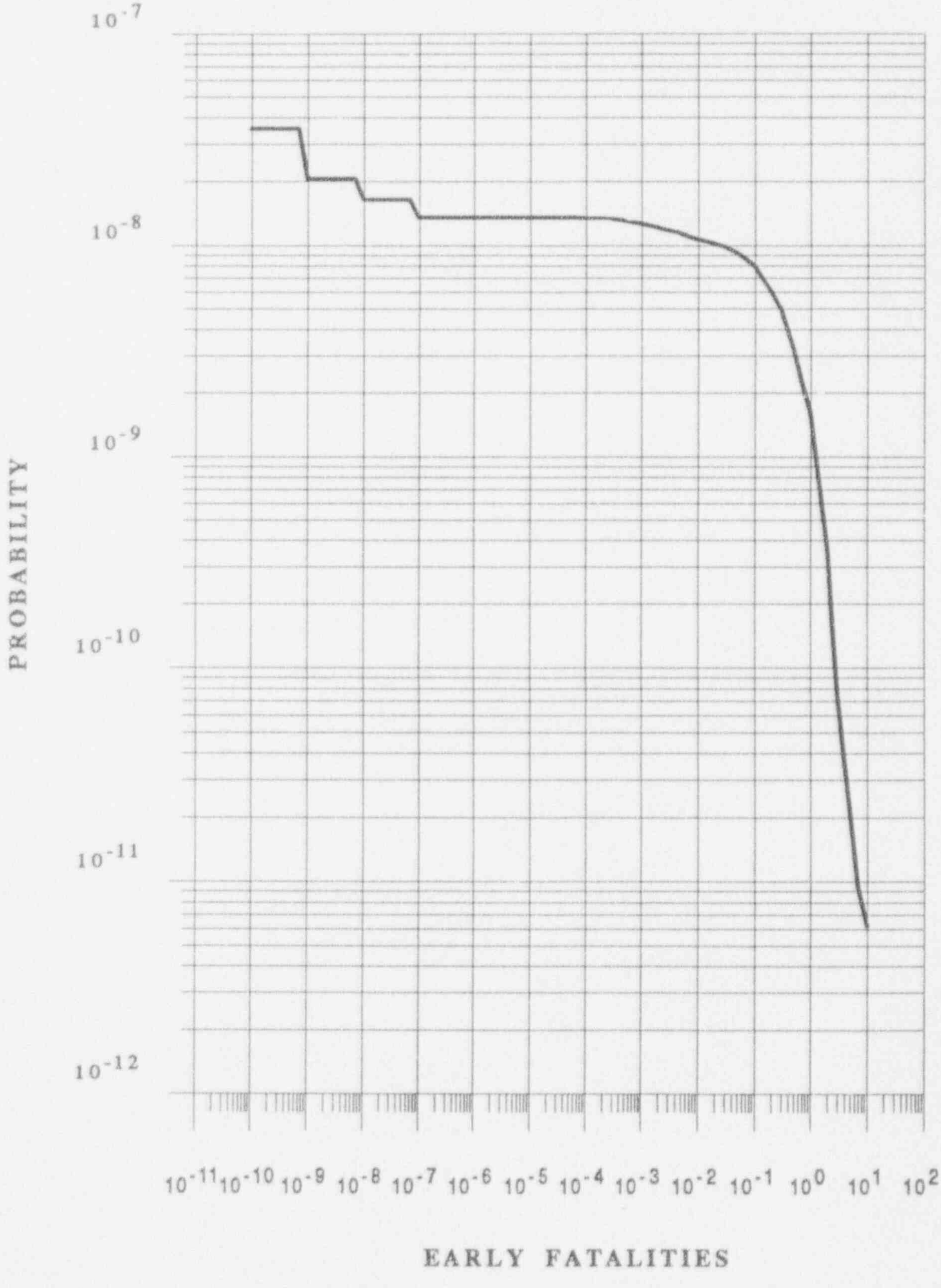


FIGURE 14.2-24

PROBABILITY OF EXCEEDANCE FOR LATENT FATALITY FOR CASE 5

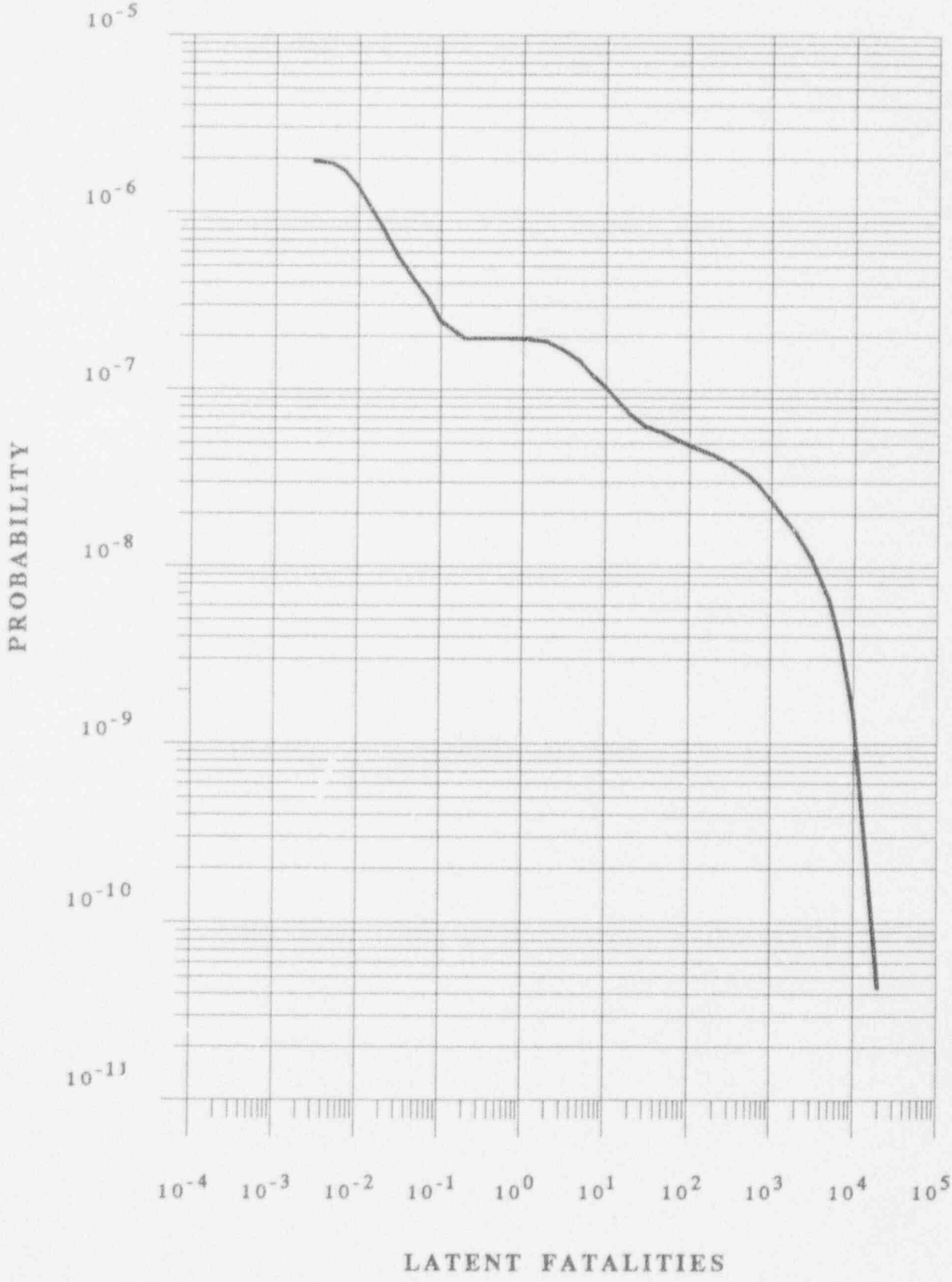


FIGURE 14.2-25

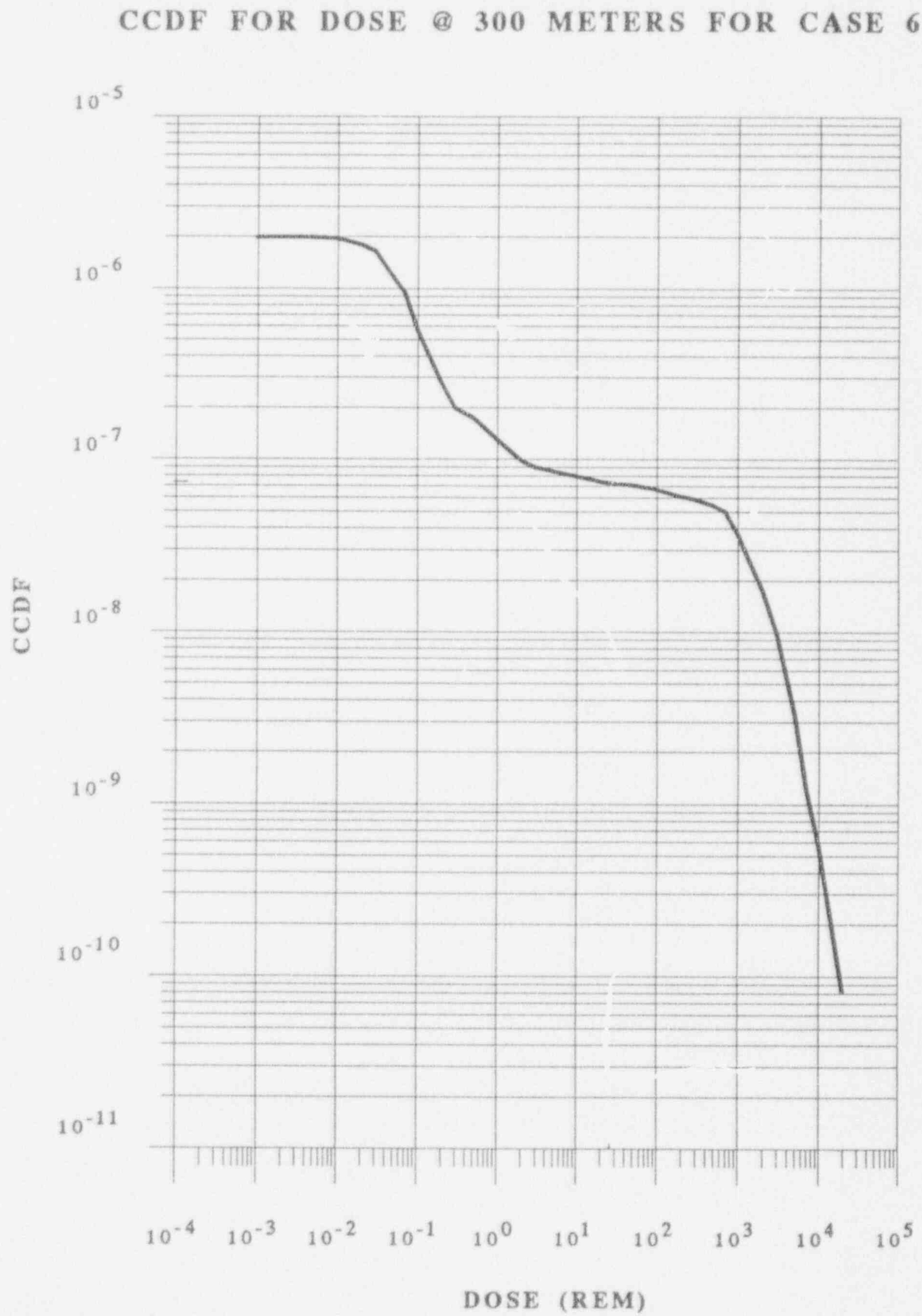


FIGURE 14.2-26

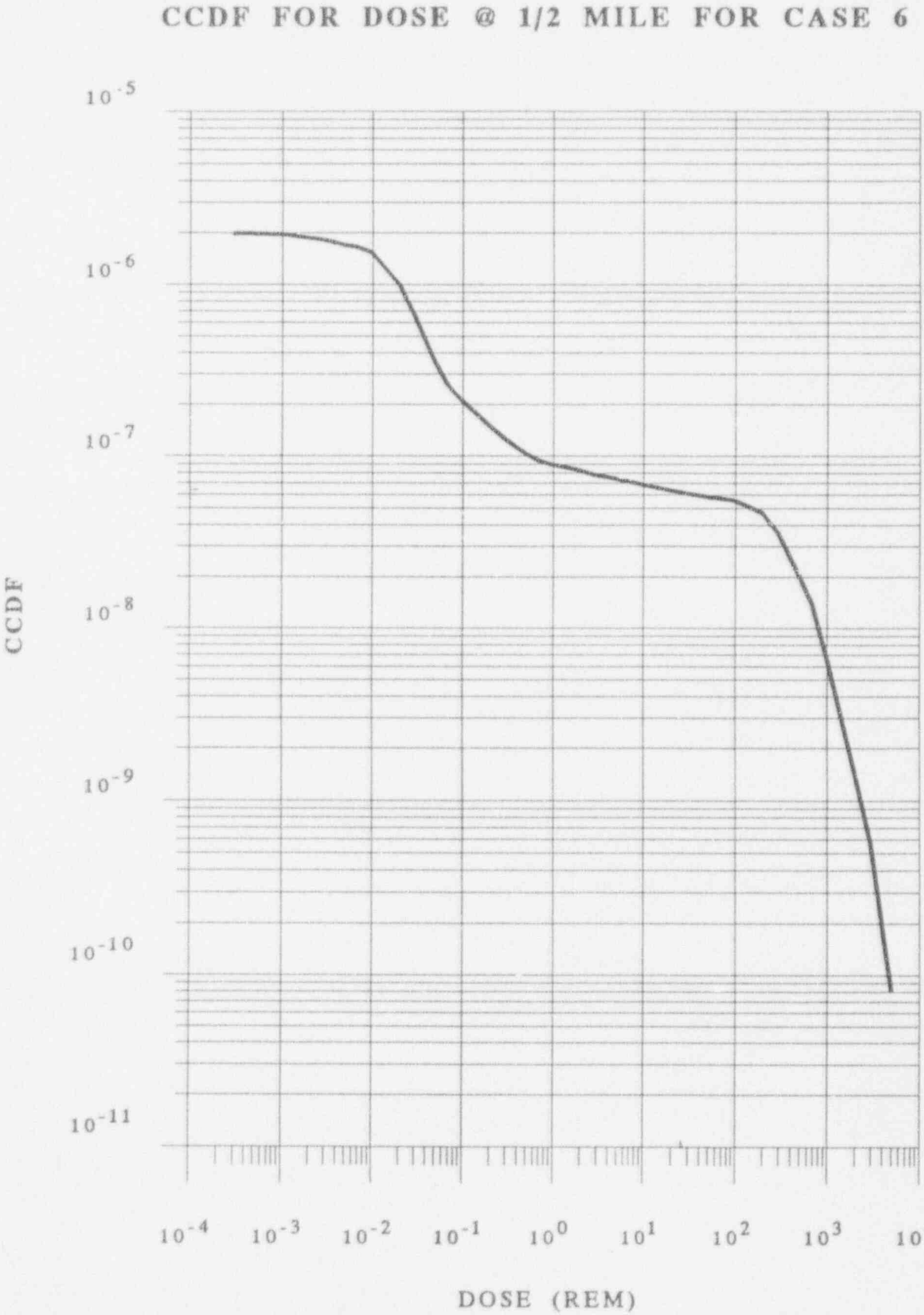


FIGURE 14.2-27
PROBABILITY OF EXCEEDANCE FOR EARLY FATALITY FOR CASE 6

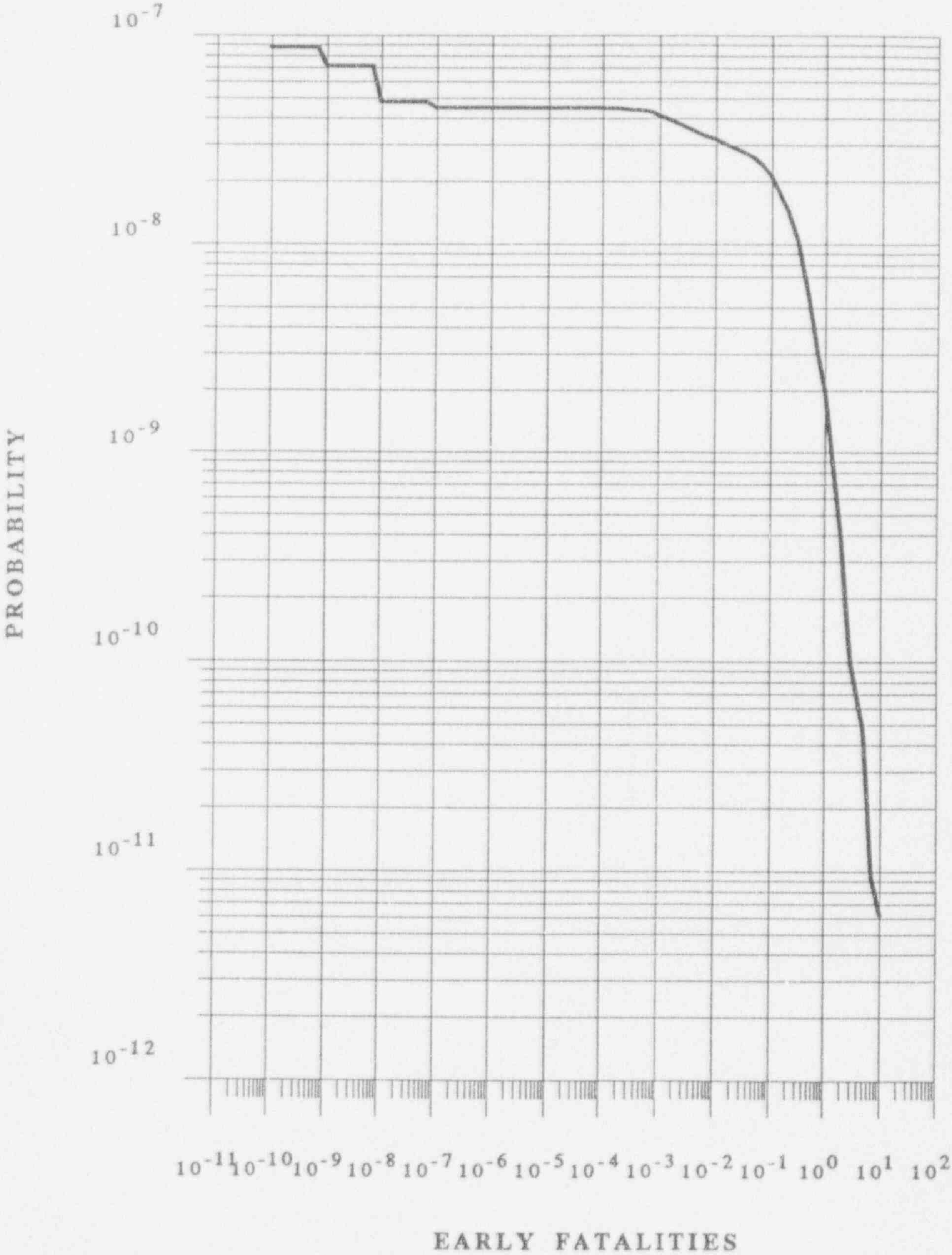
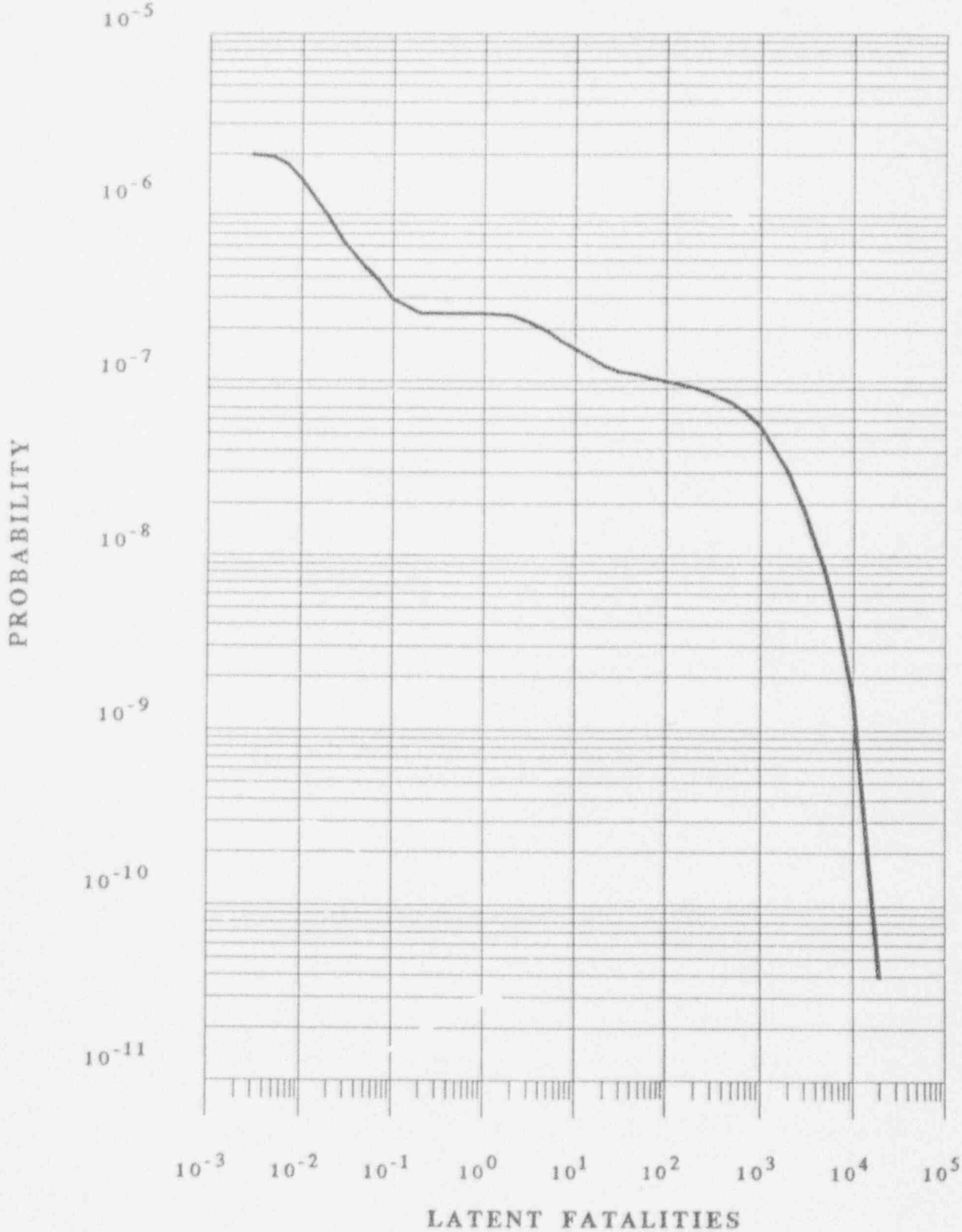


FIGURE 14.2-28

PROBABILITY OF EXCEEDANCE FOR LATENT FATALITY FOR CASE 6



19.15

~~15.0~~ SUMMARY OF PRA-BASED DESIGN INSIGHTS

This section of the report summarizes the PRA-based insights for the System 80+ design. The System 80+ PRA was performed to satisfy the objectives required for the Advanced Light Water Reactor design certification PRA. These objectives, as they relate to the System 80+ design, are:

- to assess, as realistic as possible, the risk profile of the proposed design in terms of the frequency of severe core damage accidents and their consequences;

- to develop better understanding and insights about the design strengths and relative weaknesses beyond those identified through deterministic analyses; and

- to support pre- and post-certification regulatory activities which include Design Acceptance Criteria (DAC); Inspection, Testing, Analyses, and Acceptance Criteria (ITAAC); Reliability Assurance Program (RAP); and technical specifications.

Since the System 80+ PRA is being used to support the pre- and post-certification activities, the insights gained regarding the risk contributors are very useful. Therefore, the following useful information and insights are summarized in this section of the report:

- how PRA insights influenced the design,

- what design features were added to or deleted from the design as a result of PRA insights,

- how it was determined if there were any vulnerabilities in the plant design from internal or external events,

- how the PRA was used to develop an appropriate balance of prevention and mitigation in the design.

- how to use the models, information, and results of the design for verifying some of the key assumptions of the PRA.

- how to use insights from the uncertainty, importance, and sensitivity analyses to support various activities such DAC, RAP, ITAAC, and technical specifications.

- how to use insights from the external events analyses, shutdown and low power risk analyses to support pre- and post-certification activities.

The special features that are incorporated into the System 80+ design to prevent and mitigate accidents are summarized in Section 15.1. Insights about the System 80+ design gained from the internal events risk profile and the external events risk profile are summarized in Sections 15.2 and 15.3, respectively. Shutdown and low-power operation are included as part of the System 80+ PRA, and the insights gained from the risk associated with these modes of operation are summarized in Section 15.4. The use of PRA in the design process is summarized in Section 15.5. The use of PRA results and insights to support certification and followup activities is summarized in Section 15.6.

15.1 Special Design Features

System 80+ is an evolutionary Advanced Light Water Reactor (ALWR). ABB C-E designed the System 80+ using PRA extensively in the design process to identify areas for improvement and to monitor progress toward meeting risk reduction goals. Using the Standard System 80 design as the starting point, the System 80 design evolved into System 80+. The changes were made to make the plant safer, more available, and easier to operate. Therefore, the System 80+ design contains features that reduced risk, when compared with existing generation of commercial nuclear power plants. Table 15.1-1 summarizes the System 80+ evolutionary features that affect safety and compares them to similar features of System 80.

The features summarized in Table 15.1-1 contributes to risk reduction, some more than others. These features are either preventative or mitigative. The purpose

of the preventative features is to minimize or reduce the initiation of plant transients, arrest or terminate the progression of plant transients once they occur, and prevent core damage. Likewise, the purpose of the mitigating features is to mitigate severe accidents and the consequences of core damage.

15.1.1 Design Feature for Preventing Core Damage

The following briefly describes the major features that were incorporated into the System 80+ design to limit plant transients and to prevent severe accidents from occurring.

<u>Larger Pressurizer</u>

The reason for designing a larger pressurizer volume in the System 80+, as compared the existing generation of commercial nuclear power plants, is to make the plant response to transients slower and more resilient. The larger pressure volume helps maintain a higher pressurizer pressure and water level following a turbine trip. It also helps prevent emptying the pressurizer following overcooling transients. For certain transient events, the rise in pressurizer pressure will be moderate and consequently the primary safety valves will not be challenged. A larger pressurizer volume also helps to lower the peak pressure that can be reached following an Anticipated Transient Without Scram (ATWS) event. The primary bases for increasing the pressurizer volume are:

- to prevent the draining of the pressurizer and uncovering of the heaters following a reactor or a turbine trip,

- to prevent water level surges that cause liquid or two-phase flow to reach the primary safety valves following a feedwater line break or a loss of load transient,

- to prevent the lifting of the primary safety valves following certain transient events,

- to increase the margin for a safety injection actuation signal

during a reactor or turbine trip.

- to minimize the fluctuation of the pressurizer during transient events.

Larger Steam Generators

The larger than existing steam generators of System 80+ is designed to make the plant response to transients slower and more resilient. The increased heat transfer area of the steam generators provides a 10% tube plugging margin, which helps increase the availability of the steam generator secondary heat removal. The increased downcomer volume and the 20% increase in steam generator inventory help reduce the fluctuations during transients and increase the boil-off time to dryout the steam generators. The time required to dryout the secondary inventory of the steam generators is approximately 50% longer for System 80+ than the dryout time for System 80. The improved steam generator tube materials and the reduced hot-leg temperature are designed to help reduce the frequency of steam generator tube ruptures.

Shutdown Cooling System (SCS)/Containment Spray System (CSS)

In addition to their long-term decay heat removal function, the SCS pumps are designed to perform residual heat removal injection and cooling of the In-containment Water Storage Tank (IRWST). In the residual heat removal injection mode of operation, the SCS is used (in conjunction with the Rapid Depressurization System) as a backup to the Safety Injection System (SIS) to inject borated water into the reactor core. To provide operating flexibility, the design pressure of the SCS is much higher for System 80+ than Standard System 80. The SCS pumps can also be used as backup to the CSS pumps to perform IRWST cooling during "feed and bleed" operations (beyond design basis events). The two-train redundancy for each of these systems, coupled with the interchangeable SCS and CSS pumps, enhance the availability of these systems.

Multiple Independent Connection to Grid and Turbine/Generator Runback Capability

The System 80+ design includes a main switchyard for incoming and outgoing electric power and a separate and independent backup switchyard that is tied to the grid at some distance from the main switchyard. In addition, the System 80+ turbine generator system and the associated buses are designed to runback to maintain hotel load on a loss of grid event. These features are intended to reduce the frequency of Loss of Offsite Power (LOOP) events and station blackout events.

Separate Startup and Emergency Feedwater System

The use of a non-safety related Startup Feedwater System (SFWS) for normal startup and shutdown operations helps reduce the demands on the Emergency Feedwater System (EFWS). In addition, the SFWS provides an independent means of supplying feedwater to the steam generators for removing heat from the Reactor Coolant System (RCS) during emergency conditions when the main feedwater is not available (the SFWS is automatically actuated upon loss of main feedwater and prevents the need to actuation the EFWS).

Improved Control Room Design

The System 80+ control room design (Nuplex 80+) is intended to improve existing control rooms while maintaining their strengths. In that respect it is an evolutionary design that is expected to provide more and better information to the operator than the Standard System 80 design.

Improved Normally Operating Component Cooling Water System (CCWS)

The Component Cooling Water System (CCWS) is a closed-loop system that provides cooling water to remove heat from plant systems, components, and structures. Heat from the CCWS is rejected to the ultimate heat sink through the open-loop Station Service Water System (SSWS). Each of these systems consists of a separate and redundant division. Each division contains two pumps: one is normally operating, while the other pump is in standby and starts automatically

if the operating pump trips. This configuration eliminates the demand failures of pumps and valves that were found to be significant contributors to risk in the System 80 design with standby CCWS/SSWS configurations.

Facilities Designs

Facilities are designed to provided physical separation of systems or trains of system that perform redundant safety-related functions. This increases the availability of systems due to their protection from failures associated with internal fires, internal floods, and similar common-cause failures. This contributes to risk reduction when compared to existing plant designs.

Safety Injection System (SIS) with Diract Vessel Injection

The primary function of the Safety Injection System (SIS) is to inject borated water into the RCS for inventory and reactivity control during severe accidents such as Loss of Coolant Accidents (LOCAs) and ATWS. The SIS can be used in conjunction with the Rapid Depressurization System for "feed and bleed" operation. For continuous long-term post-LOCA (large) cooling of the reactor core, the SIS pumps are realigned to provided simultaneous hot-leg and direct vessel injection to prevent boron crystallization. The following are major evolutionary characteristics of the System 80+ SIS:

- four high-pressure 100% capacity pumps,
- four safety injection tanks (SITs),
- direct vessel injection (pumps take suction from the IRWST and deliver borated water to the reactor vessel downcomer via the DVI lines),
- elimination of need for low pressure pumps,
- elimination of need to realign pump suction,

- hot side injection into each hot-leg.

- capability to test pumps at design flow.

- "feed and bleed" cooling of the RCS (in conjunction with the Safety Depressurization System for beyond design basis events).

These evolutionary characteristics help reduce the unavailability of the System 80+ SIS to levels below those for existing generation of commercial nuclear power plants. This was achieved by reducing or eliminating several contributors to SIS unavailability. For example: (1) a four-train (as compared to a two-train) SIS, reduces the contribution to the system unavailability that is due to outages for testing, repair and maintenance; (2) the elimination of the low-pressure pumps eliminates the failures to start for these pumps; (3) the elimination of the need to realign the suction of the pumps eliminates the contribution of the failure to do so; (4) the provision for cold-leg DVI increases the time for SIS response during a small break LOCA.

Safety Depressurization System (SDS)

The Safety Depressurization System (SDS) consists of two sub-systems: (1) the Reactor Coolant Gas Vent System (RCGVS) and (2) the Rapid Depressurization System (RDS) or bleed system. The RCGVS provides a safety-related means of venting non-condensable gases from the pressurizer and the reactor vessel upper head. Likewise, the RDS provides a manual safety-related means of rapidly depressurizing the RCS so that the SIS can deliver borated water to the reactor core, when long-term decay heat removal fails via the Shutdown Cooling System or via the steam generator secondary heat removal. Rapid depressurization of the RCS is manually accomplished and is often exercised in conjunction with the "feed" function which is provided by the SIS. This is a significant risk reduction feature added to the System 80+ design.

Emergency Feedwater System (EFWS)

The Emergency Feedwater System (EFWS) provides an independent safety-related

means of supplying feedwater to the steam generators during the early phase of secondary heat removal in the event that both the main feedwater and the startup feedwater are lost. The EFWS consists of two divisions each of which is aligned to deliver feedwater to its respective steam generator. Each division contains a motor-driven train and a turbine-driven train. The steam required to operate the turbine-driven pump is supplied from the associated steam generator that feedwater is delivered to, and the cross-connect of steam supply to the EFWS turbine pumps is not a design characteristic. For station blackout sequences, the turbine-driven trains of the EFWS are available to remove decay heat from the RCS. Because of the redundancy and diversity of the emergency feedwater trains, this system is a significant contributor to risk reduction.

Two Emergency Diesels and Standby Combustion Turbine

Each of the two divisions of class 1E AC power is supplied with emergency standby power from an emergency diesel generator (DG). Each DG is provided with a dedicated 125 VDC battery. The emergency DGs start and load automatically following a LOOP event. In addition to the two emergency DGs, the System 80+ design has an alternate standby onsite AC power source. This is a non-safety combustion turbine power source provided to cope with station blackout scenarios. The alternate power source is independent and diverse from the DGs. Once started, the combustion turbine is manually loaded to power one division of class 1E AC loads when the associated DG is unavailable.

Vital Batteries

Six independent and separate 125 VDC batteries are included in the System 80+ design, in comparison to four batteries for the System 80 design. For System 80+, each battery can supply the continuous emergency load of its own load group for a period of 4 hours. In addition, the batteries provide a station blackout coping capability assuming manual load shedding or the use of a load management program. This permits operating the instrumentation and control loads associated with the turbine-driven emergency feedwater pumps for 8 hours.

In-Containment Refueling Water Storage Tank

Sufficient borated water is stored in the In-Containment Refueling Water Storage Tank (IRWST) to meet all post-accident safety injection pumps and containment spray pumps operation requirements. The volume of borated water is also sufficient to flood the refueling pool during normal refueling operations. The IRWST eliminates the need for switching over from injection mode to recirculation mode during emergency core cooling operations and therefore, eliminates failures associated with switch-over in existing commercial nuclear power plants.

15.1.2 Design Features for Mitigating Consequences of Core Damage

The following briefly describes the major features that were incorporated into the System 80+ design to mitigate severe accidents and the consequences of core damage.

Large Spherical Steel Containment

The most important advantages of this feature are listed below:

- Enhanced containment atmospheric mixing and dilution of post accident hydrogen gases, reduces the potential for developing detonable concentrations of hydrogen under severe accident conditions.

- High containment pressure capacity values are several times higher than the design pressure. (The containment pressure capacity is sufficiently large that the containment loads associated with early challenges, e.g., hydrogen combustion and direct containment heating, are at or below Service Level C value.) This assures an extremely low probability of containment failure for such challenges, the high-pressure capacity also significantly delays the time of release for late containment failure challenges.

In-Containment Water Storage System

This system performs water collection, delivery, storage, and heat sink functions inside the containment during normal operation and accident conditions. It comprises the IRWST, the holdup volume tank (HVT), the steam relief system (SRS), and the Cavity Flooding System (CFS). Containment spray water, RCS breakflow, and condensed water on containment structures will drain first into the HVT, and eventually to the IRWST through spillways connecting the IRWST and HVT. The IRWST provides water from steam condensation and fission product scrubbing before vessel breach, and water for reactor cavity flooding through the CFS. The in-containment water storage system is, therefore, significant for severe accident progression in its ability to reduce containment pressure (through steam condensation), to reduce fission product release (through pool scrubbing), and to reduce the probability of core concrete interaction through cavity flooding.

Safety Depressurization System (SDS)

In addition to the core damage prevention function, discussed earlier, the Rapid Depressurization System (RDS) of the SDS also serves a mitigative function. Specifically, actuation of the RDS prior to the core debris penetrating the vessel, can reduce or eliminate the potential for direct containment heating and large hydrogen combustion events at vessel breach and, thus, reduce the probability of early containment failure. The RDS also reduce the amount of fission product release associated with reactor vessel breach at high pressure, since the RDS flow is discharged directly into a sparger network in the IRWST and not into the containment atmosphere.

Reactor Cavity Design for Corium Disentrainment

The reactor cavity of the System 80+ design minimizes debris dispersal to the upper compartment of the containment after a high pressure vessel breach, and thereby reduces the potential for containment failure caused by direct containment heating. The path from the cavity to the upper containment is convoluted so that the corium will be disentrained and removed from the atmosphere before reaching the upper containment region. This design feature

reduces the quality of corium available for dispersal into the upper compartment and, therefore, the pressure rise associated with direct containment heating. In conjunction with the high containment pressure capacity for the System 80+ design, the retentive cavity design serves to further reduce the probability of containment failure as a result of direct containment heating.

Reactor Cavity Design for Debris Coolability

Another feature of the reactor cavity that is important to severe accident progression is the ability to quench and cool core debris in the cavity. The flow area in the System 80+ design meets the EPRI debris spreading criterion to enhance the potential for debris cooling. In addition, the reactor Cavity Flooding System (CFS) is designed to flood the reactor cavity in the event of a severe accident for the purpose of covering the core debris with water and maintaining a long-term debris coolability. The CFS also serves to scrub fission products. The CFS is designed to flood the reactor cavity in the event of a severe accident using water from the HVT portion of the containment water storage system.

Hydrogen Mitigation System

The System 80+ design incorporates a deliberate ignition system to maintain containment hydrogen concentrations below a detonable limit. The system uses igniters of the glow plug design and is manually remote controlled. Because of the proven design of the glow plug igniters and the reliability of the electrical distribution system used in the System 80+ design, the Hydrogen Mitigation System (HMS) is a significant risk reduction contributor to containment failure.

Table 15.1-1

Major System 80+ Preventive and Mitgative Design Features

TYPE OF FEATURE	DESIGN FEATURE
PREVENTIVE Transient Prevention	Larger pressurizer
	Larger steam generators
	High-pressure Shutdown Cooling System (SCS) - functionally interchangeable SCS and Containment Spray System (CSS) pumps
	Multiple independent connections to the grid and turbine-generator runback capability
	Dedicated startup feedwater system
	Improved control room design
	Improved normally operating Component Cooling Water System (CCWS)/Station Service Water System (SSWS)
Transient Mitigation/ Severe Accident Prevention	Four train Safety Injection System (SIS) with direct vessel injection
	Safety Depressurization System (SDS)
	Four train Emergency Feedwater System
	Two emergency diesel generators and a standby alternate AC source (combustion turbine)
	Six vital batteries
	In-containment Refueling Water Storage Tank (IRWST)
	Cross-connected CSS and SCS trains
	Improved control room design
MITIGATIVE	Large spherical containment Reactor cavity designed for corium disentrainment
	Reactor cavity designed for debris coolability
	IRWST and SDS
	Hydrogen mitigation system

15.2 Internal Events Risk Profile Insights

15.2.1 Core Damage Frequency of the Level I PRA

The Level I portion of the Sytem 80+ PRA addresses the internal (and the external) initiators of accident sequences which lead to core damage. Core damage is assumed to occur if the collapsed level in the reactor has decreased such that the active fuel in the core is uncovered and a temperature of 2200 °F or higher is reached in any node of the core as defined in best-estimate thermo-hydraulic calculations. The methodology for the Level I portion of the PRA for internal events complies with the recommendations of the PRA Key Assumptions and Groundrules of EPRI ALWR requirements document[7]. The small event tree/large fault tree (fault tree linking) approach was used to evaluate core damage frequency. Generic industry data, as presented in the EPRI ALWR requirements document[7], was used to perform the PRA. The methods used for the external events evaluation are summarized in Section 15.3.

The core damage frequency for internally initiated events is summarized in Section 15.2.1.1. The dominant accident sequences and their major contributors to core damage frequency for internal events are summarized in Section 15.2.1.2. In Section 15.2.1.3, the impact of several System 80+ design features, as they relate to the reduction of core damage frequency for internal events, are presented. Finally, the insights drawn from the uncertainty, sensitivity, and importance analyses are presented in Section 15.2.1.4.

15.2.1.1 Core Damage Frequency by Inititating Events

The estimated core damage frequency attributable to internal events for the System 80+ plant is 1.7E-06. Table 15.2-1 presents the core damage frequency contributions by initiating events (for comparison, the core damage frequency for the Standard System 80 design is also presented). The core damage frequency of the System 80+ design is a factor of 128 less than the core damage frequency of the System 80 design. This factor was derived using the same groundrules of the System 80 PRA. The relative contributions (percent of total) of the various internal events to the total core damage frequency are tabulated in Table 15.2-1

and shown graphically in Figure 15.2-1. For the System 80 design, loss of offsite power and station blackout dominates (46%) the core damage frequency profile. This is followed by the LOCAs (31%) and then transients (14%). The contribution by ATWS is relatively small (6%). For the System 80+ design, the LOCA categories of initiating events dominate (47%) the core damage frequency profile. This is followed by the transient category (35%) of events. The contribution from loss of offsite power (including station blackout) is relatively small because of the following features that were incorporated into the design:

- multiple independent connections to the grid,

- turbine-generator runback capability to maintain hotel loads,

- alternate standby AC source (combustion turbine), and

- six vital 125 VDC batteries.

The contribution from ATWS is also relatively saml1.

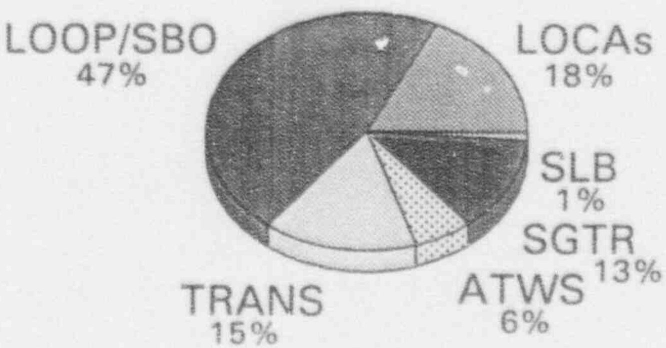
Table 15.2-1
COMPARISON OF CORE DAMAGE FREQUENCY CONTRIBUTIONS
BY INITIATING EVENT

	SYSTEM 80		SYSTEM 80+	
INITIATING EVENT	CDF/yr	% OF TOTAL	CDF/yr	% OF TOTAL
LARGE LOCA	1.8E-6	1.9	1.1E-7	6.6
MEDIUM LOCA	3.6E-6	4.4	3.1E-7	
MEDIUM LOCA1	N/S	N/S	1.4E-7	8.3
MEDIUM LOCA2	N/S	N/S	1.7E-7	9.8
SMALL LOCA	9.4E-6	11.6	2.0E-7	11.8
STEAMLINE/LARGE SEC. SIDE BREAK	9.0E-7	1.1	1.9E-9	0.1
STEAM GENERATOR TUBE RUPTURE	1.1E-5	12.9	2.9E-7	16.9
TRANSIENTS	1.2E-5	15.4	5.9E-7	
LOSS OF FEED. FLOW	N/S	N/S	5.1E-7	30.1
LOSS OF COMPONENT CLG WATER	N/A	N/S	4.9E-10	0.0
LOSS OF 125 VDC BUS	N/S	N/S	1.6E-10	0.0
LOSS OF 4.16 KV BUS	N/S	N/S	3.5E-10	0.0
LOSS OF HVAC	N/C	N/S	6.5E-9	0.4
OTHER	N/S	N/S	7.6E-8	4.5
LOSS OF OFFSITE POWER	3.8E-5	46.4	1.9E-8	1.1
SBO/BATTERY DEPLETION	L/S	L/S	2.1E-8	1.3
ATWS	4.8E-6	5.9	5.4E-8	3.2
INTERFACING SYSTEM LOCA	4.5E-9	0.0	5.2E-10	0.0
VESSEL RUPTURE	1.0E-7	0.3	1.0E-7	5.9
TOTAL	8.1E-5	100.0	1.7E-6 [6.3E-7]*	100.0

CDF Core Damage Frequency
N/S means Not Calculated Separately
N/A means Not Applicable
N/C means Not Considered
L/S means included with loss of offsite power
* This value is used for comparison purposes only. It was derived using the same groundrules of the System 80 PRA.

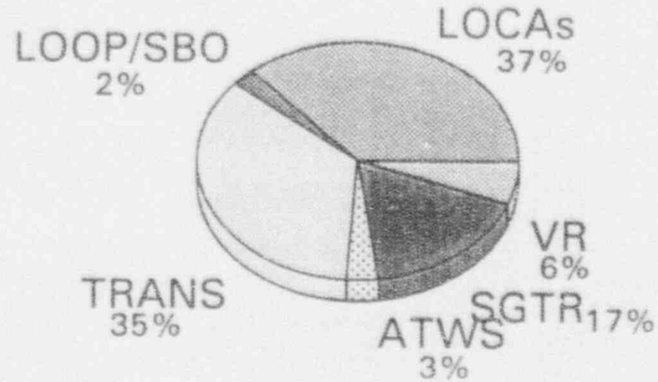
Figure 15.2 -1

RELATIVE CONTRIBUTIONS OF INTERNAL EVENTS TO TOTAL CDF



SYSTEM 80

(CDF = 8.1E-5/YR)



SYSTEM 80+

(CDF = 6.3E-7/YR)*

* This value is used for comparison purposes only. It was derived using the ground rules of System 80 PRA.

15.2.1.2 Dominant Accident Sequences for Internal Events

The dominant accident sequences due to internal initiated events and their contributions to the core damage frequency for the System 80+ design are presented in Table 15.2-2. The dominant accident sequences are described below.

LOSS OF FEED FLOW EVENTS

Three dominant accident sequences were identified for loss of main feedwater flow events, LOFW-9, LOFW-4, LOFW-8, and LOFW-5. The calculate frequencies for these accident sequences are 4.6E-07, 3.6E-08, 2.1E-08, and 5.6E-09 per year respectively.

- LOFW-9 is an accident sequence which involves loss of main feedwater to the steam generators followed by failure of the Emergency Feedwater System (EFWS), and failure of the Safety Depressurization System (SDS) to perform bleed operation. The EFWS is the preferred means available for removing decay heat via the steam generators. If the EFWS fails then once-through cooling or "feed and bleed" is the alternate means of removing decay heat from the reactor core. However in this sequence, the "bleed" portion of "feed and bleed" fails and consequently core damage occurs. The dominant contributors to this sequence are: (1) common cause failure of EFW distribution line check valves, (2) common cause failure of EFW pump discharge check valves, (3) common cause failure of the "bleed" valves, and (4) operator fails to initiate "feed and bleed".

- LOFW-4 is an accident sequence which involves loss of main feedwater to the steam generators followed by successful deliver of emergency feedwater to the steam generators, failure of long-term decay heat removal, and failure of "feed" operation. Long-term cooling provides continued decay heat removal, once the initial cooldown by the EFWS is completed. However for this sequence, long-term cooling fails and "feed and bleed" becomes the other alternative for removing decay heat from the reactor core to prevent core damage from occurring. Because the "feed" portion of "feed and bleed" also fails core damage eventually occurs. The dominant

contributors to this sequence are: (1) common cause failure of the safety injection line check valves, (2) independent failure of the CST makeup valve (check or manual), and (3) operator fails to align CST to EFW storage tanks.

- LOFW-8 is an accident sequence that is similar to LOFW-9 except that the "bleed" portion of "feed and bleed" fails instead of "feed". The dominant contributors to this sequence are: (1) common cause failure of the EFW distribution line check valve, (2) common cause failure of the EFW pump discharge check valve, (3) common cause failure of safety injection line check valves, (4) common cause failure of the safety injection line motor operated valves, and (5) common cause failure of the safety injection pumps.

- LOFW-5 is an accident sequence that is similar to LOFW-4 except that the "bleed" portion of "feed and bleed" fails instead of "feed". The dominant contributors to this sequence are: (1) common cause failure of the SCS suction valves, (2) independent failure of the CST makeup valve (check or manual), (3) operator fails to initiate "feed and bleed", (4) common cause failure of "bleed" valves, (5) operator fails to align CST to EFW storage tanks.

STEAM GENERATOR TUBE RUPTURE (SGTR) EVENTS

Four dominant accident sequences were identified, SGTR-17, SGTR-16, SGTR-12, SGTR-15, and SGTR-6. The calculated frequencies for these dominant sequences are 2.7E-07, 1.5E-08, 6.3E-09, 1.2E-09, and 4.4E-09 per year respectively.

- SGTR-17 is an accident sequence which involves a steam generator tube rupture event followed by failure of the Safety Injection System (SIS) and the inability to aggressively cooldown the secondary side of the plant to initiate shutdown cooling injection. Failure of SIS results in loss of the preferred way of making-up and controlling the lost RCS inventory. By aggressively cooling down and depressurizing the RCS, the SCS can be used to provide the necessary makeup to the reactor core. Therefore, a failure

of SIS and failure to establish aggressive cooldown results in core damage. The dominant contributors to this sequence are: (1) common cause failure of the safety injection line check valves, (2) common cause failure of safety injection line motor operated valves, (3) common cause failure of safety injection pumps to start or run, and (4) operator fails to perform aggressive cooldown.

- SGTR-16 is an accident sequence which involves a steam generator tube rupture event followed by failure of the SIS and failure of the SCS to inject borated water into the RCS. Once SIS fails, an aggressive cooldown of the secondary side is successfully accomplished. The aggressive cooldown also decreases the RCS pressure so that the SCS pumps can be used to provide the necessary makeup to the RCS. However for this sequence, the SCS fails to provide the necessary makeup and consequently core damage occurs. The dominant contributors to this sequence are: (1) common cause failure of the safety injection line check valves, (2) common cause failure of the safety injection to start or run, and (3) operator fails to align SCS for injection.

- SGTR-12 is an accident sequence which involves a steam generator tube rupture event followed by failure of the EFWS and the SDS. Following the SGTR event, the SIS provides makeup for the lost reactor coolant (through the ruptured tube). However, the removal of decay heat via the intact steam generator is not accomplished because the EFWS fails to deliver feedwater to the steam generator. Although the removal of decay heat via the steam generator is lost (the preferred means), "feed and bleed" would then be used as the alternate means of removing decay heat from the RCS. However for this sequence, the SDS which provides the "bleed" portion of "feed and bleed" also fails and consequently core damage occurs. The dominant contributors to this sequence are: (1)common cause failure of the EFW distribution line check valves, (2) common cause failure of the EFW pump discharge check valves, (3) operator fails to initiate "feed and bleed", and (4) common cause failure of the "feed and bleed" valves.

- SGTR-15 is an accident sequence which involves a steam generator tube rupture event followed by failure of the SIS, failure to isolate the ruptured steam generator, and failure re-fill the IRWST. Following the tube rupture event the SIS fails to provide the necessary makeup. Aggressive cooldown of the secondary side is then accomplished and the SCS is used to provide makeup (inject borated water) to the RCS. Normally the ruptured steam generator would be identified and isolated as soon as practicable to minimize the need to replenish the make source (the IRWST). However for this sequence, the ruptured steam generator is not isolated and re-filling of the IRWST is also not accomplished. As a result, core damage eventually occurs. The dominant contributors to this sequence are: (1) failure of the main steam safety valves to re-seat, (2) common cause failure of the safety injection line check valves, (3) common cause failure of the safety injection line motor operated valves, and (4) operator fails to align CVCS to re-fill the IRWST following SGTR.

- SGTR-9 is an accident sequence which involves a steam generator tube rupture event followed by failure of RCS pressure control, failure to isolate the ruptured steam generator, and failure to re-fill the IRWST. Following the tube rupture event, the SIS successfully provided the necessary make to the RCS. Also, the EFWS delivered feedwater successfully to the intact steam generator for decay heat removal via the steam generator. However for this sequence, RCS pressure control is not provided, the ruptured steam generator is not isolated, and re-filling of the IRWST is also not accomplished. As a result, core damage eventually occurs. The dominant contributors to this sequence are: (1) failure of the main steam safety valves to re-seat, (2) operator fails to throttle safety injection pumps, and (3) operator fails to align CVCS to re-fill IRWST following SGTR.

SMALL LOCA EVENTS

Two dominant accident sequences were identified for small Loss of Coolant Accident (LOCA) events, SL-11, SL-10, and SL-4. The calculated frequencies for these dominant accident sequences are 1.6E-07, 9.0E-09, and 9.0E-09 per year,

respectively.

- SL-11 is an accident sequence which involves a small break loss of coolant followed by failure of SIS and failure to aggressive cooldown the secondary side. To mitigate a small LOCA, makeup of the lost reactor coolant must be provided. Normally the SIS provides this function, but for this sequence it fails. The other line of defence would be to aggressively cooldown the secondary side and use the SCS pumps to provide the needed makeup. Aggressive cooldown also fails and the lost coolant is not restored. Consequently, core damage occurs. The dominant contributors to this sequence are: (1) common cause failure of the safety injection line check valves, (2) common cause failure of the safety injection line motor valves, (3) common cause failure of the safety injection pumps to start or run, and (4) operator fails to perform aggressive cooldown following a small LOCA.

- SL-10 is an accident sequence which involves a small break loss of coolant followed by failure of the SIS and failure of the SCS to provide injection. Makeup of the lost reactor coolant must be provided to mitigate a small LOCA. Normally this function is provided by the SIS, but for this sequence, it fails. Aggressive cooldown of the secondary side is then accomplished successfully. In cooling down the secondary side, RCS pressure also decrease and the SCS would then be aligned to inject borated water into the RCS. Although aggressive cooldown was successful for this sequence, the SCS fails to provide the necessary makeup and consequently core damage occurs. The dominant contributors to this sequence are: (1) common cause failure of the safety injection line check valves, (2) common cause failure of the safety injection line motor valves, (3) operator fails to align SCS for injection, and (4) common cause failure of the safety injection pumps to start or to run.

- SL-4 is an accident sequence which involves a small break loss of coolant followed by failure of long-term decay heat removal and failure of the SDS. For this sequence, the SIS provided the necessary makeup to the RCS and the EFWS successfully provided feedwater to the steam generators

during the first phase of plant cooldown. Long-term decay heat removal would normally be initiated to continue the plant cooldown process, but this function is not accomplished in this sequence. "Feed and bleed" is the next alternative. However, the failure of the SDS causes the "bleed" portion to be unsuccessful and consequently core damage occurs. The dominant contributors to this sequence are: (1) common cause failure of the SCS/CCW valves, and (2) failure of the "bleed" valves.

MEDIUM LOCA EVENTS

Two dominant accident sequences were identified for medium LOCA events, ML2-3 and ML1-3. The calculated frequencies for these accident sequences are 1.6E-07 and 1.4E-07, respectively. (To facilitate the definition and evaluation of the Plant Damage States (PDS) for the containment response analysis, the medium LOCA category of events were split into two separate events bases on the estimated in-vessel pressure at the time of the onset of core damage.)

- ML2-3 is an accident sequence which involves a medium LOCA event followed by failure of the SIS. Failure of the SIS during the early phase of the medium LOCA (i.e., during direct vessel injection) results in failure to provide makeup to the reactor core and also to remove heat from the core. Failure of the SIS during the latter phase of the medium LOCA (i.e., simultaneous hot-leg and direct vessel injection) results in boron crystallization which blocks flow through the core and consequently core damage occurs. The dominant contributors to this sequence are: (1) common cause failure of the hot-leg isolation valves, (2) common cause failure of the safety injection line check valves, (3) common cause failure of hot-leg check valves, (4) operator fails to initiate hot-leg injection, and (5) common cause failure of the safety injection line motor valves.

- ML1-3 is an accident sequence which is similar to ML2-3, except that the in-vessel pressure at the onset of core damage is less than the in-vessel pressure associated with ML2-3 sequence.

LARGE LOCA EVENTS

Two dominant accident sequences were identified for large LOCA events, LL-3 and LL-4. The calculated frequencies for these accident sequences are 1.1E-07 and 4.7E-09, respectively.

- LL-3 is an accident sequence which involves a large loss of coolant followed by failure of the SIS. The inventory of the Safety Injection Tanks (SITs) discharges into the RCS to provide the instantaneous makeup required following the rapid depressurization of the RCS caused by the large LOCA. The SIS then failed to provide its function. Failure of the SIS during the early phase of the medium LOCA (i.e., during direct vessel injection) results in failure to provide makeup to the reactor core and also to remove heat from the core. Failure of the SIS during the latter phase of the medium LOCA (i.e., simultaneous hot-leg and direct vessel injection) results in boron crystallization which blocks flow through the core and consequently core damage occurs. The dominant contributors to this sequence are: (1) common cause failure of the safety injection line check valves, (2) common cause failure of the safety injection line motor valves, (3) common cause failure of the hot-leg check valves, (4) common cause failure of hot-leg isolation valve, and (5) operator fails to initiate hot-leg injection.

- LL-4 is an accident sequence which involves a large loss of coolant followed by failure of the SITs. The required instantaneous makeup of reactor coolant is not provided and consequently core damage occurs. The dominant contributors to this sequence are: (1) common cause failure of safety injection line check valves, and (2) common cause failure of SIT discharge check valves.

VESSEL RUPTURE EVENT

This event is defined as any breach of the primary pressure boundary that causes loss of reactor coolant in excess of the capacity of the SIS. If this event were to occur, it leads directly to core damage and there are no sequences associated

with it. The estimate frequency for this event is 1.0E-07 per year.

OTHER TRANSIENT EVENTS

Other transients are those non-LOCA events other than loss of main feedwater in which a process parameter perturbation leads to a reactor trip. Three dominant accident sequences were identified for other transient events, TOTH-4, TOTH-5, and TOTH-9. The calculated frequencies for these accident sequences are 6.9E-08, 6.9E-09, and 2.7E-09 per year, respectively.

- TOTH-4 is an accident sequence which involves a transient other than loss of main feedwater followed by failure of long-term decay heat removal and failure of the SIS. The EFWS is used during the early phase of plant cooldown to remove decay heat from the RCS. Once SCS entry conditions are met, long-term decay heat removal is normally initiated to continue the cooldown process. This function is not accomplished for this sequence because of failure of the SCS during initiation or during operation and makeup to the emergency feedwater storage tanks is not provided. Without the added makeup the EFWS cannot continue the decay heat removal process. Failure of long-term decay heat removal using the SCS and the EFWS would then cause operator to initiate "feed and bleed". However for this sequence, the SIS also fails and "feed" cannot be accomplish. As a result, the decay heat removal process is terminated and core damage eventually occurs. The dominant contributors to this sequence are: (1) common cause failure of safety injection line check valves, (2) independent failure of CST make-up valve (check or manual), and (3) operator fails to align CST to EFW storage tanks.

- TOTH-5 is an accident sequence which involves a transient other than loss of main feedwater followed by failure of long-term decay heat removal and failure of the SDS. This sequence is similar to TOTH-4, except that "bleed" fails instead of "feed". The dominant contributors to this sequence are: (1) common cause failure of the SCS suction valves, (2) failure of the CST make-up valve (check or manual), (3) operator fails to initiate "feed and bleed", (4) common cause failure of the "feed and

bleed" valves, and (5) operator fails to align CST to EFW storage tanks.

- TOTH-9 is an accident sequence which involves a transient other than loss of main feedwater followed by failure of the EFWS, the Startup Feedwater System (SFWS), and the SDS. The inability to deliver feedwater to the steam generators terminates the preferred means of removing decay heat. "Feed and bleed" operation then becomes the next alternative for removing heat from the core. Both the "feed" portion and the "bleed" portion must function for this operation to be successful. For this sequence, the SDS also fails to provide "bleed" and therefore all mean of removing decay heat from the core is lost. Consequently, core damage eventually occurs. The dominant contributors to this sequence are: (1) common cause failure of the EFW distribution line check valves, (2) common cause failure of the EFW pump discharge check valves, (3) failure of the start-up feedwater pump to start, and (4) operator fails to initiate "feed and bleed".

<u>ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS) EVENTS</u>

Two dominant accident sequences were identified for ATWS events, ATWS-29 and ATWS-9. The calculated frequencies for these accident sequences are 4.7E-08 and 2.1E-09 per year, respectively.

- ATWS-29 is an accident sequence which involves a transient without scramming the reactor in conjunction with an adverse moderator temperature coefficient (MTC). For this sequence, the MTC is not negative enough to reverse by reactivity feedback the increasing temperature and pressure of the RCS. As a result the reactor vessel eventually fails.

- ATWS-9 is an accident sequence which involves ATWS followed by successful delivery of emergency feedwater to the steam generators, failure to deliver borated water to the RCS, and failure of "bleed" operation. Failure to borate the RCS prohibits the ability to decrease the reactivity, and therefore reactivity control cannot be accomplished. The failure of "bleed" prevents the depressurization of the RCS to allow the injection of borated water by the safety injection pumps. The dominant

contributors to this sequence are: (1) operator fails to initiate boron delivery via the CVCS, (2) operator fails to initiate "feed and bleed", and (3) common cause failure of the "bleed" valves.

LOSS OF OFFSITE POWER (LOOP)/STATION BLACKOUT (SBO) EVENTS

Three dominant accident sequences were identified for LOOP/SBO events, LOOP-12, LOOP-9, and SBO. The calculated frequencies for these sequences are 1.3E-08, 3.8E-09, and 2.1E-08 per year, respectively.

- LOOP-12 is an accident sequence which involves a loss of offsite power followed by failure of the primary safety valves to reseat and failure of the SIS. Because of the pressure transient associated with this LOOP event, the primary safety valves open to relief primary pressure. However, the valves do not re-seat as required and a PSV induced LOCA occurs. The SIS would normally provide makeup for the lost reactor coolant under these conditions, but the SIS also fails and consequently core damage eventually occurs. The dominant contributors to this sequence are: (1) failure of the primary safety valves to re-seat, (2) common cause failure of the safety injection line check valves, (3) common cause failure of the safety injection line motor valves, and (4) common cause failure of the safety injection pumps to start or to run.

- LOOP-9 is an accident sequence which involves a loss offsite power followed by failure of the EFWS, and failure of the SDS. For this sequence, the primary safety valves open and re-seat as required following the plant trip. Because the EFWS has failed, the removal of decay heat from the reactor core via the steam generators cannot be accomplished. "Feed and bleed" would be the next means of removing decay heat, but the SDS also fails. "Bleed" is not accomplish because of SDS failure. As a result, all means of removing decay heat from the core is now lost and core damage eventually occurs. The dominant contributors to this sequence are: (1) common cause failure of the EFW distribution line check valves, (2) common cause failure of the EFW pump discharge check valves, (3) operator fails to initiate "feed and bleed", and (4) common cause failure

of the "bleed" valves.

- SBO is treated as a special LOOP event. For this event, the offsite power sources, the onsite emergency diesel generators, and the alternate AC source are unavailable. Therefore, the only mitigating system available is the EFWS, using the turbine-driven pumps. After eight hours the batteries would be depleted if an AC power source is not restored. Once the batteries are depleted, long-term decay heat removal would be terminated and core damage eventually occurs. The dominant contributors to this sequence are: (1) common cause failure of the emergency diesel generators on demand, (2) failure to start and load standby AC source, (3) common cause failure of DG sequencers, and (4) failure of the emergency diesels to start and load.

LOSS OF HVAC EVENTS

One dominant accident sequence was identified for loss of HVAC events, LHV-5. The calculated frequency for this accident sequence is 3.6E-09 per year.

- LHV-5 is an accident sequence which involves loss of one division of HVAC followed by failure of long-term decay heat removal and failure of SDS. The emergency and startup feedwater systems are used to remove decay heat from the RCS until shutdown cooling entry conditions are met. Once shutdown cooling conditions are met, the SCS would then be used to remove decay heat. However for this sequence, the SCS fails either during initiation or during operation and the EFWS cannot continue to remove decay heat because makeup to the emergency feedwater storage tanks is not provided. Failure of the feedwater and shutdown cooling systems causes the preferred means of removing decay heat to be lost. "Feed and bleed" would then be the alternate means of removing decay heat from the core. The SDS is used to provide the "bleed" function in the "feed and bleed" operation, but this system also fails. Failure of both the preferred and the alternate means of removing decay heat eventually causes core damage to occur. The dominant contributors to this sequence are: (1) common cause failure of the "bleed" valves, (2) operator fails to initiate "feed

and bleed", (3) failure of the SCS suction valve, (4) failure of the CST makeup valve (check or manual), and (5) operator fails to align CST to EFW storage tanks.

LARGE SECONDARY SIDE BREAK (LSSB) EVENTS

One dominant accident sequence was identified for LSSB events, LSSB-9. The calculated frequency for this sequence is 2.2E-09 per year.

- LSSB-9 is an accident sequence which involves a large secondary side break event followed by failure of the EFWS and failure of the SDS. For this sequence, the EFWS fails to deliver feedwater to the intact steam generator. If decay heat cannot be removed via the steam generator, the "feed and bleed" would be the next alternative. Both the "feed" portion and the "bleed" portion must operate successfully to remove decay heat. The SDS which performs the "bleed" operation also fails in this sequence. The removal of decay heat from the core is therefore terminated and core damage eventually occurs. The dominant contributors to this sequence are: (1) common cause failure of the EFW distribution line check valves, (2) common cause failure of the EFW pump discharge check valves, (3) operator fails to initiate "feed and bleed", and (4) common cause failure of the "bleed" valves.

15.2.1.3 Risk-Reduction Design Features

The System 80+ possesses several features, as described in Section 15.1.1, that are beneficial to reducing the frequency of core damage. The major reduction in core damage frequency (CDF), when compared with System 80, are associated with:

- LOOP/SBO events (CDF reduced from 3.8E-05 to 4.0E-08)

- Transient (loss of main feedwater and other transient) events (CDF reduced from 1.2E-05 to 5.9E-07)

- Steam generator tube rupture events (CDF reduced from 1.1E-05 to 2.9E-07)

- Small LOCA events (CDF reduced from 9.4E-06 to 2.0E-07)

- ATWS events (CDF reduced from 4.8E-06 to 5.4E-08)

The following features of the System 80+ design are the major contributors to the reduction of the core damage frequency (with respect to the System 80 design) for the above categories of internally imitated events.

- LOOP/SBO - multiple offsite power sources, alternate standby AC power source, dedicated batteries for each emergency diesel generator, four trains of emergency feedwater (two with turbine-driven pumps), turbine-generator capable of running back to hotel loads.

- Transients - four trains of emergency feedwater (two with turbine-driven pumps), redundant sources of emergency feedwater, high reliable normally running component cooling water/station service water systems, separate startup feedwater system (actuated before EFWS), turbine generator capable of running back to hotel loads, two redundant and diverse emergency feedwater actuation systems, once through cooling capability ("Feed and Bleed").

- Steam generator tube rupture - four trains of emergency feedwater system (two with turbine-driven pumps), four trains of safety injection, once through cooling capability ("Feed and Bleed").

- Small LOCA - In-containment refueling water storage tank (eliminated the need for switch-over of pump suction), four trains of emergency feedwater system (with two turbine-driven pumps), four trains of safety injection system, once through cooling capability ("Feed and Bleed").

- ATWS - larger pressurizer, larger steam generators, safety depressurization system, diverse protection system.

Table 15.2-2

CORE DAMAGE FREQUENCY CONTRIBUTIONS FOR DOMINANT ACCIDENT SEQUENCES BY INITIATING INTERNAL EVENT

SEQUENCE NUMBER	SEQUENCE	MEAN CORE DAMAGE FREQUENCY CONTRIBUTION EVENTS/YEAR	ERROR FACTOR
LOFW-9	(LOFW)(Emergency Feedwater Fails)(Safety Depressurization for Bleed Fails)	4.6E-07	5.4
SGTR-17	(SGTR)(Injection Fails)(Agressive Secondary Cooldown Fails)	2.7E-07	18.9
SL-11	(SLOCA)(Safety Injection Fails)(Agressive Cooldown Fails)	1.6E-07	17.9
ML2-3	(Medium LOCA 2)(Safety Injection Fails)	1.6E-07	5.1
ML1-3	(Medium LOCA 1)(Safety Injection Fails)	1.4E-07	5.6
LL-3	(LLOCA)(SITs Inject OK)(Safety Injection Fails)	1.1E-07	5.3
VR	Vessel Rupture	1.0E-07	10.0
TOTH-4	(Other Transients)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(Safety Injection for Feed Fails)	6.9E-08	7.5
ATWS-29	(ATWS)(Adverse MTC)	4.7E-08	8.5
LOFW-4	(LOFW)(Emergency Feedwater OK)(Long-term Decay Heat Removal Fails)(Bleed OK)(Safety Injection for Feed Fails)	3.6E-08	6.9
SBO	Station Blackout with Battery Depletion	2.1E-08	9.2
LOFW-8	(LOFW)(Emergency Feedwater Fails)(Bleed OK)(Safety Injection for Feed Fails)	2.1E-08	4.5
SGTR-16	(SGTR)(Safety Injection Fails)(Agressive Cooldown OK)(RHR Injection Fails)	1.5E-08	9.2
LOOP-12	(LOOP)(PSV Fails to Reseat)(Safety Injection Fails)	1.3E-08	8.5

Table 15.2-2 (Cont'd)

CORE DAMAGE FREQUENCY CONTRIBUTIONS FOR DOMINANT ACCIDENT SEQUENCES BY INITIATING INTERNAL EVENT

SEQUENCE NUMBER	SEQUENCE	MEAN CORE DAMAGE FREQUENCY CONTRIBUTION EVENTS/YEAR	ERROR FACTOR
SL-10	(SLOCA)(Safety Injection Fails)(Agressive Cooldown OK) (RHR Injection Fails)	9.0E-09	8.3
SL-4	(SLOCA)(Safety Injection OK)(Deliver Feedwater OK) (Long-term Decay Heat Removal Fails)(Safety Depressurization - Bleed Fails)	8.9E-09	5.4
TOTH-5	(Other Transients)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(Safety Depressurization Fails)	6.9E-09	7.5
SGTR-12	(SGTR)(Safety Injection OK)(Feedwater Fails)(Safety Depressurization - Bleed Fails)	6.3E-09	7.5
LOFW-5	(LOFW)(Emergency Feedwater OK)(Long-term Decay Heat Removal Fails)(Safety Depressurization for Bleed Fails)	5.6E-09	9.3
LL-4	(LLOCA)(SITs Fail to Inject)	4.7E-09	6.3
SGTR-9	(SGTR)(Safety Injection OK)(Deliver Feedwater OK)(RCS Pressure Control Fails)(SG not Isolated)(Failure to Refill IRWST)	4.4E-09	12.0
LOOP-9	(LOOP)(Failure to Deliver Emergency Feedwater)(Safety Depressurization for Bleed Fails)	3.8E-09	9.1
LHV-5	(LHVAC)(Deliver Feedwater OK)(Long-term Decay Heat Removal Fails)(Safety Depressurization for Bleed Fails)	3.6E-09	17.2
TOTH-9	(Other Transients)(Feedwater Fails)(Safety Depressurization Fails)	2.7E-09	7.1
LSSB-9	(LSSB)(Safety Injection OK)(Failure to Deliver Feedwater) (Safety Depressurization for Bleed Fails)	2.2E-09	15.5
ATWS-9	(ATWS)(PSVs Open and Re-close OK)(No Consequential SGTR)(Deliver Feedwater OK)(Failure to Deliver Boron by Charging Pumps)(Safety Depressurization Fails)	2.1E-09	17.6

Table 15.2-2 (Cont'd)

CORE DAMAGE FREQUENCY CONTRIBUTIONS FOR DOMINANT ACCIDENT SEQUENCES BY INITIATING INTERNAL EVENT

SEQUENCE NUMBER	SEQUENCE	MEAN CORE DAMAGE FREQUENCY CONTRIBUTION EVENTS/YEAR	ERROR FACTOR
SGTR-15	(SGTR)(Safety Injection Fails)(Aggressive Cooldown OK)(SCS Injection OK)(Unisolable Leak in Ruptured SG)(Failure to Re-fill IRWST)	1.2E-09	13.3

15.2.1.4 Insights from the Uncertainty, Sensitivity, and Importance Analyses

15.2.1.4.1 Insights from the Uncertainty Analysis

The System 80+ PRA results for internal events, as presented Table 15.2-2 of this report, are summarized in terms of mean value and associated error factor, which is a measure of the uncertainty. (The error factor is the average ratio between the 95th percentile to the 50th percentile and the 50th percentile to the 5th percentile.) The first 14 accident sequences listed in Table 15.2-2 contribute more than 90% of the overall total core damage frequency for internal events of the System 80+ design. The magnitude of uncertainties that characterize the first 14 dominant accident sequences is shown in Table 15.2-3. This table lists the dominant core damage sequences, the mean core damage frequency and associated error factor for each of the dominant core damage sequences, the major contributors to each dominant sequence, and the error factor associated with each of the major contributors. The major insights from the uncertainty analysis are listed below:

- 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.

- 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 included hardware failures such as common cause failure of the safety injection pumps (EF = 16.1), common cause failure of the emergency DG sequencers (EF = 11.2), independent failure of the CST manual makeup valve (EF = 15), and vessel rupture (EF = 10.0). The human errors with large uncertainties include operator fails to initiate "feed and bleed" (EF = 11.7), operator fails to perform aggressive secondary side cooldown following SGTR (EF = 15.0), and operator fails to perform aggressive cooldown following a small LOCA (EF = 16.6).

- The hardware failures and human errors with large uncertainties associated with them are the major contributors to the uncertainty associated with the calculated core damage frequency for internal events.

15.2.1.4.2 Insights from Level I Sensitivity Analyses

Several cases of sensitivity analyses were preformed for the System 80+ design to determine what impact, if any, the following issues would have on the core damage frequency. The sensitivity insights are summarized in Table 15.2-4.

<u>OVERALL OPERATOR ERROR RATE</u>

The probability that the operator fails to perform a specified task was determined using the SHARP[39] methodology. It has been shown in previous PRAs that the core damage frequency can be sensitive to human error probabilities. As a result, a sensitivity was performed to determine the impact on the core damage frequency for internal events. All operator error probabilities and non-recovery probabilities were increase by a factor of 10 and the total core damage frequency for internal events was then requantified. The results for this case are presented in Table 15.2-4 as case No. 1. For this case, the results show that the "present" core damage frequency would increase by a factor of approximately 6. This implies that the core damage frequency for the System 80+ design is somewhat sensitive to operator error probabilities.

<u>CONTROL ROOM RESPONSE</u>

Several types of operator actions are performed during the progression of an accident sequence. Actions are performed inside the control room as well as outside the control room. An issue has arisen regarding the capability of the operator to perform mitigating actions outside the control room once an accident or transient has occurred. To address this issue, a sensitivity analysis was performed which credited only the operator actions that took place from the control room. The results, case No. 2 of Table 15.2-4, show that the "present" core damage frequency would increase by a two orders of magnitude. Consequently,

the System 80+ design is extremely sensitive to operator actions which are performed outside the control room during the progression of an accident sequence.

MOTOR-OPERATED VALVE FAILURE RATE

A large number of motor-operated valves are used in safety related systems of the System 80+ design. In general, these valves are required to change position in order for the systems to perform their safety related functions. There has been concern that the failure rates of motor-operated valves have been under-estimated. Consequently, a sensitivity analysis was performed to address this issue. The results for this case are presented in Table 15.2-4 as case No. 3. The results show that the "present" core damage frequency would increase by a factor of approximately 3 and the core damage frequency for internal events is not highly sensitive to the failure rates of motor-operated valves.

SITs FOR MEDIUM LOCA

For the System 80+ PRA, a best-estimate thermo-dynamic analysis was performed to confirm the belief that the Safety Injection Tanks (SITs) were not needed to prevent core damage following a medium LOCA event. In previous PRAs, it was assumed that three of the four SITs must inject to prevent core damage from occurring following a large or medium LOCA event. This assumption is included only in the large LOCA model and not in the medium LOCA models, and a sensitivity analysis was therefore performed to address this issue. The results for this case are presented in Table 15.2-4 as case No. 4. There were no measurable changes in the "present" core damage frequency when the SITs were credited for preventing core damage following a medium LOCA event. The System 80+ design is not sensitive to this issue.

FEASIBILITY OF AGGRESSIVE SECONDARY SIDE COOLDOWN

For small LOCAs and Steam Generator Tube Ruptures (SGTRs), it was assumed that if safety injection was not available for inventory control, the RCS could be depressurized via a rapid cooldown of the secondary side then aligning the SCS

pumps to provide injection for RCS inventory control. This assumption was based on analyses for the System 80 plants documented in CEN-239[9,10,11]. A confirmatory analysis was performed to demonstrate that these analyses are applicable to the System 80+ design. However, the impact on core damage if aggressive cooldown is not feasible still remained an issue and was therefore addressed via sensitivity analysis. The results for this case are presented in Table 15.2-4 as case No. 5. The results indicate that the "present" core damage frequency would increase by a factor of approximately 5, and the System 80+ PRA results for internal events are not highly sensitively to the ability to perform aggressive cooldown of the secondary side following a small LOCA or SGTR event.

RCP SEAL FAILURE FOLLOWING STATION BLACKOUT EVENT

With a loss of station AC power (Station Blackout), cooling water to the seals of the Reactor Coolant Pump (RCP) will be lost. The NRC has postulated in their evaluation of Station Blackout[57] that under these conditions, the seals will begin to degrade and gross seal leakage on the order of several hundred gallons per minute may occur. The CEOG contends that this in not credible for pumps used in C-E plants[58]. A sensitivity analysis was performed to assess the impact on core damage frequency if there is a finite probability that the System 80+ RCP seals will fail following a Station Blackout (SBO) or loss of cooling water event.

In the System 80+ PRA, SBO is defined as a Loss of Offsite Power (LOOP) with demand failure of both emergency diesel generators and failure of the standby AC power source. If following an SBO, RCP failures occur, core damage can be prevented if offsite power is restored and the injection pumps are started before the core is uncovered. The available time to recover offsite power is a function of the RCP seal leak rate. For this sensitivity analysis, it was assumed that the RCP seal leak rate was such that the time available to recover offsite power before the core would become uncovered was one hour.

Loss of component cooling water, as an initiating event, will also result in the loss of cooling water flow to the seals of two of the four RCPs. The results for this case are presented in Table 15.2-4 as case No. 6.

The modeling and quantification of the effects of RCP seal failure on core damage frequency show that the "present" core damage frequency for internal events would increase by less than 1%. Thus, the System 80+ core damage frequency for internal events is not sensitive to RCP seal failure following a SBO or loss of cooling water event.

COMPONENTS UNAVAILABLE DUE TO MAINTENANCE

Components in safety related systems are periodically tested per technical specification requirements. In some cases, the components may be unable to perform their safety related function during the test. In addition, if the component is found to be failed during the test, it is taken out of service for maintenance. While the component is out of service for maintenance, it is unable to perform its safety related function. Component unavailability due to test and maintenance was included in the System 80+ PRA models. A sensitivity analysis was performed to evaluate the impact on core damage frequency if it was assumed that all components were able to perform their safety related function while in test and that no maintenance was performed on safety related equipment while the plant was at power.

The results for this case are presented in Table 15.2-4 as case No. 7. The results show that the core damage frequency for internal events did not change from its "present" value.

ADVERSE MTC

An ATWS is an event in which an anticipated transient occurs but the reactor is not shutdown by automatic insertion of the control rods. One factor that influences the progression of an ATWS event is the Moderator Temperature Coefficient (MTC). If the MTC is more positive than a calculated critical value, the peak RCS pressure will exceed the level C stress limit pressure and a non-mitigatable LOCA is assumed to occur. For System 80+, the critical MTC was calculated to be -0.30E-4 $\Delta\rho/°F$. For MTC values more positive than -0.30E-4 $\Delta\rho/°F$, the peak RCS pressure will exceed the level C stress limit pressure if less than tree PSVs open. For System 80+, it has been determined that the MTC

value should be less than -0.30E-4 $\Delta\rho/^{\circ}F$ for 99% of the core life. The dominant ATWS core damage sequence is (ATWS occurs) AND (MTC is adverse). A sensitivity analysis was performed to assess the impact on core damage frequency if the MTC was found to be adverse over a larger fraction of the core life.

The results for this case are presented in Table 15.2-4 as case No. 8. The results show that the "present" core damage frequency for internal events would increase by a factor of 1.3. This indicates that the System 80+ core damage frequency for internal events is not very sensitive to the adverse MTC probability.

LOSS OF OFFSITE POWER FREQUENCY

For the System 80+ PRA, the core damage frequency contribution attributable to events initiated by LOOP was calculated to be 2.0E-08 per year. This represents approximately 1% of the total core damage frequency for internal events. In past PRAs, the core damage frequency attributable to LOOP has been greater in both absolute value and relative contribution to the total. There are a number of reasons for the reduction in the core damage frequency contribution for LOOP for System 80+. These include: (1) the capability of the main turbine/generator to runback and pickup hotel load on loss of offsite power, (2) two separate switchyards for incoming power, (3) a four train EFW system with two 100% capacity turbine driven pumps, (4) 6 vital batteries which provide an 8 hour coping capability, and (5) a standby combustion turbine which can backup the emergency diesel generators.

Based on the first three features, a LOOP initiating event was defined as a loss of site power which required the startup and loading of the emergency diesel generators. The LOOP frequency calculated for System 80+ is 5.0E-3 per year. This is almost an order of magnitude lower than a LOOP frequency based purely on the failure of the grid. This value, as presented in the KAG[7], is 3.5E-2/year. A sensitivity analysis was performed to determine the impact on the "present" core damage frequency if the LOOP frequency was increased by an order of magnitude. The results for this case are presented in Table 15.2-4 as case No. 9. The results show that the core damage frequency would increase by

approximately 18%. This indicates that the System 80+ overall core damage frequency is somewhat sensitive to the frequency of LOOP.

An additional LOOP sensitivity analysis was performed to evaluate the effect of changing the base Loss of Grid frequency from 0.035 per year to 0.15 per year. The effective System 80+ LOOP frequency for this case was 2.32E-2 per year as opposed to the base case frequency of 5.0E-3 per year. For this case, the "present" core damage frequency, shown as case No. 9A in Table 15.2-4, would increase by approximately 7% which is not significant.

VESSEL RUPTURE

Vessel rupture was originally evaluated in WASH 1400. It is typically defined as a rupture of the vessel or a large LOCA in excess of the ECCS capabilities. Vessel rupture is assumed to directly lead to core damage. This event and its initiating frequency have essentially been accepted as is since WASH 1400 because it has little impact on the overall core damage frequency for existing plants. However, it contributes approximately 6% of the core damage frequency for internal events. With current materials and current manufacturing methods, it has been questioned as to whether or not vessel rupture is a credible event for an ALWR. A sensitivity analysis was therefore performed to evaluate the impact on plant core damage frequency if vessel rupture was assumed not to be credible. As expected, the "present" core damage frequency for internal events would decrease, by approximately 6%. The results for this case are presented in Table 15.2-4 as case No. 10A.

COMMON CAUSE FAILURES

As shown in Section 15.2.1.4.3, the System 80+ plant core damage frequency for internal events is dominated by common cause failures. It has been contended that with: (1) complete divisional separation, (2) improved staff training, and (3) improved maintenance techniques and proper selection of components; the potential for common mode failure can be essentially eliminated. A sensitivity analysis was performed to evaluate the impact on plant core damage frequency of the assumption that all common mode failures except for diesels and batteries

were eliminated. The results for this case are presented as case No. 10B of Table 15.2-4. The results of this analysis show that the "present" core damage frequency would decrease by a factor of approximately 8. This is somewhat of a modest decrease. Combining the assumptions that vessel rupture is not credible and that common mode failure of equipment other than the diesel generators and batteries are not credible would decrease the "present" core damage frequency by a factor of 14. The combined impact of these two assumptions on core damage frequency would be significant.

15.2.1.4.3 Insights from the Importance Analyses

Importance analyses were performed at the system level and the component level for the System 80+ design. At the system level, the risk achievement and risk reduction measures of importance were calculated for the mitigating systems of the System 80+ design. The risk achievement worth is expressed as the ratio giving the factor by which risk increases due to the system of concern not being available. Likewise, the risk reduction worth is a measure of the risk that would be reduced by decreasing the unavailability of the system to zero (i.e., the system is always operating or is operable when demanded). The system level measure of importance are provided in Table 15.2-5. The results are sorted on the risk achievement measure, with the most important system (highest risk achievement worth listed first followed by the next and the least important (lowest risk achievement worth) listed last. Some insights gained from the system level importance analyses are listed below:

- The systems that would adversely impact (increase) the overall core damage frequency for internal events the most include the electrical distribution system, the emergency feedwater system, the safety injection system, and the component cooling water/station service water systems.

- The system that would be the most beneficial (i.e., system with a high reliability) in reducing the overall core damage frequency for internal events is the emergency feedwater system followed by the safety injection system and then the safety depressurization system.

Several importance measures were also calculated for the events of the System 80+ PRA. The calculated component importance measures are:

- FUSSELL-VESELY - The Fussell-Vesely importance gives the risk associated with a given component or event. This importance measure determines how much the component is contributing to the overall core damage frequency.

- BIRNBAUM - The Birnbaum importance measures the difference in core damage frequency with and without the occurrence of an event. The Birnbaum gives the increase in risk associated with the failure of a component.

- CRITICALITY - The Criticality importance measure gives the core damage frequency that the system failure is a result of the failure of a critical component.

- RISK ACHIEVEMENT WORTH - The Risk Achievement Worth is expressed as a ratio giving the factor by which risk increases due to a component not being available.

- RISK REDUCTION WORTH - The Risk Reduction Worth is a measure of the risk that would be reduced by reducing the unavailability of the component of interest to zero.

- UNCERTAINTY IMPORTANCE - The Uncertainty Importance is the measure of the standard deviation about the mean frequency. By reducing the uncertainty of a given event, the uncertainty of the overall core damage frequency will also decrease. This importance measure identifies those events whose uncertainty contributes the most to the overall core damage frequency.

The importance measures are calculated to analyze how each event influences the overall core damage frequency and to analyze how the events ranks against one another.

A list of component importance measures is presented in Table 15.2-6. The components are sorted according to their risk achievement worth. The most important (highest risk achievement worth) component is listed first followed by the next important component. Finally, the least important (lowest risk achievement worth) component is listed last. The following general insights are provided for the component importance measures:

- Of the top fifty components listed, with respect to risk achievement worth, common cause failures are the most important category (36), followed by initiating events (8), then independent faults (3), and then human errors (3).

- Because of the redundancy and diversity of the mitigating systems, independent faults are not the most important (high risk achievement worth) events that would adversely impact the core damage frequency for internal events.

- Common cause failures of Electrical Distribution System components are important (high risk achievement worth) events. These events include: (1) common cause failure of the 125 VDC class 1E buses, (2) common cause failure of the 480 VAC load center transformers, (3) common cause failure of the 4.16 KV class 1E buses, (4) common cause failure of 480 VAC class 1E load centers, and (5) common cause failure of 480 VAC motor control centers.

- Common cause failures of the distribution line check valves and the pump discharge check valves in the Emergency Feedwater System are also important (high risk achievement worth) events.

- Several operator errors also have high risk achievement worth. Such events include: (1) operator fails to initiate hot-leg injection, (2) operator fails to align the CST to the emergency feedwater storage tanks, and (3) operator fails to initiate "feed and bleed".

- The most important (high risk achievement worth) independent faults

are associated with failure of the CST valves. These valves are used primarily to provide makeup to the emergency feedwater storage tanks in order to continue and maintain the decay heat removal from the reactor core.

- Loss of main feedwater is the most important (highest risk reduction worth) event in terms of reducing the "present" core damage frequency for internal events.

- Common cause failure of the safety injection line check valves (which are used during safety injection, SCS injection, and shutdown cooling operations) is the next important event in terms of risk reduction. The "present" core damage frequency of internal events would decrease by approximately 31%, if common cause failure of the safety injection line check valves never occurred.

- Operator errors also play a big part in reducing the "present" core damage frequency for internal events. The most important operator errors in terms of risk reduction worth are: (1) operator fails to provide aggressive cooldown following a steam generator tube rupture event - 23% reduction, (2) operator fails to provide aggressive cooldown following a small LOCA event - 20% reduction, and (3) operator fails to initiate "feed and bleed" - 20% reduction.

Table 15.2-3

Major Contributors to the Uncertainty of CDF (Internal Events) for System 80+

DOMINANT SEQUENCE	MEAN CDF/yr	EF	MAJOR CONTRIBUTORS	
			Description	EF
LOFW-9	4.6E-07	5.4	Loss of main feedwater event	3.0
			Common cause failure of EFW distribution line check valves	3.2
			Common cause failure of EFW pump discharge check valves	3.2
			Common cause failure of "bleed" valves	1.6
			Operator fails to initiate "feed and bleed"	11.7
SGTR-17	2.7E-07	18.9	Steam generator tube rupture event	5.0
			Common cause failure of safety injection line check valves	3.2
			Common cause failure of safety injection line motor valves	1.2
			Common cause failure of safety injection pumps to start	16.1
			Common cause failure of safety injection pumps to run	5.8
			Operator fails to perform aggressive secondary side cooldown	15.0
SL-11	1.6E-07	17.9	Small LOCA event	5.0
			Common cause failure of safety injection line check valves	3.2
			Common cause failure of safety injection line motor valves	1.2
			Common cause failure of safety injection pumps to run	5.8
			Common cause failure of safety injection pumps to start	16.1
			Operator fails to perform aggressive cooldown	16.6
ML2-3	1.7E-07	5.1	Medium LOCA 2 event	5.0
			Common cause failure of hot-leg injection valves	1.6
			Common cause failure of safety injection line check valves	3.2
			Common cause failure of hot-leg check valves	3.2
			Common cause failure of safety injection line motor valves	1.2
			Operator fails to initiate hot-leg injection	4.1
ML1-3	1.4E-07	5.6	Medium LOCA 1 event	5.0
			Common cause failure of hot-leg injection valves	1.6
			Common cause failure of safety injection line check valves	3.2
			Common cause failure of hot-leg check valves	3.2
			Common cause failure of safety injection line motor valves	1.2
			Operator fails to initiate hot-leg injection	4.1

Table 15.2-3

Major Contributors to the Uncertainty of CDF (Internal Events) for System 80+

DOMINANT SEQUENCE	MEAN CDF/yr	EF	MAJOR CONTRIBUTORS	
			Description	EF
LL-3	1.1E-07	5.3	Large LOCA event	5.0
			Common cause failure of safety injection line check valves	3.2
			Common cause failure of safety injection line motor valves	1.2
			Common casue failure of hot-leg check valves	3.2
			Common cause failure of hot-leg injection valves	1.6
			Operator fails to initiate hot-leg injection	4.1
VR	1.0E-07	10.0	Vessel rupture event	10.0
TOTH-4	6.9E-08	7.5	Transient event other than loss of main feedwater	3.0
			Common cause failure of safety injection line check valves	3.2
			CST manaul makeup valve fails to open	15.0
			CST makeup check valve fails to open	3.2
			Operator fails to align CST to EFW storage tanks	3.4
ATWS-29	4.7E-08	8.5	ATWS event	6.3
			Adverse moderator temperature coefficient	3.0
LOFW-4	3.9E-08	6.9	Loss of main feedwater event	3.0
			Common cause failure of safety injection line check valves	3.2
			CST manual makeup valve fails to open	15.0
			CST makeup check valve fails to open	3.2
			Operator fails to align CST to EFW storage tanks	3.4
SBO	2.1E-08	9.2	Loss of offsite power event	5.0
			Common cause demand failure of the emergency DGs	3.2
			Common cause failure of DG sequencers	11.2
			Common cause failure of DG room dampers	2.3
			Emergency DG fails to start	3.2
			Failure to start and load standby AC power	3.0
			Operator fails to recover offsite power in 10 hours	1.0

Table 15.2-3

Major Contributors to the Uncertainty of CDF (Internal Events) for System 80+

DOMINANT SEQUENCE	MEAN CDF/yr	EF	MAJOR CONTRIBUTORS	
			Description	EF
LOFW-8	2.1E-08	4.5	Loss of main feedwater event	3.0
			Common cause failure of EFW distribution line check valves	3.2
			Common cause failure of EFW pump discharge check valves	3.2
			Common cause failure of safety injection line check valves	3.2
			Common cause failure of safety injection line motor valves	1.2
			Common cause failure of safety injection pumps to run	5.8
SGTR-16	1.5E-08	9.2	Steam generator tube rupture event	5.0
			Common cause failure of safety injection line check valves	3.2
			Common cause failure of safety injection pumps to run	5.8
			Common cause failure of safety injection pumps to start	16.1
			Operator fails to align SCS for injection	4.0
LOOP-12	1.3E-08	8.5	Loss of offsite power event	5.0
			Failure of primary safety valve to reseat	3.0
			Common cause failure of safety injection line check valves	3.2

Table 15.2-4
SUMMARY OF SYSTEM 80+ PRA SENSITIVITY ANALYSIS RESULTS

CASE No.	DESCRIPTION	MODELED AS	PRESENT CDF	SENSITIVITY CDF	CHANGE FACTOR
1	Operator Error Rate - General	Increase all operator error rates by factor of 10	1.7E-06	9.8E-06	5.9
2	Operator Error Rate - Control Room Response	Set operator error rate for all actions performed outside of the control room to 1.0	1.7E-06	1.8E-04	108.4
3	Motor-Operated Valve Failure Rate	Increase generic failure rates for all MOV failures by factor of 10. Increase MOV common cause failure rates by factor of 10.	1.7E-06	5.7E-05	3.4
4	SIT Injection for Medium LOCA	Add SIT injection failure cutsets for MLOCA1 and MLOCA2 based on large LOCA SIT failure	1.7E-06	1.6E-06	-
5	Aggressive Secondary Cooldown not Feasible	Delete Small LOCA sequences 9 and 10 and SGTR sequences 15 and 16. Set operator error rate for failure to initiate aggressive secondary cooldown to 1.0	1.7E-06	7.7E-06	4.6
6	RCP Seal Failure on Station Blackout or Loss of Seal Injection or Seal Cooling	Create models for core damage due to RCP seal failure on SBO and Loss of Component Cooling Water	1.7E-06	1.7E-06	-
7	Components Unavailable Due to Maintenance	Set unavailability due to test and maintenance to 0.0 for all components	1.7E-06	1.7E-06	-
8	Adverse MTC More Probable	Increase probability of having an adverse MTC when an ATWS occurs by a factor of 10	1.7E-06	2.1E-06	1.3
9	LOOP Frequency higher	Increase LOOP frequency by factor of 10	1.7E-06	2.0E-06	1.2
9A	LOOP Frequency Higher	Increase Loss of Grid Frequency to 0.15/year	1.7E-06	1.8E-06	1.1
10A	Vessel Rupture Not Credible	Set Vessel Rupture Rate to 0.0	1.7E-06	1.6E-06	-1.1
10B	Common Mode Failure Not Credible	Set all Common Mode Failure Ratess except for Diesels and Batteries to 0.0	1.7E-06	2.2E-07	-7.7
10C	Vessel Rupture and Common Mode Not Credible	Set Vessel Rupture and all Common Mode Failure Rates Except for Diesels and Batteries to 0.0	1.7E-06	1.2E-07	-14.4

Table 15.2-5
SYSTEM IMPORTANCE FOR SYSTEM 80+ PRA FOR INTERNAL EVENTS
(* - Sorted by this measure)

SYSTEM NAME	* RISK ACHIEVEMENT WORTH	RISK REDUCTION WORTH
ELECTRICAL DISTRIBUTION SYSTEM	4.12E+05	1.04
EMERGENCY FEEDWATER SYSTEM	1.73E+05	2.47
SAFETY INJECTION SYSTEM	9.76E+03	2.22
COMPONENT COOLING/STATION SERVICE WATER SYSTEM	8.73E+03	1.00
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM	3.84E+03	1.01
SHUTDOWN COOLING SYSTEM	4.65E+02	1.09
SAFETY DEPRESSURIZATION SYSTEM	2.13E+02	1.34
SAFETY INJECTION TANKs	4.50E+01	1.00
CONTAINMENT SPRAY SYSTEM	1.39E+01	1.00
STEAM REMOVAL SYSTEM	1.00E+01	1.00
RCS PRESSURE CONTROL SYSTEM	3.93E+00	1.00
START-UP FEEDWATER SYSTEM	3.05E+00	1.00
CHEMICAL & VOLUME CONTROL SYSTEM	1.98E+00	1.00
INSTRUMENT AIR SYSTEM	1.00E+00	1.00

Table 15.2-6

COMPONENT IMPORTANCES FOR SYSTEM 80+ PRA FOR INTERNAL EVENTS
(* - Sorted by this measure)

Event Name	Prob	Fuss-Vese	.Risk Ach.	Risk Red.	Birnbaum	Critical	Unc. Imp.
VR	9.94E-08	6.61E-02	8.88E+05	1.09E+00	1.00E+00	6.61E-02	2.20E-07
ISLOCA	5.51E-10	2.39E-04	7.94E+05	1.00E+00	8.78E-01	2.39E-04	1.04E-08
ELCX125C1E	1.06E-07	3.21E-02	3.21E+05	1.04E+00	4.19E-01	3.21E-02	5.29E-08
ATWS	4.69E-06	3.78E-02	9.11E+03	1.04E+00	1.06E-02	3.78E-02	7.20E-08
EXLXESF	3.68E-07	2.80E-03	7.62E+03	1.00E+00	9.72E-03	2.80E-03	4.11E-09
ELBX416ESF	1.11E-07	7.90E-04	7.25E+03	1.00E+00	9.19E-03	7.90E-04	-1.42E-19
ELLXESF	1.08E-07	7.19E-04	6.91E+03	1.00E+00	8.76E-03	7.19E-04	-4.93E-18
ELMXMCC1E	1.05E-07	7.24E-04	6.91E+03	1.00E+00	8.76E-03	7.24E-04	2.33E-09
AVCXEFWP	2.81E-05	1.26E-01	5.00E+03	1.17E+00	7.23E-03	1.26E-01	1.56E-07
AVCXDIST	2.80E-05	1.27E-01	5.00E+03	1.17E+00	7.23E-03	1.27E-01	1.56E-07
MLOCA1	6.96E-05	1.03E-01	1.72E+03	1.13E+00	2.01E-03	1.03E-01	1.65E-07
MLOCA2	7.69E-05	1.11E-01	1.72E+03	1.15E+00	2.01E-03	1.11E-01	1.87E-07
LLOCA	6.90E-05	8.16E-02	1.36E+03	1.10E+00	1.55E-03	8.16E-02	1.20E-07
JVCXD-SET2	1.45E-04	6.70E-02	5.00E+02	1.09E+00	6.96E-04	6.70E-02	8.09E-08
HVCXD-SET6	5.50E-04	1.93E-01	4.03E+02	1.31E+00	7.63E-04	1.93E-01	3.21E-07
HVMXD-SET3	1.38E-04	5.37E-02	3.95E+02	1.06E+00	7.53E-04	5.37E-02	2.78E-08
HPSXR-SET2	5.60E-05	2.07E-02	3.94E+02	1.02E+00	7.51E-04	2.07E-02	5.49E-08
HPSXD-SET2	3.46E-05	1.21E-02	3.92E+02	1.01E+00	7.48E-04	1.21E-02	7.09E-08
HVCXD-SET5	2.82E-05	1.09E-02	3.91E+02	1.01E+00	7.46E-04	1.09E-02	1.68E-08
HVCXD-SET7	2.87E-05	1.11E-02	3.91E+02	1.01E+00	7.46E-04	1.11E-02	1.65E-08
HBDXD-SET2	1.35E-05	5.25E-03	3.90E+02	1.01E+00	7.46E-04	5.25E-03	-2.36E-19
ECBX125C1E	9.90E-07	2.87E-04	2.91E+02	1.00E+00	3.75E-04	2.87E-04	4.54E-10
HHFFHOTLEG	1.39E-04	2.21E-02	1.65E+02	1.02E+00	2.16E-04	2.21E-02	3.06E-08
HVCXD-SET4	1.41E-04	2.24E-02	1.65E+02	1.02E+00	2.16E-04	2.24E-02	2.23E-08
AVNAEF-215	3.81E-04	3.68E-02	1.30E+02	1.05E+00	1.67E-04	3.68E-02	1.76E-07
AVCAEF-214	2.02E-04	2.51E-02	1.29E+02	1.03E+00	1.66E-04	2.51E-02	2.65E-08
AHFDCST	1.07E-04	1.34E-02	1.27E+02	1.01E+00	1.64E-04	1.34E-02	1.51E-08
HVMXD-SET2	7.97E-04	9.49E-02	1.21E+02	1.11E+00	1.59E-04	9.49E-02	3.53E-08
AVCDEF-214	7.87E-06	5.58E-04	7.33E+01	1.00E+00	8.96E-05	5.58E-04	7.66E-10
LVCXD-SET1	3.45E-05	1.94E-03	5.76E+01	1.00E+00	7.33E-05	1.94E-03	1.58E-09
HVCXD-SET1	3.47E-05	1.99E-03	5.76E+01	1.00E+00	7.33E-05	1.99E-03	-2.30E-18
HBDXD-SET1	1.55E-05	8.36E-04	5.47E+01	1.00E+00	6.90E-05	8.36E-04	1.23E-09
HVCXD-SET2	3.49E-05	1.89E-03	5.47E+01	1.00E+00	6.90E-05	1.89E-03	2.50E-09
HPSXD-SET1	6.13E-05	2.96E-03	5.47E+01	1.00E+00	6.90E-05	2.96E-03	1.22E-08
HPSXR-SET1	7.09E-05	3.93E-03	5.47E+01	1.00E+00	6.90E-05	3.93E-03	6.48E-09
HVMXD-SET1	1.87E-04	1.00E-02	5.47E+01	1.01E+00	6.90E-05	1.00E-02	4.45E-09
HVCXD-SET3	6.00E-04	3.12E-02	5.47E+01	1.03E+00	6.90E-05	3.12E-02	3.31E-08
SLOCA	3.09E-03	8.82E-02	3.70E+01	1.22E+00	6.79E-05	8.82E-02	2.40E-07
SGTR	4.56E-03	1.20E-01	3.61E+01	1.28E+00	6.38E-05	1.20E-01	3.55E-07
VHFFFEEDBLEED	8.99E-03	1.37E-01	1.93E+01	1.20E+00	2.60E-05	1.37E-01	2.68E-07
VVMXBLDV	4.84E-03	8.37E-02	1.85E+01	1.10E+00	2.47E-05	8.37E-02	3.43E-08
VCIXBLDV	9.32E-05	1.47E-03	1.69E+01	1.00E+00	2.25E-05	1.47E-03	2.25E-09
APAXDEFWP102-104	2.27E-04	2.76E-03	1.47E+01	1.00E+00	2.15E-05	2.76E-03	1.27E-08
EBDXDDG201	1.34E-05	1.68E-04	1.35E+01	1.00E+00	1.52E-05	1.68E-04	6.19E-09
EWFXDDGRM	2.71E-05	3.38E-04	1.35E+01	1.00E+00	1.52E-05	3.38E-04	-1.09E-18
AVMXEF102-103	2.00E-04	2.33E-03	1.26E+01	1.00E+00	1.85E-05	2.33E-03	1.86E-09
EDDXDDGA-B	2.84E-04	2.98E-03	1.17E+01	1.00E+00	1.30E-05	2.98E-03	2.00E-09
AVNXEF218-219	3.76E-05	4.43E-04	1.17E+01	1.00E+00	1.46E-05	4.43E-04	1.77E-09
ESXXSEQ	2.25E-04	2.29E-03	1.16E+01	1.00E+00	1.28E-05	2.29E-03	5.58E-09
EJEXDGRM	1.35E-04	1.47E-03	1.15E+01	1.00E+00	1.27E-05	1.47E-03	2.13E-09
APAXREFWP102-104	5.94E-05	6.12E-04	1.03E+01	1.00E+00	1.43E-05	6.12E-04	3.47E-09
JVMXSI-651/654	8.01E-04	5.43E-03	7.71E+00	1.01E+00	1.03E-05	5.43E-03	2.39E-09
JVCXD-SET1	3.56E-05	2.27E-04	7.43E+00	1.00E+00	9.82E-06	2.27E-04	3.21E-09
JVCXD-SET3	3.58E-05	2.27E-04	7.43E+00	1.00E+00	9.82E-06	2.27E-04	1.96E-09
LOOP	5.06E-03	2.80E-02	6.83E+00	1.03E+00	7.05E-06	2.80E-02	4.09E-08

Table 15.2-6 (Cont'd)

COMPONENT IMPORTANCES FOR SYSTEM 80+ PRA FOR INTERNAL EVENTS
(* - Sorted by this measure)

Event Name	Prob	Fuss-Vese	*Risk Ach.	Risk Red.	Birnbaum	Critical	Unc. Imp.
ABDXEFWP102-104	3.01E-05	1.47E-04	5.89E+00	1.00E+00	7.70E-06	1.47E-04	6.43E-09
JHFDRHRI	3.32E-03	1.56E-02	5.72E+00	1.02E+00	6.57E-06	1.56E-02	2.16E-08
JHFDSCSLTC	1.14E-04	5.46E-04	5.60E+00	1.00E+00	7.38E-06	5.46E-04	-1.28E-17
DVMXD-ADVSG2	2.00E-04	8.21E-04	5.14E+00	1.00E+00	5.80E-06	8.21E-04	-7.38E-18
DVMXD-ADVSG1	2.01E-04	8.37E-04	5.14E+00	1.00E+00	5.80E-06	8.37E-04	-3.16E-18
SE-PSVFTO	8.37E-05	3.28E-04	4.83E+00	1.00E+00	4.69E-06	3.28E-04	-7.47E-18
SE-MTC	9.86E-03	3.53E-02	4.79E+00	1.04E+00	4.69E-06	3.53E-02	3.33E-08
SE-PSV	2.78E-03	9.62E-03	4.43E+00	1.01E+00	4.45E-06	9.62E-03	9.12E-09
GVCXSI-157/158	3.60E-05	1.22E-04	4.40E+00	1.00E+00	4.86E-06	1.22E-04	-1.66E-17
PHFFSIPUMP	2.07E-04	6.47E-04	3.93E+00	1.00E+00	4.13E-06	6.47E-04	-1.46E-17
APTXDP101-103	1.19E-03	3.30E-03	3.80E+00	1.00E+00	4.48E-06	3.30E-03	4.18E-09
AHFFASCSGTR	6.61E-02	9.99E-02	3.65E+00	1.23E+00	3.91E-06	9.99E-02	6.82E-07
AHFFASCSLOCA	6.82E-02	7.60E-02	2.89E+00	1.20E+00	2.64E-06	7.60E-02	5.18E-07
EBGPTT	8.09E-04	1.54E-03	2.88E+00	1.00E+00	2.49E-06	1.54E-03	1.32E-09
AVDXEF108-109	2.53E-04	3.76E-04	2.65E+00	1.00E+00	2.60E-06	3.76E-04	-6.55E-18
AVDXEF112-113	2.58E-04	4.16E-04	2.65E+00	1.00E+00	2.60E-06	4.16E-04	-9.57E-18
CVDXCC102-202	1.37E-04	1.73E-04	2.44E+00	1.00E+00	1.89E-06	1.73E-04	6.11E-09
HVMASI-609	3.97E-03	5.76E-03	2.36E+00	1.01E+00	1.79E-06	5.76E-03	2.38E-09
HVMASI-604	3.97E-03	5.76E-03	2.36E+00	1.01E+00	1.79E-06	5.76E-03	2.38E-09
HVMASI-331	3.97E-03	5.76E-03	2.36E+00	1.01E+00	1.79E-06	5.76E-03	2.38E-09
HVMASI-321	3.97E-03	5.76E-03	2.36E+00	1.01E+00	1.79E-06	5.76E-03	2.38E-09
HVCASI-532	2.02E-04	2.59E-04	2.30E+00	1.00E+00	1.71E-06	2.59E-04	2.47E-09
HVCASI-533	2.02E-04	2.59E-04	2.30E+00	1.00E+00	1.71E-06	2.59E-04	2.47E-09
HVCASI-522	2.02E-04	2.59E-04	2.30E+00	1.00E+00	1.71E-06	2.59E-04	2.47E-09
HVCASI-523	2.02E-04	2.59E-04	2.30E+00	1.00E+00	1.71E-06	2.59E-04	2.47E-09
FSSXSIAS	3.01E-03	3.68E-03	2.26E+00	1.00E+00	2.25E-06	3.68E-03	4.81E-09
JVMXD-SET1	2.00E-04	2.46E-04	2.24E+00	1.00E+00	1.89E-06	2.46E-04	9.34E-10
JVMXD-SET2	1.97E-04	2.44E-04	2.24E+00	1.00E+00	1.89E-06	2.44E-04	-2.46E-18
APTXRP101-103	1.26E-04	1.58E-04	2.12E+00	1.00E+00	1.86E-06	1.58E-04	5.94E-09
LSSB	1.58E-03	1.35E-03	1.86E+00	1.00E+00	1.16E-06	1.35E-03	3.93E-09
FHFFSIAS	4.62E-03	3.68E-03	1.80E+00	1.00E+00	1.35E-06	3.68E-03	3.56E-09
VCIPRC-409	4.85E-04	3.71E-04	1.77E+00	1.00E+00	1.09E-06	3.71E-04	1.48E-09
VCIPRC-407	4.85E-04	3.71E-04	1.77E+00	1.00E+00	1.09E-06	3.71E-04	1.48E-09
VVMARC-407	2.39E-02	1.85E-02	1.76E+00	1.02E+00	1.11E-06	1.85E-02	7.60E-09
VVMARC-409	2.39E-02	1.88E-02	1.76E+00	1.02E+00	1.11E-06	1.88E-02	7.42E-09
VCIPRC-408	4.85E-04	3.69E-04	1.76E+00	1.00E+00	1.09E-06	3.69E-04	1.89E-09
VCIPRC-406	4.85E-04	3.69E-04	1.76E+00	1.00E+00	1.09E-06	3.69E-04	1.89E-09
VVMARC-408	2.43E-02	1.87E-02	1.76E+00	1.02E+00	1.11E-06	1.87E-02	6.63E-09
VVMARC-406	2.40E-02	1.87E-02	1.76E+00	1.02E+00	1.11E-06	1.87E-02	7.24E-09
JVMXSI-655/656	2.02E-04	1.31E-04	1.63E+00	1.00E+00	1.01E-06	1.31E-04	4.68E-09
JVMXCC-111/211	2.00E-04	1.29E-04	1.63E+00	1.00E+00	1.01E-06	1.29E-04	4.40E-09
AVMDEF-106	9.32E-04	5.66E-04	1.61E+00	1.00E+00	8.58E-07	5.66E-04	1.48E-09
LOFW	4.07E-01	3.13E-01	1.60E+00	1.65E+00	1.21E-06	3.13E-01	3.57E-07
AHFF-RSEFW	1.04E-03	5.75E-04	1.60E+00	1.00E+00	7.17E-07	5.75E-04	5.76E-10
GVCXSI-100/101	3.68E-05	2.22E-05	1.58E+00	1.00E+00	6.97E-07	2.22E-05	6.64E-09
FSEX-EFAS	2.80E-03	1.26E-03	1.46E+00	1.00E+00	6.85E-07	1.26E-03	2.03E-09
CVCXCC1303-2303	3.53E-05	1.53E-05	1.43E+00	1.00E+00	5.50E-07	1.53E-05	6.63E-09
CVMXSW121-221	2.00E-04	8.56E-05	1.43E+00	1.00E+00	5.50E-07	8.56E-05	6.40E-09
CPBXDCCWP1B-2B	1.42E-04	5.57E-05	1.43E+00	1.00E+00	5.49E-07	5.57E-05	-1.72E-17
AVCAEF-202	2.02E-04	8.12E-05	1.41E+00	1.00E+00	5.93E-07	8.12E-05	1.90E-09
MPMJSFP-101	2.09E-03	7.53E-04	1.38E+00	1.00E+00	5.26E-07	7.53E-04	-6.49E-18
MPMKSFP-101	5.80E-04	2.19E-04	1.37E+00	1.00E+00	5.07E-07	2.19E-04	3.32E-09
RCVRSBAC	5.05E-02	1.62E-02	1.31E+00	1.02E+00	4.00E-07	1.62E-02	1.31E-08
RCVRPWR10	4.90E-02	1.55E-02	1.30E+00	1.02E+00	3.93E-07	1.55E-02	9.38E-09
ESXRDGB	4.74E-03	2.21E-03	1.29E+00	1.00E+00	3.51E-07	2.21E-03	4.04E-09

Table 15.2-6 (Cont'd)

COMPONENT IMPORTANCES FOR SYSTEM 80+ PRA FOR INTERNAL EVENTS
(* - Sorted by this measure)

Event Name	Prob	Fuss-Vese	.Risk Ach.	Risk Red.	Birnbaum	Critical	Unc. Imp.
ESXRDGA	4.74E-03	2.18E-03	1.28E+00	1.00E+00	3.47E-07	2.18E-03	3.53E-09
FHFFEFWS	4.65E-03	1.26E-03	1.28E+00	1.00E+00	4.10E-07	1.26E-03	-1.82E-18
EDDJDGB	1.36E-02	4.91E-03	1.28E+00	1.01E+00	3.43E-07	4.91E-03	3.04E-09
ECBVS28801	9.69E-04	2.47E-04	1.27E+00	1.00E+00	3.31E-07	2.47E-04	-7.70E-18
ECBPS28801	9.34E-04	2.65E-04	1.27E+00	1.00E+00	3.31E-07	2.65E-04	-1.34E-17
EDDJDGA	1.36E-02	4.83E-03	1.27E+00	1.01E+00	3.36E-07	4.83E-03	2.20E-09
ECBPS18801	9.34E-04	2.54E-04	1.27E+00	1.00E+00	3.28E-07	2.54E-04	-1.26E-17
ECBVS18801	9.69E-04	2.92E-04	1.27E+00	1.00E+00	3.28E-07	2.92E-04	-3.86E-18
JVMASI-651	3.97E-03	1.06E-03	1.27E+00	1.00E+00	4.11E-07	1.06E-03	9.67E-10
AVMAEF-102	3.97E-03	1.06E-03	1.27E+00	1.00E+00	3.93E-07	1.06E-03	-3.03E-18
JVMASI-653	3.97E-03	1.04E-03	1.26E+00	1.00E+00	4.03E-07	1.04E-03	1.60E-09
MSXRSFWS	3.17E-04	7.76E-05	1.25E+00	1.00E+00	3.52E-07	7.76E-05	3.50E-09
EDDVDGB	5.58E-03	1.15E-03	1.22E+00	1.00E+00	2.74E-07	1.15E-03	2.96E-09
EDDVDGA	5.88E-03	1.20E-03	1.22E+00	1.00E+00	2.72E-07	1.20E-03	3.46E-09
EBDBDGSBS201	2.99E-04	6.06E-05	1.20E+00	1.00E+00	2.41E-07	6.06E-05	6.71E-09
EBDAPNSSBS2011	3.17E-04	6.04E-05	1.20E+00	1.00E+00	2.41E-07	6.04E-05	3.63E-09
UHFDRFIRWSTSGTR	3.47E-03	5.98E-04	1.20E+00	1.00E+00	2.80E-07	5.98E-04	6.28E-09
EBDBDGSAS201	2.99E-04	5.94E-05	1.20E+00	1.00E+00	2.39E-07	5.94E-05	6.69E-09
EBDAPNSSAS2011	3.17E-04	5.70E-05	1.20E+00	1.00E+00	2.38E-07	5.70E-05	3.73E-09
FSMOMSIS	1.64E-04	3.12E-05	1.20E+00	1.00E+00	2.90E-07	3.12E-05	5.94E-09
MVCXDCCKV	1.46E-04	2.90E-05	1.20E+00	1.00E+00	2.90E-07	2.90E-05	5.94E-09
MVCASF-004	2.02E-04	3.91E-05	1.20E+00	1.00E+00	2.90E-07	3.91E-05	5.96E-09
AVSDEF-104	1.12E-02	2.17E-03	1.19E+00	1.00E+00	2.87E-07	2.17E-03	3.45E-09
CPWKSSWP1A	7.06E-04	1.16E-04	1.18E+00	1.00E+00	2.44E-07	1.16E-04	-4.73E-17
AVMAEF-103	3.97E-03	5.98E-04	1.15E+00	1.00E+00	2.27E-07	5.98E-04	-4.28E-19
APAKEFWP-104	3.55E-03	5.75E-04	1.14E+00	1.00E+00	2.15E-07	5.75E-04	1.56E-09
APAKEFWP-102	3.55E-03	5.76E-04	1.14E+00	1.00E+00	2.14E-07	5.76E-04	1.29E-09
APAJEFPW-104	3.32E-03	6.95E-04	1.14E+00	1.00E+00	2.08E-07	6.95E-04	3.84E-09
APAJEFWP-102	3.32E-03	6.72E-04	1.14E+00	1.00E+00	2.07E-07	6.72E-04	3.81E-09
APAVEFWP-102	1.94E-03	2.22E-04	1.11E+00	1.00E+00	1.71E-07	2.22E-04	-8.39E-18
APAVEFWP-104	2.04E-03	2.37E-04	1.11E+00	1.00E+00	1.71E-07	2.37E-04	-7.41E-18
GVMXSI-300/301	2.00E-04	2.08E-05	1.11E+00	1.00E+00	1.28E-07	2.08E-05	7.68E-09
AVCAEF-200	2.02E-04	2.01E-05	1.10E+00	1.00E+00	1.60E-07	2.01E-05	6.36E-09
APTJEFWP-101	1.49E-02	1.76E-03	1.10E+00	1.00E+00	1.61E-07	1.76E-03	2.63E-09
UFLTBAFILTER	2.35E-04	2.21E-05	1.09E+00	1.00E+00	1.38E-07	2.21E-05	5.90E-09
UVCACH-668	2.02E-04	1.82E-05	1.09E+00	1.00E+00	1.38E-07	1.82E-05	6.51E-09
UVNACH-649	3.81E-04	3.03E-05	1.09E+00	1.00E+00	1.38E-07	3.03E-05	3.98E-09
UVNACH-126	3.81E-04	3.03E-05	1.09E+00	1.00E+00	1.38E-07	3.03E-05	3.98E-09
APTJEFWP-103	1.49E-02	1.62E-03	1.09E+00	1.00E+00	1.47E-07	1.62E-03	2.98E-09
AVSDEF-105	1.12E-02	9.94E-04	1.08E+00	1.00E+00	1.34E-07	9.94E-04	7.87E-10
CVMXCC107-207	1.99E-04	1.65E-05	1.08E+00	1.00E+00	1.01E-07	1.65E-05	7.70E-09
HVCASI-543	2.02E-04	1.58E-05	1.08E+00	1.00E+00	1.37E-07	1.58E-05	6.50E-09
JVCASI-178	2.02E-04	1.58E-05	1.08E+00	1.00E+00	1.37E-07	1.58E-05	6.50E-09
HVCASI-247	2.02E-04	1.58E-05	1.08E+00	1.00E+00	1.37E-07	1.58E-05	6.50E-09
HVCASI-143	2.02E-04	1.58E-05	1.08E+00	1.00E+00	1.37E-07	1.58E-05	6.50E-09
JHFMBYPASS1	1.73E-04	1.66E-05	1.08E+00	1.00E+00	1.37E-07	1.66E-05	5.82E-09
APTKEFWP-101	7.23E-03	8.36E-04	1.08E+00	1.00E+00	1.27E-07	8.36E-04	3.69E-09
MVMASF-005	3.97E-03	3.10E-04	1.08E+00	1.00E+00	1.09E-07	3.10E-04	-2.43E-18
MVMASF-002	3.97E-03	3.10E-04	1.08E+00	1.00E+00	1.09E-07	3.10E-04	-2.43E-18
CFLTSSW1A	2.35E-04	2.14E-05	1.08E+00	1.00E+00	1.13E-07	2.14E-05	6.29E-09
APTKEFWP-103	7.23E-03	7.99E-04	1.07E+00	1.00E+00	1.14E-07	7.99E-04	1.18E-09
LHVAC	3.95E-02	2.75E-03	1.07E+00	1.00E+00	1.06E-07	2.75E-03	8.75E-09
APTVEFWP-101	4.94E-03	2.72E-04	1.06E+00	1.00E+00	9.14E-08	2.72E-04	-3.73E-18
APTVEFWP-103	4.65E-03	2.36E-04	1.05E+00	1.00E+00	8.04E-08	2.36E-04	-8.80E-18
FSERAPS	2.56E-02	1.26E-03	1.05E+00	1.00E+00	7.32E-08	1.26E-03	1.37E-09

Table 15.2-6 (Cont'd)

COMPONENT IMPORTANCES FOR SYSTEM 80+ PRA FOR INTERNAL EVENTS
(* - Sorted by this measure)

Event Name	Prob	Fuss-Vese	.Risk Ach.	Risk Red.	Birnbaum	Critical	Unc. Imp.
UHFFBORONRCS	3.35E-02	1.78E-03	1.05E+00	1.00E+00	6.92E-08	1.78E-03	4.41E-09
EDDXRDGA-B	4.28E-03	1.96E-04	1.05E+00	1.00E+00	5.83E-08	1.96E-04	-9.98E-18
RCVRMOV	1.82E-01	9.88E-03	1.04E+00	1.01E+00	7.27E-08	9.88E-03	9.84E-09
TOTH	5.86E-01	4.70E-02	1.04E+00	1.06E+00	1.16E-07	4.70E-02	5.22E-08
HVNDSI-130	8.11E-04	2.84E-05	1.03E+00	1.00E+00	3.90E-08	2.84E-05	6.41E-09
HVNDSI-435	8.11E-04	2.84E-05	1.03E+00	1.00E+00	3.90E-08	2.84E-05	6.41E-09
HVNDSI-254	8.11E-04	2.84E-05	1.03E+00	1.00E+00	3.90E-08	2.84E-05	6.41E-09
HVMDSI-308	9.32E-04	3.11E-05	1.03E+00	1.00E+00	3.90E-08	3.11E-05	5.50E-09
HPSKSIP3	1.18E-03	3.61E-05	1.03E+00	1.00E+00	3.90E-08	3.61E-05	5.46E-09
HPSVSIP3	1.95E-03	6.65E-05	1.03E+00	1.00E+00	3.90E-08	6.65E-05	2.83E-09
HPSJSIP3	1.05E-03	3.96E-05	1.03E+00	1.00E+00	3.89E-08	3.96E-05	4.57E-09
AVDAEF-108	2.99E-03	9.11E-05	1.03E+00	1.00E+00	5.24E-08	9.11E-05	-3.59E-17
AVDAEF-112	2.99E-03	9.11E-05	1.03E+00	1.00E+00	5.24E-08	9.11E-05	-3.59E-17
JVMACC-111	3.97E-03	1.07E-04	1.03E+00	1.00E+00	4.27E-08	1.07E-04	5.06E-09
JVMASI-601	3.97E-03	1.07E-04	1.03E+00	1.00E+00	4.27E-08	1.07E-04	5.06E-09
JVMASI-655	3.97E-03	1.07E-04	1.03E+00	1.00E+00	4.27E-08	1.07E-04	5.06E-09
JVMASI-310	3.97E-03	1.07E-04	1.03E+00	1.00E+00	4.27E-08	1.07E-04	5.06E-09
CPWJSSWP1B	2.24E-03	6.06E-05	1.03E+00	1.00E+00	3.31E-08	6.06E-05	-1.99E-17
AVDAEF-109	2.99E-03	7.49E-05	1.02E+00	1.00E+00	4.14E-08	7.49E-05	-1.31E-17
AVDAEF-113	2.99E-03	7.49E-05	1.02E+00	1.00E+00	4.14E-08	7.49E-05	-1.31E-17
AVNDEF-341	8.11E-04	1.43E-05	1.02E+00	1.00E+00	2.63E-08	1.43E-05	6.52E-09
AVNDEF-340	8.11E-04	1.43E-05	1.02E+00	1.00E+00	2.63E-08	1.43E-05	6.52E-09
AHFLEF-340	8.67E-04	1.56E-05	1.02E+00	1.00E+00	2.63E-08	1.56E-05	6.38E-09
AVMDEF-107	9.32E-04	1.87E-05	1.02E+00	1.00E+00	2.63E-08	1.87E-05	6.02E-09
AHFLEF-341	8.83E-04	1.71E-05	1.02E+00	1.00E+00	2.63E-08	1.71E-05	6.40E-09
CHFLSW-1305	8.84E-04	1.58E-05	1.02E+00	1.00E+00	2.34E-08	1.58E-05	6.07E-09
CPWVSSWP1B	1.74E-03	3.30E-05	1.02E+00	1.00E+00	2.34E-08	3.30E-05	6.38E-09
CPWKSSWP1B	7.06E-04	2.78E-05	1.02E+00	1.00E+00	2.34E-08	2.78E-05	6.43E-09
L416	1.53E-03	2.38E-05	1.02E+00	1.00E+00	2.66E-08	2.38E-05	5.99E-09
LCCW	2.27E-03	3.85E-05	1.02E+00	1.00E+00	2.66E-08	3.85E-05	4.92E-09
LVRBMSSVS	5.47E-02	6.81E-04	1.01E+00	1.00E+00	1.69E-08	6.81E-04	2.60E-09
JVMASI-654	3.97E-03	3.75E-05	1.01E+00	1.00E+00	1.54E-08	3.75E-05	6.77E-09
EDDKDGB	5.78E-02	5.81E-04	1.01E+00	1.00E+00	1.01E-08	5.81E-04	1.91E-09
HVMASI-636	3.97E-03	2.89E-05	1.01E+00	1.00E+00	8.65E-09	2.89E-05	7.19E-09
RCVRPWR12	3.10E-02	1.93E-04	1.01E+00	1.00E+00	7.85E-09	1.93E-04	9.39E-09
DHFFRECLOSEADV	3.76E-03	1.81E-05	1.00E+00	1.00E+00	7.00E-09	1.81E-05	6.99E-09
DVMASG-184	3.97E-03	2.05E-05	1.00E+00	1.00E+00	6.78E-09	2.05E-05	7.55E-09
DVMASG-185	3.97E-03	2.05E-05	1.00E+00	1.00E+00	6.78E-09	2.05E-05	7.55E-09
DVMASG-178	3.97E-03	2.05E-05	1.00E+00	1.00E+00	6.78E-09	2.05E-05	7.55E-09
DVMASG-179	3.97E-03	2.05E-05	1.00E+00	1.00E+00	6.78E-09	2.05E-05	7.55E-09
JVMASI-652	3.97E-03	1.88E-05	1.00E+00	1.00E+00	7.68E-09	1.88E-05	7.32E-09
EDDKDGA	5.78E-02	3.39E-04	1.00E+00	1.00E+00	5.24E-09	3.39E-04	1.96E-09
SE-CSGTR	9.97E-02	2.65E-04	1.00E+00	1.00E+00	3.43E-09	2.65E-04	2.45E-09
GPXKECSBS	4.80E-02	4.30E-05	1.00E+00	1.00E+00	1.08E-09	4.30E-05	9.39E-09
RCVRPWR04	2.40E-01	2.68E-04	1.00E+00	1.00E+00	1.44E-09	2.68E-04	9.39E-09
ZQE-CLBRK	2.67E-01	2.87E-04	1.00E+00	1.00E+00	1.30E-09	2.87E-04	9.39E-09
RCVRPWR05	1.80E-01	7.33E-05	1.00E+00	1.00E+00	4.94E-10	7.33E-05	9.39E-09

15.2.2 Analysis of Containment Performance and Source Terms - Level II PRA

The objective of the containment response analysis is to ascertain the likelihood, magnitude, and timing of radiological releases to the environment following a severe accdient. In order to determine the consequences of an accident, each of the accident sequences identified as leading to core damage in the Level I PRA is further analyzed to determine if ti will lead loss of containment integrity, and if so, the nature of its associated release. The mode and timing of the containment failure and the nature of the releases are affected by the physical phenomena of the accident. The combinations of these physical phenomena define specific event sequence with unique consequences. These combinations of consequences are the potential Plant Damage States (PDSs) resulting from the accident. The parameters used to define the PDSs are based on factors that have the greatest effect on the public and include the following:

- Source term magnitude and isotopic content,
- Energy of the release,
- Duration of the release, and
- Warning time for evacuation.

Based on the qualitative evaluation of the containment response phenomena for System 80+ design, a set of eight PDS parameters and their associated values were defined. Several (6480) possible combinations of parameter values for the eight PDS parameters were noted, all of which were not physically possible. Therefore, a set of rules were developed to delete the combinations that were physically impossible. After applying the deletion rules, 245 logical PDSs survived the screening. In order to ascertain the information for mapping the core damage sequences into the appropriate PDS, six containment safeguard states were developed. These containment safeguard states were linked with the dominant core damage sequences to form plant accident states which contained enough information to be mapped directly into the PDSs. Twenty six of the 89 accident sequences for internal events and 3 of the 9 accident sequences due to tornado strikes were

linked with the containment safeguard states to form plant accident states. Approximately 66% of the plant accident states had frequencies below a certain screening value and were therefore eliminated from further evaluation. The elimianted plant accident states represent 1% of the overall core damage frequency.

To provide some initial insight into the potential severe accident progression, the relative split of the PDSs between parameter values for major parameters was evaluated and summarized below (the percentages presented are based on the core damage frequency contribution and not sequence count):

RCS PRESSURE AT CORE DAMAGE

Approximately 18% of the PDSs are low pressure sequences, 32% are medium pressure sequences, and 50% are high pressure sequences. Of the high pressure sequences, 27% have a cycling relief valve release rate and 23% have a small LOCA release rate.

CORE MELT TIMING

Approximately 78% of the sequences resulted in core damage within 8 hours (early). Sequences resulting in core damage between 8 and 24 hours account for 22%. Only one PDS with two sequences had core damage after 24 hours. This PDS is a negligible contributor of core damage.

CAVITY CONDITION

The cavity was not flooded in only 7% of the sequences. For the remaining 93% of the sequences, the cavity was flooded.

CONTAINMENT SPRAY STATUS

For 82% of the sequence, the containment spray system was available. The containment spray system was not available for the remaining 18% of the sequences.

RELEASE POINT

Approximately 38% of the sequences resulted in initial releases directly to the containment. Initial releases were to the IRWST for 47% of the sequences. Approximately 15% of the sequences resulted in initial releases through the steam generators. The containment bypass sequence accounted for less than 1% of the core damage frequency.

15.2.2.1 Containment Failure Frequency

The results from the containment response analysis show that the System 80+ containment is robust and quite capable of accommodating severe accident challenges. The conditional probability for intact containment releases, RC1, is 0.902. Likewise, the conditional probabilities for the various types of containment failures are: (1) 0.076 for late containment failure releases, RC2, (2) 0.011 for early containment failure releases, RC3, (3) 0.011 for releases due to containment isolation failure, RC4, and (4) 0.0 for containment bypass failure releases, RC5. The Conditional Containment Failure Probability (CCFP) is shown to be low. The above results as shown in Table 15.2-7 and Figure 15.2-2. Combining the CCFP with the core damage frequency calculated from the Level I portion of the PRA (2.0E-06 per year) results in a very low frequency (2.0E-07 per year, seismic events excluded) of containment failure from severe accidents.

A further breakdown of the contributors to the conditional containment failure probability due to the timing of core damage is shown in Tables 15.2-8 and 15.2-9. The relative contributions to CCFP for early (i.e., less than 8 hours after the initiating event occurs) core damage sequences are provided in Table 15.2-8. Similarly, the relative contributions to CCFP for mid (8 - 24 hours after the initiating event occurs) core damage sequences are provided in Table 15.2-9.

The insights and leading contributors to containment failure are summarized in Section 15.2.2.2.

15.2.2.2 Dominant Contributors to Containment Failure

Table 15.2-7 shows the fractional contributions by containment failure modes to the core damage frequency for internal and external events, excluding seismic events. The containment remained intact for about 90.2% of the core damage sequences. Of the 9.8% of core damage sequences in which the containment does fail, the late containment failure mode is the most dominant contributor followed by early containment failures, containment isolation failures and then containment bypass failures.

The breakdown of contributors to containment failure is also shown as a pie chart in Figure 15.2-2. The majority of containment failures (80.8%) involve late containment failures. An additional 11.2% of the containment failures is due to the early containment failure mode followed by containment isolation failures and containment bypass failures with 7.9% and 0.1%, respectively.

LATE CONTAINMENT FAILURES

The leading contributor to containment failure following a severe accident is "Late Containment Failure". This failure mode represents approximately 81% of the overall containment failure frequency. The frequency for this containment failure mode is 1.5E-7 per year. The major contributor to this failure mode includes base mat melt-through. All dry cavity cases result in containment failure and a very small percentage of the wet cavity cases are assume to result in containment melt-through.

EARLY CONTAINMENT FAILURES

Early containment failures are the second largest contributor to the frequency of containment failure. The frequency of this failure mode is 2.2E-08 per year and it accounts for approximately 11% of the overall containment failure frequency. The major contributors to this failure mode includes rapid steam generation and steam explosions, missile or rocket impingement, hydrogen burns, and direct containment heating.

CONTAINMENT ISOLATION FAILURES

Containment isolation failures are the third largest contributor to the frequency of containment failure. The frequency of this failure mode is 2.2E-08 per year and it accounts for approximately 8% of the overall containment failure frequency.

CONTAINMENT BYPASS

The containment bypass sequences are virtual non-contributors to the overall frequency of containment failure. The frequency for this failure mode is 2.0E-10 per year, the majority of which is cause by the interfacing system LOCA sequence.

15.2.2.3 Fission Product Release Characteristics

The insights regarding the relative significance of the various fission product release classes are provided below.

RELEASE CLASSES

A total of 57 release classes with non-zero frequencies were identified. This includes 36 early core melt frequency release classes, 20 mid core melt sequence release classes and 1 late core melt sequence release class. Of these release classes, only those that had frequencies above a certain screening value (i.e., 1.0E-09 per year) were considered for further evaluation. A noted exception to the screening is the containment bypass release class. This release class is caused by an interfacing system LOCA. Although the frequency of this release class is less than the screening value, it is retained for further evaluation. The release classes that were eliminated from further evaluation represent less than 1% of the total core damage frequency.

RELEASE CHARACTERISTICS

Each of the end-points of the containment event tree (CET) uniquely specifies the status of the containment following a severe accident, whether it is intact or

breached and if breached the mode of containment failure. The status of the various phenomena which have the potential to affect the source term characteristics is also uniquely specified for each end-point of the CET. As a result, each end-point of the CET represents a distinct release class which can be fully characterized by the following parameters:

- Frequency of occurrence,
- Isotopic content and magnitude of the release,
- Energy of the release,
- Time of the release,
- Duration of the release, and
- Location of the release.

Values for the above parameters are summarized in Tables 15.2-10, except for the isotopic content and magnitude of the release which is summarized in Table 15.2-11. As shown in Table 15.2-11, release class RC4.18L has the highest release fractions for the radionuclide groups. This release class represents a SGTR and containment isolation failure followed by a dry cavity and the containment sprays being unavailable. Other release classes with high release fractions include RC4.12E (similar to RC4.18L, except that the containment sprays are available) and RC4.4E (similar to RC4.18L, except that the containment sprays are available and the cavity is flooded).

15.2.2.3 Insights from Level II Sensitivity Analyses

The results from the containment response analysis show that the System 80+ containment is robust and quite capable of accommodating severe accident challenges. To assess the impact of certain assumptions that were made in performing the containment response analysis, the Level II portion of the PRA, several sensitivity analyses were performed. These analyses determined what

effect certain assumptions may have on the containment failure modes and associated conditional failure probabilities. The results from the sensitivity analyses are summarized in Table 15.2-12.

The major insights from the Level II sensitivity analyses are presented below:

- Hydrogen ignitors are provided to prevent the build-up of hydrogen inside the containment following a severe accident. However, the conditional probabilities for the various containment failure modes are insensitive to the availability of the hydrogen ignitors following a severe accident.

- The System 80+ containment characteristics do not favor deflagration to detonation transition and the release classes are not sensitive to deflagration to detonation transition.

- Late containment failure releases are somewhat sensitive to low heat transfer rate from the corium to the cavity water. This 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.

- Late containment failure releases are sensitive to the reliability of the emergency containment heat removal system and the recovery of containment heat removal following a severe accident.

- The conditional probabilities for System 80+ release classes are not sensitive to temperature induced creep failure of the RCS piping and the depressurization of the RCS, using the Safety Depressurization System.

- The frequency of 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.

Table 15.2-7

Overall Containment Failure Modes

CONTAINMENT FAILURE MODE	CCFP	CTMT FAILURE FREQUENCY (Per Year)
Late containment failures	0.076	1.5E-07
Early containment failures	0.011	2.2E-08
Containment isolation failures	0.011	2.2E-08
Containment bypass failures	< 0.001	2.0E-10
TOTAL	0.098	2.0E-07

Table 15.2-8

Containment Failure Modes for Early Core Damage Sequences*

CONTAINMENT FAILURE MODE	DOMINANT CONTRIBUTORS	CCFP (%)
Late containment failures	Base mat meltthrough Steam overpressurization	79.5
Early containment failures	Rapid steam generation Alpha mode Hydrogen burns Direct containment heating	11.1
Containment isolation failures		9.1
Containment bypass failures	Interfacing system LOCA	0.3

* The CCFP due to early core damage sequences is 0.0103.

Table 15.2-9

Containment Failure Modes for Mid Core Damage Sequences*

CONTAINMENT FAILURE MODE	DOMINANT CONTRIBUTORS	CCFP (%)
Late containment failures	Base mat meltthrough Steam overpressurization	85.8
Early containment failures	Rapid steam generation Alpha mode Hydrogen burns Direct containment heating	11.6
Containment isolation failures		2.5
Containment bypass failures	Interfacing system LOCA	0.1

* The CCFP due to mid core damage sequences is 0.0084.

TABLE 15.2-10 RELEASE PARAMETER DATA FOR SYSTEM 80+ RELEASE CLASSES							
RELEASE CLASS	CORE MELT TIME	CONTAINMENT STATUS	RELEASE CLASS FREQUENCY	RELEASE START TIME	RELEASE DURATION	RELEASE HEIGHT	RELEASE ENERGY (Watts)
RC1.1E	EARLY	CONTAINMENT INTACT	1.36E-06	5 HOURS	24.0 HOURS	16.6M	2.09E+04
RC1.1M	MID	CONTAINMENT INTACT	3.81E-07	17 HOURS	24.0 HOURS	16.6M	2.09E+04
RC2.1E	EARLY	LATE FAILURE	3.46E-09	11 HOURS	0.056 HOURS	16.6M	1.67E+09
RC2.2E	EARLY	LATE FAILURE	2.04E-09	11 HOURS	0.056 HOURS	16.6M	1.67E+09
RC2.4E	EARLY	BASEMAT MELTTHROUGH	3.64E-08	65 HOURS	24.0 HOURS	0 M	4.19E+04
RC2.5E	EARLY	BASEMAT MELTTHROUGH	2.84E-08	65 HOURS	24.0 HOURS	0 M	4.19E+04
RC2.6E	EARLY	BASEMAT MELTTHROUGH	3.45E-08	65 HOURS	24.0 HOURS	0 M	4.19E+04
RC2.7E	EARLY	BASEMAT MELTTHROUGH	1.62E-08	65 HOURS	24.0 HOURS	0 M	4.19E+04
RC2.2M	MID	LATE FAILURE	4.05E-09	65 HOURS	0.056 HOURS	16.6 M	1.67E+09
RC2.5M	MID	BASEMAT MELTTHROUGH	3.95E-09	65 HOURS	24.0 HOURS	0 M	4.19E+04
RC2.6M	MID	BASEMAT MELTTHROUGH	9.08E-09	65 HOURS	24.0 HOURS	0 M	4.19E+04
RC2.7M	MID	LATE FAILURE	1.22E-08	65 HOURS	0.056 HOURS	16.6 M	1.67E+09
RC3.1E	EARLY	EARLY FAILURE	6.58E-09	5 HOURS	0.056 HOURS	16.6 M	8.37E+07
RC3.2E	EARLY	EARLY FAILURE	3.08E-09	5 HOURS	0.056 HOURS	16.6 M	8.37E+07
RC3.4E	EARLY	EARLY FAILURE	6.73E-09	5 HOURS	0.056 HOURS	16.6 M	8.37E+07
RC3.6E	EARLY	EARLY FAILURE	3.12E-09	5 HOURS	0.056 HOURS	16.6 M	8.37E+07

TABLE 15.2-10
RELEASE PARAMETER DATA FOR SYSTEM 80+ RELEASE CLASSES

RELEASE CLASS	CORE MELT TIME	CONTAINMENT STATUS	RELEASE CLASS FREQUENCY	RELEASE START TIME	RELEASE DURATION	RELEASE HEIGHT	RELEASE ENERGY (Watts)
RC3.2M	MID	EARLY FAILURE	1.80E-09	17 HOURS	0.056 HOURS	16.6 M	8.37E+07
RC3.6M	MID	EARLY FAILURE	1.81E-09	17 HOURS	0.056 HOURS	16.6 M	8.37E+07
RC4.4E	EARLY	ISOLATION FAILURE (SGTR)	5.98E-09	4 HOURS	24.0 HOURS	19.7 M	4.19E+04
RC4.8E	EARLY	ISOLATION FAILURE (TRANSIENT)	1.12E-09	5 HOURS	24.0 HOURS	16.6 M	2.51E+05
RC4.12E	EARLY	ISOLATION FAILURE (SGTR)	6.54E-09	4 HOURS	24.0 HOURS	19.7 M	4.19E+04
RC4.18L	LATE	ISOLATION FAILURE (SGTR)	5.56E-09	25 HOURS	24.0 HOURS	19.7 M	4.19E+04
RC5.1E	EARLY	CONTAINMENT BYPASS	5.10E-10	2 HOUR	24.0 HOURS	0 M	2.09E+04

TABLE 15.2-11 RELEASE FRACTIONS BY RELEASE CLASS									
RELEASE CLASS	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
RC1.1M	5.00E-03	2.08E-05	9.72E-08	4.66E-08	1.12E-08	1.90E-09	8.87E-10	3.05E-09	1.12E-08
RC2.1E	1.00E+00	8.45E-05	1.00E-03	1.86E-05	3.87E-06	8.10E-07	2.46E-07	1.16E-06	4.38E-06
RC2.2E	1.00E+00	1.09E-04	1.00E-03	1.86E-05	3.54E-06	9.03E-07	2.18E-07	1.01E-06	3.92E-06
RC2.4E	5.44E-02	3.45E-04	1.00E-04	1.99E-07	6.15E-08	4.39E-09	6.36E-09	1.06E-08	5.45E-08
RC2.5E	5.44E-02	3.15E-04	1.00E-04	1.37E-07	4.38E-08	3.40E-09	4.26E-09	6.70E-09	3.63E-08
RC2.6E	5.44E-02	3.15E-04	1.00E-04	3.56E-07	1.27E-07	7.48E-09	1.41E-08	1.98E-08	1.08E-07
RC2.7E	5.44E-02	3.15E-04	1.00E-04	1.37E-07	4.38E-08	2.70E-09	4.26E-09	6.70E-09	3.63E-08
RC2.2M	1.00E+00	5.64E-05	1.00E-03	3.56E-06	8.20E-07	1.89E-07	5.62E-08	2.86E-07	8.79E-07
RC2.5M	1.00E+00	6.39E-03	1.00E-04	3.01E-04	1.19E-04	3.13E-06	1.32E-05	1.45E-05	9.33E-05
RC2.6M	4.46E-01	2.89E-03	1.00E-04	3.67E-05	1.72E-05	4.32E-07	1.76E-06	2.11E-06	1.38E-05
RC2.7M	1.00E+00	6.68E-03	1.00E-03	3.14E-04	1.24E-04	3.35E-06	1.38E-05	1.51E-05	9.72E-05
RC3.1E	1.00E+00	1.37E-02	1.15E-02	5.53E-03	1.10E-03	2.14E-04	6.10E-05	2.64E-04	1.25E-03
RC3.2E	1.00E+00	2.02E-02	1.76E-02	8.42E-03	1.49E-03	3.84E-04	8.16E-05	3.53E-04	1.66E-03
RC3.4E	1.00E+00	2.26E-02	1.50E-02	7.11E-03	1.84E-03	2.29E-04	1.45E-04	3.65E-04	1.83E-03
RC3.6E	1.00E+00	2.35E-02	1.64E-02	8.27E-03	2.05E-03	3.15E-04	1.68E-04	3.96E-04	2.01E-03
RC3.2M	1.00E+00	4.95E-02	3.96E-02	1.95E-02	3.45E-03	9.12E-04	2.14E-04	1.05E-03	3.81E-03
RC3.6M	1.00E+00	6.34E-02	4.97E-02	2.43E-02	5.73E-03	9.60E-04	4.49E-04	1.32E-03	5.63E-03
RC4.4E	1.00E+00	2.42E-01	2.20E-01	1.03E-01	2.30E-02	5.96E-03	1.57E-03	8.32E-03	2.53E-02

TABLE 15.2-11
RELEASE FRACTIONS BY RELEASE CLASS

RELEASE CLASS	FISSION PRODUCT RELEASE FRACTIONS								
	NOBLE GASES	IODINE	CESIUM	TELLURIUM	BARIUM	STRONTIUM	RUTHENIUM	LANTHANUM	CERIUM
RC4.8E	1.00E+00	7.80E-03	6.92E-04	3.29E-04	8.65E-05	1.06E-05	6.78E-06	1.76E-05	8.55E-05
RC4.12E	1.00E+00	2.48E-01	2.21E-01	1.03E-01	2.31E-02	5.96E-03	1.58E-03	8.33E-03	4.75E-03
RC4.18L	1.00E+00	3.47E-01	3.23E-01	1.60E-01	4.26E-02	7.54E-03	3.46E-03	1.17E-02	4.23E-02
RC5.1E	1.00E+00	5.77E-02	5.64E-02	2.75E-02	8.82E-03	6.20E-04	9.86E-04	1.53E-03	8.10E-03

TABLE 15.2-12

SUMMARY OF CONTAINMENT RESPONSE SENSITIVITY ANALYSIS
RESULTS FOR SYSTEM 80+

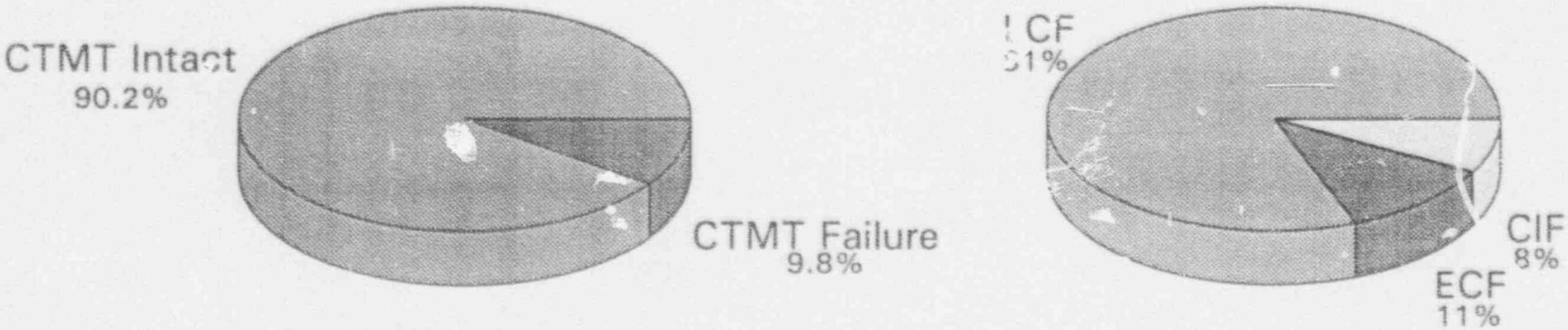
CASE	DESCRIPTION	MODELED AS	CONDITIONAL PROBABILITY OF RELEASE CLASS				
			RC1	RC2	RC3	RC4	RC5
BASE	AS DESCRIBED IN THE PRA	AS MODELED IN THE PRA	0.902	0.076	0.011	0.011	0.0
1	H_2 IGNITORS UNAVAILABLE	CHANGE PROBABILITY OF "IGFAIL" FROM CURRENT VALUE TO 1.0 OF ALL PDSs.	0.902	0.076	0.011	0.011	0.0
2	CONTAINMENT CHARACTERISTICS FAVOR DEFLAGRATION TO DETONATION TRANSITION	INCREASE THE PROBABILITY OF "DDTOK" FROM 0.0 OR 0.01 TO 0.05 FOR ALL PDSs.	0.902	0.076	0.011	0.011	0.0
3	LOW HEAT TRANSFER FROM CORIUM TO CAVITY WATER	INCREASE THE PROBABILITY OF "LOWHTXFER" FROM 0.01 TO 0.02 FOR ALL PDSs ASSOCIATED WITH "WET" CAVITY.	0.890	0.088	0.011	0.011	0.0
4	INSUFFICIENT HEAT TRANSFER FROM CORIUM TO CAVITY WATER	CHANGE THE PROBABILITY OF "WETCAVITY" FROM 1.0 TO 0.5 FOR ALL PDSs ASSOCIATED WITH "WET" CAVITY.	0.445	0.538	0.006	0.011	0.0
5	CONTAINMENT HEAT REMOVAL LESS LIKELY TO BE RECOVERED	INCREASE THE PROBABILITY OF "NCHRECOV" FROM 0.1 TO 0.2.	0.895	0.083	0.011	0.011	0.0
6A	INDUCED FAILURE OF RCS PIPING ALWAYS OCCUR	CHANGE THE PROBABILITY OF "HSINTACT" FROM 0.65 TO 1.0 FOR PDSs WITH RCS LEAK RATE = "CRV" AND RCS PRESSURE = "HIGH"	0.902	0.076	0.011	0.011	0.0
6B	INDUCED FAILURE OF RCS PIPING NEVER OCCURS	CHANGE THE PROBABILITY OF "HSINTACT" FROM 0.65 TO 0.0 FOR PDSs WITH RCS LEAK RATE = "CRV" AND RCS PRESSURE = "HIGH"	0.902	0.076	0.011	0.011	0.0

TABLE 15.2-12 (Cont'd)

SUMMARY OF CONTAINMENT RESPONSE SENSITIVITY ANALYSIS
RESULTS FOR SYSTEM 80+

CASE	DESCRIPTION	MODELED AS	CONDITIONAL PROBABILITY OF RELEASE CLASS				
			RC1	RC2	RC3	RC4	RC5
7A	THE RCS IS NOT DEPRESSURIZED BY THE SDS	CHANGE PROBABILITY OF "NOSDSDP" FROM 0.2 TO 1.0 FOR PDSs WITH RCS LEAK RATE = "CRV" AND RCS PRESSURE = "HIGH".	0.902	0.076	0.011	0.011	0.0
7B	THE RCS IS DEPRESSURIZED BY THE SDS	CHANGE PROBABILITY OF "NOSDSDP" FROM 0.2 TO 0.0 FOR PDSs WITH RCS LEAK RATE = "CRV" AND RCS PRESSURE = "HIGH".	0.902	0.076	0.011	0.011	0.0
8	CONTAINMENT IS LESS LIKELY TO BE ISOLATED	CHANGE PROBABILITY OF "ISOL" FROM 2.1E-03 TO 1.0E-02.	0.893	0.078	0.011	0.017	0.0

Figure 15.2-2
Contributions to Containment Failure Frequency



Core Melt Frequency *
(2.0 x 10^{-6} per year)

Containment Failure Frequency *
(2.0E-7 per year)

* For internal and external events, excluding seismic events.

15.2.3 Release Consequences Assessment - Level III PRA

The characteristics of the release classes determined by the containment response analysis are the primary input to the consequence analysis which calculates the risk measures, including:

- The effective Dose Equivalent (EDE) whole-body dose at 300 meters from the reactor site boundary,

- The EDE whole-body dose at one half mile from the reactor site boundary,

- Early fatalities, and

- Latent cancer fatalities.

The Complementary Cumulative Distribution Functions (CCDFs) for the above risk measures were generated. These distributions are presented in Figures 15.2-3 through 15.2-6. Figure 15.2-3 presents the total CCDF for the EDE whole body dose at 300 meters. This CCDF represents the total frequency of exceeding a given EDE whole body dose at a radius of 300 meters from the reactor. Figure 15.2-4 presents the total CCDF for the EDE whole body dose at 0.5 miles. This CCDF represents the total frequency of exceeding a given EDE whole body dose at a radius of one half mile from the reactor. Figure 15.2-5 presents the total CCDF for Early Fatalities. This CCDF represents the total frequency of exceeding a given number of early fatalities. Figure 15.2-6 presents the total CCDF for Latent Cancer Fatalities. This CCDF represents the total frequency of exceeding a given number of latent cancer fatalities.

The large offsite-release goal adopted by Combustion Engineering for the System 80+ Standard Design is:

> "In the event of a severe accident, the dose beyond a one-half mile radius from the reactor shall not exceed 25 rem. The mean frequency of occurrence for higher offsite doses shall be less than once per million

reactor-years, considering both internal and external events."

This goal is consistent with the ALWR large offsite release goal established by EPRI in the ALWR Utility Requirements Document[7].

As can be seen from Figure 15.2-4, the frequency with which a whole body dose of 25 rem is exceeded at 0.5 miles is approximately 1.9E-8 per year. Also, as can be seen from Figure 15.2-3, the frequency with which a whole body dose of 25 rem is exceeded at 300 meters is approximately 2.3E-8 per year. In both cases, the frequency of exceeding 25 rems is less than the goal of 1.0E-6 per year. Therefore, the System 80+ Standard Design meets the large offsite release goal with substantial margin.

In their policy statement on safety goals for the operation of nuclear power plants, the NRC provided two quantitative health effect objectives. They are:

> "The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed." and

> "The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes."

The commission did not define the sum of cancer fatality risks or the sum of prompt fatality risks. They did, however, provide a general performance guideline for use by the staff. This guideline is:

> "Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of a large release of

radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."

The magnitude of a large release has not been specifically defined. However, it is generally interpreted to be a release that is likely to result in one early fatality. Figure 15.2-5 presents the total CCDF for Early Fatalities. This CCDF represents the total frequency of exceeding a given number of early fatalities. As can be seen from this figure, the probability of exceeding one prompt fatality for System 80+ is approximately 2.1E-09 per year. This is well within the 1.0E-6 per year goal. In addition, as shown on Figure 15.2-6, the probability of exceeding one latent cancer death is approximately 2.0E-07. Thus, System 80+ meets the NRC's large release guideline and by implication, the quantitative health objectives.

15.2.3.1 Dominant Contributors to Risk

As mentioned in the previous section, four risk measures were calculated for the System 80+ design. The dominant contributors to each of the four risk measures are identified by release class and are described below.

DOSE AT 300 METERS

Four dominant release classes were identified for the effective dose equivalent whole-body dose of 20 Rem at a distance of 300 meters from the reactor site boundary. These release classes include RC4.12E, RC4.4E, RC4.18L, and RC3.4E. The probabilities of exceeding 20 Rem at 300 meters for these release classes are 6.5E-09, 6.0E-09, 5.6E-09, and 1.0E-09 respectively. A description for each of these release classes is provided below.

- RC4.12E covers the releases associated with a containment isolation failure with vaporization releases and re-vaporization releases for sequences in which core damage occurs within the first eight hours. In-vessel scrubbing is successful, in addition to scrubbing of the vaporization and re-vaporization releases.

The dominant PDSs for this release class are PDS184 and PDS181. PDS184 is characterized by a steam generator tube rupture with failure of safety injection and failure of aggressive secondary cooldown. For this release class, it is assumed that the ruptured steam generator is not successfully isolated. Core damage is assumed to occur within four hours after the initiating event with vessel failure occurring within the next hour. The containment spray system is available and the cavity is flooded. The releases are via the unisolated ruptured steam generator.

PDS181 is similar to PDS184, except that the cavity is not flooded. The releases for this class are assumed to start at the time of core damage and last for 24 hours. The release to environment is assumed to occur at an elevation of 19.7 meters above grade.

- RC4.4E covers the releases associated with a containment isolation failure with vaporization releases but no re-vaporization releases for sequences in which core damage occurs within 8 hours after the initiating event. Scrubbing of in-vessel fission products is successful as well as scrubbing of vaporization releases.

  The dominant PDS for this release class is PDS184 which is described above. The releases for this class were assumed to start at the time of core damage and last for 24 hours. The release to the environment was assumed to occur at an elevation of 19.7 meters above grade.

- RC4.18L covers the releases associated with a containment isolation failure with vaporization releases and re-vaporization releases for sequences in which the core damage occurs after 24 hours. Scrubbing of in-vessel fission products is successful, but scrubbing of vaporization and re-vaporization releases is not accomplished.

  The dominant PDS for this release class is PDS194. This PDS is characterized by a steam generator tube rupture with successful safety injection and successful operation of the emergency feedwater water system. However, the ruptured steam generator is not isolated and RCS

pressure control is also not established, and the leakage from the primary side to the secondary side remains high. The inventory of the IRWST is not replenished and is therefore depleted in approximately 25 hours. Core damage is assume to occur once the IRWST is depleted followed by vessel failure approximately one hour later. For this PDS, the cavity is dry and the containment spray system is unavailable due to depletion of the IRWST inventory. Releases occur via the unisolated ruptured steam generator.

The releases for this PDS were assumed to start at the time of core damage and continue for approximately 24 hours. The release to the environment was assumed to occur at an elevation of 19.7 meters above grade.

- RC3.4E covers the releases associated with an early containment failure with vaporization releases but not re-vaporization releases for accident equences in which core damage occurs within the first 8 hours after the initiating event. The scrubbing of in-vessel fission products is successful as well as the scrubbing of vaporization releases.

The dominant PDSs for this release class include PDS235, PDS20, PDS3, and PDS85. The releases for PDS85 and PDS20 are bounded by the releases for PDS3. Therefore, this release class is dominated by PDS235 and PDS3. PDS235 is characterized by a loss of main feedwater and failure of the emergency feedwater system. The "bleed" portion of "Feed and Bleed" also fails for this PDS. Core damage is assumed to occur within 4 hours after the initiating event followed by vessel failure within the next hour. The containment spray system is available and the cavity is flooded for this PDS.

PDS3 is characterized by a large LOCA with failure of safety injection. Core damage is assumed to occur within 4 hours after the initiating event, followed by vessel rupture an hour later. For PDS3, the containment spray system is also available and the cavity is also flooded.

The releases for this release class are assumed to start at the time of containment failure which occurs immediately after vessel failure and last

for approximately 200 seconds. The release to the environment is assumed to occur at an elevation of 16.6 meters above grade.

DOSE AT ONE-HALF MILE

Three dominant release classes were identified for the effective dose equivalent whole-body dose of 25 Rem at a distance of one-half mile from the reactor site boundary. These release classes include RC4.12E, RC4.4E, and RC4.18L. The probabilities of exceeding 25 Rem at one-half mile for these release classes are 6.0E-09, 5.7E-09, 5.6E-09 respectively. A description for each of the dominant release classes is provided above in the sub-section on "DOSE AT 300 METERS".

EARLY FATALITIES

Three dominant release classes were identified for large releases that resulted in early fatalities. These release classes include RC4.12E, RC4.18L, and RC4.4E. The probabilities of exceeding one early fatality due to a large release are 8.4E-10, 6.3E-10, 1.5E-10 respectively. A description for each of the dominant release classes is provided above in the sub-section on "DOSE AT 300 METERS".

LATENT FATALITIES

Five dominant release classes were identified for large releases that resulted in latent fatalities. These release classes include RC2.4E, RC2.6E, RC2.5E, RC2.7E, and RC2.7M. The probabilities of exceeding one latent fatality due to a large release are 3.6E-08, 3.4E-08, 2.8E-08, 1.6E-08, and 1.2E-08 respectively. A description for each of the dominant release classes is provided below.

- RC2.4E covers the releases associated with late containment failure with vaporization releases but no re-vaporization releases for accident sequences in which core damage occurs within 8 hours after the initiating event. The scrubbing of in-vessel fission products is successful as well as the scrubbing of vaporization releases.

  The dominant PDSs for this release class incluce PDS233, PDS83, PDS18,

PDS3, and PDS1. The releases from PDS83, PDS18, and PDS3 are bounded by the releases from PDS1. Therefore, this release class is characterized by PDS233 and PDS1. PDS233 is characterized by a loss of main feedwater and failure of the emergency feedwater system. Failure of the "bleed" portion of "Feed and Bleed" also occurs for this PDS. Core damage is assumed to occur within 4 hours after the initiating event followed by vessel rupture within the next hour. The containment spray system is available and the cavity is not flooded for this PDS.

PDS1 is characterized by a large LOCA with failure of safety injection. Core damage is assumed to occur within 4 hours after the initiating event followed by vessel failure within the next hour. The containment spray system is available and the cavity is not flooded for this PDS.

The releases for this release class are assumed to start at the time of containment failure (65 hours after the initiating event) and continue for 24 hours. The release to the environment is assumed to occur at grade level

RC2.6E covers the releases associated with a late containment failure with both vaporization releases and re-vaporization releases for accident sequences in which core damage occurs within 8 hours after the initiating event. Scrubbing of in-vessel fission products is successful, and scrubbing of vaporization and re-vaporization releases is also successful.

The dominant PDSs for this release class are PDS181, PDS199, and PDS233. For this release class, PDS181 and PDS199 are equivalent and PDS199 is therefore used to represent both PDSs. PDS199 is characterized by a small LOCA with failure of safety injection and failure of aggressive cooldown of the secondary side. Core damage is assumed to occur within 4 hours after the initiating event followed by vessel failure one hour later. For this PDS, containment spray is available but the cavity is not flooded.

PDS133 is characterized by a transient with failure of the emergency feedwater system and failure of the "bleed" portion of "Feed and Bleed"

operation. Core damage is also assumed to occur within 4 hours after the initiating event, followed by vessel failure one hour later. For this PDS, containment spray is available but the cavity is not flooded.

The releases for this release class are assumed to start at the time of containment failure (65 hours after the initiating event) and continue for 24 hours. The release to the environment is assumed to occur at grade level.

- RC2.5E covers the releases associated with late containment failure with vaporization releases but no re-vaporization releases for accident sequences in which core damage occurs within the first 8 hours after the initiating event. Scrubbing of in-vessel fission products is successful, but scrubbing of vaporization releases is not.

  The sole contributor to this release class is PDS241. PDS241 is characterized by loss of main feedwater with failure of emergency feedwater and failure of the "bleed" portion of "Feed and Bleed" operation. Core damage is assumed to occur within 4 hours after the initiating event, followed by vessel failure one hour later. For this PDS, the containment spray system is available and the cavity is not flooded.

  The releases for this release class are assumed to start at the time of containment failure (65 hours after the initiating event) and continue for 24 hours. The release to the environment is assumed to occur at grade level.

- RC2.7E covers the releases associated with late containment failure with vaporization releases and re-vaporization releases for accident sequences in which core damage occurs within the first 8 hours after the initiating event. Scrubbing of in-vessel fission products and re-vaporization releases is successful, but scrubbing of vaporization releases is not.

  The sole contributor to this release class is PDS241. PDS241 is

characterized by loss of main feedwater with failure of emergency feedwater and failure of the "bleed" portion of "Feed and Bleed" operation. Core damage is assumed to occur within 4 hours after the initiating event, followed by vessel failure one hour later. For this PDS, the containment spray system is available and the cavity is not flooded.

The releases for this release class are assumed to start at the time of containment failure (65 hours after the initiating event) and continue for 24 hours. The release to the environment is assumed to occur at grade level.

- RC2.7M covers the releases associated with a late containment failure with vaporization releases and re-vaporization releases for accident sequences in which core damage occurs within 8 to 24 hours after the initiating event. Scrubbing of in-vessel fission products and re-vaporization releases is successful, but scrubbing of vaporization releases is not.

  The dominant contributors for this release class are PDS145 and PDS242. The releases for PDS145 are considered to be bounded by the releases for PDS242. Therefore, PDS242 is use to characterize this release class.

  PDS242 is characterized by a station blackout with successful operation of the emergency feedwater system until the batteries are depleted 8 hours after the station blackout event occurs. Core damage is assumed to occur approximately 10 hours after the blackout event started, followed by vessel failure one hour later. For this PDS, the containment spray system is not available and the cavity is not flooded.

  The releases for this release class are assumed to start at the time of containment failure (65 hours after the initiating event) and continue for 24 hours. The release to the environment is assumed to occur at grade level.

15.2.3.2 Insights from Level III Sensitivity Analyses

The results of the consequence analysis were compared with the NRC's health objectives and risk goals. The comparison shows that the System 80+ design meets the NRC's large release guidelines and by implication the health objectives are also met. For the base case consequence analysis, releases due to containment isolation failure were the dominant contributors to equivalent whole-body dose at distance (i.e., 300 meter and one-half mile from the reactor site boundary) and to early fatalities. Latent fatalities were dominated by releases caused by late containment failures. To assess the impact of certain assumptions that were made in performing the consequence analysis which is the level III portion of the PRA, several sensitivity analyses were performed. The results of the sensitivity analyses are summarized in Table 15.2-13.

The major insights from the Level III sensitivity analyses are presented below:

- Latent fatalities and the equivalent whole-body dose of 25 Rem at various distances from the reactor site boundary are not significantly affected by the point of release of radioactive materials to the environment. However, early fatalities would tend to increase slightly if all releases occurred at grade level. There would be a noticeable decrease in early fatalities if all releases occurred at the top of the containment.

- The overall risk of the System 80+ design, as characterized by the risk measures described in this report, is relatively insensitive to containment bypass releases that are not scrubbed prior to their release into the environment.

- The reliability of the containment isolation function can have a significant impact on the overall risk of the System 80+ design.

- Latent fatality is the risk measure that would be most affected if the frequency of basemat melt-through occurred more frequently than currently anticipated.

- Because of 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.

- The risk of the System 80+ design is sensitive to the isotopic content that is used to characterize the various release classes.

TABLE 15.2-13

SUMMARY OF SENSITIVITY RESULTS OF RISK CONSEQUENCES FOR SYSTEM 80+

CASE No.	DESCRIPTION	PROBABILITY OF EXCEEDING			
		25 Rems @ 300 M	25 Rems @ ½ Mi	1 EARLY FATALITY	1 LATENT FATALITY
BASE	AS DESCRIBED IN SECTION 13	2.3E-08	1.8E-08	1.6E-09	1.9E-07
1A	RELEASES OCCUR AT THE TOP OF THE CONTAINMENT (173.2 ft OR 52.8 M)	2.0E-08	1.8E-08	7.0E-10	1.9E-07
1B	RELEASES OCCUR AT GRADE LEVEL	1.9E-08	1.9E-08	1.7E-09	1.9E-07
2	INCREASED IODINE AND CESIUM RELEASE FRACTIONS	3.6E-08	2.3E-08	4.3E-09	2.1E-07
3	CONTAINMENT BYPASS RELEASES UNSCRUBBED	2.2E-08	1.8E-08	1.8E-09	1.9E-07
4	CONTAINMENT ISOLATION SYSTEM IS LESS RELIABLE	1.9E-07	1.7E-07	1.6E-08	3.7E-07
5	BASEMAT MELT-THROUGH OCCURS MORE FREQUENTLY	2.3E-08	1.8E-08	1.6E-09	3.2E-07
6	ISLOCA OCCURS MORE FREQUENTLY	7.2E-08	6.2E-08	2.0E-09	2.4E-07

Figure 15.2-3

TOTAL CCDF @ 300 METERS FOR ALL RELEASE CLASSES

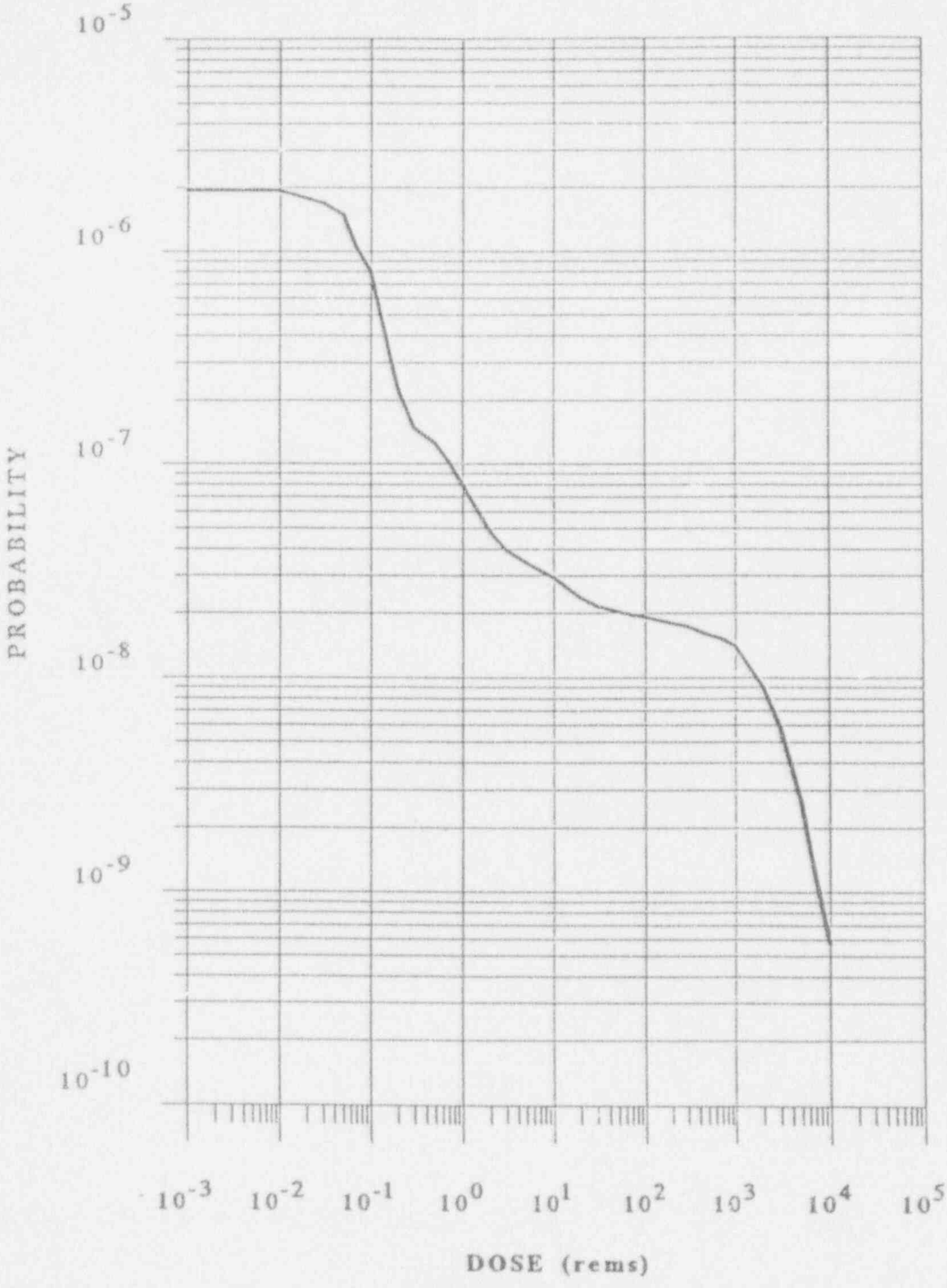


Figure 15.2-4

TOTAL CCDF @ 1/2 MILE FOR ALL RELEASE CLASSES

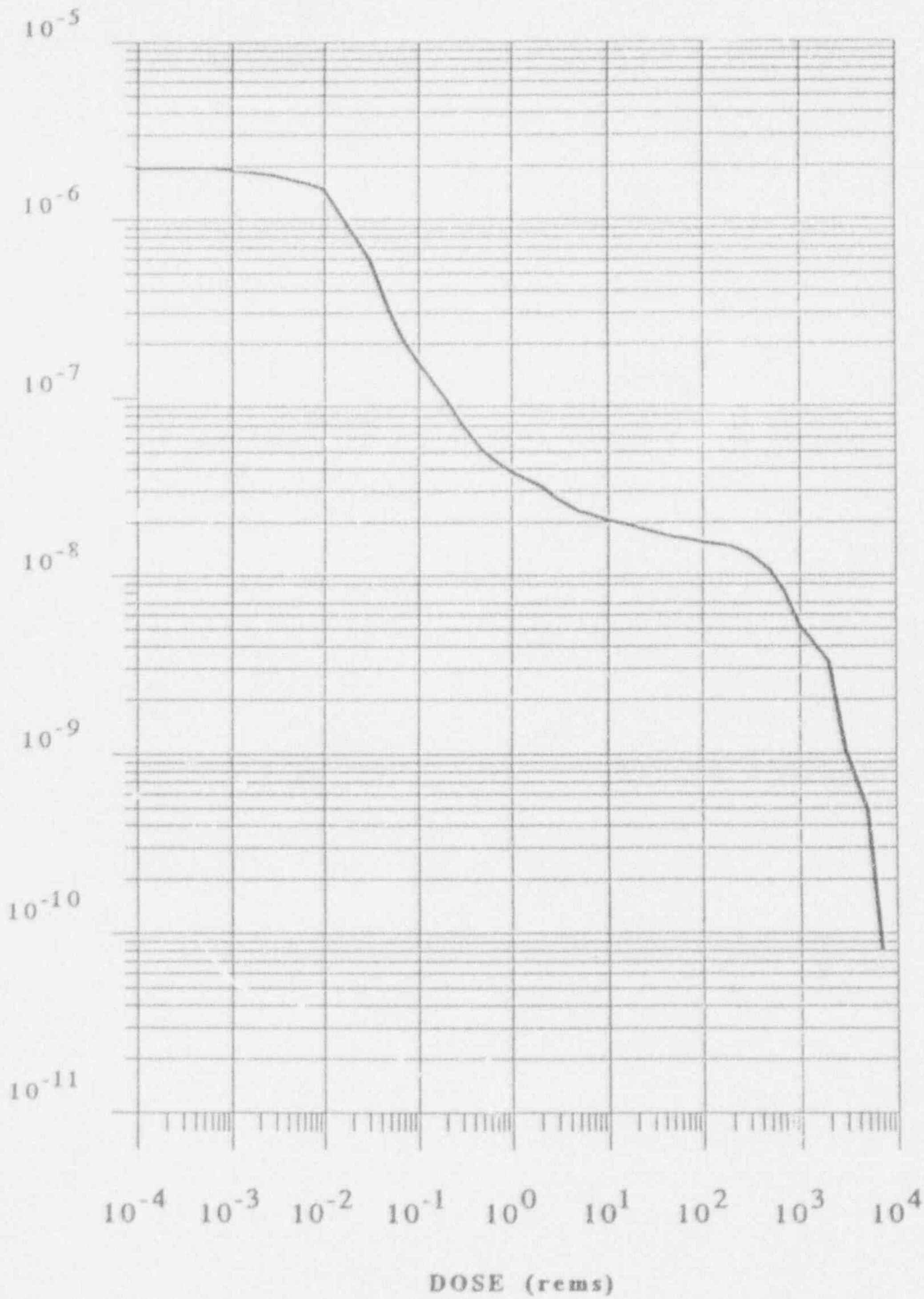


Figure 15.2-5

TOTAL CCDF FOR EARLY FATALITY FOR ALL RELEASE CLASSES

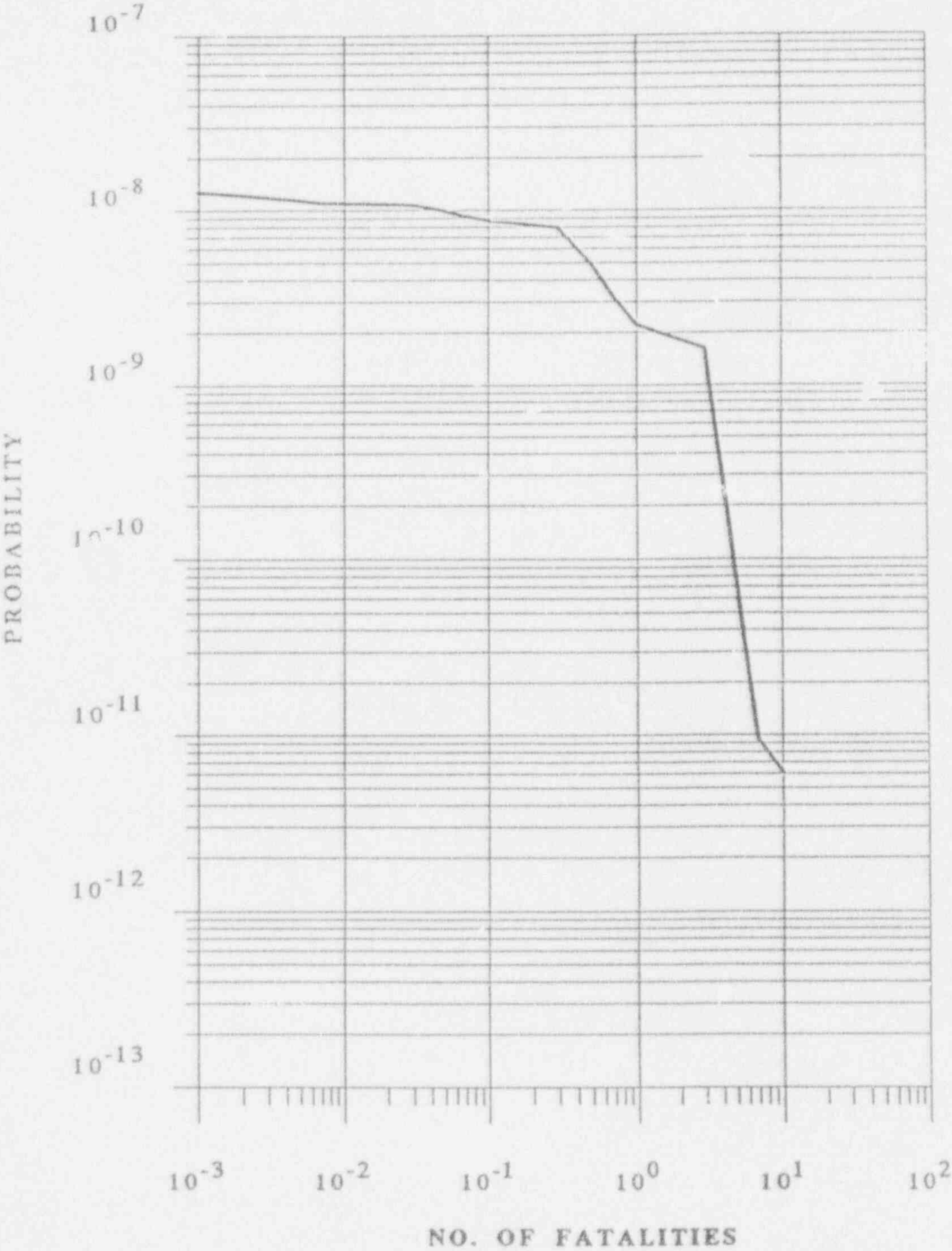
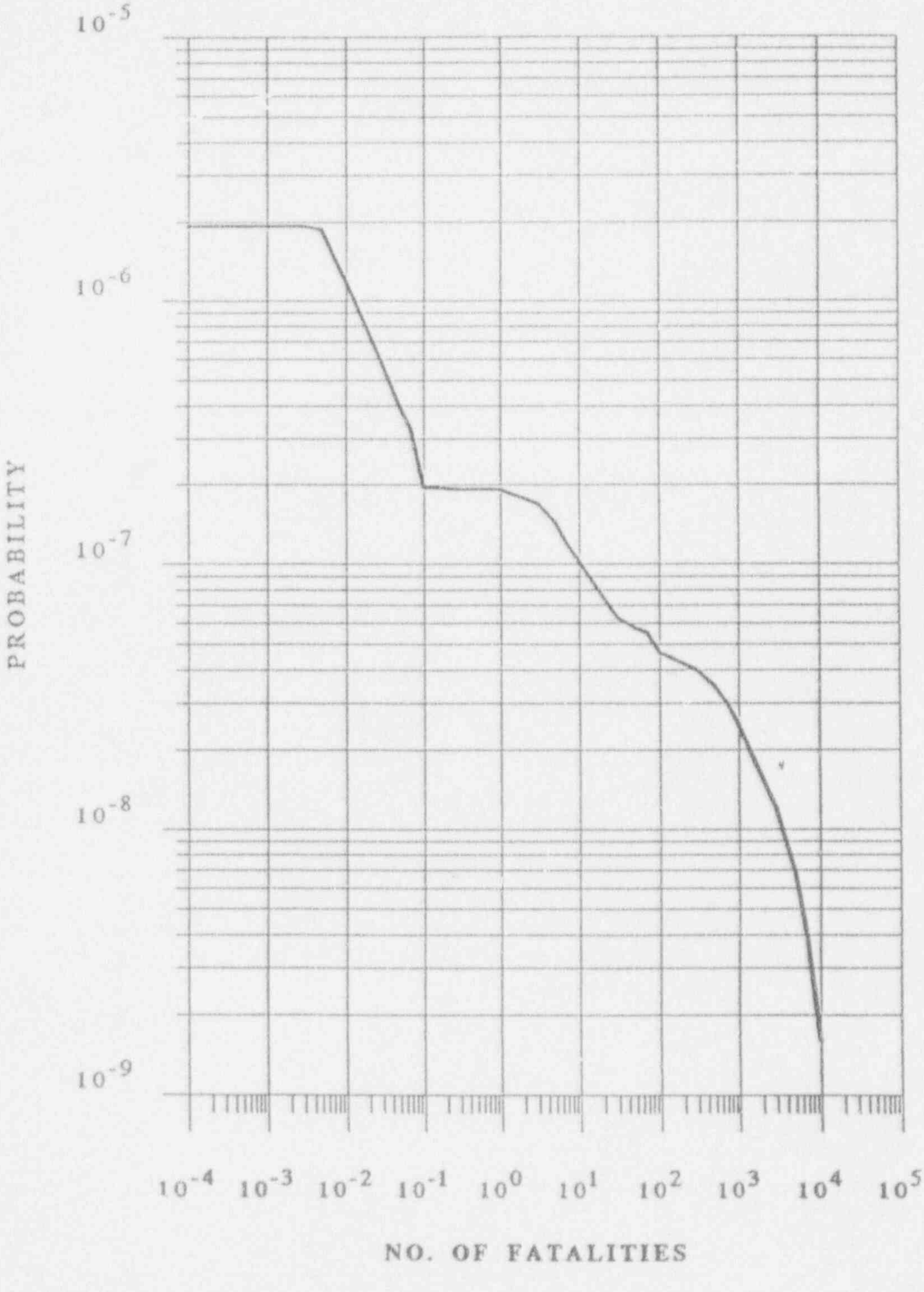


Figure 15.2-6

TOTAL CCDF FOR LATENT FATALITY FOR ALL RELEASE CLASSES



15.3 EXTERNAL EVENTS RISK PROFILE INSIGHTS

The external events analyses for the System 80+ design included both qualitative and quantitative analyses. Bounding site characteristics were used for the quantitative analyses to minimize potential future restrictions on plant siting. The qualitative external events evaluation involved the following: (1) identification of the external events to be considered, (2) grouping of events with similar plant effects and consequences, (3) establishment of screening criteria to eliminate events that are insignificant contributors to risk, and (4) identification of events that require further quantitative evaluation. Based on the qualitative evaluation, most of the external events were eliminate from further quantitative evaluation. Four external events (tornado, fire, flood, and seismic) were identified as having the potential to induce system failures and therefore required further quantitative evaluation.

The major findings and insights obtained for tornados, fires, and floods are provided in Sections 15.3.1, 15.3.2, and 15.3.3 respectively. The major insights for seismic events is not currently available and will be provided later.

15.3.1 Insights from the Tornado Strike Analysis

The core damage frequency due to tornado strike events is calculated to be 2.5E-07 per year. The dominant contributors to the core damage frequency of tornado strike events are provided in Table 15.3-1. For the System 80+ PRA it is assumed offsite power will be lost for more than 24 hour due to a tornado strike event. It is also assumed that the turbine-generator will be unable to runback and pick up hotel loads following a tornado strike event. Thus, the tornado strike considered in the analysis is an event which requires actuation of the emergency diesel generators due to the loss of offsite power. The alternate standby AC source is assumed to be unavailable due to the tornado strike and is therefore not credited in the analysis.

Three dominant accident sequences were identified for tornado strike events: TRND-4, TRND-SBO, and TRND-5. The calculated frequencies for these accident sequences are 2.5E-07, 1.7E-8, and 4.1E-9 per year respectively.

- TRND-4 is an accident sequence which involves a tornado strike event followed by successful opening and reseating of the primary safety valves, failure of decay heat removal, and failure of the SIS. For this sequence, the EFWS is used to remove decay heat from the RCS until shutdown cooling entry conditions are met. Once shutdown cooling entry conditions are met, the SCS would be aligned for decay heat removal. However, the SCS fails to perform its function and consequently the only other means of removing decay heat from the core is via "feed and bleed" operation. In order for "feed and bleed" to be successful both the "bleed" portion and the "feed" portion must operate. In addition to SCS failure, "feed" also fails for this sequence. This results in the termination of decay heat removal and consequently core damage occurs. The dominant contributors to this sequence are: (1) common cause operating failure of the emergency diesel generators, (2) operating independent failure of the emergency diesel generators.

- TRND-SBO is an accident sequence which involves a tornado strike event followed by station blackout with battery depletion. For this accident sequence, the emergency diesel generators also fail and the only mitigating system available is the EFWS, using the turbine-driven pumps. After eight hours of operation, the batteries would be depleted and long term decay heat removal would be terminated. Consequently, core damage occurs. The dominant contributors to this sequence are: (1) blockage of the station service water intake structure due to tornado generated debris, (2) common cause failure of the emergency diesel generators, and (3) independent demand failures of the emergency diesel generators.

- TRND-5 is an accident sequence which involves a tornado strike event followed by failure of long-term decay heat removal and failure of the SDS. This accident sequence is similar to TRND-4, except that "bleed" fails instead of "feed". The dominant contributors to this accident sequence are: (1) blockage of the station service water intake structure, (2) failure of the operator to initiate "feed and bleed", (3) common cause operating failure of the emergency diesel generators.

The major insight gained from the tornado strike evaluation is that the single most dominant contributor to core damage is caused by blockage of the intake structure. Blockage of the intake structure for the station service water pumps is caused tornado generated debris. Blockage would result in loss of cooling water to the emergency diesel generators and all safety-related motor-driven pumps. The EFWS, using the turbine-driven pumps, would be the only means of removing decay heat from the core until the batteries are depleted.

15.3.2 Insights from the Fire Risk Assessment

A qualitative fire risk assessment was performed for the System 80+ design. The evaluation addressed each of the fire areas defined, except the containment area and the control room area. Each fire area was analyzed to assure that in the event that all the active equipment in the area affected by a fire were rendered inoperable, redundant systems, trains, or channels would be available in another fire area. This would enable safe shutdown to be achieved and maintained. In performing the fire assessment is was assumed that the fire would not spread to adjacent areas due to the presence of 3-hour fire barriers between each of the areas. It also was assumed that no challenges to redundant systems would be experienced as a result of the fire. Finally it was assumed that redundant systems and equipment are not rendered inoperable by events not related to the fire.

A quantitative assessment of the risk due to internal fires can not be made at this time because detailed design information for cable routing and the fire detection and fire suppression system is not presently available. However, a scoping evaluation is performed to assess the risk due to internal fires in areas of the Nuclear Annex other than the containment or the control room. Two types of fires were considered in the scoping evaluation: (1) a fire in an area which could disable safety-related equipment in that area and which has the potential for initiating a transient, and (2) a fire in an area which by itself could disable safety-related equipment but would require would require the penetration of a fire barrier in order to initiate a transient. The first type of fire is designated as type "a" and the second type as type "b". The fire ignition sources and frequencies by applicable areas are presented in Table 15.3-2.

Although a detailed quantitative analysis of internal fires was not performed at this stage of the System 80+ design, a scoping estimate of the risk due to fire was calculated by using a conservative scoping valve (2.7E-02 per year) for fire event frequency and by assuming that the effects on plant systems would be the same as a loss of one division of component cooling water/station service water. Using this approach, the estimate core damage frequency due to internal fires is 1.9E-08 per year.

15.3.3 Insights from Internal Flood Analysis

The System 80+ plant design emphasizes the elimination and minimization of potential flood sources within safety-related areas as a means of flood protection. For example, station service water and component cooling water heat exchangers are located outside the Nuclear Annex. Water-cooled components within the Nuclear Annex are cooled by Component Cooling Water, with the exception of HVAC equipment which is cooled by chilled water systems. These cooling water systems are closed systems with a defined volume of water. The safety related cooling water systems are separated by division with no open cross connections, thus eliminating the possibility of a single pipe break from flooding one division and the other division being lost due to loss of pressure boundary integrity. Condenser circulating water is also located outside of the Nuclear Annex. These features reduce in-plant cooling water to a limited volume which can be easily accommodated to limit the extent of flooding.

The System 80+ control complex is protected from flooding in that no water lines are routed above or through the control room or computer room. Water lines routed to HVAC air handling units, around the control room, are contained in rooms with curbs which prevent any potential water leakage from entering the control room or computer room.

Protection from external flooding is provided by elevated building entrances. Secondary flooding sources located in the Turbine Building are confined to that building. Entrances from the Turbine Building to the Nuclear Annex are sufficiently elevated to allow operator action to isolate a break in the Condenser Circulating Water System before the water level from the Turbine

Building flood reaches the Nuclear Annex entrance. Lengths of high energy and moderate energy piping have been minimized by equipment location. Equipment is located in quadrants around the spherical containment to minimize the lengths of piping runs. The subsphere provides further close proximity of equipment to reduce piping runs from containment.

Flood barriers have been integrated into the design to provide further flood protection while minimizing the impact on maintenance accessibility. The primary means of flood control in the Nuclear Annex is provided by the structural wall which serves as a barrier between redundant divisions of safe shutdown systems and components.

Each half of the subsphere is compartmentalized to separate redundant safe shutdown components to the extent practical, while maintaining accessibility requirements. The subsphere, which houses the front line safety systems is compartmentalized into quadrants, with two quadrants on either side of the divisional structural wall. Flood barriers provide separation between the quadrants, while maintaining equipment removal capability.

The detailed information needed to identify the potential flood sources and flood levels, such as pipe routing, flood curbs and flood barriers, is not available at this time. However, for estimating the flood event frequency, it was assumed that flooding could occur due to: (1) the rupture of pipes connected to the water sources such as In-containment Refueling Water Storage Tank (IRWST), Emergency Feedwater Storage Tank (EFWST), and Component Cooling Water System (CCWS); (2) the rupture of the EFWST itself; or (3) the rupture of the main feedwater piping. In addition, it was assumed that pipe ruptures could potentially occur in either the suction leg or the discharge leg of an ESF pump. Therefore, there are 2 pipe sections associated with each ESF pump that could be a potential source of flooding. The tank rupture was assumed to be a catastrophic failure, resulting in spillage of all its contents to the floor immediately. Isolation of the flooding source was not credited in determining the flood event frequency. The worst credible flood event would affect only one division of the ESF equipment.

Although a detailed quantitative analysis of internal floods was not performed

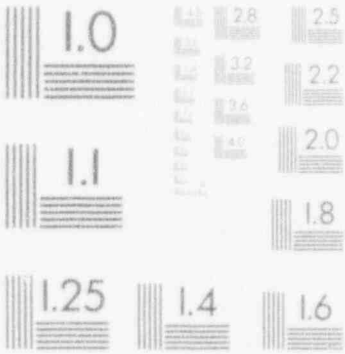
at this stage of the System 80+ design, a scoping estimate of the risk due to flood was calculated by using a conservative scoping valve (1.0E-02 per year) for flooding event frequency and by assuming that the effects on plant systems would be the same as a loss of one division of component cooling water/station service water. Using this approach, the estimate core damage frequency due to internal fires is 6.9E-09 per year.

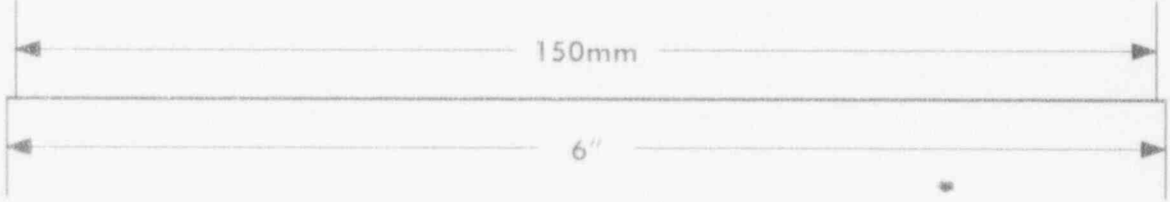
Table 15.3-1
CORE DAMAGE FREQUENCY CONTRIBUTIONS FOR DOMINANT ACCIDENT SEQUENCES
BY INITIATING EXTERNAL EVENT

SEQUENCE NUMBER	SEQUENCE	MEAN CORE DAMAGE FREQUENCY CONTRIBUTION EVENTS/YEAR	ERROR FACTOR
TRND-4	(TRND)(PSV Reseat)(Delivery of Emergency Feedwater OK)(Long-term Decay Heat Removal Fails)(Safety Depressurization for Bleed OK) (Safety Injection for Feed Fails)	2.50E-07	6.36
TRND-SBO	(TRND)(Station Blackout with Battery Depletion)	1.69E-08	6.99
TRND-5	(TRND)(PSV Reseat)(Delivery of Emergency Feedwater OK)(Long-Term Heat Removal Fails) (Safety Depressurization for Bleed Fails)	4.08E-09	8.99

1

IMAGE EVALUATION TEST TARGET (MT-3)





1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

1

IMAGE EVALUATION TEST TARGET (MT-3)

1.0 1.1 1.25 1.4 1.6 1.8 2.0 2.2 2.5 2.8 3.2 3.6 4.0

150mm

6"

Table 15.3-2

FIRE IGNITION SOURCES AND FREQUENCIES BY APPLICABLE FIRE AREAS

FIRE AREAS/ROOMS	FIRE IGNITION SOURCES	FIRE TYPE	FIRE FREQUENCY (EVENT/YEAR)
Auxiliary Building	Electrical Cabinets Pumps	a	1.9E-02 1.9E-02
Switchgear Room	Electrical Cabinets	a	1.5E-02
Cable Spreading Room	Electrical Cabinets	a	3.2E-03
Diesel Generator Room	Diesel Generators Electrical Cabinets	b	2.6E-02 2.4E-03
Battery Room	Batteries	b	3.2E-03
Reactor Building	Electrical Cabinets Pumps	x (see Note)	
Control Room	Electrical Cabinets	x	
Intake Structure	Electrical Cabinets Fire Pumps & Other	x	
Turbine Building	T/G Excitor, T/G Oil, T/G Hydrogen, Electrical Cabinets, Other Pumps, Boiler Main Feedwater Pumps	x	
Radwaste Area	Miscellaneous Components	x	
Transformer Yard	Yard Transformers	x	
Plant-Wide Components	Fire protection panels, Non-qualified cable run, junction box in qualified cable, Transformers, Battery Cahrgers, H_2 Tanks, Gas Turbines, Air Compressors, Ventilation Sub-systems, Dryers, etc.	x	

NOTE: For System 80+ plant, fires in these areas may initiate a transient but would not disable safety-related equipment and, therefore, are excluded from the scoping evaluation.

15.4 SHUTDOWN AND LOW-POWER OPERATION RISK INSIGHTS

A study of the risk associated with the low power and shutdown modes of operation was performed (see Section 8.0). The scope of this assessment included the evaluation of both internal events and external events occurring during low-power and shutdown modes of operation.

Event trees were developed and quantified for loss of decay heat removal (DHR) and loss of coolant inventory as initiating events during Modes 4 through 6. The core damage frequencies associated with loss of offsite power, fire, and floods were also quantified. In quantifying the core damage frequency (CDF), emphasis was placed on the human errors because they have been shown, in earlier studies, to be dominant contributors to shutdown risk. The system failure probabilities were evaluated using modifications to the fault trees presented in Section 6.0.

The results obtained from the shutdown and low-power risk evaluation are summarized in Section 15.4.1. Similar to the PRA that was performed during power operation, insights were gained for the shutdown risk evaluation. Such insights are summarized in Section 15.4.2.

15.4.1 Results

The estimate core damage frequency attributable to internal and external events during shutdown and low-power modes of operation is 8.4E-07 per year. Table 15.4-1 identifies the contributors to core damage frequency (CDF) by shutdown and low-power modes of operation and by initiating event. Reduced inventory operation during mode 5 (Mode 5R) accounts for 48% of the internal risk. Loss of DHR is the dominant initiator for accident sequence in this mode of operation. (This mode of operation and sequence were also identified in earlier PRAs as being the dominant contributors to CDF.) Internal events occurring during modes 6E and 6I are the second leading contributors (29%) to CDF. During mode 6E or 6I, the IRWST is empty and therefore not available as a source of coolant for any makeup or feed and bleed operation. LOCA is the dominant initiator for accident sequence in this mode of operation because the capability to provide makeup to the core is limited. The third leading contributors to CDF occur during Modes

4, 5, and 6F (IRWST full). Loss of offsite power is the dominant initiator for accident sequence during this mode of operation because of the long time interval spent in this configuration and the need to restore AC in about two hours.

The importance of the various internal initiating event types in terms of CDF contribution is also shown in Table 15.4-1. Loss of offsite power (LOOP) is the leading contributor (39%) to CDF for internally initiated events during low-power and shutdown modes of operations. Even with requirements for two switch-yards and two standby generators to be available during shutdown, the risk was not negligible. Loss of DHR was the second largest initiating event type in terms of contribution to CDF (36% of internal CDF).

Fires (3.0E-07 per year) account for 36% of the total internal and external CDF during shutdown and low-power operations. The flooding CDF was calculated to be 8.2E-8 per year and was modeled as part of the LOCA risk.

Table 15.4-2 compares the CDF for low-power and shutdown events with those for power operation. The CDF for low-power and shutdown modes is 30% of the total CDF. The CDF for shutdown and low-power events is significantly less than the EPRI goal of 1.0E-05 per year. The total System 80+ CDF is also less than the EPRI goal.

Table 15.4-3 lists the dominant sequences leading to core damage. Loss of DHR in reduced inventory accounts for 24% of the internal event CDF. LOOP in Modes 4, 5, and 6F is the second largest sequence. LOCAs in Modes 6E, and 6I when the IRWST and SIS is not necessarily available are the third largest sequence.

Table 15.4-4 compares the results of this study with other shutdown PRAs. In all these studies, loss of DHR during reduced inventory is the largest single contributor. The lower total CDF for System 80+ is due to design improvements. The System 80+ has two dedicated Shutdown Cooling Systems (SCS) that can also be used to feed coolant from the IRWST in a LOCA or during reduced inventory operations. The Containment Spray System (CSS) pumps are installed as spares to the SCS pumps. The System 80+ has a four train Safety Injection System (SIS) and a new technical specification requires that two trains be available during most

shutdown modes. This coupled with an IRWST gives added LOCA protection and a DHR path using feed and bleed.

This study was performed as a part of the System 80+ design process and had an impact on the design. For example, the cross-connect valves for the CSS were replaced with MOVs which resulted in an improvement in the restoration of DHR. This PRA has been used to help develop technical specifications. For example, a new technical specification for two SIS trains to be available in modes when the IRWST is available is being considered.

15.4.2 Insights

The insights from the low-power and shutdown modes of operations are listed below. General insights are provided as well as more specific insights as they relate to the type of initiator.

<u>General Insights</u>

The general insights for low-power and shutdown modes of operation are listed below.

- The initiating event frequencies are higher than Mode 1 operation because of the greater opportunity for operator errors during outages. Operator training and management control of plant configuration is important to reducing shutdown events and risk.

- The operation and maintenance personnel must have the procedures, training and spare parts to restore DHR in a timely manner. SCS operation is the true end-state for shutdown sequences.

- Most systems are manually started or aligned in shutdown modes. Training is especially important because the operator must be able to cope with the plant in an unplanned configuration.

- The concept of defense in depth applies to shutdown modes as well as

Mode 1. The more ways that the operator can maintain coolant inventory and remove decay heat, the lower the risk. The presence of SIS capability in shutdown is an example of added defense in depth.

Loss of DHR Insights

The major insights from the loss of decay heat removal during low-power and shutdown modes of operation are listed below:

- Reduced inventory is the most critical operation. The operator should be aware of this and plant activities should be scheduled accordingly. Use of nozzle dams is encouraged as a method of limiting the time spent in this mode.

- The operator must have procedures and training to align the SCS train to the IRWST and use it to makeup inventory or do a feed and bleed operation.

- Failure of the standby SCS train for either DHR or feed operation is dominated by failure of control valves and MOVs. An aggressive valve testing and maintenance program on the SCS and CSS would reduce shutdown risk.

- The use of the CVCS to makeup inventory is an important recovery action in reduced inventory operation. It also acts as a temporary (about 12 hours) cooling technique. The operator should have procedures and training on its use.

- Safety injection in conjunction with bleed is an important means of removing decay heat during shutdown modes. Having two of the four SIS trains available during most shutdown modes is an important new technical specification.

- The CSS pumps are used as backup to the SCS pumps. The operator

must therefore be properly trained in performing the procedure(s) to alignment the CSS pumps for operation if the SCS pumps should fail. Again, valve maintenance and testing is important for shutdown risk reduction.

LOCA Insights

The major insights for LOCAs during low-power and shutdown modes of operation are listed below:

- SCS injection is an important means of makeup following a LOCA. Therefore, the development of appropriate procedures and the training of the operator to perform these procedures are necessary and important in mitigating LOCA events during low-power and shutdown modes of operation.

- The dominant failure mode for SCS feed is failure of control valves and MOVs. A valve maintenance program is important.

- The use of the SIS to provide injection during a LOCA important. Since manual actuation is required, training and procedures are required to properly accomplish this task.

- The CVCS is another important means of makeup following a LOCA and with proper training and procedures this system will most likely be available when required.

- For LOCAs located in the containment, the IRWST acts as a sump and makes the coolant available for injection. Procedures are needed to ensure that flow paths are maintained during the outage.

LOOP Insights

The insights for loss of offsite power during low-power and shutdown modes of

operation are listed below:

- The new technical specification for having two of the three standby and emergency generators available reduces the CDF.

- The reliability of the two switchyards is important to risk reduction for LOOP. Procedures to control maintenance in both these areas at the same time should be considered.

- Because of the nature of LOOP events, the importance of restoring a source of AC to the nuclear facility must be emphasized.

Fire Risk Insights

The major insights for fire events occurring during low-power and shutdown modes of operation are listed below:

- The frequency of fires in outages is high because of the maintenance activities and can be reduced by training.

- The owners of the facility must maintain a well trained and prepared fire brigade.

- In order to maintain the validity of the assessment of the level of fire risk associated with the System 80+ design, separation must be maintained between systems comprising the alternate success paths within a quadrant. Separation between systems implies not only separation between their major components but also separation between their power supply and control cables.

Flooding Risk Insights

The major insights for flooding events that occur during low-power and shutdown modes of operation are listed below:

- In order to maintain the validity of the assessment of the level of flooding risk associated with the System 80+ design, separation must be maintained between systems comprising the alternate success paths within a quadrant. Separation between success paths implies not only separation between their major components but also separation between the associated power supplies.

Table 15.4-1

FREQUENCY OF CORE DAMAGE FOR SHUTDOWN EVENTS

Low-power & Shutdown Modes	Internal Events			Total
	Loss of DHR CDF	LOCA CDF	LOOP CDF	
2, 3			1.5E-08	1.5E-08
4, 5, 6F	1.1E-09	1.4E-09	1.0E-07	1.1E-07
5R	1.5E-07	2.5E-08	8.2E-08	2.6E-07
6E, 6I	3.9E-08	1.1E-07	1.0E-08	1.6E-07
Total CDF for Internal Events				5.4E-07
4, 5, 6E, 6F, 6I	External Events (Fire)			3.0E-07
Total CDF for Internal and External Events During Low-power & Shutdown Modes				8.4E-07

5R Mode 5 with reduced RCS inventory
6E Mode 6 with IRWST empty and upper internals removed
6F Mode 6 with IRWST full
6I Mode 6 with IRWST empty and upper internals in place

Table 15.4-2
CORE DAMAGE FREQUENCY CONTRIBUTION BY INITIATING EVENT

INITIATING EVENT	MEAN CORE DAMAGE FREQUENCY	ERROR FACTOR	PERCENT OF TOTAL
Large Loss-Of-Coolant-Accident (LLOCA)	1.1E-07	5.8	6.6
Medium Loss-Of-Coolant-Accident (MLOCA1)	1.4E-07	5.3	8.3
Medium Loss-Of-Coolant-Accident (MLOCA2)	1.7E-07	5.8	9.8
Small Loss-Of-Coolant-Accident (SLOCA)	2.0E-07	10.9	11.8
Large Secondary Side Break (LSSB)	1.9E-09	14.1	0.1
Steam Generator Tube Rupture (SGTR)	2.9E-07	11.8	16.9
Loss of Feedwater Flow (LOFW)	5.1E-07	4.8	30.1
Other Transients (TOTH)	7.6E-08	6.0	4.5
Loss Of Offsite Power (LOOP)	1.9E-08	7.7	1.1
Station Blackout with Battery Depletion	2.1E-08	9.2	1.3
Loss of Component Cooling Water (CCW) Div 2	4.9E-10	10.2	0.0
Loss of 4.16 Kv Bus	3.5E-10	6.5	0.0
Loss of 125 VDC Vital Bus	1.8E-10	4.4	0.0
Anticipated Transient Without Scram (ATWS)	5.4E-08	8.5	3.2
Interfacing System LOCA	5.2E-10	234.0	0.0
Loss of HVAC	6.4E-09	14.9	0.4
Vessel Rupture	1.0E-07	10.0	5.9
Internal Events - Total	1.7E-06*	2.7	100.0
Tornado Strike Events	2.5E-07	7.1	90.7
Fire (scoping estimate)	1.9E-08**		6.8
Flood (scoping estimate)	6.9E-09**		2.5
External Events - Total	2.8E-07**		100.0
Internal Events	1.7E-06	2.7	60.0
External Events	2.8E-07**		10.1
Shutdown Risk & Low-power (Int. & Ext.)	8.4E-07**		29.9
TOTAL	2.8E-06**		100.0

* This value represents the mean of the combined accident sequences for internal initiators, and not the sum of the mean for each internal initiator.

** Best Estimate.

Table 15.4-3

CORE DAMAGE FREQUENCY CONTRIBUTIONS FOR DOMINANT ACCIDENT SEQUENCES BY INITIATING INTERNAL EVENT DURING SHUTDOWN & LOW-POWER OPERATION

MODE OF OPERATION	SEQUENCE	MEAN CORE DAMAGE FREQUENCY CONTRIBUTION	
		EVENTS/YEAR	% of TOTAL
5R	(LDHR)(Failure to Restore Operating SCS Train)(Failure to makeup using CVCS or SCS)(Bleed OK)(Safety Injection for Feed fails)	1.3E-07	24.0
4, 5, 6F	(LOOP)(Failure of Alternate Switchyard, Emergency Diesel)(Failure of Alternate AC Source)(Failure to Restore AC)	1.0E-07	19.0
6E, 6I	(LOCA)(Failure to Isolate LOCA)(Failure of SCS Injection)(Successful makeup by CVCS)(Failure to Restore DHR in 12 Hours)	8.6E-08	16.0
5R	(LOOP)(Failure of Alternate Switchyard, Emergency Diesel)(Failure of Alternate AC Source)(Failure to Restore AC)	8.2E-08	15.0
6I	(LOOP)(Failure to Restore SCS Operating Train)(Failure to Start Standby SCS Train)(Failure to use CSS Pumps for Heat Removal)(Failure to Restore DHR in 18 Hours)	3.1E-08	5.7
5R	(LDHR)(Failure to Restore Operating SCS Train)(Make by CVCS OK)(Failure to Start Standby SCS Train)(Bleed OK)(Failure to Recover SCS in 12 Hours)	1.6E-08	3.0
2, 3	(LOOP)(Failure of Alternate Switchyard, Emergency Diesel)(Failure of Alternate AC Source)(Failure to Restore AC)	1.5E-08	2.8
6E, 6I	(LOOP)(Failure of Alternate Switchyard, Emergency Diesel)(Failure of Alternate AC Source)(Failure to Restore AC)	1.0E-08	2.0
5R	(LOCA)(Isolation of LOCA OK)(Failure to Provide Makeup)(Bleed OK)(Safety Injection for Feed Fails)	9.9E-09	1.8

TABLE 15.4-4

COMPARISON OF SHUTDOWN PRAs

EVENT\ STUDY TYPE	SYSTEM 80+	NSAC-84	NUREG/ CR-5051	SEABROOK STUDY
TOTAL CDF	8.4E-7	1.8E-5	5.2E-5	4.5E-5
REDUCED INVENTORY	31%	61%	64%	71%
LOSS OF DHR	23%	71%	82%	61%
LOCA	16%	10%	8%	18%
LOOP	25%	0.7%	10%	6%
FIRE	36%			4%
OTHER				11%

15.5 USE OF PRA IN THE DESIGN PROCESS

Probabilistic Risk Assessment (PRA) was used extensively in the System 80+ design process. PRA was used to confirm that the System 80+ design complied with the applicable risk goals, and to select among the alternate design options.

The insights gained from past PRAs, especially the System 80, were used to identify vulnerabilities in operating plants. This information was then used to incorporate features in the System 80+ design that reduced or eliminated these vulnerabilities. PRA was then used to confirm the risk reduction associated with these improvements. Examples are the risk reduction presented in Section 15.2.1.1 for LOOP/SBO, SGTR, transients, small LOCA, and ATWS accident sequences. in Section 15.2.1.1.

Another use of the PRA in the System 80+ design process, which was also of a confirmatory nature, was to demonstrate compliance with applicable risk goals. In performing the PRA for System 80+, failure of the safety depressurization valves and the cavity flooding valves due to seismic failure of their dedicated inverters at relatively low acceleration was determined to be potentially risk significant. Therefore, a design requirement for seismic isolation of these inverters was added.

The System 80+ PRA was also used to evaluate design alternatives. The major design options are cited below.

COMPONENT COOLING WATER SYSTEM CONFIGURATION

Early in the program, System 80+ had a standby, safety related Essential Component Cooling Water System and Essential Service Water System for cooling safety related loads. Demand failure of the pump and valves in these systems were found to be significant risk contributors. As a result, the System 80+ was changed to a normally operating Component Cooling Water System (CCWS) and a Station Service Water System (SSWS) where the non-safety loads can be shed when required. The selected CCWS and SSWS have two divisions with two pumps in each division. One pump in each division is normally operating and the second pump

is in standby and will start if the operating pump in the same division trips. A subsequent evaluation was also made to determine if the standby pumps had to be automatically loaded on the emergency diesel generators and started following a LOOP event. The evaluation indicated that there would be no significant risk impact if the standby pumps were aligned to the emergency diesel generators following a LOOP event but were not started unless the previously operating pump fails to restart. Thus larger and consequently less reliable emergency diesel generators were not required.

EMERGENCY AC POWER CONFIGURATION

The System 80+ design includes two emergency diesel generators which provide power to the safety related loads following a LOOP event. In addition, there is also a standby alternate AC power source (combustion turbine) which can be aligned to either of the safety related 4.16 KV buses in the event of a failure of one of the emergency diesel generators. The alternate AC power source is sized to provide power to a set of non-safety loads which, from an operational stand-point, is desirable following a LOOP event. PRA was used to compare two configurations of emergency power: (1) two emergency diesel generators plus a combustion turbine, and (2) four emergency diesel generators. The comparison indicated that the four diesel generator configuration was slightly, but not significantly, more reliable than the configuration which included two diesels and a combustion turbine. However, the four diesel generator configuration did not provide power to the permanent non-safety loads. In addition, the four diesel generator configuration would have a significant impact on plant size, cost, and layout because of the need for two additional divisions of diesel support systems such as cooling water, starting power, and fuel supplies.

EVALUATION OF SAMDA

The System 80+ PRA was also used to evaluate eleven Severe Accident Mitigation Design Alternatives (SAMDAs). The selected alternatives were based on the Design Alternatives evaluated for Limerick and on the results of the System 80+ PRA. The design alternative analysis used a bounding technique. It was assumed that each design alternative worked perfectly and completely eliminated the

accident sequences that the design alternative addressed. This approach maximizes the benefits associated with each design alternative. The eleven design alternatives are listed below:

- A perfect containment spray system (i.e., zero failure probability) that prevents high pressure containment failures caused by slow steam pressurization.

- A filtered vent design that prevents all slow high pressure containment failures.

- An improved DC batteries and EFWS design alternative that allows for decay heat removal during a station blackout event by using the batteries and the turbine-driven pumps of the EFWS for the time period that is required (without any failures).

- A reactor coolant pump seal design that cannot withstand loss of cooling to the seals which would then lead to seal LOCA and consequently core damage.

- An auxiliary spray system alternative design that always depressurizes the primary system (during SGTR events) with sufficient speed to ensure that the SCS would always remove decay heat.

- A system of relief valves that prevents any equipment damage from a primary pressure spike following an ATWS event.

- An ideal concrete composition that prevents base-mat melt-through.

- A reactor vessel exterior cooling system that prevents vessel melt-through and direct containment heating.
- Ideal hydrogen (H_2) igniters that prevents containment failures from hydrogen burns or explosions.

- A high pressure safety injection system alternative design that eliminates most sequences with high pressure failures.

- A perfect safety depressurization system that quickly depressurizes the primary system to allow the SITs and safety injection pumps to be used for delivering coolant to the core and for removing decay heat.

The estimated cost (in millions of dollars) and benefits (in person-rems of risk reduction) for each of the above design alternatives are presented in Table 15.5-1. None of the above design alternatives was found to be cost-beneficial.

Table 15.5-1

Summary of the Risk Reductions of the Design Alternatives

DESIGN ALTERNATIVE	PERSON-REM REDUCTION (%)	NET CAPITAL BENEFIT ($M)*
PERFECT CONTAINMENT SPRAY SYSTEM	90.0	(1.5)
FILTERED VENT	86.0	(10.0)
IMPROVED DC BATTERIES AND EFWS	69.0	(2.0)
REACTOR COOLANT PUMP SEAL COOLING	16.5	(0.1)
PRESSURIZER AUXILIARY SPRAY	6.7	(5.0)
ATWS VALVES	4.2	(1.0)
IDEAL CONCRETE COMPOSITION	2.5	(5.0)
REACTOR VESSEL EXTERIOR COOLING	2.5	(5.5)
H_2 IGNITERS	0.1	(1.0)
HIGH PRESSURE SAFETY INJECTION	0.0	(20.0)
REACTOR COOLANT SYSTEM DEPRESSURIZATION	0.0	(0.5)

* Negative benefits (costs) are indicated by parentheses ( ).

15.6 USE OF PRA TO SUPPORT CERTIFICATION ACTIVITIES

The System 80+ PRA results and insights are used in support of pre- and post-certification activities. The majority of the insights are identified during the pre-certification stage of the design. As a result, this has lead to further improvements in the design to eliminate or minimize potential vulnerabilities during the review process. The following activities include the use of PRA insights in support of design certification process.

- Understanding of the design robustness to severe accidents - PRA insights are used to develop an in-depth understanding of the robustness and tolerance of the System 80+ design to severe accidents initiated by events which are either internal or external to the plant systems.

- Importance of operator interface with the design - PRA insights are used to identify risk significant human errors associated with the System 80+ design. By characterizing the risk significant human error, new operating procedures can be developed or existing procedures refined to provide better training to plant operators.

- Development and implementation of other programs - the PRA results and insights were used to systematically identify the key assumptions, major operator actions, and risk significant components that characterize the "present" risk of the System 80+ design. This information was used to support such programs as: (1) Design Acceptance Criteria (DAC), (2) Inspection, tests, analyses, and acceptance criteria (ITAAC), and (3) Reliability Assurance Program (RAP).

The PRA for the System 80+ design provides adequate models and associated data to effectively support the above mentioned certification activities.

ATTACHMENT 6

defect size, the rod can fill rapidly, but during a power increase it also expels water or steam readily without a large pressure buildup. Defects which could result in an opening in cladding are scrupulously checked for during the fuel rod manufacturing process by both ultrasonic and helium leak testing. Clad defects which could develop during reactor operation due to hydriding are also controlled by limiting those factors; e.g., hydrogen content of fuel pellets, which contributes to hydriding.

The most likely time for a waterlogging rupture incident would be after an abnormally long shutdown period. After this time, however, the startup rate is controlled so that even if a fuel rod were filled with coolant, it would "bake out", thus minimizing the possibility of additional cladding rupture. The combination of control and inspection during the manufacturing process and the limits on the rate of power change restrict the potential for waterlogging rupture to a very small number of fuel rods.

The UO_2 fuel pellets are highly resistant to attack by reactor coolant in the event cladding defects should occur. Extensive experimental work and operating experience have shown that the design parameters chosen conservatively account for changes in thermal performance during operation and that coolant activity buildup resulting from cladding rupture is limited by the ability of uranium dioxide to retain solid and gaseous fission products.

4.2.3.2.10 Fuel Burnup Experience

The C-E fuel rod design is based on an extensive experimental data base and by an extension of experimental knowledge through design application of C-E fuel rod evaluation codes. The experimental data base includes data from C-E or C-E/Kraftwerk Union (KWU) joint irradiation experiments, from C-E and KWU operating commercial plant performance and from many basic experiments conducted in various research reactors which are available in the open literature. Some of these sources are discussed below. Evidence currently available indicates that Zircaloy and UO_2 fuel performance is satisfactory to exposures in excess of ~~55~~ 59,000 MWd/MTU (Reference 78).

A. Public Information

General fuel performance information available in the open literature has provided part of the C-E fuel rod design data base. Particular experiments that have been cited in the past as key references are:

1. Determination of the effect of fuel-cladding gap on the linear heat rating to melting for UO_2 fuel rods, conducted in the Westinghouse test reactor.

is 197

at full power ~~is 197~~

times in cycle. Also, the effect of full insertion of the part-strength CEA group on the axial peaking factor is negligible for steady state operation. The three-dimensional peaking factor, $F^n q$, expected during steady-state operation is then just the product of the planar radial peaking factor ($F^n r$) and the axial peaking factor. The maximum expected value of $F^n q$ ~~is 2.00~~ during the first cycle and, as can be seen from the above figures, occurs near the beginning-of-cycle for steady-state, base loaded operation with no full-strength or part-strength CEA insertion.

Figures 4.3-24 through 4.3-35 show typical fuel cycle loading patterns, initial burnup distributions, and planar radial power distributions for the second and third cycles based on a refueling interval of approximately 18-months. The expected power distributions for these cycles are similar to those of the first cycle except for reduced power in fuel assemblies located on the periphery of the core and consequently higher radial peaking factors in the interior region of the core. The expected power distributions are well within the nuclear design limits described in Section 4.3.2.2.2. The uncertainty associated with these calculated power distributions is discussed in Section 4.3.3.1.2.2.6.
3

The capability of the core to follow load transients without exceeding power distribution limitations depends on the margin to operating limits compared to the margin required for base loaded, unrodded operation. In order to illustrate the core maneuvering capability, the results of calculations of the power distributions and power peaking factors during load following transients are discussed below. The axial power distributions are calculated by VISIONS (Reference 2), a three-dimensional neutron diffusion code that considers the effects of the temporal and spatial variations of xenon and iodine concentration, CEA positions, fuel temperature and moderator temperature distributions, soluble boron concentration, and burnup. The nuclear peaking factors $F^n q$ and $F^n r$ are synthesized in VISIONS using the calculated three-dimensional coarse-mesh power distribution and input pin-to-box factors from MC (see Section 4.3.3.1.1.3). Figures 4.3-36 through 4.3-39 show the calculated axial power distributions and associated nuclear peaking factors during a typical day of a maneuvering transient to 50 percent of the full power conditions. Figures 4.3-36 and 4.3-37, which represent maneuvering transients near beginning-of-cycle and end-of-cycle, respectively, also show the locations of full-strength and part-strength CEA groups during the transients. The transients begin with the lead part-strength CEA group fully inserted. Throughout the calculation of the

power distribution during these transients it is assumed that the part-strength CEA groups are available for control of the axial power distribution. The two part-strength CEA groups are moved to positions that minimize the difference between the current shape index and the reference value of shape index that existed prior to the initiation of the maneuver. In addition, the positions of the part-strength CEA groups supplement reactivity control provided by full-strength regulating rods, so that the calculated maneuvering transients can be accomplished without changing soluble boron concentration to compensate for reactivity changes due to power level and xenon.

The detailed radial power distribution within any assembly is a function of the location of that assembly within the core as well as the time in life, CEA insertion, and other considerations. The normalized assembly power distribution used for the sample DNB calculation discussed in Section 4.4.2.2. is shown on Figure 4.3-40. In Section 4.3.3.1.2, the accuracy of calculations of the power distribution within a fuel assembly is discussed.

4.3.2.2.4 Allowances and Uncertainties on Power Distributions

In comparing the expected power distributions and implied peak linear heat generation rate (PLHGR) produced by analysis with the design limits stated in Section 4.3.2.2.2, consideration must be given to the uncertainty and allowances associated with on-line monitoring by COLSS.

The COLSS uncertainty analysis, as applied to System 80, is described in Section 7.7 and in Reference 1. For monitoring linear heat rate, COLSS applies an overall uncertainty factor for linear heat rate measurement, in addition to a power level uncertainty factor of 1.02. These factors are applied to the COLSS monitoring of $F^n q$, such that a COLSS-measured $F^n q$ of 2.28, the chosen design limit for $F^n q$ given in Section 4.3.2.2.2, will not result in exceeding the design limit for PLHGR. The allowances and uncertainties applied for the COLSS monitoring of thermal margin to the DNBR limit are also described in Section 7.7 and Reference 1.

4.3.2.2.5 Comparisons Between Limiting and Expected Power Distributions

1.97,

As discussed in Section 4.3.2.2.3, the maximum expected unrodded $F^n q$ that occurs during the first cycle at full power is 2.00. ~~Augmenting this value by the required calculational uncertainty (Reference 3) provides an upper limit on~~ ~~$F^n q$ of 2.11~~ which is well below the design target of 2.28. Additionally, the calculations described in Section 4.3.2.2.3 show that, with

is reduced. The buildup of equilibrium xenon produces a net negative change of -0.3×10^{-4} $\Delta\rho/°F$ in the moderator temperature coefficient; this change is due mainly to the accompanying reduction in critical soluble boron. The changing fuel isotopic concentrations and the changing neutron spectrum during fuel depletion also contribute a small negative component to the moderator temperature coefficient.

The dependence of the moderator temperature coefficient on moderator temperature at BOC and EOC (at constant soluble boron) is shown in Figures 4.3-42 and 4.3-43, respectively. These figures also show the expected moderator temperature coefficient at reduced power levels (corresponding to reduced moderator temperature) based on power reductions accomplished with soluble boron only and with CEAs only. These two modes of power reduction result in the most positive and most negative moderator temperature coefficients expected to occur at reduced power levels. These figures show the expected moderator temperature coefficient for the full range of expected operating conditions and accident conditions addressed in Chapter 15.

4.3.2.3.3 Moderator Density Coefficient

The moderator density coefficient is the change in reactivity per unit change in the core average moderator density at constant moderator temperature. A positive moderator density coefficient translates into a negative contribution to the total moderator temperature coefficient, which is defined in Section 4.3.2.3.2. The density coefficient is always positive in the operating range, although the magnitude decreases as the soluble boron level in the core is increased. The calculated density coefficient is shown in Table 4.3-4, and curves of density coefficient as a function of density for several soluble boron concentrations are presented in Figure 4.3-44. These curves are based upon 3-D ROCS calculations and have been generated over a wide range of core conditions. The density coefficients explicitly used in the accident analyses are based upon core conditions with the most limiting temperature coefficients allowed by the technical specifications. Table 4.3-3 shows a comparison of the expected values of the moderator temperature coefficients with those actually used in the accident analyses.

4.3.2.3.4 Moderator Nuclear Temperature Coefficient

The moderator nuclear temperature coefficient is the change in reactivity per unit change in core average moderator temperature, at constant moderator density. The source of this reactivity dependence is the spectral effects associated with the change in thermal scattering properties of water molecules as the internal energy, which is represented by the bulk water temperature, is

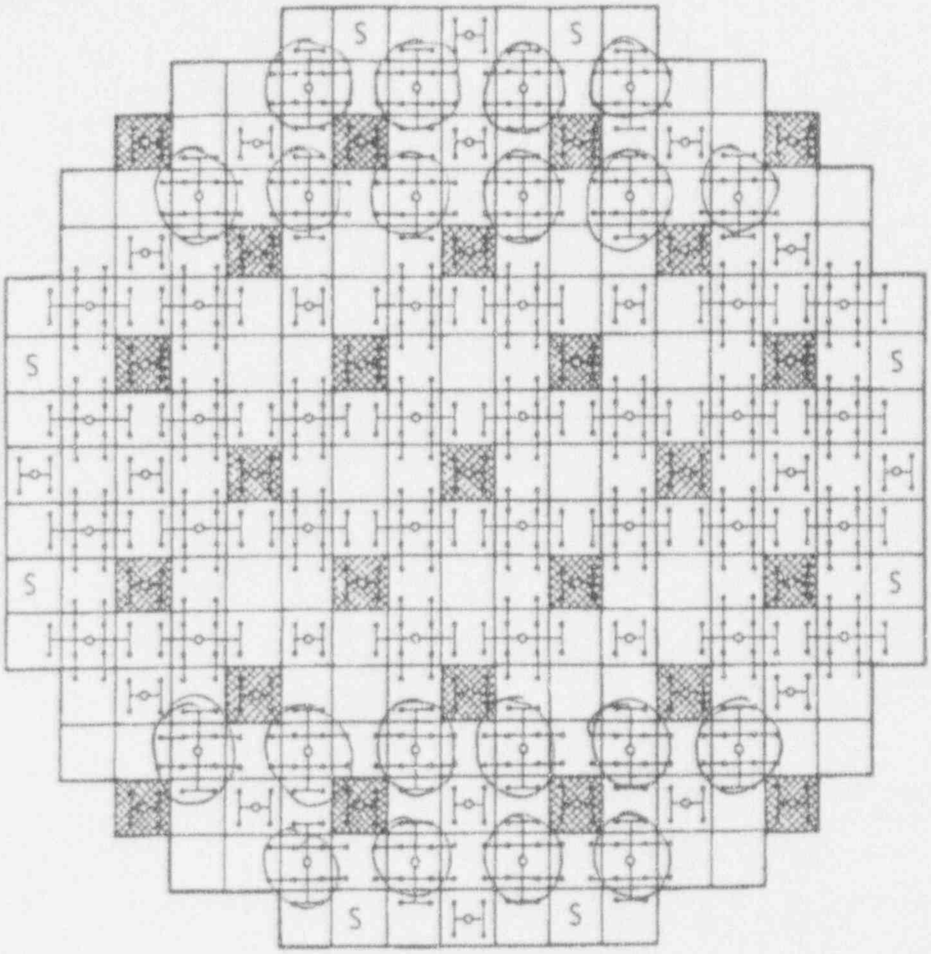
of the stuck CEA) to the reactivity worth requirements from Table 4.3-6. All required biases and uncertainties have been included in the CEA worths of Table 4.3-7. Section 4.3.3 presents detailed information on biases and uncertainties.

4.3.2.5 Control Element Assembly Patterns and Reactivity Worths

The locations of all CEAs are shown in Figure 4.3-46. The CEAs designated as regulating control rods are divided into three groups, the shutdown CEAs are divided into three groups, and the part-strength CEAs (PSCEAs) are divided into two groups. These groups are identified, for first cycle operation, in Figure 4.3-47. All CEAs in a group are withdrawn or inserted quasi-simultaneously. Shutdown groups are inserted after the regulating groups are inserted and are withdrawn before the regulating groups are withdrawn. The reactivity worths of sequentially inserted CEA groups are shown in Table 4.3-5 near the beginning and end of the first cycle where the maximum rod radial peaking factors (F_r^n) for these configurations occur. The values of F_r^n for these times are shown in Table 4.3-8.

It is expected that the core will operate essentially unrodded during full power, base-load operation, except for limited insertion of the lead PSCEA group or the lead regulating group in order to compensate for minor variations in moderator temperature and boron concentration. If the plant is required to perform load follow operations, such as planned load cycles, the full power operation of the core may involve full insertion of the lead PSCEA group to enable control strategies for power changes which can remove the need for soluble boron level changes. Movement of the PSCEAs will be restricted only by their effect on axial power distribution. For operation with substantial insertion of regulating CEAs, the relationship between power level and maximum permitted CEA insertion is typified in Figure 4.3-48. This figure also illustrates the regulating group insertion order (3-2-1) and the 40% fixed overlap between successive regulating groups. Compliance with the power dependent insertion limits throughout the cycle insures that adequate shutdown margin is maintained and that the core conditions are no more severe than the initial conditions assumed in the accident analyses described in Chapter 15.

Reactivity insertion rates for the safety analysis of the core are presented in Chapter 15. The full power CEA ejection accident considers the ejection of one CEA from the maximum insertion of the lead regulating bank allowed by the PDIL. The ejected CEA worth is calculated by taking the difference between the pre-ejection and post-ejection reactivity of the core computed by static methods. The maximum ejection CEA worth at hot full power used in the safety analysis is conservative since



Rotate Circled CEAs by 90°

12 ELEMENT FULL STRENGTH CEA

4 ELEMENT FULL STRENGTH CEA

4 ELEMENT PART STRENGTH CEA

S DENOTES SPARE CEA LOCATIONS FOR OPEN MARKET PLUTONIUM RECYCLE

Amendment ~~#~~ O
~~[illegible]~~
May 1, 1993

SYSTEM 80+™	CONTROL ELEMENT ASSEMBLY LOCATIONS	Figure 4.3-46

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

	MAX. VALUE	IN BOX
BOX RPD	1.27	25
MAX PIN	1.46	25

					1 C0 0.68 1.15	2 C1 0.94 1.32	3 C1 0.99 1.35	4 B0 0.90 1.21
			5 C0 0.60 1.01	6 C7 0.81 1.24	7 C7 1.08 1.32	8 B6 1.12 1.31	9 B6 1.14 1.21	10 B6 1.15 1.33
		11 C0 0.69 1.09	12 C7 1.00 1.30	13 A0 0.85 0.94	14 B5 1.16 1.38	15 A0 0.98 1.05	16 B6 1.19 1.33	17 A0 0.98 1.05
	18 C0 0.60 1.01	19 C7 1.00 1.29	20 A0 0.85 0.94	21 C8 1.20 1.40	22 A0 0.92 1.00	23 B5 1.19 1.38	24 A0 0.94 1.02	25 C8 1.27 1.46
	26 C7 0.81 1.24	27 A0 0.85 0.94	28 C8 1.20 1.40	29 A0 0.87 0.94	30 B8 1.03 1.22	31 A0 0.90 0.97	32 B8 1.07 1.23	33 A0 0.91 0.99
34 C0 0.68 1.15	35 C7 1.08 1.32	36 B5 1.16 1.38	37 A0 0.92 1.00	38 B8 1.03 1.22	39 A0 0.88 0.96	40 C8 1.23 1.42	41 A0 0.93 1.00	42 B5 1.18 1.34
43 C1 0.94 1.32	44 B6 1.12 1.32	45 A0 0.98 1.04	46 B5 1.19 1.38	47 A0 0.90 0.96	48 C8 1.23 1.42	49 A0 0.94 1.01	50 B6 1.17 1.29	51 A0 0.96 1.03
52 C1 0.99 1.35	53 B6 1.14 1.21	54 B6 1.19 1.33	55 A0 0.94 1.02	56 B8 1.07 1.23	57 A0 0.93 1.00	58 B6 1.17 1.29	59 A0 0.95 1.01	60 B6 1.17 1.29
61 B0 0.90 1.21	62 B6 1.15 1.33	63 A0 0.98 1.05	64 C8 1.27 1.46	65 A0 0.91 0.99	66 B5 1.18 1.34	67 A0 0.96 1.03	68 B6 1.17 1.29	69 A0 0.95 1.00

Figure 4.3-3

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

	MAX. VALUE	IN BOX
BOX RPD	1.26	25
MAX PIN	1.46	25

					1 C0 0.69 1.15	2 C1 0.94 1.32	3 C1 0.99 1.34	4 B0 0.89 1.20
			5 C0 0.62 1.03	6 C7 0.82 1.25	7 C7 1.09 1.33	8 B6 1.12 1.31	9 B6 1.13 1.21	10 B6 1.15 1.32
		11 C0 0.71 1.11	12 C7 1.02 1.31	13 A0 0.86 0.95	14 B5 1.16 1.38	15 A0 0.98 1.05	16 B6 1.18 1.32	17 A0 0.97 1.05
	18 C0 0.62 1.03	19 C7 1.02 1.31	20 A0 0.87 0.95	21 C8 1.21 1.41	22 A0 0.93 1.00	23 B5 1.19 1.38	24 A0 0.94 1.02	25 C8 1.26 1.45
	26 C7 0.82 1.25	27 A0 0.86 0.95	28 C8 1.21 1.41	29 A0 0.87 0.95	30 B8 1.04 1.22	31 A0 0.90 0.97	32 B8 1.07 1.23	33 A0 0.91 0.98
34 C0 0.69 1.15	35 C7 1.09 1.33	36 B5 1.16 1.38	37 A0 0.93 1.00	38 B8 1.04 1.22	39 A0 0.88 0.96	40 C8 1.23 1.41	41 A0 0.92 0.99	42 B5 1.17 1.33
43 C1 0.94 1.32	44 B6 1.12 1.31	45 A0 0.98 1.04	46 B5 1.19 1.38	47 A0 0.90 0.96	48 C8 1.23 1.41	49 A0 0.93 1.00	50 B6 1.15 1.28	51 A0 0.95 1.02
52 C1 0.99 1.34	53 B6 1.13 1.21	54 B6 1.18 1.32	55 A0 0.94 1.01	56 B8 1.07 1.23	57 A0 0.92 1.00	58 B6 1.15 1.28	59 A0 0.94 0.99	60 B6 1.15 1.27
61 B0 0.89 1.20	62 B6 1.15 1.32	63 A0 0.97 1.05	64 C8 1.26 1.45	65 A0 0.91 0.98	66 B5 1.17 1.33	67 A0 0.95 1.02	68 B6 1.15 1.27	69 A0 0.94 0.99

Figure 4.3-4

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

MAX. VALUE	IN BOX
1.28	25
1.46	40

					1 C0 0.65 1.10	2 C1 0.88 1.24	3 C1 0.93 1.26	4 B0 0.84 1.14
			5 C0 0.58 0.99	6 C7 0.79 1.20	7 C7 1.05 1.29	8 B6 1.08 1.27	9 B6 1.11 1.19	10 B6 1.12 1.29
		11 C0 0.67 1.07	12 C7 0.98 1.27	13 A0 0.84 0.95	14 B5 1.14 1.36	15 A0 0.98 1.05	16 B6 1.18 1.30	17 A0 0.98 1.06
	18 C0 0.58 0.99	19 C7 0.98 1.27	20 A0 0.86 0.94	21 C8 1.20 1.39	22 A0 0.94 1.02	23 B5 1.19 1.36	24 A0 0.96 1.05	25 C8 1.28 1.45
	26 C7 0.79 1.20	27 A0 0.84 0.95	28 C8 1.20 1.40	29 A0 0.89 0.96	30 B8 1.07 1.25	31 A0 0.93 1.01	32 B8 1.11 1.28	33 A0 0.95 1.03
34 C0 0.65 1.10	35 C7 1.05 1.28	36 B5 1.14 1.36	37 A0 0.94 1.01	38 B8 1.07 1.25	39 A0 0.92 1.01	40 C8 1.28 1.46	41 A0 0.97 1.04	42 B5 1.22 1.37
43 C1 0.88 1.24	44 B6 1.08 1.27	45 A0 0.98 1.04	46 B5 1.19 1.36	47 A0 0.93 1.00	48 C8 1.28 1.46	49 A0 0.98 1.05	50 B6 1.21 1.32	51 A0 1.00 1.07
52 C1 0.93 1.26	53 B6 1.11 1.19	54 B6 1.18 1.30	55 A0 0.96 1.03	56 B8 1.11 1.28	57 A0 0.97 1.05	58 B6 1.21 1.32	59 A0 0.99 1.05	60 B6 1.20 1.31
61 B0 0.84 1.14	62 B6 1.12 1.29	63 A0 0.98 1.06	64 C8 1.28 1.45	65 A0 0.95 1.03	66 B5 1.22 1.37	67 A0 1.00 1.07	68 B6 1.20 1.31	69 A0 0.99 1.05

Figure 4.3-5

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

	MAX. VALUE	IN BOX
BOX RPD	1.26	25
MAX PIN	1.46	25

					1 C0 0.67 1.14	2 C1 0.92 1.29	3 C1 0.97 1.32	4 B0 0.87 1.18
			5 C0 0.60 1.01	6 C7 0.81 1.23	7 C7 1.09 1.32	8 B6 1.12 1.31	9 B6 1.15 1.23	10 B6 1.1[illegible] 1.33
		11 C0 0.69 1.09	12 C7 0.99 1.28	13 A0 0.86 0.96	14 B5 1.17 1.38	15 A0 1.00 1.08	16 B6 1.21 1.33	17 A0 1.00 1.08
	18 C0 0.60 1.01	19 C7 0.99 1.27	20 A0 0.86 0.94	21 C8 1.18 1.40	22 A0 0.94 1.02	23 B5 1.20 1.38	24 A0 0.96 1.05	25 C8 1.26 1.46
	26 C7 0.81 1.23	27 A0 0.86 0.96	28 C8 1.18 1.40	29 A0 0.78 0.84	30 B8 1.04 1.24	31 A0 0.92 1.00	32 B8 1.07 1.25	33 A0 0.82 0.87
34 C0 0.67 1.14	35 C7 1.09 1.32	36 B5 1.17 1.38	37 A0 0.94 1.02	38 B8 1.04 1.24	39 A0 0.90 0.99	40 C8 1.26 1.44	41 A0 0.95 1.02	42 B5 1.17 1.34
43 C1 0.92 1.29	44 B6 1.12 1.31	45 A0 1.00 1.06	46 B5 1.20 1.38	47 A0 0.92 0.99	48 C8 1.26 1.44	49 A0 0.97 1.03	50 B6 1.18 1.29	51 A0 0.97 1.05
52 C1 0.97 1.32	53 B6 1.15 1.23	54 B6 1.21 1.33	55 A0 0.96 1.04	56 B8 1.07 1.26	57 A0 0.95 1.03	58 B6 1.18 1.29	59 A0 0.96 1.02	60 B6 1.14 1.27
61 B0 0.87 1.18	62 B6 1.16 1.33	63 A0 1.00 1.08	64 C8 1.26 1.46	65 A0 0.82 0.87	66 B5 1.17 1.34	67 A0 0.97 1.05	68 B6 1.14 1.27	69 A0 0.84 0.88

Figure 4.3-5a

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

MAX. VALUE	IN BOX
1.30	40
1.45	40

					1 C0 0.61 1.04	2 C1 0.81 1.15	3 C1 0.84 1.17	4 B0 0.76 1.04
			5 C0 0.58 1.00	6 C7 0.81 1.16	7 C7 1.05 1.28	8 B6 1.06 1.20	9 B6 1.09 1.19	10 B6 1.09 1.21
		11 C0 0.67 1.08	12 C7 1.00 1.29	13 A0 0.88 1.00	14 B5 1.13 1.27	15 A0 0.97 1.04	16 B6 1.16 1.25	17 A0 0.98 1.06
	18 C0 0.58 1.00	19 C7 1.00 1.28	20 A0 0.91 1.02	21 C8 1.25 1.40	22 A0 0.97 1.04	23 B5 1.17 1.27	24 A0 0.98 1.06	25 C8 1.29 1.43
	26 C7 0.81 1.16	27 A0 0.88 1.00	28 C8 1.25 1.41	29 A0 0.97 1.05	30 B8 1.14 1.27	31 A0 0.98 1.06	32 B8 1.15 1.28	33 A0 0.98 1.05
34 C0 0.61 1.04	35 C7 1.05 1.28	36 B5 1.13 1.27	37 A0 0.97 1.04	38 B8 1.14 1.28	39 A0 0.99 1.08	40 C8 1.30 1.45	41 A0 0.98 1.06	42 B5 1.17 1.27
43 C1 0.81 1.15	44 B6 1.06 1.20	45 A0 0.97 1.04	46 B5 1.17 1.27	47 A0 0.98 1.06	48 C8 1.30 1.45	49 A0 1.00 1.09	50 B6 1.17 1.26	51 A0 0.97 1.04
52 C1 0.84 1.17	53 B6 1.09 1.19	54 B6 1.16 1.25	55 A0 0.98 1.06	56 B8 1.15 1.28	57 A0 0.98 1.06	58 B6 1.17 1.25	59 A0 0.96 1.03	60 B6 1.14 1.22
61 B0 0.76 1.04	62 B6 1.09 1.21	63 A0 0.98 1.06	64 C8 1.29 1.43	65 A0 0.98 1.05	66 B5 1.17 1.27	67 A0 0.97 1.04	68 B6 1.14 1.22	69 A0 0.95 1.01

Figure 4.3-6

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

	MAX. VALUE	IN BOX
BOX RPD	1.29	40
MAX PIN	1.44	25

					1 C0 0.63 1.09	2 C1 0.84 1.19	3 C1 0.88 1.22	4 B0 0.79 1.09
			5 C0 0.60 1.02	6 C7 0.83 1.19	7 C7 1.08 1.32	8 B6 1.11 1.24	9 B6 1.13 1.24	10 B6 1.13 1.25
		11 C0 0.68 1.09	12 C7 1.01 1.29	13 A0 0.89 1.01	14 B5 1.15 1.28	15 A0 0.99 1.06	16 B6 1.19 1.27	17 A0 1.01 1.09
	18 C0 0.60 1.02	19 C7 1.01 1.29	20 A0 0.90 1.00	21 C8 1.22 1.39	22 A0 0.96 1.04	23 B5 1.18 1.29	24 A0 0.98 1.06	25 C8 1.27 1.44
	26 C7 0.83 1.19	27 A0 0.89 1.01	28 C8 1.22 1.39	29 A0 0.85 0.92	30 B8 1.11 1.26	31 A0 0.97 1.05	32 B8 1.11 1.27	33 A0 0.85 0.92
34 C0 0.63 1.09	35 C7 1.08 1.32	36 B5 1.15 1.28	37 A0 0.96 1.04	38 B8 1.11 1.26	39 A0 0.97 1.07	40 C8 1.29 1.44	41 A0 0.96 1.04	42 B5 1.13 1.24
43 C1 0.84 1.20	44 B6 1.11 1.24	45 A0 0.99 1.06	46 B5 1.18 1.29	47 A0 0.97 1.04	48 C8 1.29 1.43	49 A0 0.99 1.08	50 B6 1.14 1.24	51 A0 0.94 1.01
52 C1 0.88 1.22	53 B6 1.13 1.23	54 B6 1.19 1.27	55 A0 0.98 1.06	56 B8 1.11 1.27	57 A0 0.96 1.04	58 B6 1.14 1.24	59 A0 0.93 1.01	60 B6 1.09 1.19
61 B0 0.79 1.09	62 B6 1.13 1.25	63 A0 1.01 1.09	64 C8 1.27 1.44	65 A0 0.85 0.92	66 B5 1.13 1.24	67 A0 0.94 1.01	68 B6 1.09 1.19	69 A0 0.81 0.87

Figure 4.3-6a

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

MAX. VALUE	IN BOX
1.28	40
1.42	40

					1 C0 0.61 1.05	2 C1 0.80 1.13	3 C1 0.83 1.15	4 B0 0.75 1.02
			5 C0 0.61 1.03	6 C7 0.84 1.19	7 C7 1.06 1.29	8 B6 1.07 1.18	9 B6 1.08 1.19	10 B6 1.08 1.19
		11 C0 0.69 1.11	12 C7 1.03 1.31	13 A0 0.90 1.02	14 B5 1.13 1.24	15 A0 0.97 1.04	16 B6 1.15 1.23	17 A0 0.98 1.06
	18 C0 0.61 1.03	19 C7 1.03 1.31	20 A0 0.94 1.05	21 C8 1.26 1.41	22 A0 0.98 1.06	23 B5 1.15 1.25	24 A0 0.98 1.06	25 C8 1.27 1.41
	26 C7 0.84 1.19	27 A0 0.90 1.02	28 C8 1.26 1.41	29 A0 0.99 1.08	30 B8 1.15 1.28	31 A0 0.98 1.06	32 B8 1.14 1.26	33 A0 0.97 1.05
34 C0 0.61 1.05	35 C7 1.06 1.28	36 B5 1.13 1.24	37 A0 0.98 1.06	38 B8 1.15 1.28	39 A0 1.00 1.09	40 C8 1.28 1.42	41 A0 0.97 1.05	42 B5 1.14 1.23
43 C1 0.80 1.13	44 B6 1.07 1.19	45 A0 0.97 1.04	46 B5 1.15 1.25	47 A0 0.98 1.06	48 C8 1.28 1.42	49 A0 0.99 1.08	50 B6 1.13 1.22	51 A0 0.94 1.02
52 C1 0.83 1.15	53 B6 1.08 1.19	54 B6 1.15 1.23	55 A0 0.98 1.06	56 B8 1.14 1.26	57 A0 0.97 1.05	58 B6 1.13 1.22	59 A0 0.94 1.01	60 B6 1.10 1.18
61 B0 0.75 1.02	62 B6 1.08 1.19	63 A0 0.98 1.06	64 C8 1.27 1.41	65 A0 0.97 1.05	66 B5 1.14 1.23	67 A0 0.94 1.02	68 B6 1.10 1.18	69 A0 0.92 0.99

Figure 4.3-7

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

	MAX. VALUE	IN BOX
BOX RPD	1.27	40
MAX PIN	1.41	25

					1 C0 0.64 1.09	2 C1 0.84 1.18	3 C1 0.87 1.20	4 B0 0.78 1.07
			5 C0 0.62 1.05	6 C7 0.86 1.22	7 C7 1.10 1.33	8 B6 1.11 1.23	9 B6 1.13 1.24	10 B6 1.13 1.24
		11 C0 0.70 1.13	12 C7 1.04 1.32	13 A0 0.92 1.03	14 B5 1.15 1.25	15 A0 0.99 1.07	16 B6 1.18 1.25	17 A0 1.00 1.08
	18 C0 0.62 1.05	19 C7 1.04 1.31	20 A0 0.94 1.04	21 C8 1.24 1.39	22 A0 0.97 1.05	23 B5 1.16 1.26	24 A0 0.97 1.06	25 C8 1.25 1.41
	26 C7 0.86 1.22	27 A0 0.92 1.03	28 C8 1.24 1.39	29 A0 0.86 0.94	30 B8 1.12 1.26	31 A0 0.97 1.05	32 B8 1.11 1.25	33 A0 0.84 0.92
34 C0 0.64 1.09	35 C7 1.10 1.32	36 B5 1.15 1.25	37 A0 0.97 1.05	38 B8 1.12 1.26	39 A0 0.99 1.07	40 C8 1.27 1.41	41 A0 0.95 1.03	42 B5 1.09 1.18
43 C1 0.84 1.18	44 B6 1.11 1.23	45 A0 0.99 1.07	46 B5 1.16 1.26	47 A0 0.97 1.05	48 C8 1.27 1.41	49 A0 0.97 1.07	50 B6 1.11 1.21	51 A0 0.92 0.99
52 C1 0.87 1.20	53 B6 1.13 1.23	54 B6 1.18 1.25	55 A0 0.97 1.06	56 B8 1.11 1.25	57 A0 0.95 1.03	58 B6 1.11 1.21	59 A0 0.91 0.99	60 B6 1.05 1.14
61 B0 0.78 1.07	62 B6 1.13 1.24	63 A0 1.00 1.08	64 C8 1.25 1.41	65 A0 0.84 0.92	66 B5 1.09 1.18	67 A0 0.92 0.99	68 B6 1.05 1.14	69 A0 0.78 0.84

Figure 4.3-7a

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

MAX. VALUE	IN BOX
1.25	21
1.38	28

					1 C0 0.64 1.06	2 C1 0.81 1.12	3 C1 0.82 1.12	4 B0 0.75 1.00
			5 C0 0.65 1.09	6 C7 0.90 1.22	7 C7 1.08 1.26	8 B6 1.07 1.17	9 B6 1.07 1.17	10 B6 1.07 1.16
		11 C0 0.73 1.15	12 C7 1.07 1.30	13 A0 0.95 1.06	14 B5 1.12 1.20	15 A0 0.97 1.05	16 B6 1.12 1.18	17 A0 0.98 1.07
	18 C0 0.65 1.09	19 C7 1.07 1.30	20 A0 0.98 1.10	21 C8 1.25 1.37	22 A0 0.99 1.08	23 B5 1.12 1.20	24 A0 0.98 1.06	25 C8 1.22 1.34
	26 C7 0.90 1.22	27 A0 0.95 1.06	28 C8 1.25 1.38	29 A0 1.01 1.10	30 B8 1.14 1.25	31 A0 0.98 1.07	32 B8 1.12 1.22	33 A0 0.97 1.06
34 C0 0.64 1.06	35 C7 1.08 1.26	36 B5 1.12 1.21	37 A0 0.99 1.08	38 B8 1.14 1.26	39 A0 1.00 1.09	40 C8 1.23 1.36	41 A0 0.97 1.05	42 B5 1.10 1.18
43 C1 0.81 1.12	44 B6 1.07 1.17	45 A0 0.97 1.05	46 B5 1.12 1.20	47 A0 0.98 1.07	48 C8 1.23 1.36	49 A0 0.98 1.08	50 B6 1.10 1.17	51 A0 0.94 1.02
52 C1 0.82 1.12	53 B6 1.07 1.16	54 B6 1.12 1.18	55 A0 0.98 1.06	56 B8 1.12 1.22	57 A0 0.97 1.06	58 B6 1.10 1.17	59 A0 0.94 1.02	60 B6 1.08 1.14
61 B0 0.75 1.00	62 B6 1.07 1.16	63 A0 0.98 1.07	64 C8 1.22 1.34	65 A0 0.97 1.06	66 B5 1.10 1.18	67 A0 0.94 1.02	68 B6 1.08 1.14	69 A0 0.93 1.00

Figure 4.3-8

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

	MAX. VALUE	IN BOX
BOX RPD	1.38	40
MAX PIN	1.59	40

					1 C0 0.61 1.04	2 C1 0.81 1.12	3 C1 0.81 1.11	4 B0 0.71 0.96
			5 C0 0.60 1.02	6 C7 0.81 1.21	7 C7 1.03 1.26	8 B6 1.02 1.20	9 B6 0.96 1.06	10 B6 0.90 0.96
		11 C0 0.70 1.12	12 C7 1.02 1.32	13 A0 0.88 1.00	14 B5 1.15 1.34	15 A0 0.94 1.02	16 B6 1.00 1.19	17 A0 0.59 0.65
	18 C0 0.60 1.02	19 C7 1.02 1.32	20 A0 0.92 1.02	21 C8 1.27 1.44	22 A0 0.98 1.06	23 B5 1.19 1.34	24 A0 0.91 1.00	25 C8 1.13 1.36
	26 C7 0.81 1.21	27 A0 0.88 1.00	28 C8 1.27 1.44	29 A0 0.97 1.05	30 B8 1.15 1.30	31 A0 0.99 1.08	32 B8 1.15 1.34	33 A0 0.97 1.06
34 C0 0.61 1.04	35 C7 1.03 1.26	36 B5 1.15 1.35	37 A0 0.98 1.05	38 B8 1.15 1.31	39 A0 1.01 1.11	40 C8 1.38 1.59	41 A0 1.06 1.14	42 B5 1.30 1.48
43 C1 0.81 1.12	44 B6 1.02 1.20	45 A0 0.94 1.02	46 B5 1.19 1.35	47 A0 0.99 1.07	48 C8 1.38 1.59	49 A0 1.09 1.17	50 B6 1.33 1.44	51 A0 1.11 1.19
52 C1 0.81 1.12	53 B6 0.96 1.06	54 B6 1.00 1.20	55 A0 0.91 1.00	56 B8 1.15 1.35	57 A0 1.06 1.15	58 B6 1.33 1.44	59 A0 1.11 1.18	60 B6 1.34 1.45
61 B0 0.71 0.96	62 B6 0.90 0.96	63 A0 0.59 0.65	64 C8 1.13 1.36	65 A0 0.97 1.06	66 B5 1.30 1.48	67 A0 1.11 1.19	68 B6 1.34 1.45	69 A0 1.12 1.19

Figure 4.3-9

FORMAT IS:

BOX NO.	FUEL TYPE
BOX	RPD
MAX	PIN

MAX. VALUE	IN BOX
1.37	40
1.58	40

					1 C0 0.64 1.09	2 C1 0.84 1.18	3 C1 0.85 1.17	4 B0 0.74 1.01
			5 C0 0.62 1.05	6 C7 0.83 1.24	7 C7 1.07 1.30	8 B6 1.06 1.24	9 B6 1.00 1.10	10 B6 0.94 1.00
		11 C0 0.71 1.14	12 C7 1.04 1.33	13 A0 0.90 1.01	14 B5 1.18 1.36	15 A0 0.96 1.04	16 B6 1.03 1.21	17 A0 0.60 0.66
	18 C0 0.62 1.05	19 C7 1.04 1.33	20 A0 0.92 1.01	21 C8 1.25 1.45	22 A0 0.98 1.06	23 B5 1.20 1.36	24 A0 0.90 0.99	25 C8 1.10 1.30
	26 C7 0.83 1.25	27 A0 0.90 1.01	28 C8 1.25 1.45	29 A0 0.85 0.92	30 B8 1.12 1.30	31 A0 0.98 1.07	32 B8 1.11 1.29	33 A0 0.84 0.92
34 C0 0.64 1.09	35 C7 1.07 1.30	36 B5 1.18 1.37	37 A0 0.98 1.06	38 B8 1.12 1.30	39 A0 0.99 1.10	40 C8 1.37 1.58	41 A0 1.03 1.12	42 B5 1.25 1.44
43 C1 0.84 1.18	44 B6 1.06 1.24	45 A0 0.96 1.04	46 B5 1.20 1.36	47 A0 0.98 1.06	48 C8 1.37 1.57	49 A0 1.08 1.16	50 B6 1.30 1.42	51 A0 1.08 1.16
52 C1 0.85 1.17	53 B6 1.00 1.10	54 B6 1.03 1.21	55 A0 0.90 0.99	56 B8 1.11 1.29	57 A0 1.03 1.12	58 B6 1.30 1.41	59 A0 1.08 1.15	60 B6 1.28 1.40
61 B0 0.74 1.01	62 B6 0.94 1.00	63 A0 0.60 0.66	64 C8 1.10 1.30	65 A0 0.84 0.92	66 B5 1.25 1.44	67 A0 1.08 1.16	68 B6 1.28 1.40	69 A0 0.96 1.01

Figure 4.3-9a

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

MAX. VALUE	IN BOX
1.35	40
1.51	40

					1 C0 0.61 1.04	2 C1 0.77 1.06	3 C1 0.76 1.06	4 B0 0.66 0.90
			5 C0 0.65 1.09	6 C7 0.88 1.23	7 C7 1.08 1.28	8 B6 1.03 1.16	9 B6 0.96 1.08	10 B6 0.89 0.96
		11 C0 0.74 1.19	12 C7 1.10 1.39	13 A0 0.95 1.07	14 B5 1.16 1.28	15 A0 0.94 1.03	16 B6 0.99 1.15	17 A0 0.59 0.66
	18 C0 0.65 1.09	19 C7 1.10 1.39	20 A0 1.00 1.12	21 C8 1.34 1.49	22 A0 1.02 1.11	23 B5 1.15 1.28	24 A0 0.91 1.01	25 C8 1.11 1.33
	26 C7 0.88 1.23	27 A0 0.95 1.07	28 C8 1.34 1.49	29 A0 1.05 1.14	30 B8 1.21 1.35	31 A0 1.01 1.11	32 B8 1.15 1.30	33 A0 0.96 1.05
34 C0 0.61 1.03	35 C7 1.08 1.28	36 B5 1.16 1.28	37 A0 1.02 1.11	38 B8 1.21 1.35	39 A0 1.05 1.15	40 C8 1.35 1.51	41 A0 1.02 1.11	42 B5 1.19 1.29
43 C1 0.77 1.07	44 B6 1.03 1.16	45 A0 0.94 1.03	46 B5 1.15 1.29	47 A0 1.01 1.11	48 C8 1.35 1.51	49 A0 1.05 1.15	50 B6 1.21 1.30	51 A0 1.02 1.09
52 C1 0.76 1.06	53 B6 0.96 1.08	54 B6 0.99 1.15	55 A0 0.91 1.01	56 B8 1.15 1.31	57 A0 1.02 1.11	58 B6 1.21 1.30	59 A0 1.02 1.09	60 B6 1.19 1.27
61 B0 0.66 0.90	62 B6 0.89 0.96	63 A0 0.59 0.66	64 C8 1.11 1.33	65 A0 0.96 1.05	66 B5 1.19 1.29	67 A0 1.02 1.09	68 B6 1.19 1.27	69 A0 1.01 1.08

Figure 4.3-10

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

	MAX. VALUE	IN BOX
BOX RPD	1.34	40
MAX PIN	1.50	40

					1 C0 0.64 1.08	2 C1 0.81 1.11	3 C1 0.80 1.11	4 B0 0.70 0.94
			5 C0 0.66 1.12	6 C7 0.90 1.26	7 C7 1.11 1.32	8 B6 1.07 1.19	9 B6 1.00 1.12	10 B6 0.93 1.00
		11 C0 0.75 1.21	12 C7 1.12 1.40	13 A0 0.97 1.08	14 B5 1.18 1.29	15 A0 0.96 1.05	16 B6 1.01 1.16	17 A0 0.60 0.67
	18 C0 0.66 1.12	19 C7 1.12 1.40	20 A0 1.00 1.11	21 C8 1.31 1.46	22 A0 1.01 1.10	23 B5 1.15 1.27	24 A0 0.90 1.01	25 C8 1.08 1.26
	26 C7 0.90 1.27	27 A0 0.97 1.08	28 C8 1.31 1.46	29 A0 0.91 1.00	30 B8 1.18 1.32	31 A0 1.00 1.10	32 B8 1.11 1.29	33 A0 0.83 0.91
34 C0 0.64 1.08	35 C7 1.11 1.32	36 B5 1.18 1.29	37 A0 1.01 1.09	38 B8 1.18 1.32	39 A0 1.04 1.14	40 C8 1.34 1.50	41 A0 1.00 1.10	42 B5 1.15 1.26
43 C1 0.81 1.11	44 B6 1.07 1.20	45 A0 0.96 1.05	46 B5 1.15 1.28	47 A0 1.00 1.10	48 C8 1.34 1.49	49 A0 1.04 1.14	50 B6 1.19 1.29	51 A0 0.99 1.07
52 C1 0.80 1.11	53 B6 1.00 1.12	54 B6 1.01 1.17	55 A0 0.90 1.00	56 B8 1.11 1.29	57 A0 1.00 1.10	58 B6 1.19 1.29	59 A0 0.99 1.07	60 B6 1.14 1.24
61 B0 0.70 0.94	62 B6 0.93 1.00	63 A0 0.60 0.67	64 C8 1.08 1.26	65 A0 0.83 0.91	66 B5 1.15 1.26	67 A0 0.99 1.07	68 B6 1.14 1.24	69 A0 0.86 0.93

Figure 4.3-10a

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

MAX. VALUE	IN BOX
1.33	21
1.47	28

					1 C0 0.63 1.05	2 C1 0.78 1.06	3 C1 0.76 1.05	4 B0 0.67 0.89
			5 C0 0.68 1.14	6 C7 0.92 1.26	7 C7 1.10 1.28	8 B6 1.03 1.15	9 B6 0.95 1.07	10 B6 0.88 0.95
		11 C0 0.77 1.22	12 C7 1.13 1.40	13 A0 0.99 1.10	14 B5 1.15 1.26	15 A0 0.94 1.04	16 B6 0.97 1.12	17 A0 0.58 0.66
	18 C0 0.68 1.14	19 C7 1.13 1.40	20 A0 1.04 1.16	21 C8 1.33 1.47	22 A0 1.02 1.13	23 B5 1.12 1.25	24 A0 0.91 1.01	25 C8 1.08 1.29
	26 C7 0.92 1.26	27 A0 0.99 1.10	28 C8 1.33 1.47	29 A0 1.06 1.17	30 B8 1.20 1.33	31 A0 1.01 1.12	32 B8 1.13 1.27	33 A0 0.96 1.05
34 C0 0.63 1.05	35 C7 1.10 1.28	36 B5 1.15 1.27	37 A0 1.02 1.12	38 B8 1.20 1.33	39 A0 1.06 1.15	40 C8 1.31 1.45	41 A0 1.01 1.11	42 B5 1.15 1.24
43 C1 0.78 1.06	44 B6 1.03 1.15	45 A0 0.94 1.04	46 B5 1.12 1.25	47 A0 1.01 1.11	48 C8 1.31 1.45	49 A0 1.04 1.15	50 B6 1.18 1.26	51 A0 1.01 1.08
52 C1 0.76 1.06	53 B6 0.95 1.07	54 B6 0.97 1.12	55 A0 0.91 1.01	56 B8 1.13 1.27	57 A0 1.01 1.11	58 B6 1.18 1.25	59 A0 1.01 1.09	60 B6 1.16 1.23
61 B0 0.67 0.89	62 B6 0.88 0.95	63 A0 0.58 0.66	64 C8 1.08 1.29	65 A0 0.96 1.05	66 B5 1.15 1.24	67 A0 1.01 1.08	68 B6 1.16 1.23	69 A0 1.00 1.08

Figure 4.3-11

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

MAX. VALUE	IN BOX
1.31	21
1.44	28

					1 C0 0.66 1.09	2 C1 0.82 1.11	3 C1 0.80 1.10	4 B0 0.70 0.94
			5 C0 0.71 1.17	6 C7 0.95 1.29	7 C7 1.14 1.32	8 B6 1.07 1.20	9 B6 1.00 1.11	10 B6 0.92 0.99
		11 C0 0.79 1.24	12 C7 1.16 1.41	13 A0 1.01 1.12	14 B5 1.18 1.27	15 A0 0.97 1.07	16 B6 0.99 1.13	17 A0 0.60 0.67
	18 C0 0.71 1.17	19 C7 1.16 1.41	20 A0 1.04 1.14	21 C8 1.31 1.43	22 A0 1.02 1.11	23 B5 1.13 1.24	24 A0 0.90 1.01	25 C8 1.05 1.21
	26 C7 0.95 1.30	27 A0 1.01 1.12	28 C8 1.31 1.44	29 A0 0.92 1.02	30 B8 1.16 1.29	31 A0 1.00 1.10	32 B8 1.09 1.25	33 A0 0.82 0.90
34 C0 0.66 1.09	35 C7 1.14 1.32	36 B5 1.18 1.27	37 A0 1.02 1.11	38 B8 1.16 1.29	39 A0 1.03 1.14	40 C8 1.30 1.44	41 A0 0.99 1.09	42 B5 1.11 1.21
43 C1 0.82 1.11	44 B6 1.07 1.20	45 A0 0.97 1.07	46 B5 1.13 1.24	47 A0 1.00 1.10	48 C8 1.30 1.43	49 A0 1.03 1.14	50 B6 1.16 1.24	51 A0 0.98 1.06
52 C1 0.80 1.11	53 B6 1.00 1.11	54 B6 0.99 1.14	55 A0 0.90 1.00	56 B8 1.09 1.25	57 A0 0.99 1.09	58 B6 1.16 1.24	59 A0 0.98 1.07	60 B5 1.11 1.20
61 B0 0.70 0.94	62 B6 0.92 0.99	63 A0 0.60 0.67	64 C8 1.05 1.21	65 A0 0.82 0.90	66 B5 1.11 1.21	67 A0 0.98 1.06	68 B6 1.11 1.20	69 A0 0.85 0.92

Figure 4.3-11a

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

	MAX. VALUE	IN BOX
BOX RPD	1.30	25
MAX PIN	1.48	25

					1 C0 0.65 1.11	2 C1 0.89 1.26	3 C1 0.94 1.30	4 B0 0.86 1.17
			5 C0 0.58 1.00	6 C7 0.81 1.21	7 C7 1.07 1.28	8 B6 1.09 1.23	9 B6 1.14 1.23	10 B6 1.16 1.32
		11 C0 0.59 0.98	12 C7 0.97 1.27	13 A0 0.86 0.98	14 B5 1.14 1.31	15 A0 0.89 0.95	16 B6 1.19 1.32	17 A0 1.02 1.10
	18 C0 0.58 1.00	19 C7 0.97 1.27	20 A0 0.87 0.96	21 C8 1.21 1.40	22 A0 0.95 1.03	23 B5 1.19 1.32	24 A0 0.98 1.07	25 C8 1.30 1.48
	26 C7 0.81 1.21	27 A0 0.86 0.98	28 C8 1.21 1.41	29 A0 0.82 0.88	30 B8 1.08 1.27	31 A0 0.96 1.04	32 B8 1.12 1.29	33 A0 0.86 0.92
34 C0 0.65 1.11	35 C7 1.07 1.27	36 B5 1.14 1.32	37 A0 0.95 1.03	38 B8 1.08 1.27	39 A0 0.94 1.03	40 C8 1.28 1.43	41 A0 0.98 1.05	42 B5 1.20 1.36
43 C1 0.89 1.26	44 B6 1.09 1.23	45 A0 0.89 0.95	46 B5 1.19 1.32	47 A0 0.96 1.03	48 C8 1.28 1.43	49 A0 0.89 0.95	50 B6 1.19 1.31	51 A0 1.01 1.08
52 C1 0.94 1.30	53 B6 1.14 1.22	54 B6 1.19 1.32	55 A0 0.98 1.06	56 B8 1.12 1.29	57 A0 0.98 1.06	58 B6 1.19 1.31	59 A0 0.99 1.05	60 B6 1.18 1.30
61 B0 0.86 1.17	62 B6 1.16 1.32	63 A0 1.02 1.10	64 C8 1.30 1.48	65 A0 0.86 0.92	66 B5 1.20 1.36	67 A0 1.01 1.08	68 B6 1.18 1.30	69 A0 0.88 0.93

Figure 4.3-12

FORMAT IS:

BOX NO.	FUEL TYPE
BOX	RPD
MAX	PIN

MAX. VALUE	IN BOX
1.27	25
1.43	25

					1 C0 0.65 1.10	2 C1 0.85 1.19	3 C1 0.88 1.22	4 B0 0.80 1.09
			5 C0 0.63 1.07	6 C7 0.88 1.23	7 C7 1.10 1.29	8 B6 1.10 1.20	9 B6 1.14 1.25	10 B6 1.15 1.25
		11 C0 0.62 1.04	12 C7 1.05 1.34	13 A0 0.93 1.05	14 B5 1.14 1.25	15 A0 0.88 0.95	16 B6 1.17 1.26	17 A0 1.02 1.10
	18 C0 0.63 1.07	19 C7 1.05 1.34	20 A0 0.95 1.06	21 C8 1.26 1.41	22 A0 0.98 1.06	23 B5 1.15 1.25	24 A0 0.98 1.07	25 C8 1.27 1.43
	26 C7 0.88 1.23	27 A0 0.93 1.05	28 C8 1.26 1.41	29 A0 0.88 0.97	30 B8 1.14 1.27	31 A0 0.98 1.06	32 B8 1.12 1.26	33 A0 0.86 0.94
34 C0 0.65 1.10	35 C7 1.10 1.29	36 B5 1.14 1.25	37 A0 0.98 1.06	38 B8 1.14 1.27	39 A0 0.98 1.07	40 C8 1.26 1.42	41 A0 0.95 1.04	42 B5 1.11 1.20
43 C1 0.85 1.20	44 B6 1.10 1.20	45 A0 0.88 0.95	46 B5 1.15 1.25	47 A0 0.98 1.06	48 C8 1.26 1.41	49 A0 0.86 0.95	50 B6 1.10 1.19	51 A0 0.93 1.00
52 C1 0.88 1.22	53 B6 1.14 1.25	54 B6 1.17 1.26	55 A0 0.98 1.07	56 B8 1.12 1.27	57 A0 0.95 1.04	58 B6 1.10 1.18	59 A0 0.91 0.98	60 B6 1.06 1.15
61 B0 0.80 1.09	62 B6 1.15 1.25	63 A0 1.02 1.10	64 C8 1.27 1.43	65 A0 0.86 0.94	66 B5 1.11 1.20	67 A0 0.93 1.00	68 B6 1.06 1.15	69 A0 0.80 0.86

Figure 4.3-13

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

MAX. VALUE	IN BOX
1.26	21
1.39	28

					1 C0 0.67 1.11	2 C1 0.86 1.19	3 C1 0.88 1.21	4 B0 0.80 1.08
			5 C0 0.67 1.12	6 C7 0.93 1.26	7 C7 1.12 1.29	8 B6 1.10 1.18	9 B6 1.13 1.23	10 B6 1.14 1.24
		11 C0 0.64 1.07	12 C7 1.08 1.35	13 A0 0.97 1.08	14 B5 1.14 1.23	15 A0 0.88 0.96	16 B6 1.15 1.23	17 A0 1.01 1.10
	18 C0 0.67 1.12	19 C7 1.08 1.35	20 A0 0.99 1.10	21 C8 1.26 1.39	22 A0 0.99 1.08	23 B5 1.12 1.22	24 A0 0.98 1.07	25 C8 1.23 1.38
	26 C7 0.93 1.26	27 A0 0.97 1.08	28 C8 1.26 1.39	29 A0 0.89 0.98	30 B8 1.13 1.24	31 A0 0.98 1.07	32 B8 1.10 1.23	33 A0 0.85 0.93
34 C0 0.67 1.11	35 C7 1.12 1.29	36 B5 1.14 1.24	37 A0 0.99 1.08	38 B8 1.13 1.25	39 A0 0.99 1.07	40 C8 1.22 1.37	41 A0 0.95 1.04	42 B5 1.07 1.15
43 C1 0.86 1.19	44 B6 1.10 1.19	45 A0 0.88 0.95	46 B5 1.12 1.22	47 A0 0.98 1.07	48 C8 1.22 1.37	49 A0 0.85 0.94	50 B6 1.06 1.15	51 A0 0.92 1.00
52 C1 0.88 1.21	53 B6 1.13 1.23	54 B6 1.15 1.23	55 A0 0.98 1.07	56 B8 1.10 1.24	57 A0 0.95 1.04	58 B6 1.06 1.14	59 A0 0.90 0.97	60 B6 1.04 1.12
61 B0 0.80 1.08	62 B6 1.14 1.24	63 A0 1.01 1.10	64 C8 1.23 1.38	65 A0 0.85 0.93	66 B5 1.07 1.15	67 A0 0.92 1.00	68 B6 1.04 1.12	69 A0 0.79 0.86

Figure 4.3-14

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

MAX. VALUE	IN BOX
1.36	40
1.53	40

					1 C0 0.65 1.10	2 C1 0.86 1.19	3 C1 0.86 1.19	4 B0 0.76 1.03
			5 C0 0.62 1.06	6 C7 0.84 1.26	7 C7 1.08 1.28	8 B6 1.05 1.19	9 B6 1.01 1.09	10 B6 0.96 1.02
		11 C0 0.63 1.05	12 C7 1.05 1.35	13 A0 0.91 1.03	14 B5 1.17 1.33	15 A0 0.86 0.94	16 B6 1.01 1.18	17 A0 0.60 0.66
	18 C0 0.62 1.06	19 C7 1.05 1.35	20 A0 0.93 1.03	21 C8 1.28 1.47	22 A0 0.99 1.07	23 B5 1.19 1.34	24 A0 0.91 1.00	25 C8 1.12 1.32
	26 C7 0.84 1.26	27 A0 0.91 1.02	28 C8 1.28 1.47	29 A0 0.87 0.94	30 B8 1.14 1.31	31 A0 0.99 1.08	32 B8 1.13 1.30	33 A0 0.86 0.94
34 C0 0.65 1.10	35 C7 1.08 1.28	36 B5 1.17 1.33	37 A0 0.99 1.06	38 B8 1.14 1.31	39 A0 1.00 1.10	40 C8 1.36 1.53	41 A0 1.04 1.13	42 B5 1.27 1.46
43 C1 0.86 1.19	44 B6 1.05 1.19	45 A0 0.86 0.94	46 B5 1.19 1.34	47 A0 0.99 1.07	48 C8 1.36 1.53	49 A0 0.96 1.03	50 B6 1.29 1.42	51 A0 1.09 1.18
52 C1 0.86 1.19	53 B6 1.01 1.08	54 B6 1.01 1.18	55 A0 0.91 1.00	56 B8 1.13 1.31	57 A0 1.04 1.13	58 B6 1.29 1.42	59 A0 1.09 1.16	60 B6 1.30 1.42
61 B0 0.76 1.03	62 B6 0.96 1.02	63 A0 0.60 0.66	64 C8 1.12 1.32	65 A0 0.86 0.94	66 B5 1.27 1.46	67 A0 1.09 1.18	68 B6 1.30 1.42	69 A0 0.98 1.03

Figure 4.3-15

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

MAX. VALUE	IN BOX
1.34	21
1.49	40

					1 C0 0.65 1.09	2 C1 0.82 1.13	3 C1 0.81 1.12	4 B0 0.71 0.96
			5 C0 0.67 1.14	6 C7 0.92 1.28	7 C7 1.12 1.29	8 B6 1.06 1.16	9 B6 1.01 1.11	10 B6 0.94 1.02
		11 C0 0.67 1.11	12 C7 1.12 1.43	13 A0 0.98 1.10	14 B5 1.17 1.31	15 A0 0.85 0.94	16 B6 1.00 1.13	17 A0 0.61 0.67
	18 C0 0.67 1.14	19 C7 1.12 1.42	20 A0 1.02 1.14	21 C8 1.34 1.48	22 A0 1.02 1.11	23 B5 1.15 1.29	24 A0 0.91 1.01	25 C8 1.10 1.29
	26 C7 0.92 1.28	27 A0 0.98 1.10	28 C8 1.34 1.48	29 A0 0.94 1.03	30 B8 1.20 1.33	31 A0 1.02 1.11	32 B8 1.13 1.30	33 A0 0.85 0.93
34 C0 0.65 1.09	35 C7 1.12 1.29	36 B5 1.17 1.31	37 A0 1.02 1.11	38 B8 1.20 1.34	39 A0 1.04 1.14	40 C8 1.33 1.49	41 A0 1.00 1.09	42 B5 1.16 1.27
43 C1 0.82 1.13	44 B6 1.06 1.17	45 A0 0.85 0.94	46 B5 1.15 1.29	47 A0 1.02 1.11	48 C8 1.33 1.49	49 A0 0.92 1.02	50 B6 1.18 1.26	51 A0 1.00 1.08
52 C1 0.81 1.12	53 B6 1.01 1.11	54 B6 1.00 1.13	55 A0 0.91 1.01	56 B8 1.13 1.30	57 A0 1.00 1.09	58 B6 1.18 1.26	59 A0 0.99 1.06	60 B6 1.16 1.25
61 B0 0.71 0.96	62 B6 0.94 1.02	63 A0 0.61 0.67	64 C8 1.10 1.29	65 A0 0.85 0.93	66 B5 1.16 1.27	67 A0 1.00 1.08	68 B6 1.16 1.25	69 A0 0.88 0.94

Figure 4.3-16

FORMAT IS:

BOX NO.	FUEL TYPE
BOX RPD	
MAX PIN	

	MAX. VALUE	IN BOX
BOX RPD	1.34	21
MAX PIN	1.46	28

					1 C0 0.67 1.11	2 C1 0.83 1.12	3 C1 0.81 1.12	4 B0 0.72 0.96
			5 C0 0.72 1.19	6 C7 0.97 1.31	7 C7 1.14 1.31	8 B6 1.06 1.16	9 B6 1.00 1.10	10 B6 0.94 1.01
		11 C0 0.69 1.14	12 C7 1.16 1.44	13 A0 1.02 1.14	14 B5 1.17 1.29	15 A0 0.85 0.95	16 B6 0.97 1.09	17 A0 0.60 0.67
	18 C0 0.72 1.19	19 C7 1.16 1.44	20 A0 1.05 1.17	21 C8 1.34 1.46	22 A0 1.03 1.13	23 B5 1.12 1.25	24 A0 0.90 1.01	25 C8 1.06 1.24
	26 C7 0.97 1.31	27 A0 1.02 1.14	28 C8 1.34 1.46	29 A0 0.94 1.05	30 B8 1.19 1.30	31 A0 1.01 1.12	32 B8 1.11 1.26	33 A0 0.84 0.92
34 C0 0.67 1.10	35 C7 1.14 1.31	36 B5 1.17 1.29	37 A0 1.03 1.13	38 B8 1.19 1.31	39 A0 1.04 1.14	40 C8 1.29 1.44	41 A0 0.99 1.08	42 B5 1.12 1.22
43 C1 0.83 1.12	44 B6 1.06 1.16	45 A0 0.85 0.95	46 B5 1.12 1.26	47 A0 1.01 1.12	48 C8 1.29 1.44	49 A0 0.91 1.01	50 B6 1.14 1.22	51 A0 0.99 1.07
52 C1 0.81 1.12	53 B6 1.00 1.10	54 B6 0.97 1.09	55 A0 0.90 1.01	56 B8 1.11 1.27	57 A0 0.99 1.08	58 B6 1.14 1.22	59 A0 0.98 1.06	60 B6 1.13 1.21
61 B0 0.72 0.96	62 B6 0.94 1.01	63 A0 0.60 0.67	64 C8 1.06 1.24	65 A0 0.84 0.92	66 B5 1.12 1.22	67 A0 0.99 1.07	68 B6 1.13 1.21	69 A0 0.87 0.94

Figure 4.3-17

ATTACHMENT 7

079: UNANALYZED REACTOR VESSEL THERMAL STRESS DURING NATURAL CONVECTION COOLDOWN

ISSUE

Generic Safety Issue (GSI) 079 in NUREG-0933 (Reference 1), identifies the potential for the stresses in the reactor vessel flange area or studs to exceed the allowable during its design lifetime because of a previously unanalyzed thermal stress introduced by the natural convection cooldown event.

A natural convection cooldown event occurred at the St. Lucie 1 nuclear power generating station. During the course of this event, steam voiding occurred in the reactor vessel head area. Upon analysis, concern was raised over previously unanalyzed reactor vessel thermal stresses. The concern focused on the possible existence of an axial temperature gradient of 150 to 200 degrees F in the vessel flange and studs.

The safety concern arises because this event could produce thermal stresses in the flange area or in the studs that may exceed the ASME B&PV, Section III Code (Reference 2) allowables when added to the stresses already considered. Moreover, the cycling of these temperature gradients over the life of the plant has the potential to cause a reduction in the fatigue margin of the vessel.

ACCEPTANCE CRITERIA

The acceptance criterion for the resolution of GSI 079 is that the design of the reactor pressure vessel (including the head and studs) shall accommodate the thermal stresses caused by a natural convection cooldown event. These thermal stresses, when added to stresses from events that are presently analyzed, shall not exceed the stress limits specified in the ASME B&PV Code, Section III.

RESOLUTION

Stress analyses were performed to determine the effects of a natural circulation cooldown event (similar to that of the St. Lucie occurrence) on both the St Lucie "class" reactor vessel and the System 80 "class" reactor vessel. The analyses concluded that should natural circulation cooldown of the reactor coolant system be required and should vessel head voiding subsequently occur, the resulting thermal stresses would not cause any thermal, hydraulic, or fatigue damage to the reactor vessel and its integral components over their design lifetime. The analyses showed that for an NCC event with 100 drain (void) and fill cycles, the usage factor would be less than 0.0002 for the System 80 reactor vessel.

ECGB

Furthermore, the System 80+ reactor vessel, which is designed to the ASME B&PV Code, Section III (see CESSAR-DC, Section 5.3), is essentially identical to the System 80 reactor vessel. Specifically, the vessels have the same material composition and overall dimensions and are of similar geometry (with the exception of the direct vessel injection nozzles) as described in CESSAR-DC, Table 1.3-1, and Figure 3.9-9. Because the reactor vessels for both "classes" of plants are virtually the same and since the stress analyses consider the materials, dimensions and geometry of the vessel, the analyses performed subsequent to the St. Lucie 1 event apply to the System 80+ reactor vessel.

In summary, the addition of the dynamic, thermal and fatigue effects of a natural convection cooldown on the System 80+ reactor vessel does not result in the vessel stresses or fatigue usage factor exceeding the allowable limits specified in the ASME B&PV Code, Section III. Therefore, this issue is resolved for the System 80+ Standard Design.

REFERENCES

1. NUREG-0933, "A Status Report on Unresolved Safety Issues", U.S. Nuclear Regulatory Commission, April 1989.

2. American Society of Mechanical Engineers, Boiler & Pressure Vessel Code, Section III (Nuclear).

The fluid conditions in the reactor vessel upper head during draining and refilling for NCC are essentially the same. In fact, the stresses on the System 80+ vessel may be less due to a lower initial hot leg and reactor vessel upper head temperature than System 80.

For System 80+, 30 natural convection cooldown events are included in the design bases events for thermal, hydraulic and fatigue analyses. See CESSAR-DC Table 3.9-1, Amendment K. The 30 events are applicable to the 60-year plant design life. Even if all 30 events included the 100 vessel head drain-and-fill cycles described above, the usage factor would be less than 0.006. The 30 NCC events included in the System 80+ design bases are considered conservative in light of the Generic Letter 92-02 statement that NCC events occur infrequently.